Generic Transportation Dose Assessment

NWMO TR-2012-06

October 2012

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ABSTRACT

Title: **Generic Transportation Dose Assessment**
Report No.: **NWMO TR-2012-06**
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Abstract

Canada, following a comprehensive decision making process, has chosen to develop a deep geological repository for the long term management of its used nuclear fuel. That used fuel will have to be safely and securely transported from its current interim storage facilities to the repository. The Nuclear Waste Management Organization (NWMO) is responsible for the transportation of the used fuel and has completed an update of an existing Dose Assessment Report.

The updated information will assist NWMO in evaluating the impact that transport of used nuclear fuel will have on the public and the environment. The findings from this updated assessment will provide a base case for NWMO as it plans a safe and secure transportation system, and will allow the Canadian Nuclear Safety Commission (CNSC) to evaluate compliance with its regulations for protecting public health and safety. The findings from this update will be used to support mode and route analyses, and defining transportation system operating procedures.

The Used Fuel Transportation Package (UFTP) is the reference transportation package for this updated assessment. NWMO is using the UFTP for its reference case in planning and engineering the repository design. NWMO, in order to perform a diligent assessment, conducted an analysis of potential doses that could be received by the public during road and railroad transport and for a hypothetical accident scenario.

In conducting this assessment, three road transport scenarios were identified which represent the range of pathways where the public potentially could receive levels of exposure or dose.

- Individuals residing or working along the transportation route
- Drivers or passengers riding in vehicles stuck in traffic gridlock and sitting next to a truck carrying an UFTP, and
- Persons sharing a rest area with a truck carrying a UFTP.

Similar scenarios were assessed for rail transportation.

The analysis in this report is generic, based on representative assumptions at this stage of planning where specific routes or geographic location of the repository site are unknown. The information will assist NWMO to assess the general level of exposure to the public from the UFTP operating under normal transportation conditions and to highlight opportunities where NWMO can incorporate ALARA principles to further reduce the exposure to the public.

The analysis is not intended to provide an exhaustive analysis of any specific set of scenarios. Scenarios can be defined that could result in different dose levels than those presented in this report. However, a detailed assessment of transportation impact scenarios is premature until

such factors as repository location, transportation mode and routes, population densities, transport package design, etc. are known.

The CNSC radiation dose limit for a member of the public was used to interpret the results of this assessment. The annual maximum dose or exposure rate to the public is specified in the regulations as 1 milliSievert per year (1 mSv/y). To put this number into perspective, the average Canadian receives approximately 1.8 mSv from background or natural radiation sources each year, and the dose from an abdominal X-ray is approximately 0.7 mSv.

A summary of the annual dose which a representative person[1] could receive based on typical transportation scenarios is shown in Table S-1 and Figure S-1. The CNSC regulatory limit and average annual dose from natural background radiation sources in the environment is shown.

The potential dose rates from the transport of used nuclear fuel are well below the CNSC regulatory limit and the dose received from natural sources.

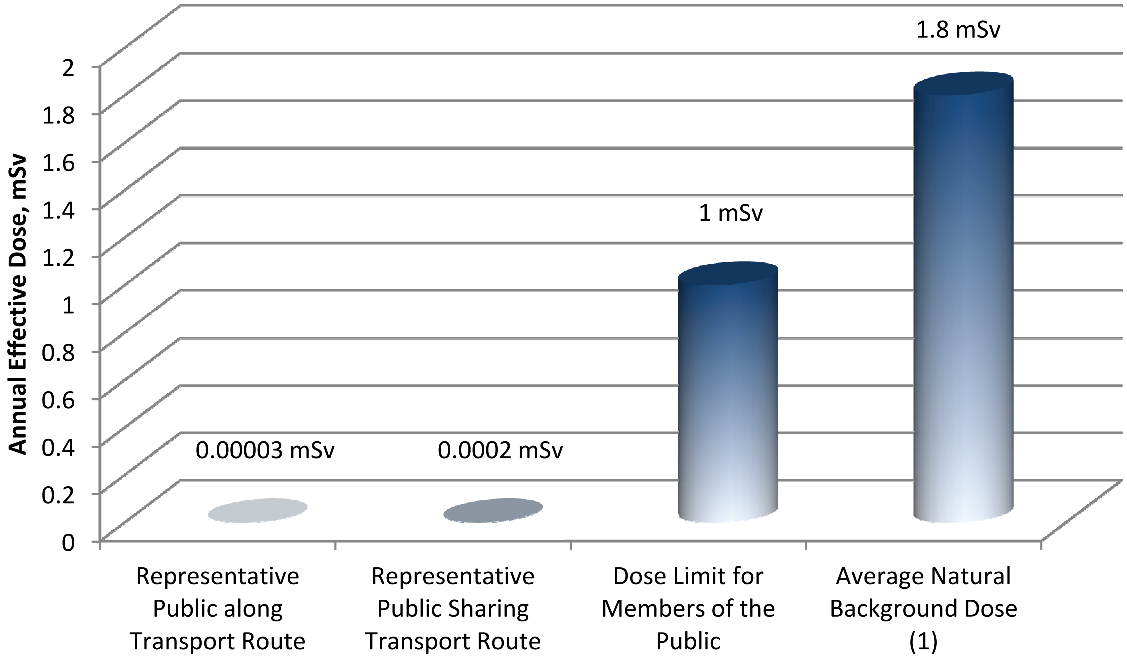
In completing the assessment of potential exposure to the general public, a hypothetical, yet highly unlikely, accident scenario was described and evaluated. This involved a potential release of radioactive material from the UFTP. Again, the standards established by the CNSC were used to conduct the assessment. The potential dose received by a person located near the accident was found to be dominated by direct external radiation exposure from the used fuel inside the transport cask. The dose contribution from potential releases was negligible in comparison.

In all cases, the maximum individual dose to the public under routine transport and accident conditions assessed was found to be less than the CNSC regulatory limit of 1 mSv per year. The public dose values calculated compare well with international experience.

Table S-1: Estimated Dose to a Representative Person due to Road Transport

	Dose [mSv/y]	Assumptions / Comments
Public along Transport Route	0.000032	Person living along route experiencing all 620 shipments
Public in Vehicle sharing Route	0.00018	Individual in vehicle travelling in front or behind cask for 1 hour for 1 shipment per month
	0.00022	Individual in vehicle travelling beside cask for 1 hour (i.e. traffic jam) for 2 shipments per year
Public in Vicinity of Rest Stop	0.00011	Person 10 m from cask for 30 minutes twice per year
Public in Vicinity of Unplanned Stop	0.00013	Person present near cask for 10 hours

1 Recently, the International Commission on Radiological Protection defined the 'representative person' as an individual with characteristics that reflect those of the group that receives the highest doses from a particular source for a given radionuclide. The representative person is equivalent to, and replaces, the 'average member of the critical group', which was previously used as the basis for determining compliance with dose constraints for the public. This new terminology emphasizes that the value is based on a dose calculation for a person, who is almost always a hypothetical construct, since no actual person corresponds precisely to the group average.



Reference:

1. Average Natural Background Dose: Grasty, R.L. & LaMarre, J.R. 2004. The Annual Effective Dose from Natural Sources of Ionising Radiation in Canada. Radiation Protection Dosimetry (2004), Vol. 108, No. 3, pp. 215-226.

Figure S-1: Comparison of Dose to a Member of the Public

ACRONYMS AND DEFINITIONS

ALARA	As low as reasonably achievable. A radiation safety principle for minimizing radiation doses and releases of radioactive materials by employing all reasonable methods, taking into account social and economic factors. ALARA is not only a sound safety principle, but is a regulatory requirement for all radiation safety programs.
APM	Adaptive Phased Management. Canada's plan for long-term management of used nuclear fuel.
Burnup	A measure of how much energy has been extracted from a fuel bundle.
CANDU	CANada Deuterium Uranium. Canadian-invented, pressurized heavy water reactor. The acronym refers to its deuterium-oxide (heavy water) moderator and its use of natural uranium fuel.
CNSC	Canadian Nuclear Safety Commission, Canada's nuclear regulatory agency.
Conveyance	Any vehicle such as a truck, train or ship used to transport radioactive material.
DGR	Deep geological repository.
Effective Dose	A quantity calculated by multiplying the equivalent dose received by every significantly irradiated tissue in the body by a respective tissue weighting factor (this factor reflects the risk of radiation-induced cancer to that tissue) and summing together the individual tissue results to obtain the effective dose. Equivalent dose is the product of the absorbed dose (the energy transferred per unit mass) and a radiation weighting factor, which accounts for differences in biological impact of different radiation types.
IAEA	International Atomic Energy Agency, United Nations nuclear regulatory authority.
ICRP	International Commission on Radiological Protection, an advisory body providing recommendations and guidance on radiation protection.
MCNP	Monte Carlo N-Particle Transport Code. The software code used to calculate dose rates emitted by the UFTP. See Section 3.2 and Appendix A.
mSv	milliSievert. One thousandth ($^1/_{1000}$) of a Sievert. The Sievert is the International System of Units (SI) derived unit of radiation dose.
NEW	Nuclear Energy Worker. A person who is required, in the course of the person's business or occupation in connection with a nuclear substance or nuclear facility, to perform duties in such circumstances that there is a reasonable probability that the person may receive a dose of radiation that is greater than the prescribed limit for the general public.
Type B	Type B packages are required for the transport of highly radioactive material including used nuclear fuel. These packages must withstand the conditions for normal transport and conditions expected in severe accidents without breach of containment.
UFTP	Used Fuel Transportation Package. The transportation package (or cask) used to assess the radiation dose reported in this study. The UFTP is currently certified by the CNSC as a Type B package. The cask was formerly referred to as the Irradiated Fuel Transportation Cask or IFTC.

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1. INTRODUCTION

Used nuclear fuel discharged from Canadian nuclear power reactors is currently stored at licensed interim storage facilities located at the reactor sites where it is produced. Adaptive Phased Management (APM), Canada's plan for long-term management of used nuclear fuel will place the used nuclear fuel into a deep geological repository (DGR). Implementation of APM requires the transport of the used nuclear fuel from the interim storage locations to the repository site.

A previous safety assessment of the used fuel transportation program examined public and occupational doses due to used fuel transportation by the three surface modes of transport, namely road, rail and water. The assessment formed part of the Environmental Impact Statement [1] on the deep geological repository concept for the long term management of Canada's used nuclear fuel which was subject to a federal environmental review process and public hearings in 1996 and 1997.

Recently, the ICRP [2] defined the 'representative person' as an individual with characteristics that reflect those of the group that receives the highest doses from a particular source for a given radionuclide. The representative person is equivalent to, and replaces, the 'average member of the critical group', which was previously used as the basis for determining compliance with dose constraints for the public [3]. This new terminology emphasizes that the value is based on a dose calculation for a person, who is almost always a hypothetical construct, since no actual person corresponds precisely to the group average. This assessment of the radiological dose to representative persons of the public (hereafter referred to as 'members of the public', 'general public', or 'public') due to used fuel transportation is a first step in assessing the radiological implications of the NWMO transportation program and its planning assumptions.

1.1 Assessment Purpose

The purpose of this work is to provide a generic assessment of the potential dose to the public due to used fuel transport in Canada. As the location of the eventual future DGR site is unknown, program specific calculations and risk assessments are premature. The intent of this assessment is to provide a starting point to address public concerns about the safety of the transportation system in a transparent manner. The information will provide a basis for decision making and program planning and to provide an input for design optimization of the components of the transportation system.

1.2 Assessment Scope

This assessment examines potential radiological individual effective dose (hereafter referred to as 'dose') received by members of the public resulting from the transportation of used nuclear fuel. Both routine operations and accident conditions are considered. For the purposes of this assessment, each transportation shipment is considered to start as the cask leaves the nuclear storage facility and is considered to end when the cask is ready to be unloaded at the APM repository site.

The scope of this assessment includes the following activities:

1. Calculation of external gamma and neutron dose rates from the Used Fuel Transportation Package (UFTP) for two reference fuel burnups of 220 and 280 MWh/kgU (see Section 3.2.5.2 for definition of burnup). The used fuel is assumed to be 30 years out-of-reactor. Both road and rail modes of transport are considered.

Note: the UFTP is certified to handle 10 year old fuel [4]. This analysis is based on the reference repository design which is based on 30 year old fuel [5].

2. Calculation of individual dose to members of the public resulting from the transportation of used nuclear fuel under normal and specified accident conditions.
3. Comparison of the results with international experience in radioactive materials shipments.

1.3 Assessment Basis

The calculation of individual dose to members of the public due to used fuel transportation is based on the following:

a) Use of the reference transport cask, the UFTP certified by the Canadian Nuclear Safety Commission (CNSC),

b) The UFTP is filled with 192 CANDU style 37 element used fuel bundles aged 30 years out-of reactor. Used fuel stored at Atomic Energy of Canada Limited sites is typically not in 37 element CANDU fuel bundles and is beyond analyses presented in this report.

c) Two used fuel burnups are considered: a burnup of 220 MWh/kgU used in calculations for normal conditions of transport and a burnup of 280 MWh/kgU used in calculations for accident conditions of transport. All irradiated fuel bundles in one UFTP are assumed to have identical burnup. Burnup is discussed in greater detail in Section 3.2.5.2.

d) The road configuration consists of a tractor-trailer loaded with a single UFTP. An annual total of 620 shipments is assumed.

e) The rail configuration consists of a train loaded with 10 UFTPs; 2 per railcar. An annual total of 62 shipments is assumed.

f) Normal and accident conditions of transport as defined by the transportation regulations [6, 7]. The transportation regulations apply a graded approach to the specification of performance standards. These are characterized in terms of three general severity levels:

   1. Routine conditions of transport – *intended to cover the use and transport of packages under everyday/routine operations (i.e. conditions of transport in which there are no minor mishaps or damaging incidents to the packages)*, [8]
   2. Normal conditions of transport – *intended to cover situations in which the package is subjected to minor mishaps or incidents ranging in severity, but would continue its journey after having been subjected to these minor mishaps* (note: routine conditions of transport are a subset of normal conditions of transport) [8], and
   3. Accident conditions of transport – *intended to cover situations in which the package is subjected to incidents or accidents ranging in severity between those having a severity greater than that covered by normal conditions of transport, up to the maximum severity levels imposed by the transportation regulations under the applicable test requirements for the type of package concerned* [8].

2. RADIATION AND RADIOLOGICAL REGULATORY CONTEXT

2.1 Radiation

Radiation emitted by used fuel is in the form of high-energy particles (alpha particles, beta particles and neutrons) and photons (gamma rays). Alpha and beta particles cannot penetrate the thick steel walls of a transport cask. Hence, only gamma and neutron radiation levels present at various distances from a loaded used fuel transport cask are assessed in this document.

In addition, gamma rays may be produced when neutrons are being absorbed when they go through a shield material. This radiation is termed neutron capture gamma radiation and is also accounted for in the assessment.

Radiation dose, measured in Sieverts (Sv)[2], is a measure of the energy absorbed in human tissue and is generally used as an indicator of the potential radiation effect on the human body. Dose rate is the measurement of radiation exposure over a period of time. The dose rate decreases as the distance from the source increases and as protective shielding is added.

2.2 Regulatory Context

In Canada, the Canadian Nuclear Safety Commission (CNSC) sets radiological dose limits to protect the health and safety of persons and the environment. This is done by following the recommendations of the International Commission on Radiological Protection, which comprises some of the world's leading scientists and other professionals in the field of radiation protection, and by using many of the standards and guides of the International Atomic Energy Agency.

Dose limits are established for:

- a member of the general public,
- a "nuclear energy worker", defined as a person who is required, in the course of the person's business or occupation in connection with a nuclear substance or nuclear facility, to perform duties in such circumstances that there is a reasonable probability that the person may receive a dose of radiation that is greater than the prescribed limit for the general public, and
- a transportation package.

2.2.1 Regulatory Dose Limits

Effective dose limits for the public and nuclear energy workers set in the Radiation Protection Regulations [9] are shown in Table 1, below.

Table 1: Regulatory Individual Effective Dose Limits

Person	Period	Effective Dose
Nuclear energy worker, including a pregnant nuclear energy worker	(*a*) One-year dosimetry period	50 mSv
	(*b*) Five-year dosimetry period	100 mSv
Pregnant nuclear energy worker	Balance of the pregnancy	4 mSv
A person who is not a nuclear energy worker	One calendar year	1 mSv

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[2] The unit used in this assessment is the milliSievert (mSv), which is equal to one thousandth ($^{1}/_{1000}$) of a Sievert.

2.2.2 Transportation Dose Limits

The transportation regulations set dose limits for transportation packages and the conveyances in which the packages are transported. Additionally, a distinction between shipments under exclusive use and not under exclusive use is made. Exclusive use is used to define shipments by a sole consignor with control over the shipment and all initial, intermediate and final loading and unloading operations [8] and describes NWMO's current planning assumptions for operating the transportation system. Shipments where these conditions cannot be met are not under exclusive use.

2.2.2.1 Dose Limits for Transportation Packages

For transportation packages, the radiation level at any point on the external surface of a transportation package shall not exceed 2 mSv/h, and shall not exceed 0.1 mSv/h at 1 m from the surface of the package [7]. For transportation packages under exclusive use, the radiation level at any point on the external surface of the transportation package shall not exceed 10 mSv/h [7].

2.2.2.2 Dose Limits for Conveyances

For conveyances, the radiation level at any point on the external surface of the conveyance shall not exceed 2 mSv/h, and shall not exceed 0.1 mSv/h at a distance of 2 m from the surface of the conveyance [7].

3. ANALYSIS METHODOLOGY AND MODELING ASSUMPTIONS

To calculate the potential dose to members of the public resulting from used fuel transportation, the dose rates at various locations and distances are calculated using a representative model of the cask (see Appendix A) in the three-dimensional Monte Carlo N-Particle transport code MCNP5 [10]. The dose rates calculated by the computer code are used as inputs to various transportation scenarios selected to represent typical interactions between the public and the used fuel. These include exposure to individuals living or working along the transport route, individuals in vehicles sharing the road, individuals present at stops along the transport route and transport accidents.

3.1 Transport Scenarios

Radiological dose to an individual from the UFTP is dependent on a number of factors which include: radioactivity of the radiological source (the used fuel); shielding present around the source (the transport cask); proximity to the radiological source; duration of exposure; and frequency of occurrence. To illustrate the range of potential doses to the public, this assessment examines dose due to used fuel transportation in the following categories:

a) An individual living or working along a transport route;

b) An occupant of a vehicle travelling the same transport route;

c) An individual in the vicinity of a stopped used fuel transport vehicle; and

d) An individual in the vicinity of a transport accident.

An initial set of representative scenarios in each of these categories is assumed and potential dose to the public is calculated for both road and rail transport. The scenarios are also similar to those used in other radioactive material transport risk assessments, such as those discussed in Section 6.3. Rationale for the exposure time, distance and frequency assumptions used in the analysed scenarios are detailed in Appendix A-5.

Transport accidents considered in this assessment are bounded by the tests for demonstrating ability to withstand accident conditions of transport in accordance with the Canadian [6] and International Transport Regulations [7]. Assumed transport scenarios are described in more detail in Section 3.1.1 for road transport and Section 3.1.2 for rail transport. Transportation accident scenarios are described in Section 3.1.3.

3.1.1 Road Transport

The weight of the fully loaded UFTP is such that only one cask can be transported on a tractor-trailer at a time in order to meet provincial transportation regulations. Hence, the road transport vehicle configuration consists of a tractor-trailer loaded with one UFTP enclosed under a weather cover. The weather cover assumed also provides shielding for neutron radiation emitted by the used fuel in the transport cask. The assumed configuration is illustrated in Figure 1. The facilities at the repository site are designed to process approximately 120,000 used fuel bundles per year resulting in approximately 620 road shipments per year.

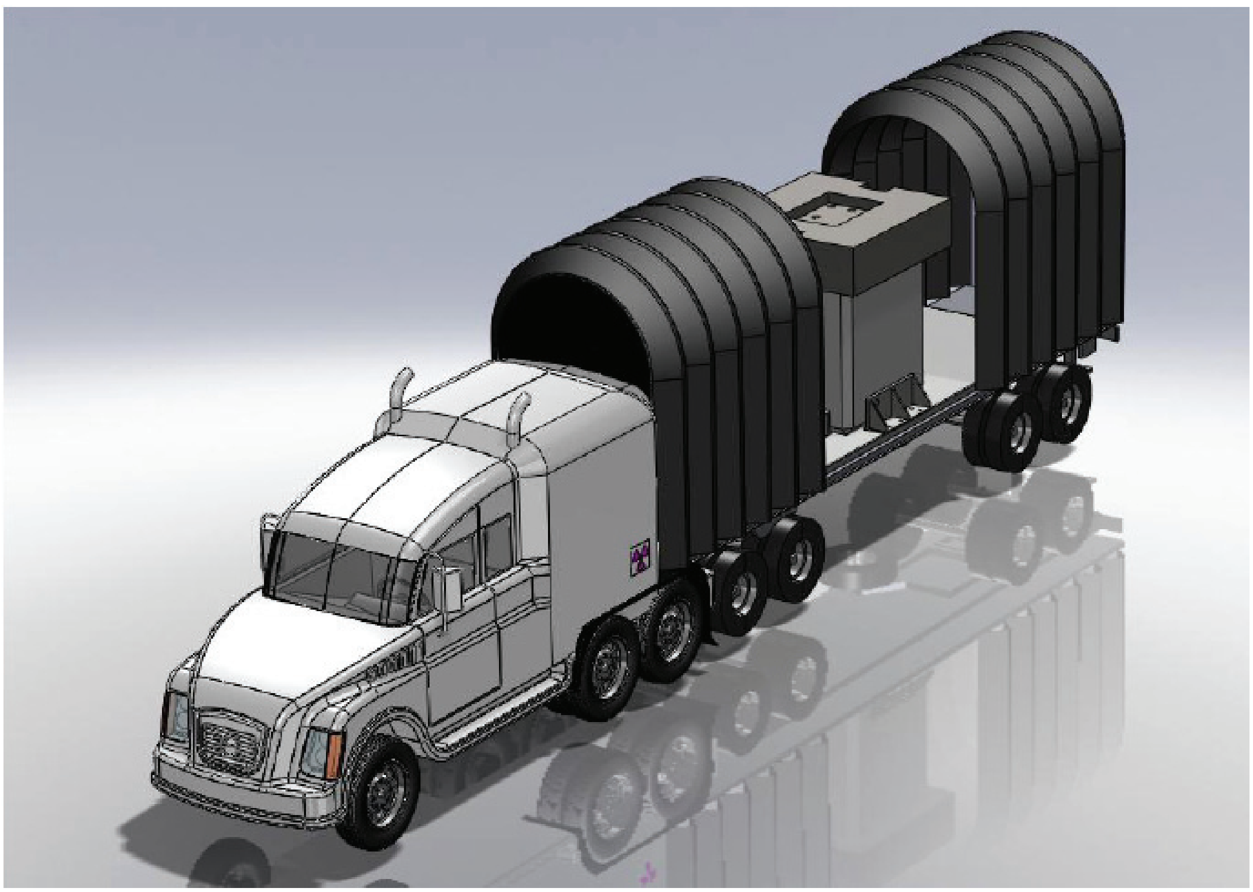


Figure 1: Typical Road Transport Configuration with Weather Cover Open

3.1.1.1 Individual along the Transport Route

Radiation exposure to an individual along a transport route varies with the speed at which the shipment passes by, the distance separating the individual and the cask as it passes by, and the number of shipments experienced by the individual over a period of time. The relationship between dose, distance and transport speed is discussed in detail in Appendix B.

For these calculations, the individual is assumed to be stationary, 30 m from the transport route. The tractor-trailer is assumed to be travelling past the individual at an average speed of 24 km/h (an assumed average speed of a vehicle travelling through a high population area). Conservatively, the individual along the route is assumed to be present for all 620 annual shipments. No credit is given for any incidental shielding (such as the walls of a building) that may be present between the resident and the UFTP.

3.1.1.2 Public Sharing Transport Route

For the public travelling in vehicles sharing the road with a used fuel shipment, two scenarios are investigated.

a) A person in a vehicle travelling directly alongside the UFTP. This scenario was selected to represent a traffic jam or gridlock situation. A vehicle occupant (driver) is assumed to be at a distance of 10 m away from and travelling in a vehicle alongside the cask for a period of 1 hour. This scenario is assumed to occur twice to the same individual over a one year period.

b) A person in a vehicle in front or behind a tractor-trailer carrying the UFTP. A vehicle occupant (driver) is assumed to be 23 m away from the cask and travelling in the same direction for a period of 1 hour. This scenario is assumed to occur six times to the same individual during a one year period.

In either scenario, no credit is given for any incidental shielding (such as the vehicle) that may be present between the receptor and the UFTP.

Exposure to individuals in oncoming traffic is considered to be of shorter duration than those in vehicles travelling in the same direction as the UFTP. Thus, only doses to individuals traveling in the same direction as the UFTP are examined as they are bounding in terms of maximum individual doses.

3.1.1.3 Public Near Rest Stop

A typical rest stop for the transport crew is assumed to be 30 minutes in duration. The tractor-trailer with one UFTP would be parked away from other vehicles and would be continuously monitored by one of the transport crew for security purposes. Rest stops would not involve vehicle refueling. Refueling operations are to take place at purpose-built “card-lock” refueling stations.

For the dose calculations, a member of the public is assumed to be located at a distance of 10 m away from the cask for the entire 30 minute rest stop. This scenario is assumed to occur twice to the same individual over a one year period.

3.1.1.4 Public Near Unplanned Stop

Unplanned stops are an inevitable component of transportation. They may be caused by mechanical breakdowns, a flat tire, weather, etc. Conservatively, an unplanned stop is assumed to be up to 10 hours in duration. For the dose calculations, a member of the public is assumed to be located at a distance of 30 m from the cask for the entire 10 hour unplanned stop. This scenario is assumed to occur once for any given individual during a one year period.

3.1.2 Rail Transport

Rail transport is expected to transport more UFTPs per shipment than the road network. The configuration of a train transporting UFTPs is assumed to consist of two locomotives, five railcars each loaded with two UFTPs with buffer cars on both ends, and a separate coach for the security escorts. Two UFTPs per rail car can readily be accommodated and a train containing 5 railcars conveying a total of 10 UFTPs is assumed. This configuration is illustrated in Figure 2. The facilities at the repository site are designed to process approximately 120,000 used fuel bundles per year resulting in approximately 62 rail shipments per year.

3.1.2.1 Individual along Transport Route

The radiation exposure to an individual living or working along a rail route varies with the speed at which the shipment passes, the distance separating the individual and the 10 casks on the train as they pass by, and the number of shipments experienced by the individual over a period of time. The relationship between dose, distance and transport speed is discussed in detail in Appendix B.

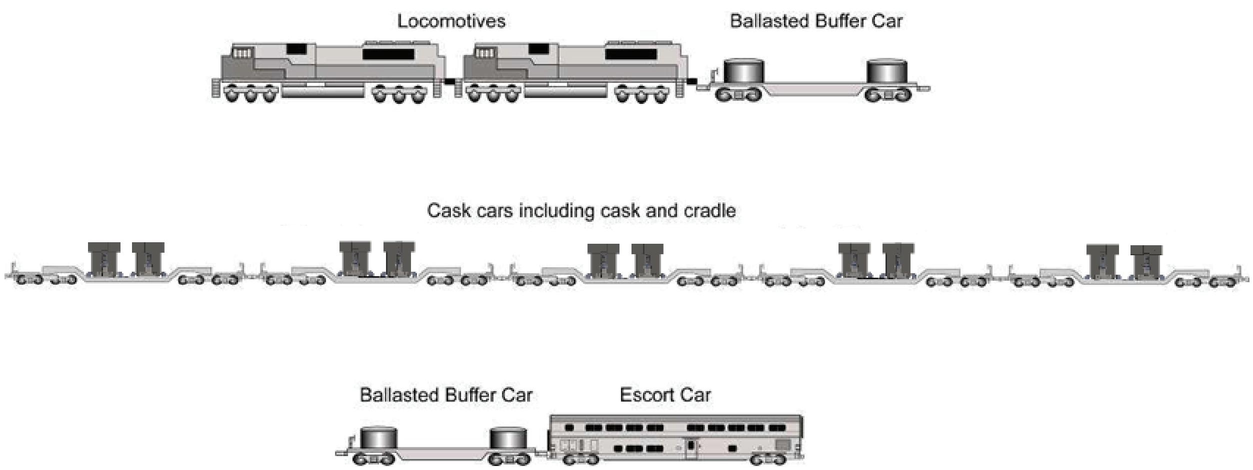


Figure 2: Typical Rail Transport Configuration

For these calculations, the individual is assumed to be stationary for the time the train passes their location (30 m from the rail route). The train is assumed to be travelling past the individual at an average speed of 24 km/h. Conservatively, the individual along the route is assumed to be present for all 62 shipments taking place over a one year period. No credit is given for any incidental shielding (such as the walls of a building) that may be present between the resident and the UFTP.

3.1.2.2 Public sharing Transport Route

In various parts of the country, rail lines run parallel to public roads. These rail lines may be in relatively close proximity to the road making it possible for an individual in a vehicle to travel adjacent to a train for a considerable distance.

For the dose calculations, a member of the public is assumed to be traveling alongside a used fuel rail shipment at a distance of 30 m from the casks in the middle of the train for 10 minutes. This scenario is assumed to occur twice to the same individual over a one year period. No credit is given for any incidental shielding (such as the vehicle) that may be present.

3.1.2.3 Public near Signal Stop

Railways throughout the world use signals to indicate to the train operator-engineer when to proceed. Signals indicate to the engineer the speed at which it is safe to travel. As such, trains will be required to stop at certain signals.

A signal stop along a rail line is assumed to be 10 minutes in duration. In general, locations of signal stops would be known and managed. For the dose calculations, a member of the public is assumed to be located at a distance of 30 m away from the casks placed in the middle of the train for the entire 10 minute signal stop. The same individual is assumed to be present for all 62 shipments taking place over a one year period.

3.1.2.4 Public near Unplanned Stop

Unplanned stops are an inevitable component of transportation. They may be caused by mechanical breakdowns, traffic incidents affecting the rail, etc. Conservatively, an unplanned rail stop is assumed to be 10 hours in duration. For the dose calculations, a member of the public is assumed to be located at a distance of 30 m from the casks placed in the middle of the train for the entire 10 hour unplanned stop. This scenario is assumed to occur once for any given individual during a one year period.

3.1.3 Transport Accidents

Transportation accidents may occur regardless of measures in place to minimize their occurrence. In this assessment, transportation accident severity is assumed to be bounded by the tests for demonstrating ability to withstand accident conditions of transport in accordance with the Canadian [6] and International Transport Regulations [7] (i.e.: the cask is assumed to remain intact with minimal release of content which is less than the regulatory limit). In this assessment, an area of 30 m radius is assumed to be immediately cordoned off around the site. Doses to members of the public located at various distances from 30 m to 800 m from the accident site over an 8 hour period are calculated.

A road accident is assumed to involve a single cask. The weather cover (which is assumed to provide neutron shielding during normal conditions) is assumed to have separated from the vehicle during the accident. A rail accident is assumed to involve two casks where the weather covers have separated from the railcar.

The IAEA regulations require that a Type B package such as the UFTP is able to withstand the cumulative effects of the following tests, in the given order:

a) a 9 m drop onto an unyielding target so as to suffer maximum damage,

b) a 1 m drop so as to suffer maximum damage onto a 15 cm diameter pin rigidly mounted perpendicular to the target, and

c) a 30 minute fire with a temperature of at least 800°C, fully engulfing the specimen.

A Type B package such as the UFTP shall also be immersed under at least 15 m of water for a period of 8 hours and 200 m of water for a period of 1 hour.

The transportation regulations acknowledge that a small amount of leakage through the seals is allowable and define limits for this leakage under both normal and accident conditions. See Appendix C. From the Advisory Material [8]:

> *The design principle embodied in the Transport Regulations is that radioactive release from a Type B(U) package should be avoided. However, since absolute containment cannot be guaranteed, the purpose of specifying maximum allowable 'activity leak' rates is to permit the specification of appropriate and practical test procedures which are related to acceptable radiological protection criteria.*

Based on the UFTP design, this potential leakage though the elastomeric seal between the body of the package and the lid and/or at the vent or drain ports [4] is also considered.

3.2 Description of the Monte Carlo N-Particle Model Components

A computer model of the UFTP was generated using the Monte Carlo N-Particle (MCNP) transport code. The Monte Carlo model used to calculate the radiation emitted by the UFTP consists of four major components: the used fuel (in bundles), the fuel module (the frame in which the used fuel bundles are held); the UFTP, and the weather cover which also provides neutron shielding. Each of these components is described in this section. The Monte Carlo model itself is described in detail in Appendix A.

3.2.1 Used Fuel Bundle

Canadian used nuclear fuel consists primarily of used CANDU fuel which is generated at commercial nuclear power reactors in Ontario, Québec and New Brunswick. In addition, there are very small quantities of used fuel from research and isotope-producing reactors in Canada. A CANDU fuel bundle consists of 28 to 37 elements containing uranium pellets. The elements are manufactured from Zircaloy 4, a zirconium metal alloy. Once filled with uranium pellets, they are sealed at both ends with zircaloy plugs and are welded together into a cylindrical matrix structure with plates at each end. A typical CANDU fuel bundle is shown in Figure 3. Each fuel bundle is approximately 500 mm long, 102 mm in diameter and weighs about 24 kg.

Figure 3: Typical CANDU Used Fuel Bundle

3.2.2 Fuel Module

Used fuel bundles are held inside the UFTP within fuel modules. The fuel module is a rack system currently used by OPG for holding used fuel bundles in interim storage. Each stainless steel fuel module contains 96 fuel bundles stored in linear pairs within 48 horizontal module tubes. These tubes are held in a rectangular framework. A typical module filled with used fuel bundles as illustrated in Figure 4 weighs approximately 2,500 kg.

3.2.3 Used Fuel Transportation Package

The UFTP was designed, tested, manufactured by Ontario Hydro in the mid 1980s. The cask design was first certified in 1985 by the Atomic Energy Control Board (the predecessor of the CNSC) under the regulations in place at that time and has been recertified under those regulations every 4 years since then. The most recent certificate was issued in July 2009 by the CNSC. An upgrade to the certificate demonstrating that the cask design meets current regulatory requirements is being prepared.

The UFTP is in the shape of a rectangular box and is designed to accommodate two stacked modules containing a total of 192 fuel bundles. The body and lid of the cask are manufactured from Type 304L stainless steel. The body is constructed from a single piece forging. The single piece lid is bolted to the body using 32 bolts. The impact limiter is made from redwood encased in a stainless steel shell. It is bolted to the top of the cask to protect the lid/body seal by reducing impact loading in the event of an accident. The impact limiter also serves as a heat shield to protect the cask seal during a fire [4]. An illustration of the UFTP is provided in Figure 4.

The overall cask dimensions are 1,881 x 1,556 x 1,824 mm (excluding protrusions and attachments) and the cask weighs approximately 35 tonnes when loaded with used fuel. The two long walls are 272 mm thick and the base, lid, and short walls are 267 mm thick. The cask can be lifted using trunnions on either side of the body and the lid is lifted via a central flange.

3.2.4 Weather Cover (Neutron Shield)

The stainless steel construction of the UFTP provides a very effective shield for gamma radiation; however, it is less effective for shielding neutrons. Although the UFTP design meets all regulatory dose requirements, a neutron shield (doubling as a weather cover and subsequently referred to as such) could be added around the cask to significantly reduce public and worker exposure. The doses from a UFTP are calculated with and without a weather cover in place. The weather cover assumed is a 10 cm thick high density polyethylene cover placed over top of the UFTP during normal transport operations. For accident scenarios, it is assumed that the weather cover has separated from the transport cask. All dose values for normal transportation conditions are provided with the weather cover in place.

3.2.5 Used Fuel Properties

All used fuel bundles inside the UFTP are assumed to be intact (undamaged) regular-length 37 element bundles. All used fuel bundles are assumed to have uniform decay and discharge burnup (see Sections 3.2.5.1.and 3.2.5.2, below).

3.2.5.1 Used Fuel Cooling Time

All fuel bundles inside the UFTP are assumed to be 30 years out-of-reactor.

3.2.5.2 Fuel Discharge Burnup

The radiation emitted by each fuel bundle after discharge from the reactor is dependent on the length of time spent in the reactor and is characterized by the fuel burnup. Burnup is a measure of how much fission energy has been produced per unit mass of fuel. At long decay time, irradiated fuel bundles with higher fuel burnup will emit more radiation.

Bundle-discharge burnups for Pickering A, Pickering B, Bruce A, Bruce B, and Darlington stations have been examined. A burnup of 220 MWh/kgU represents the highest of the average burnups at these plants [11]. A burnup of 280 MWh/kgU represents the 95th percentile of all discharged fuel bundles from Pickering A, Pickering B, and Bruce A stations; and the 99th percentile of all discharged bundles from Darlington and Bruce B stations.

The burnup for used fuel from Gentilly 2 and Point Lepreau stations is generally below 190 MWh/kgU [12], and hence is bounded by the OPG/Bruce Power values.

For the purposes of this assessment, a burnup of 220 MWh/kgU is assumed for annualized dose calculations. As an accident is assumed to represent a single occurrence, a burnup of 280 MWh/kgU is used for dose calculations in accident scenarios.

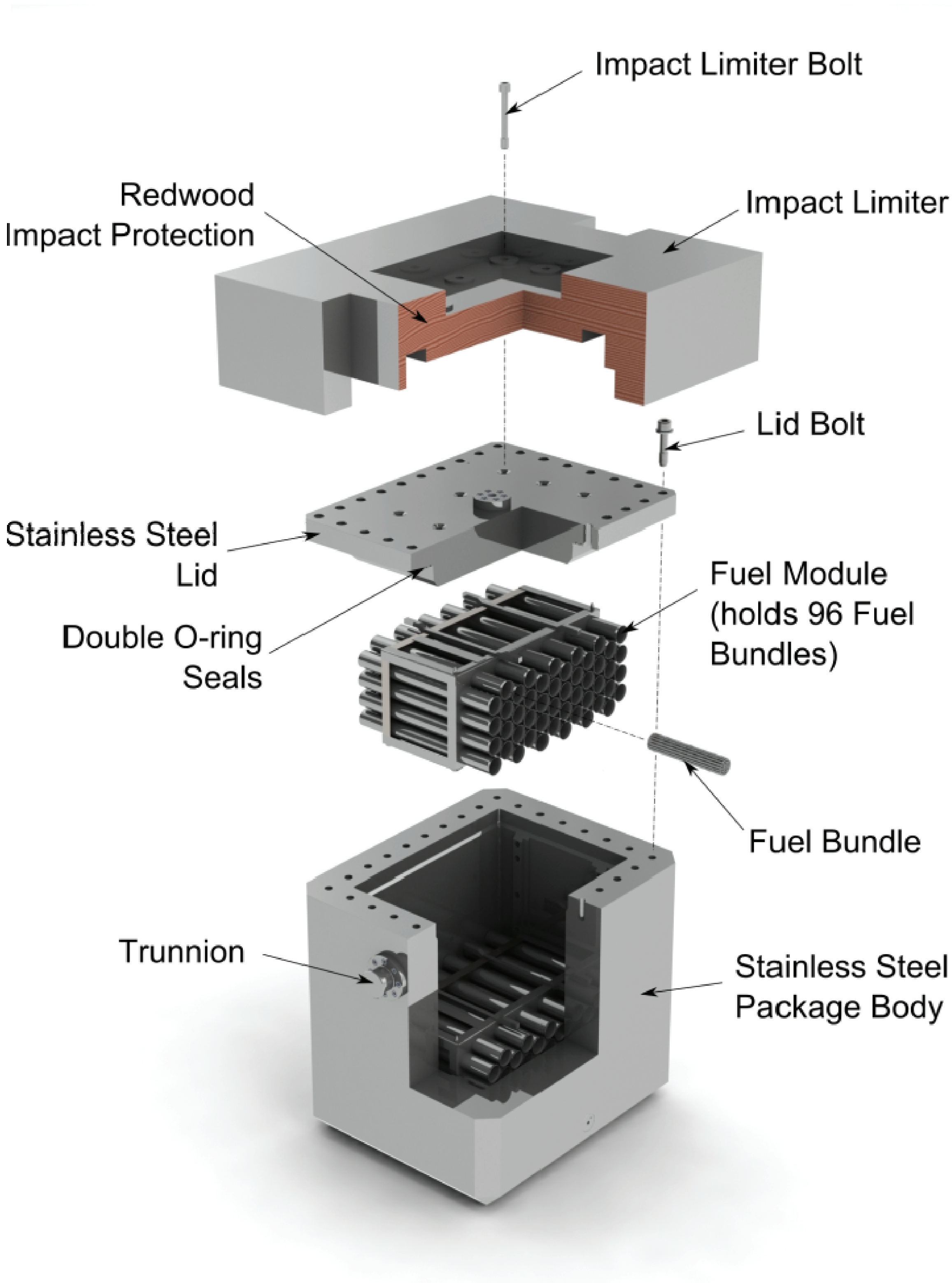


Figure 4: UFTP Assembly

3.2.5.3 Used Fuel Composition

For the purpose of shielding calculations, the fuel bundle physical composition is modeled as pure uranium oxide, i.e., fuel that has not undergone irradiation in a reactor. This assumption has no impact on the self-shielding effect on gamma rays but the neutron self-shielding effect of unirradiated uranium is slightly lower. This results in a conservatism of approximately 10% in the calculation of neutron doses.

3.3 Receptor Locations

The Monte Carlo model calculates dose rates external to the UFTP at discrete distances along lines radiating from 8 receptor locations on two planes: the cask mid-height (shown in red in Figure 5), and the mid-height of the lower fuel module (shown in blue in Figure 5) . The receptor lines are illustrated in Figures 5 and 6. The receptors are located at the cask mid-plane and edge of both the short and long sides. This provides a dose profile of the cask with respect to the used fuel inside. At each receptor location, doses are calculated at the surface, and at distances of 0.3 m, 1 m, 2 m, 3 m, 10 m, and 30 m from the surface.

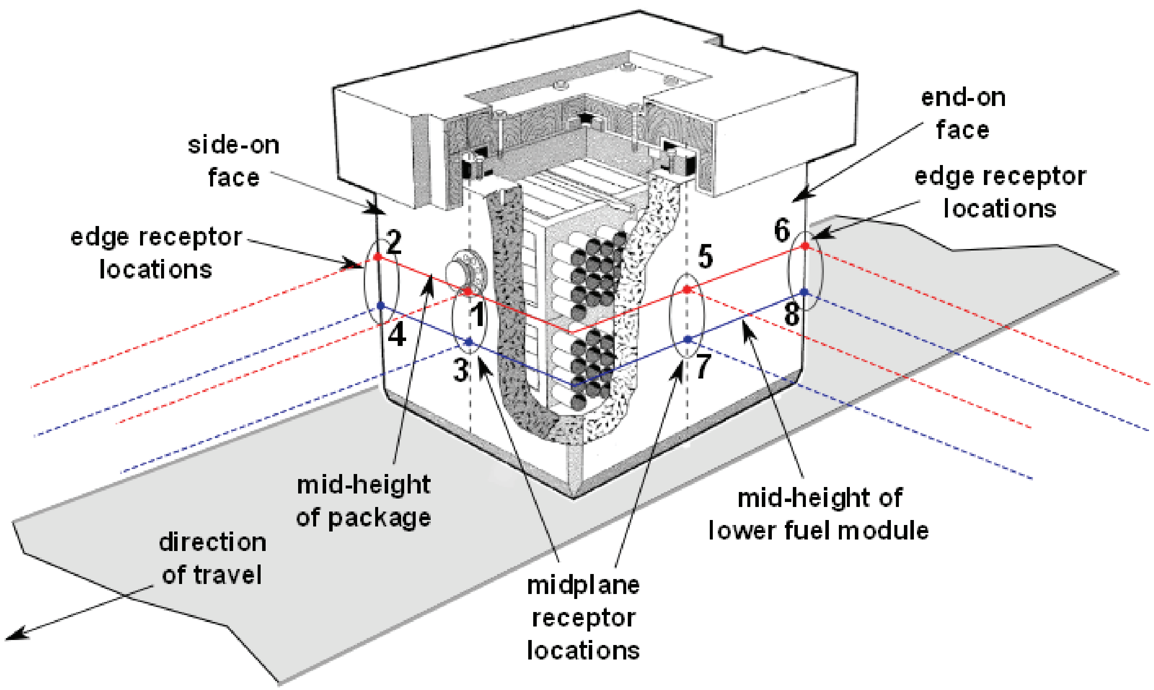


Figure 5: Dose Receptor Locations

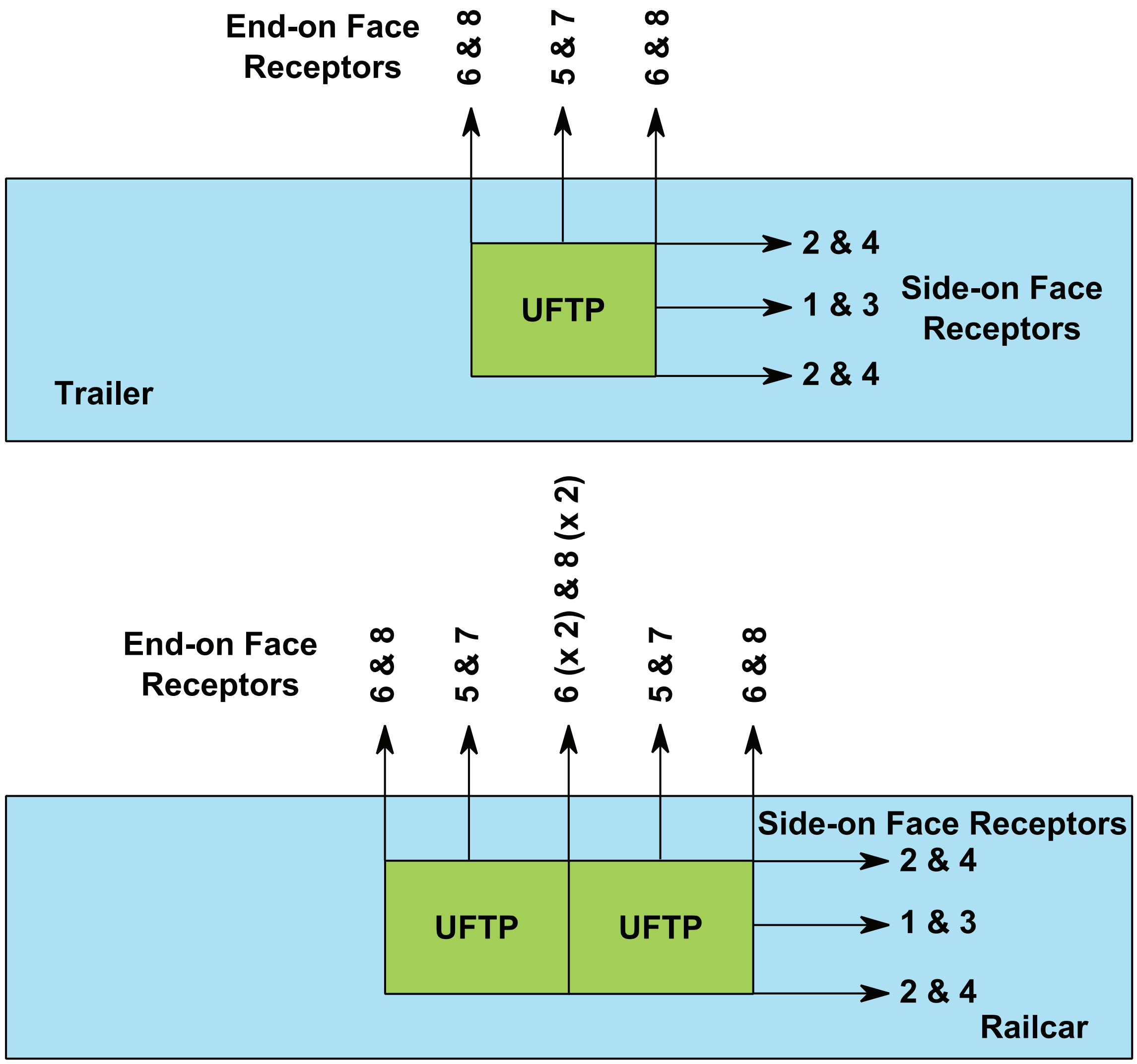


Figure 6: Monte Carlo Model Configuration and Receptor Locations

Note that the fuel bundles are situated parallel to the short wall of the UFTP and the ends of the fuel bundles face the long wall. Doses from the short wall are termed "side-on" and the long wall, "end-on" reflecting the relative location to the used fuel bundle orientation.

In both road and rail configurations, the UFTP is assumed to be situated with the side-on face in the direction of travel. Thus in the case of road transport, individuals travelling in front or behind the tractor-trailer would be exposed to the side-on face of the UFTP, whereas individuals along the transport route and individuals in passing vehicles would be exposed to the end-on face of the UFTP. Dose rates decrease as the receptor location moves from the centre of a UFTP wall because of the increased effective shielding thickness as one moves away from the perpendicular (see Figure 7). Hence if an individual were positioned facing a UFTP corner, even though both walls would be visible, the dose rate would be less than if they were perpendicular to a wall.

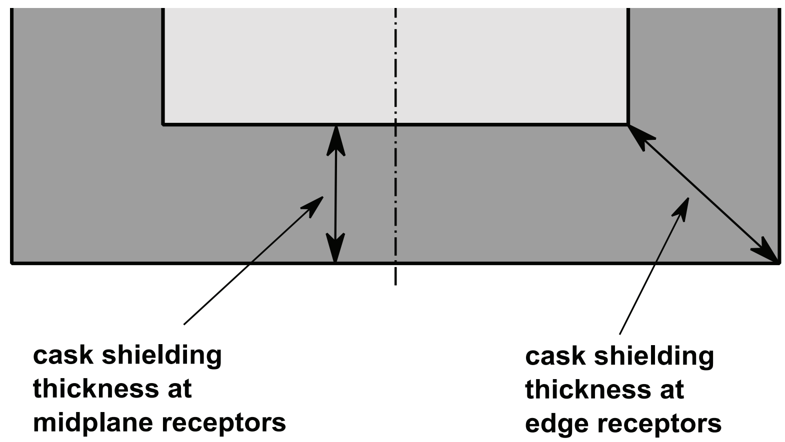


Figure 7: Shielding Thicknesses at Mid-plane and Edge Receptor Locations

3.4 Road and Rail Transportation Configurations

As described in Section 3.1.1, the road configuration consists of a single UFTP per tractor-trailer. The end-on faces of the UFTP are parallel to the direction of travel and the side-on faces are in line with the ends of the truck.

The rail configuration described in Section 3.1.2 consists of two UFTPs per rail car. Only a single railcar is modeled for simplicity. The cask orientation on the railcar is the same as the road transport arrangement: the end-on faces are parallel to the direction of travel and the side-on faces are in line with the ends of the train. Conservatively, the two casks are assumed to be touching each other as shown in Figure 6.

Radiation from the exposed side-on faces of the two cask rail configuration is assumed to be the same as for the single cask configuration. Radiation from the UFTP at the back must pass through its own shielding, in addition to the two shield walls of the UFTP in front and its used fuel contents. It has been shown that the radiation from the back UFTP would be negligible by the time it gets to the front UFTP's external forward face (see Appendices A-1.8 and A-4).

Dose rates at the edge of the cask are used to determine the contributions from a neighbouring cask. Hence, the dose rates at the plane between the casks are doubled to account for the dose-rate contribution from the adjacent cask.

4. MONTE CARLO MODEL DOSE RATE RESULTS

4.1 UFTP Dose Rate Results

The dose rates calculated at the various receptor locations described above are presented in this section. Both road and rail configurations are presented with and without the neutron shielding weather cover in place. These dose rate data are used to calculate individual public dose for the normal transport scenarios defined in Section 3.1 and external doses for the specified accident scenarios.

A comparison of the contact and 1 m dose limits specified in the transport regulations (see Section 2.2.2 and reference [7]) and the calculations for the UFTP (without the weather cover) is provided in Table 2 for 220 MWh/kgU burnup fuel. For the cask, the maximum contact and 1 m dose rates have been calculated to be 0.061 mSv/h and 0.014 mSv/h respectively. These dose rates are significantly less than the regulatory limits of 2 mSv/h at contact and 0.1 mSv/h at 1 m presented in Section 2.2.2.

For 280 MWh/kgU burnup fuel, the maximum calculated contact and 1 m dose rates from the cask without the weather cover are 0.10 mSv/h and 0.026 mSv/h respectively (see Appendix A-3). Again, the dose rates are significantly less than the regulatory limits.

Table 2: Maximum Dose Rates from UFTP (without Weather Cover)

	Dose Rate [mSv/h]		
Distance from Cask	**Regulations**	**220 MWh/kgU**	**280 MWh/kgU**
Contact	2 or 10, under exclusive use	0.061	0.10
1 m	0.1	0.014	0.026

It was found that the dose rates calculated at the mid-height of the package (shown in red in Figure 5) exceeded the dose rates at mid-height of the lower fuel module (shown in blue in Figure 5). Thus, dose rates at package mid-height receptors (receptors 1, 2, 5 & 6) are used in public dose calculations.

4.1.1 Single UFTP on Tractor-Trailer

At the receptor distances assumed in the assessment, the maximum dose rates for the road configuration with the weather cover are calculated to occur at receptor location 1 (centre of the side-on face). The doses at various distances from this location for used fuel with a burnup of 220 MWh/kgU are tabulated in

Table 3 3 below and graphed in Figure 8. Similar tables for the remaining 7 receptor locations are presented in Appendix A-3. Note, all distances are measured from the external surface of the UFTP (the external surface of the weather cover is 0.377 m from the external surface of the UFTP).

Table 3: Dose Rates from UFTP at Receptor Location 1 (with Weather Cover)

	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)		
Distance from Cask	**Neutron**	**Gamma (including capture gammas)**	**Total (±4%)**
1 m	0.00045	0.0056	0.0060
2 m	0.00017	0.0022	0.0024
3 m	0.000094	0.0011	0.0012
10 m	0.000018	0.00011	0.00013
30 m	0.000003	0.000012	0.000015

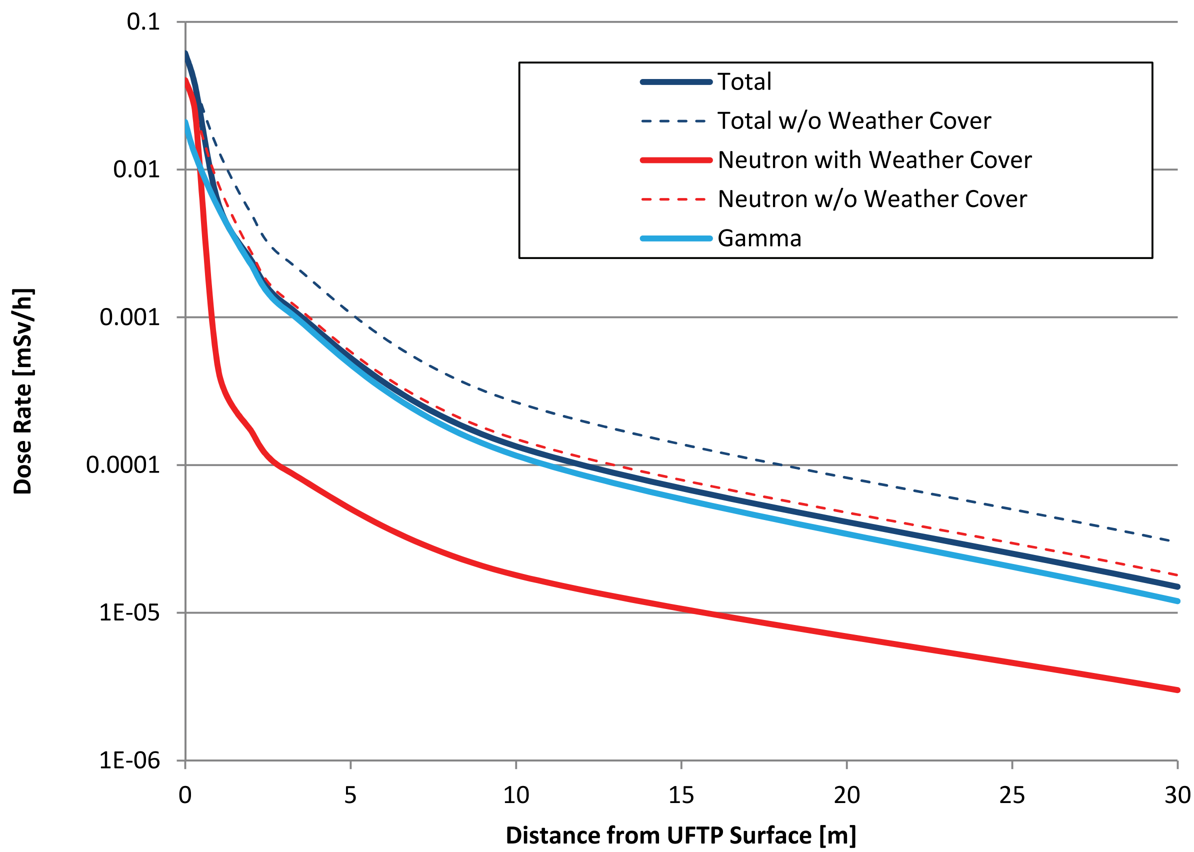


Figure 8: Dose Rates from UFTP at Receptor Location 1

4.1.2 Two UFTPs on Railcar

As discussed in Section 3.4, the dose rates on the side-on face are the same as in the tractor-trailer configuration. See receptor locations 1 through 4 in Figure 6 and Appendix A-3. The dose rate profile along the length of the platform of the rail car is represented by end-on dose rates. From the tables in Appendix A-3, the dose rates at the cask mid-height represent the bounding values.

Comparing the dose rates at receptor location 5 (Appendix A-3, Table A-3-5a) with double the dose rates at receptor location 6 (Appendix A-3, Table A-3-6a), it can be seen that the highest dose rates close to the UFTP occur at receptor location 5, directly facing the mid-plane of the cask, rather than in between the 2 casks. This is due to the fact that the contact dose rate at the edge receptors dips because of the increase in shielding thickness at the corners of the cask (see Figure 8). However, with increasing distance from the cask, the maximum dose rate shifts to the plane between the two casks at receptor location 6 (x 2). A comparison of dose rates at receptor locations 1, 2, 5, 6 and double the dose at receptor location 6 is provided in Figure 9.

All dose rates are significantly less than the regulatory limits for transportation packages (see Section 2.2.2).

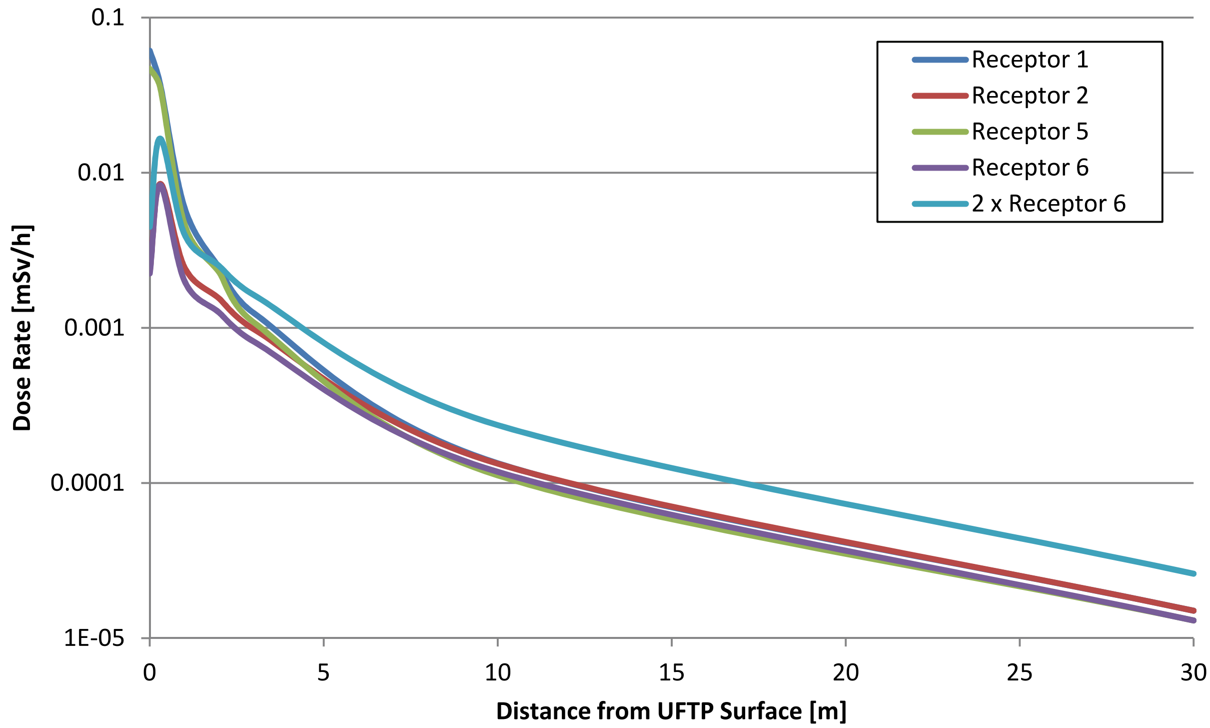


Figure 9: Comparison of Dose Rates at Receptor Locations 1, 2, 5 and 6

5. DOSE CALCULATIONS FOR TRANSPORTATION SCENARIOS

The dose rate results from the Monte Carlo model calculated in Section 4 are used to calculate public dose for each of the normal conditions of transport exposure scenarios presented in Section 3.1 and external radiation dose for the specified accident scenarios. To calculate individual external dose for a given scenario, the dose rate at the distance defined in the scenario is simply multiplied by the exposure time.

For example, from Table 3, the dose rate at 10 m from cask at Receptor location 1 is 0.00013 mSv/h. This means an individual standing 10 m away from the long side of the cask for one hour would receive a dose of 0.00012 mSv. If the individual is present at that location for 15 minutes, they would receive a dose of 0.000033 mSv, one quarter of the hourly dose. Likewise, an individual present at that location for 3 hours would receive a dose of 0.00039 mSv, or three times the hourly dose.

Exposure frequency must also be considered to calculate annual dose. From the previous example, an individual located 10 m away from the long side of the cask for 15 minutes receives a dose of 0.000033 mSv. Assuming the individual experiences this scenario 5 times during the span of a year, the individual then receives 5 times the dose received in the 15 minute exposure, or 0.00016 mSv per year.

As mentioned in Section 3.2.4, the weather cover provides additional shielding for neutron radiation. The weather cover is an integral part of the transportation system and the UFTP will be covered during all shipments.

Calculation of dose to an individual along a transportation route differs somewhat from the calculations described above. In this case, the individual is assumed to be stationary at a given distance from the transport route. The dose rate the individual is exposed to will increase as the vehicle approaches, reaches a maximum when the transport cask(s) is (are) closest to the individual and then decreases as the vehicle continues on its route. The dose received is also dependent on vehicle speed; the faster the vehicle speed, the lower the dose because of the reduced exposure time.

The methodology used to determine dose to an individual along the transport route is presented in Appendix B. A vehicle speed of 24 km/h (the average speed of a vehicle in a high population area) is assumed for the calculation of public dose.

5.1 Dose due to Normal Road Transport

Dose rates used in the calculation of individual dose for each of the road transportation scenarios presented in Section 3.1.1 are tabulated in Table 4 below. These dose rates have been extracted from the Monte Carlo dose calculations for each receptor location presented in Appendix A-3. The dose rates calculated for each road transport scenario are provided in Table 5, and the resulting annual dose is presented in Table 6. An illustration depicting the road transportation scenarios is provided in Figure 10.

In all cases, dose due to the transportation of used nuclear fuel inside a UFTP is well below the dose limits set by the CNSC. The maximum exposure to a member of the public is estimated to occur during a traffic jam. The individual is assumed to be at distance of 10 m from the cask for one hour, twice per year. For this scenario, the dose is a factor of over 8,000 times less than the average annual natural background radiation dose in Canada.

Table 4: Dose Rates used in Road Mode Individual Dose Calculations

Fuel Bundle Burnup 220 MWh/kgU		
Receptor Location	**Distance from Cask**	**Dose Rate (mSv/h)**
1	10 m	0.00013
1	23 m	0.000030
1	30 m	0.000015
5	10 m	0.00011
5	15 m	0.000060
5	30 m	0.000013

Table 5: Public Dose for Road Transport Scenarios

Scenario	Distance from Cask	Exposure	Dose [mSv]	Assumptions / Comments
Public along Transport Route	30 m	1 shipment	0.000000051	Used fuel transport passing a stationary individual at 24 km/h (Location 5)
Public in Vehicle sharing Route	23 m	1 hour	0.000030	Individual travelling in vehicle behind transport cask for 1 hour (Location 1)
	10 m	1 hour	0.00011	Car driving beside cask for 1 hour (i.e. traffic jam) (Location 5)
Public in Vicinity at Rest Stop	10 m	30 minutes	0.000056	Person present near cask (Location 5)
Public in Vicinity of Unplanned Stop	30 m	10 hours	0.00015	Person present near cask (Location 1)

Table 6: Annual Public Dose due to Normal Road Transport

	Distance from Cask	Exposure (Number of Shipments)	Dose [mSv/y]	Assumptions / Comments
Public along Transport Route	30 m	620	0.000032	Individual along route experiencing all 620 shipments
Public in Vehicle sharing Route	23 m	6	0.00018	Individual in vehicle travelling in front or behind cask for 1 hour six times per year
	10 m	2	0.00022	Individual in vehicle travelling beside cask for 1 hour (i.e. traffic jam) for 2 shipments per year
Public in Vicinity at Rest Stop	10 m	2	0.00011	Person 10 m from cask for 30 minutes twice per year

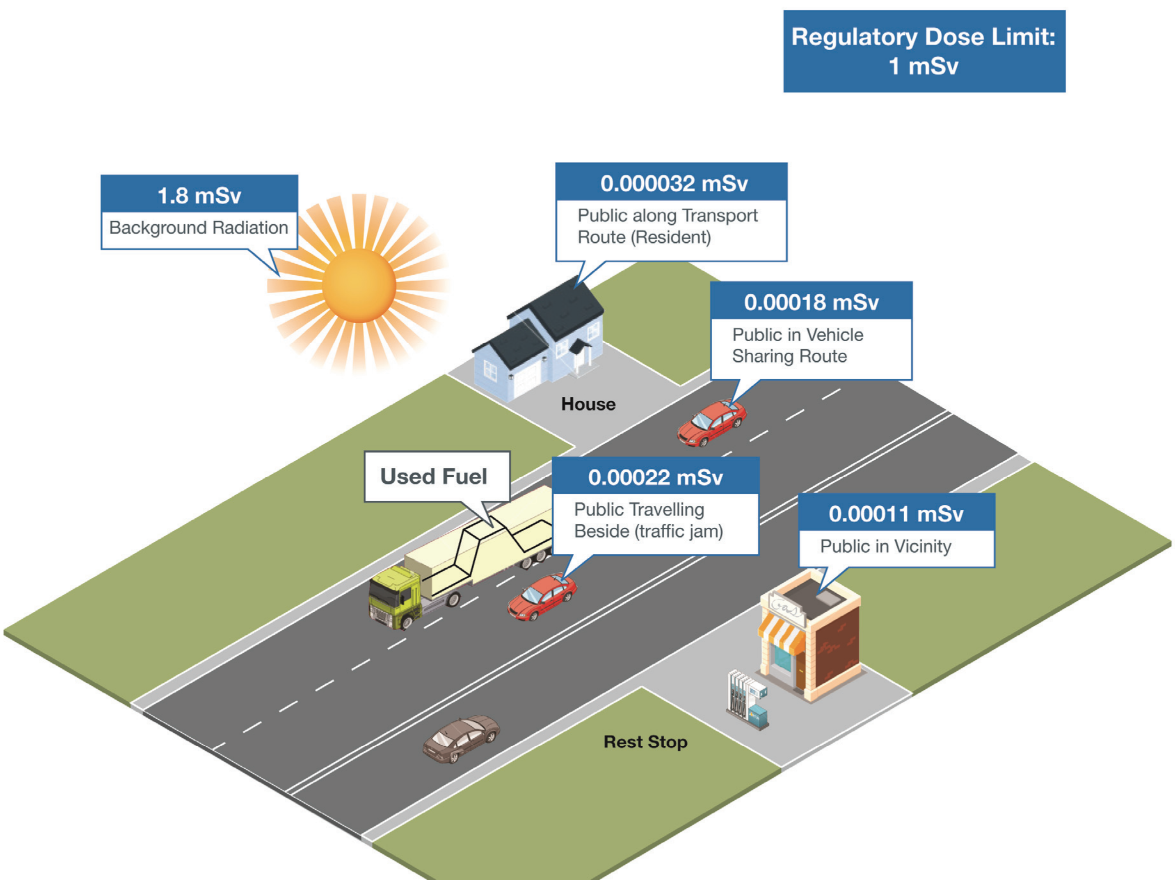


Figure 10: Annual Dose due to Road Transport

5.2 Dose due to Normal Rail Transport

Calculation of individual dose for each of the rail transportation scenarios presented in Section 3.1.2 is based on a single receptor location, Receptor 5 at a distance of 30 m from the middle railcar. The dose rate at this location is tabulated in Table 7 below. The doses calculated for each rail transport scenario are provided in Table 8, and the resulting annual dose is presented in Table 9.

In all cases, dose due to the transportation of used nuclear fuel inside the UFTP is well below the dose limits set by the CNSC. The maximum dose to a member of the public is calculated to be 0.00028 mSv per year to a person near a signal stop. The individual is assumed to be at distance of 30 m from the casks in the middle of the train for the 10 minute stop and to experience all 62 shipments per year. Even in this scenario, the dose is a factor of over 6,400 times less than the average annual natural background radiation dose in Canada.

Table 7: Dose Rate used in Rail Mode Individual Dose Calculations (with Weather Cover)

Fuel Bundle Burnup 220 MWh/kgU		
Receptor Location	**Distance from Cask**	**Dose Rate (mSv/h)**
6 (x 2)	30 m	0.000027

Table 8: Public Dose for Rail Transport Scenarios

Scenario	Distance from Cask	Exposure	Dose [mSv]	Assumptions / Comments
Public along Transport Route	30 m	1 shipment	0.00000051	Used fuel transport passing a stationary individual at 24 km/h
Public in Vehicle sharing Route	30 m	10 minutes	0.0000045	Individual travelling in vehicle alongside train for 10 minutes
Public in Vicinity of Signal Stop	30 m	10 minutes	0.0000045	Person present near casks
Public in Vicinity at Unplanned Stop	30 m	10 hours	0.00027	Person present near casks

Table 9: Annual Public Dose due to Normal Rail Transport

Scenario	Distance from Cask	Exposure (Number of Shipments)	Dose [mSv/y]	Assumptions / Comments
Public along Transport Route	30 m	62	0.000032	Individual along route experiencing all 62 shipments
Public in Vehicle sharing Route	30 m	2	0.000009	Individual travelling in vehicle alongside train for 10 minutes
Public in Vicinity at Signal Stop	30 m	62	0.00028	Person living near signal stop as all 62 shipments pass by

5.3 Dose due to Transport Accident

The ability of the cask to withstand the regulatory tests designed to simulate real world accident conditions was demonstrated through physical and analytical testing in the safety report [4]. Therefore in this assessment, the cask is assumed to be intact following an accident. The potential radiological dose to a member of the public in the unlikely event of an accident was assessed consistent with the regulatory test conditions and the assessment considers the following sources of radiation:

- radiation from the used fuel within the cask;
- radiation from an allowed minimal release of gas though the elastomeric seals; and
- an internal source of radiation due to the inhalation of this material.

The calculation to determine external dose is identical to the calculations for dose due to normal road transport in Section 4.1 with two exceptions. In this assessment, the cask is assumed to contain used fuel with a burnup of 280 MWh/kgU for the accident analysis rather than the burnup of 220 MWh/kgU as used in the analysis for normal conditions of transport, see Section 3.2.5.2. And the weather cover is assumed to have separated from the cask and is no longer able to provided neutron shielding. Dose rates at various distances along each receptor for both used fuel burnups are tabulated in Appendix A-3.

As the accident conditions stipulated in the transportation regulations are not expected to breach the cask seal, only permeation leakage of radionuclides in a gaseous form from within the cask is considered. From the Safety Analysis Report for the cask [4], two gaseous radioisotopes present in the cask, namely Krypton-85 (Kr-85) and Hydrogen-3 (H-3, known as ‘tritium’), were found to be important. The UFTP radioactive material inventory used in this assessment includes additional radioisotopes of elements that will or may be gaseous under the accident conditions considered, however due to their relative low radioactivities compared to Kr-85 and H-3 and very low dose per activity uptake, the dose contribution from these radionuclides would be much less than from Kr-85 and H-3.

The assumed leakage rates of Kr-85 and H-3 under accident conditions is based on the testing and assessment of the UFTP design [4]. These calculations have been adjusted to account for the UFTP radioactive material inventory of fuel aged 30 years out-of-reactor [11] and more recent effective free inventory fraction assumptions made in references [13] and [14]. The gaseous radioactive material would be released at a very slow rate. It is assumed that the public would be cleared from the immediate area, thus the closest member of the public exposed would be located at a distance of 30 m. It is also assumed that the public within an 800 m radius from the accident site would be exposed to any potential release for 8 hours.

Using the calculated release rate and exposure time, the methodology detailed in the Canadian Standards Association (CSA) Guidelines N288.2-M91 (Reaffirmed 2008) [15] was used to calculate doses to members of the public under the specified accident conditions. Gaseous material released from the cask is assumed to be dispersed by ambient weather conditions. Conservatively, weather conditions leading to minimal dispersion (i.e. minimal dilution of radioactivity concentration in the air; less dispersion equals greater dose) were assumed in this assessment.

Calculation of public dose resulting from accident conditions are described in detail in Appendix C. The results for an accident involving a UFTP containing used fuel of burnup 280 MWh/kgU for road transport are shown in Table 10 and Figure 11 for the UFTP only (without weather cover). As stated in Section 3.2.4, the weather cover is assumed to separate from the cask during the accident and is not credited for providing any additional shielding.

Table 10: Potential Doses from Specified Road Transport Accident Conditions

Distance – UFTP to Receptor	30 m	40 m	50 m	75 m
Dose from H-3 [mSv]	0.000026	0.000016	0.000011	0.000005
Dose from Kr-85 [mSv]	0.00000007	0.00000004	0.00000003	0.00000002
External Dose from material remaining in UFTP [mSv]	0.00046	0.00026	0.00016	0.000073
Total Dose [mSv]	0.00048	0.00027	0.00018	0.000078

Distance – UFTP to Receptor	100 m	300 m	500 m	800 m
Dose from H-3 [mSv]	0.000003	0.000000	0.000000	0.000000
Dose from Kr-85 [mSv]	0.00000001	0.00000000	0.00000000	0.000000
External Dose from material remaining in UFTP [mSv]	0.000041	0.000005	0.000002	0.000001
Total Dose [mSv]	0.000044	0.000005	0.000002	0.000001

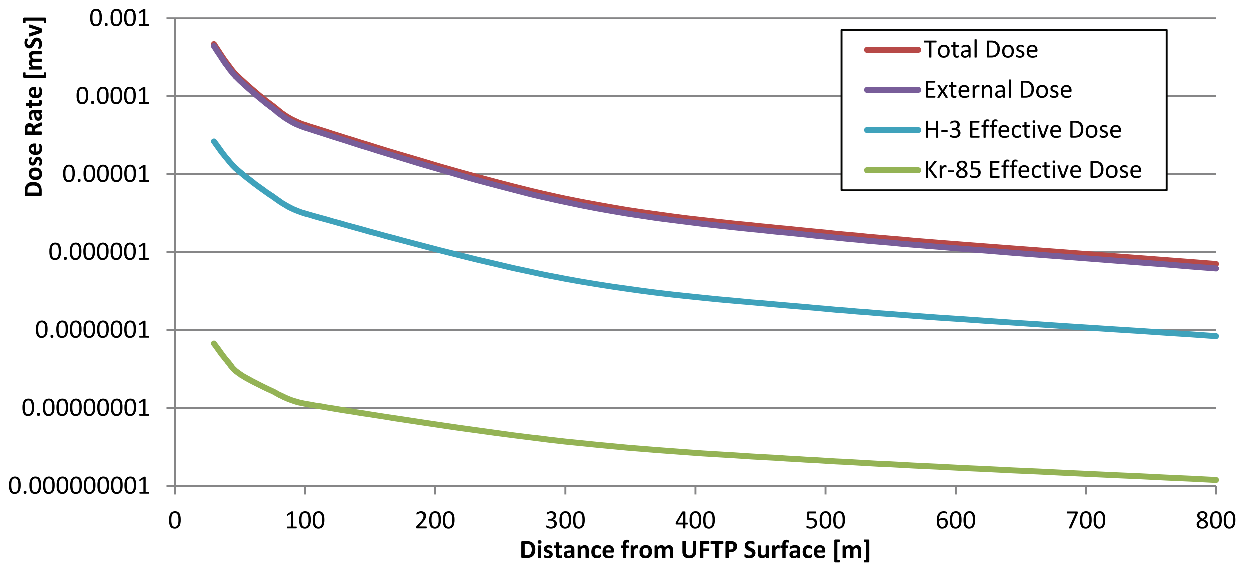


Figure 11: Potential Doses from Specified Road Transport Accident Conditions

For rail transport, the same methodology as for road transport was used, with the exception that two UFTPs are assumed to be affected by the accident (even if all 10 casks transported on the train to be affected, the dose contribution from potential releases is less than the direct dose rate from one cask). The potential gaseous radioactive material release inventories are doubled for the two casks affected. The results for a rail accident involving UFTPs containing used fuel of burnup 280 MWh/kgU for road transport are shown in Table 11 and Figure 12. The total dose resulting from an accident, either road or rail, is dominated by direct external radiation from radioactive material remaining within the UFTPs; the dose contribution from potential releases is negligible in comparison.

Table 11: Potential Doses from Specified Rail Transport Accident Conditions

Distance – UFTP to Receptor	30 m	40 m	50 m	75 m
Dose from H-3 [mSv]	0.000053	0.000032	0.000021	0.000010
Dose from Kr-85 [mSv]	0.00000013	0.00000008	0.00000005	0.00000003
External Dose from material remaining in UFTP [mSv]	0.00091	0.00051	0.00033	0.00015
Total Dose [mSv]	0.00097	0.00055	0.00035	0.00016

Distance – UFTP to Receptor	100 m	300 m	500 m	800 m
Dose from H-3 [mSv]	0.000006	0.000001	0.000000	0.000000
Dose from Kr-85 [mSv]	0.00000002	0.00000001	0.000000	0.000000
External Dose from material remaining in UFTP [mSv]	0.000082	0.000009	0.000003	0.000001
Total Dose [mSv]	0.000088	0.000010	0.000004	0.000001

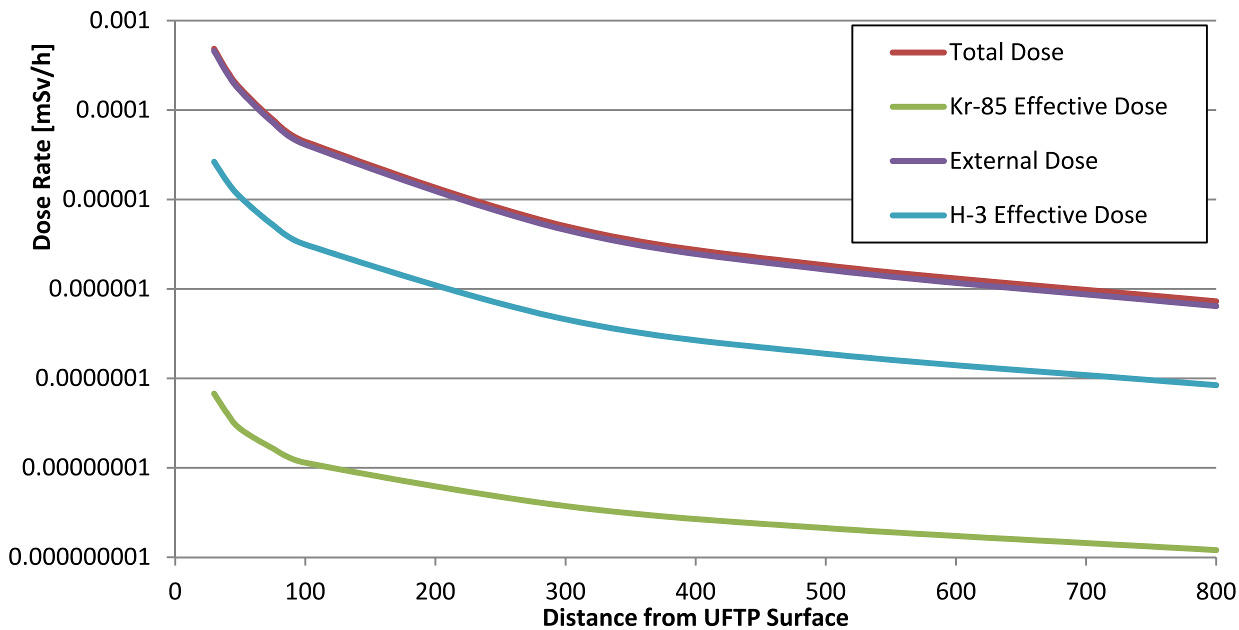


Figure 12: Potential Doses from Specified Rail Transport Accident Conditions

The highest dose to a member of the public due to the transportation accident scenarios assessed results from an individual located at a distance of 30 m from a rail accident for 8 hours. Even in this scenario, the dose exposure is a factor of over 1,800 times less than the average annual natural background radiation dose in Canada.

In all cases, public dose from the specified accident conditions due to the transportation of used nuclear fuel are well below the dose limit of 1 mSv per year set by the CNSC.

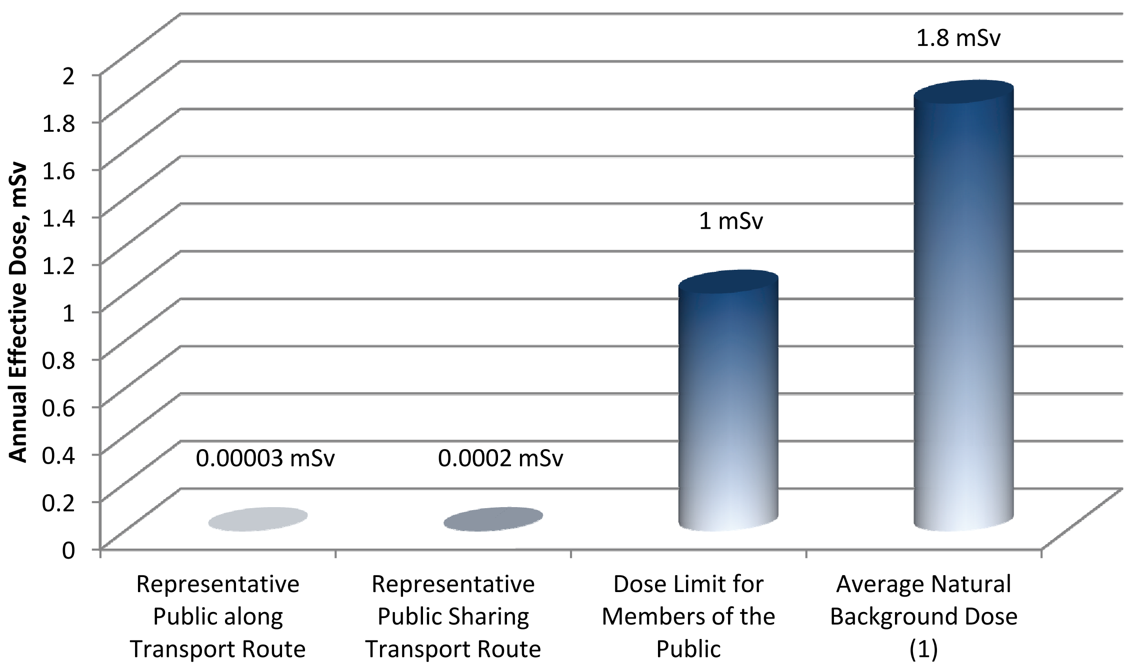
6. DISCUSSION OF RESULTS

6.1 Dose Assessment Summary

As described in Section 5, doses calculated for individuals due to the transportation of used nuclear fuel are very low and well below regulatory limits and natural or background radiation levels.

As shown in Tables 5 and 8, the dose to an individual along a transportation route exposed to all shipments is considerably less than that to an individual sharing the transport route or at a transport stop. Even though the number of casks the individual along the transport route is exposed to is high (620 casks), the total exposure time is very small. See Appendix B.

An individual along a transportation route has been calculated to receive a dose of 0.000032 mSv per year due to transportation, a factor of safety of 30,000 compared to the regulatory limit of 1 mSv. The relationship between regulatory dose, average natural background dose, and potential annual doses to an individual along a transportation route and a member of the public in a vehicle sharing the transportation route is shown in Figure 13.



Reference:

1. Average Natural Background Dose: Grasty, R.L. & LaMarre, J.R. 2004. The Annual Effective Dose from Natural Sources of Ionising Radiation in Canada. Radiation Protection Dosimetry (2004), Vol. 108, No. 3, pp. 215-226.

Figure 13: Comparison of Dose to a Member of the Public

6.2 Additional External Shielding

The Monte Carlo calculations (see Figure 8) show that the cask itself provides very effective shielding for gamma radiation, but is less effective for neutron radiation. Although radiation levels external to the cask are well below regulatory limits, the addition of a 10 cm thick layer of high density polyethylene inside the weather cover as a neutron shield further reduces the radiation levels by approximately 75%. This additional shielding is simple and cost effective to implement and could be considered in As Low As Reasonably Achievable (ALARA) assessment of the final used fuel transportation system.

6.3 Comparison of Dose Results with Ontario Power Generation Environmental Impact Statement and International Experience

A comparison of several radioactive material transportation risk assessments was conducted and is discussed in Sections 6.3.1 to 6.3.5 below. It should be noted that only dose results are compared; the inputs and assumptions in each comparison case are different from the current assessment (e.g. type and quantity of fuel, transport container specifics, number of shipments, etc), and hence direct comparison may not be valid.

6.3.1 Ontario Hydro Nuclear 1994 Assessment

A previous safety assessment of the used fuel transportation system [1] by Ontario Hydro Nuclear (OHN, now known as Ontario Power Generation) examined transportation to a central repository site using several modes of transport including all road, mostly rail and mostly water.

The previous transportation safety analysis formed part of the Environmental Impact Statement [1] on the repository concept which was subject to a federal environmental review process and public hearings in 1996 and 1997.

The comparable OHN 1994 assessment results and corresponding current assessment results are shown in Table 12.

Table 12: Comparison of Maximum Individual Annual Dose Results – OHN 1994 Assessment

		Maximum Individual Annual Dose [mSv]	
Mode	**Receptor**	**OHN 1994 Assessment**	**Current Assessment**
Road	Person present at a truck stop	0.070	0.00011
Road	Individual along the route	0.0041	0.000032
Rail	Individual along the route	0.0003	0.000032

All of the results for the current assessment are less than those reported in the OHN 1994 assessment. The differences in the input assumptions are briefly described below.

For truck stops, the OHN 1994 assessment assumed that the maximally exposed individual would be at a distance of 25 m from the UFTP transport vehicle, for 50% of the time, and that one-third of the shipments would stop for one hour at any particular truck stop. The current assessment assumes a member of the public is located a distance of 10 m away from the package for two 30 minute rest stops per year.

For individuals along the road routes, the OHN 1994 assessment dose was calculated for a distance of 8 m from the shipment path to the individual. The corresponding distance in the current assessment is 30 m.

The assumptions used in the rail calculations for individuals along the route in the OHN 1994 assessment are not specified in the documentation.

6.3.2 United States Department of Energy (DOE) Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain

The US DOE Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain [16] included an assessment of doses from shipments of spent nuclear fuel by truck and rail from 72 commercial and 5 DOE sites throughout the United States to a geologic repository at Yucca Mountain, Nye County, Nevada.

The Yucca Mountain assessment included estimates of maximum individual doses. A comparison of relevant receptors between the Yucca Mountain and current assessments is shown in Table 13.

Table 13: Comparison of Results for Maximum Individual Annual Doses – Yucca Mountain

Mode	Receptor	Maximum Individual Annual Dose [mSv]	
		Yucca Mountain	Current Assessment
Road	Individual along route	0.0025	0.000032
Road	Person in traffic jam [a]	0.16	0.000112
Road	Rest stop – public in vicinity	0.67	0.00011
Rail	Individual along route	0.00067	0.000032

[a] Single shipment dose rather than annual dose.

All of the results for the current assessment are less than those of the Yucca Mountain assessment.

The time spent in the traffic jam for the Yucca Mountain assessment was not specified, and may be greater than that assumed for the current assessment. For the rest stop – public in vicinity, the Yucca Mountain assessment assumed that the same person is present at the same stops for 50% of all shipments, whereas this assessment assumes a member of the public is located a distance of 10 m away from the package for two 30 minute rest stops per year. Input assumptions used in the Yucca Mountain assessment for the other scenarios compared were not found available in the public domain.

6.3.3 United Kingdom National Radiological Protection Board Assessment of Radiation Risks to Transport Accident Responders

The United Kingdom National Radiological Protection Board (NRPB) conducted an assessment of radiation risks to emergency service personnel responding to radiation transport accidents for road and rail routes [17]. It was found that the maximum dose from the release of radioactive materials to a responder at the outer cordon set up at a distance of 45 m around an accident involving MAGNOX reactor used fuel in a Type B container would be 0.0000009 mSv. This can also be considered the maximum dose to a member of the public. The corresponding dose at this distance in this assessment is 0.000013 mSv. The UK assessment assumed an exposure time of 1 hour, whereas the current assessment assumed 8 hours. Factoring in the difference in exposure time, the current assessment value would be approximately twice the NRPB estimate.

6.3.4 World Nuclear Transport Institute Study

The World Nuclear Transport Institute (WNTI) conducted a global assessment in 2006 of the likely doses to various types of workers in nuclear fuel cycle transport operations and also to members of the public based mainly on the extensive experience of actual operations [18].

A comparison of relevant receptors between the WNTI and current assessments is shown in Table 14.

Table 14: Comparison of Results for Maximum Individual Annual Doses – WNTI

Mode	Maximum Individual Annual Dose [mSv]	
	WNTI	Current Assessment
Road	<0.0040	0.00022
Rail	<0.0060	0.00028

The current assessment numbers are smaller than the WNTI assessment numbers, however the WNTI values are ‘less than’ (denoted by the symbol ‘<’).

6.3.5 Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Paper on Transport of Radioactive Material in Germany

GRS conducted a study on radiation exposures of workers and the public associated with the transport of radioactive material in Germany [19].

A comparison of relevant receptors between the GRS and current assessments is shown in Table 15.

Table 15: Comparison of Results for Maximum Individual Annual Doses – GRS

Mode	Maximum Individual Annual Dose [mSv]	
	GRS	Current Assessment
Road	0.01	0.00022
Rail	0.01	0.00028

The results for the current assessment are less than the GRS assessment values. Details on the relevant GRS factors (e.g. distances, exposure times, etc) were not provided in the GRS paper; hence direct comparisons may not be valid.

6.3.6 Summary of Comparison with Dose Results with International Experience

A summary of the comparisons presented in this section is provided in Table 16.

Table 16: Summary of Comparison with Dose Results with International Experience

Mode	Receptor	Maximum Individual Annual Dose [mSv]					
		Current Assessment	OHN 1994	Yucca Mountain	UK NRPB	WNTI	GRS
Road	Individual along route	0.000032	0.0041	0.0025	-	-	-
Road	Person in traffic jam [a]	0.00013	-	0.16	-	-	-
Road	Rest stop – public in vicinity	0.00011	0.070	0.67	-	-	-
Road	Individual near accident	0.000013	-	-	0.0000009	-	-
Road	Maximum of all receptors	0.00022	-	-	-	<0.0040	0.01
Rail	Individual along route	0.000032	0.0003	0.00067	-	-	-
Rail	Maximum of all receptors	0.00028	-	-	-	<0.0060	0.01

[a] Single shipment dose rather than annual dose.

7. CONCLUSIONS

7.1 Dose Rates from the UFTP

Dose rate profiles of the UFTP containing used fuel with burnups of 220 and 280 MWh/kgU were calculated using the Monte Carlo N-Particle computer code. It was determined that the dose rates emitted by the UFTP for both used fuel burnups are extremely low and are significantly below CNSC regulatory limits.

7.2 Public Dose due to Used Fuel Transport

Doses for groups of most exposed individuals were calculated including: individuals along the transport route; persons sharing the transport route; and persons at rest/signal stops. A maximum individual dose of 0.00028 mSv was calculated to result from residing near a rail transportation signal stop, significantly below average natural or background radiation levels [20] and typical X-ray examinations [21].

Dose resulting from potential transport accidents bounded by the conditions defined in the Canadian and International Transport Regulations were also assessed. It was determined that the potential dose received by a person near such an accident is dominated by direct external radiation exposure from the used fuel inside the transport cask, with the dose contribution from potential releases being negligible in comparison.

In all cases, the maximum individual dose to the public under routine transport and accident conditions was assessed to be less than the regulatory limit of 1 mSv per year.

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APPENDIX A THE MONTE CARLO MODEL OF THE UFTP AND DOSE RATE RESULTS

CONTENTS

A-1 Description of the Monte Carlo Model

Monte Carlo N-Particle (MCNP) transport code is a general-purpose computer code used for calculation of neutron, photon, electron, or coupled neutron/photon/electron transport. The code is used in many areas of application, including radiation protection, radiation shielding, radiography, medical physics, and nuclear criticality safety. In this case, the code is used to calculate dose rates at various locations and distances from the UFTP based on the flux spectra. A detailed description of the Monte Carlo model is provided in this appendix. MCNP5 version 1.3 is used in the calculations.

MCNP was originally developed at the Los Alamos National Labs (LANL) in the late 1970’s and has since become an international standard for neutron and photon radiation transport simulations. Though currently MCNP is not formally qualified under the AMEC NSS established processes for qualification of computer codes used in safety analyses, its continuous refinement and extensive international use over the last three decades affirms that it has all the attributes of a qualified code and is considered the best tool for the current task of assessing neutron and gamma dose rates from UFTP. Further, MCNP is frequently used in the field of Medical Physics for calculations of beta, gamma and neutron dose rates to human tissue; as such, its accuracy for the current task is sufficient.

The Monte Carlo model of the UFTP is developed in 2 stages.

1. UFTP without the Impact Limiter
2. UFTP with the Impact Limiter and covered completely inside the weather cover

For the UFTP, all the short walls, bottom and lid are modeled to be 267 mm thick whereas the long walls are 272 mm thick. For simplicity, no gaps, joints, liners, crevices, or drain tubes are included in the model.

The UFTP is assumed to be fully loaded with each of the 48 module tubes within both fuel modules occupied by 2 bundles stored end to end. No empty module tubes are assumed. The module tubes are arranged in a 12-cm triangular pitch. The centres of the top and bottom rows of the 2 modules are separated by 2 lattice pitches (i.e., 24 cm). The rectangular structural frame holding the module tubes is omitted from the simulation model. Omission of this frame is a conservative assumption as the framework provides extra attenuation of photon and neutron particles emitting from the irradiated fuel bundles inside the UFTP.

A-1.1 UFTP Model with No Impact Limiter

Figures A-1-1 and A-1-2 show the Monte Carlo model of the UFTP containing 192 irradiated fuel bundles. Figure A-1-1 is the end-on sectional view of the UFTP showing the 2 groups of fuel bundles stored in the upper and lower storage modules.

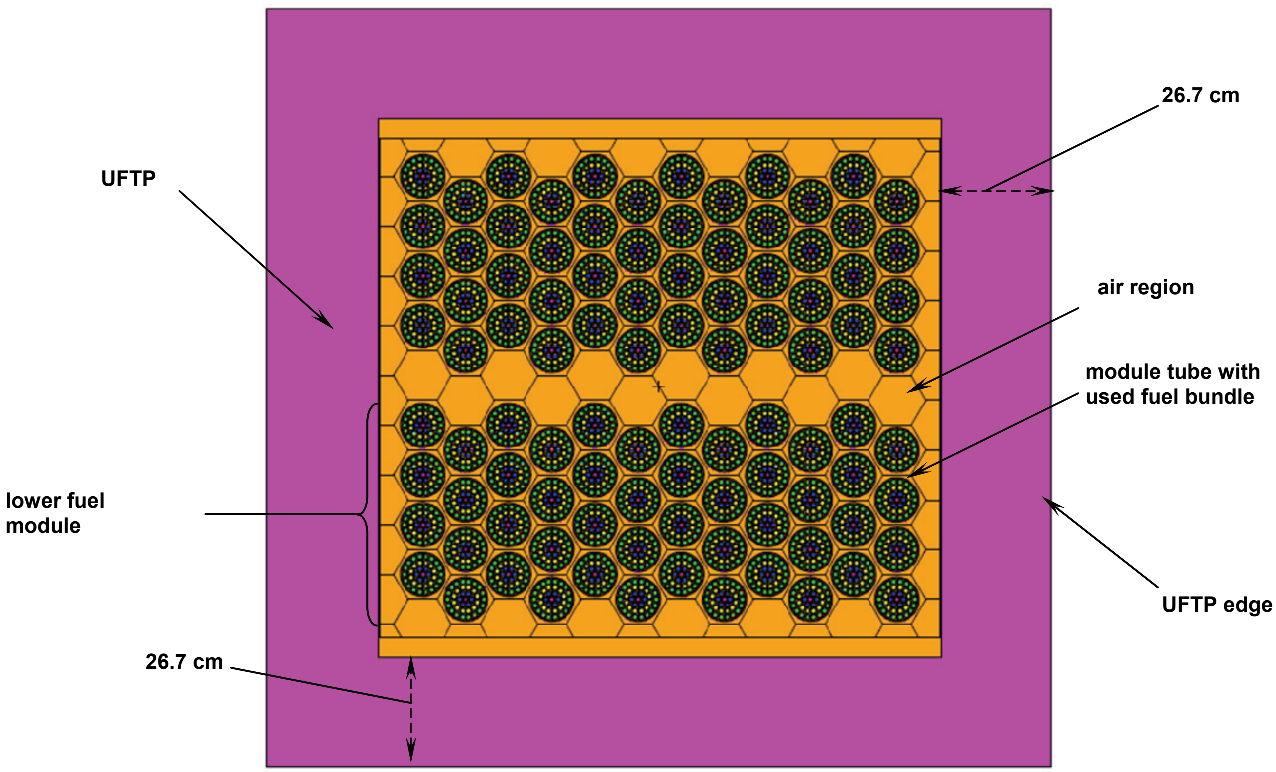

Figure A-1-1: Section View of the UFTP Model Showing Used Fuel in 2 Modules

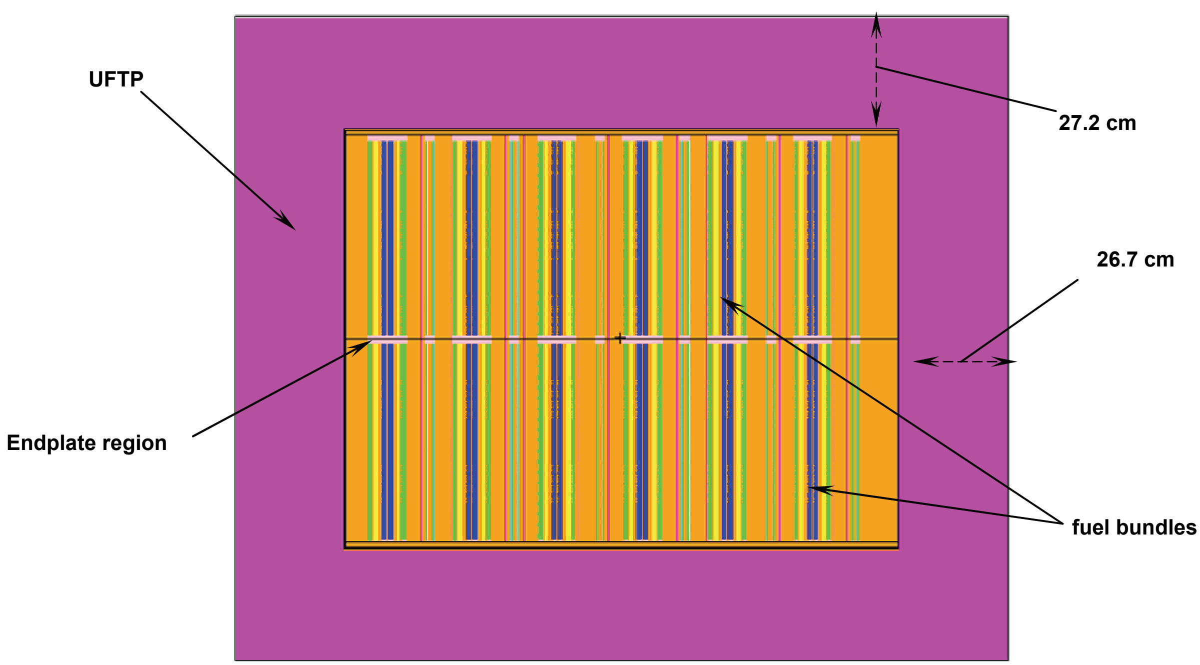

Figure A-1-2: Top View of the UFTP Model

A-1.2 Fuel Bundle and Module Tube

Each used fuel bundle is discretely modeled inside a module tube. This assembly is collectively defined as a hexagonal lattice with a lattice pitch of 12 cm (the centre to centre distance between fuel bundles) and a length of 49.53 cm. The lattice is a perfect hexagon with all 6 sizes of equal length.

A pictorial representation of the Monte Carlo model of the fuel bundle is illustrated in Figure A-1-3. The fuel elements representing the separate fuel rings are all modelled the same as unirradiated natural uranium dioxide (see Section 3.2.5.3). The interfacing material the fuel elements and module tubes is modeled as air at room temperature (293°K). The module tube material is modelled as Type 304L stainless steel.

All bundles are modelled in the exact same azimuthal orientation inside the transfer tube as shown in Figure A-1-3. The module tube is concentric with the fuel-bundle axis with a finite gap separating the fuel elements in the outer fuel ring from the inner surface of the tube. The endcaps and endplate of the bundle are modelled as one homogenized region with a thickness of 7.15 mm and a radius of 50 mm.

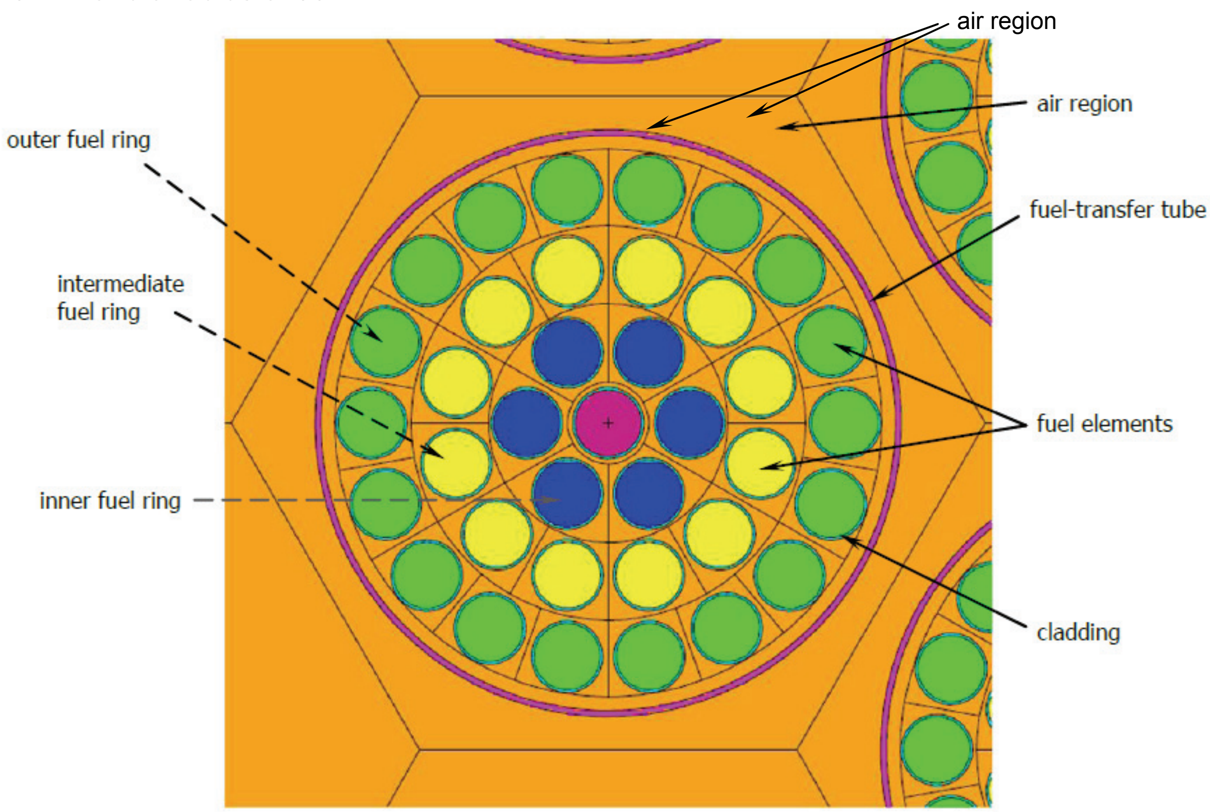

Figure A-1-3: Monte Carlo model of a 37-Element CANDU Fuel Bundle inside a Module Tube

A-1.3 Material Compositions

The UFTP, module and used fuel bundle are manufactured from various different materials as described in the applicable subsections of Section 3.2. These materials include uranium dioxide UO2, zircaloy Zr-4, Type 304L stainless steel and air. The endplate regions of each fuel bundle are approximated as a homogenized mixture of air, end cap and endplate based on the zircaloy mass in [A1]. The compositions of different materials and the endplate region used in the Monte Carlo model are listed in Table A-1-1.

The end-cap and endplate materials of the used fuel bundle are represented as a homogenized lower density Zr-4 material.

Table A-1-1: Material Compositions

Material	Element/ Nuclide	Weight Percent, wt-%	Material	Element	Weight Percent, wt-%
UO_2 (ρ=10.6 g/cm^3)	^{234}U	4.78×10^{-3}	ASTM 304L (ρ =7.89 g/cm^3)	C	0.03
	^{235}U	6.29×10^{-1}		Cr	19
	^{238}U	8.78×10^{1}		Mn	2
	^{16}O	1.15×10^{1}		Ni	10
Zr-4 (ρ =6.55 g/cm^3)	Sn	1.7×10^{-2}		N	0.1
	Fe	2.4×10^{-3}		P	0.045
	Cr	1.3×10^{-3}		Si	1
	Ni	7.0×10^{-4}		S	0.03
	Zr	0.9786[3]		Fe	67.795
Endplate (ρ =2.32 g/cm^3)	Zr-4	reduced density	Air (ρ =0.0012 g/cm^3)	O_2	23.2
				N_2	75.47
				Ar	1.28

A-1.4 Source Definition

The dose-rate calculations are performed in a fixed-source mode. That is to say, the source activity profiles described in Appendix A-2 are inputs to MCNP. Based on these source profiles, particles are tracked from the fuel elements to the exterior of the UFTP model at different locations.

In the MCNP input files, source terms are explicitly defined as a volumetric source for each fuel element. The source term in each element is spatially constant in both axial and radial directions. Each element has a diameter of 12.3 mm and a length of 481 mm.

Note that while the neutron and gamma source terms are defined explicitly as fixed sources, the source for neutron capture gammas is generated from the neutron calculations through capture reactions along the neutron transport media. The spatial and energy profiles of the capture gammas are implicitly generated within the neutron transport calculations and are accounted separately from direct gamma radiation.

A-1.5 Cross Section Libraries

All nuclide cross sections used in the calculation are from the default MCNP cross-section libraries. The identification numbers used in the calculations are 42c, 50c, 60c, and 66c. All

[3] Other trace impurities have been ignored in the calculation and are grouped as Zr.

material properties are assumed to be at 20°C (room temperature) with no additional temperature adjustment for neutron scattering.

In defining the material compositions, only elements with significant contents are included. Omission of trace materials does not have any impact on the neutron and photon transport process, hence the dose rates exterior of the UFTP are also unaffected.

A-1.6 Dose-Rate Conversions

With a nominal steel thickness no less than 26.7 cm, gamma shielding provided by the UFTP is so substantial that the expected gamma dose rates outside the UFTP due to decay gammas from irradiated fuel bundles are low. As a result, gamma dose rates are calculated using a partial deterministic approach with the use of the point-detector estimator coupled with the dose-rate conversion factors. The point detector estimator allows dose rates at distances far away from the UFTP to converge faster; although close to the UFTP surface the dose rates tend to fluctuate more because of higher scatter events.

All calculated fluxes at the dose receptors are multiplied by the neutron and photon fluence-to-dose-rate conversion coefficients from ICRP Publication 74 for neutrons (Table A-1-2) and photons (Table A-1-3). The coefficients used are for the antero-posterior phantom geometry[4] which gives the highest dose rates.

Table A-1-2: Neutron Fluence-to-Dose-Rate Conversion Coefficients

Energy, MeV	$\mu Sv.cm^2.s.h^{-1}$	Energy, MeV	$\mu Sv.cm^2.s.h^{-1}$
1.00×10^{-9}	0.018864	1.00×10^{-2}	0.06588
1.00×10^{-8}	0.02358	5.00×10^{-2}	0.1386
2.50×10^{-8}	0.02736	1.00×10^{-1}	0.21528
1.00×10^{-7}	0.03582	3.00×10^{-1}	0.4788
5.00×10^{-7}	0.04608	5.00×10^{-1}	0.6768
1.00×10^{-6}	0.04968	7.00×10^{-1}	0.8316
5.00×10^{-6}	0.054	1	1.0152
1.00×10^{-5}	0.05436	1.2	1.116
5.00×10^{-5}	0.05328	2	1.3788
1.00×10^{-4}	0.05256	3	1.5552
5.00×10^{-4}	0.05112	4	1.6488
1.00×10^{-3}	0.05112	5	1.7064
5.00×10^{-3}	0.05652	6	1.7388

[4] antero-posterior phantom geometry is where the radiation is incident directly face-on to a standing person's body.

Table A-1-3: Photon Fluence-to-Dose-Rate Conversion Coefficients

Energy, MeV	μSv.cm2.s.h-1	Energy, MeV	μSv.cm2.s.h-1
0.1	1.862×10^{-3}	1.5	2.205×10^{-2}
0.15	2.708×10^{-3}	2	2.696×10^{-2}
0.2	3.615×10^{-3}	3	3.559×10^{-2}
0.3	5.430×10^{-3}	4	4.326×10^{-2}
0.4	7.185×10^{-3}	5	5.040×10^{-2}
0.5	8.876×10^{-3}	6	5.755×10^{-2}
0.6	1.047×10^{-2}	8	7.171×10^{-2}
0.8	1.342×10^{-2}	10	8.554×10^{-2}
1	1.614×10^{-2}		

A-1.7 Road and Rail arrangement Geometries

The UFTP walls adjacent to the fuel bundle ends ('end-on') have an external dimension of 1881 mm, see Figure 4: UFTP Assembly in Section 3.2.3. The UFTP walls adjacent to the fuel bundle sides ('side-on') have an external dimension of 1566 mm.

Dose rates at the edge of the cask are calculated to determine the contributions from 2 UFTPs placing adjacent to each other for the rail transportation. For the rail transportation, the dose rates along the plane separating the 2 UFTPs can be estimated by doubling the edge dose rates of one UFTP to account for radiation coming from two UFTPs.

Geometry for road transport – single UFTP with its long side parallel to the direction of travel, and its short side (side with trunnions) in line with the ends of the truck.

Geometry for rail transport – two UFTPs per rail car. As in the truck case, its long side is parallel to the direction of travel, and its short side (side with trunnions) is in line with the ends of the train. Conservatively, no gap is assumed between the packages. The rail configuration is shown in Figure A-1-4.

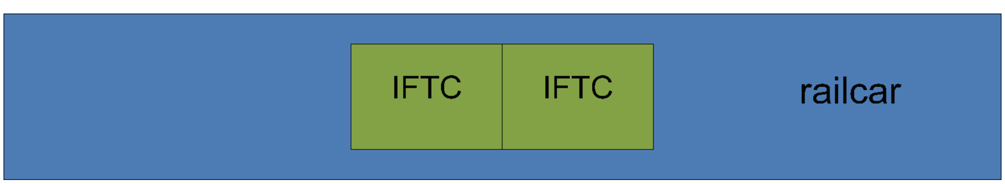

Figure A-1-4: Gamma Dose Rate Model Configuration for Railcar

Radiation is highest when one is directly facing the midplane of an UFTP. Dose rates at any distance away from the UFTP midplane or away from the UFTP wall surface will drop due to increase shield penetration and geometric attenuation respectively.

The regulatory limit of the UFTP is 0.1 mSv/h at 1 m. Radiation from an UFTP located behind another has to go through its own shield walls in addition to that of the UFTP in front plus the fuel inside. Attenuation provided by 2 UFTP shield walls is more than 1E+05 for gammas and 4E+03 for neutrons, see Appendix A-4. If it is assumed that all radiation coming from the UFTP located immediately behind another one is due to neutrons, the extra radiation caused by that UFTP is not expected to add more than 0.000025 mSv/h after the radiation has travelled past the cask in front using the 1 m regulatory dose-rate limit as the starting source. This anticipated dose-rate increase does not even account for the added attenuation provided by the fuel inside the UFTP and the distance factor.

As such, it is justifiable to exclude the contribution from all UFTP's that are located behind the first row in the calculations.

A-1.8 UFTP Model with Impact Limiter

Based on the bare UFTP model, the Impact Limiter and the weather cover are added to the model to assess the added neutron shielding provided by the 2 structures.

A schematic of the Impact Limiter is illustrated in Figure A-1-5. The Impact Limiter is made with redwood blocks encased inside a steel frame. The modelling assumptions of the Impact Limiter are listed as follows.

- Impact limiter is made of dry American redwood blocks
- Density of dry American redwood is 450 kg/m^3
- Redwood is assumed to be all cellulose with a chemical formula $(C_6H_{10}O_5)_n$
- No gap exists in the Impact Limiter
- Steel frame has a uniform thickness of 6 mm

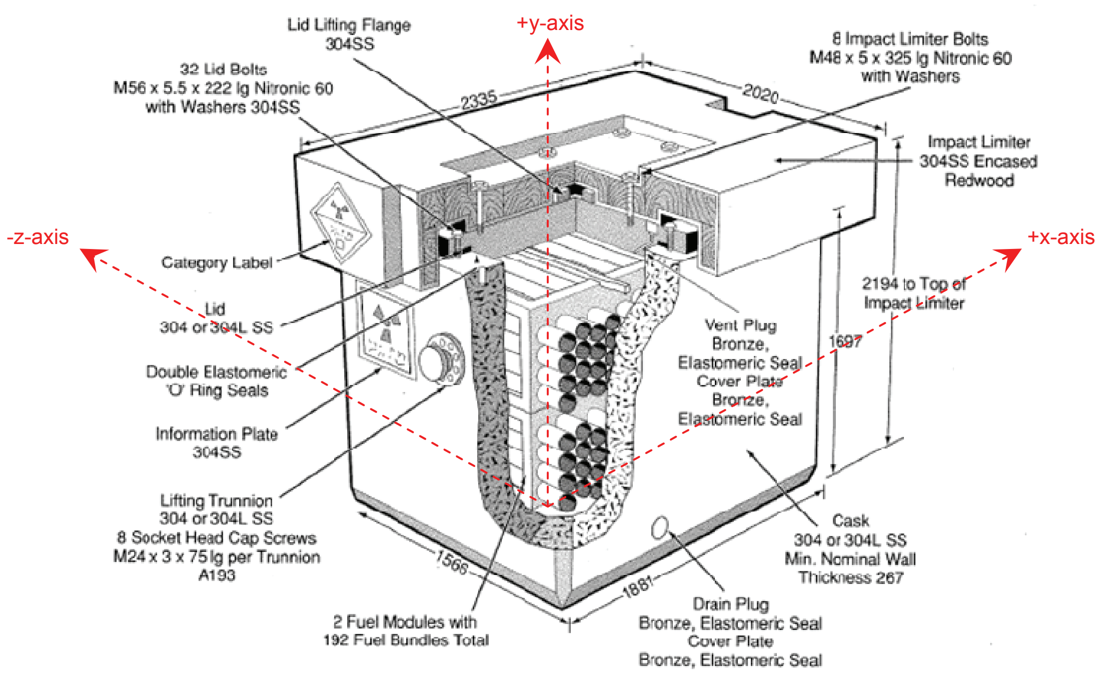

Figure A-1-5
UFTP – CANDU Used Fuel Transportation Package with Impact Limiter

Dimensions of the Impact Limiter are listed in Table A-1-4.

Table A-1-4: Dimensions of the Impact Limiter

Description		Dimensions, mm
Impact Limiter	Width (z-axis)	2020
	Depth (x-axis)	2335
Recessed Region	Width (z-axis)	1145
	Depth (x-axis)	830
Overall thickness including UFTP side overlap		640
Exterior height of UFTP	Without Impact Limiter	1824
	With Impact Limiter	2194
Steel liner thickness		6
Thickness along y-axis	General	370
	Recessed portion	225

A-1.9 Weather Cover Lined with Polyethylene

During road transportation the UFTP is covered by a weather shield. Whilst the frame of the weather shield does not provide any significant radiation shielding, there is a layer of polyethylene at the inner surface of the cover which can provide extra control of the neutron dose rates outside the UFTP.

The layer of high density polyethylene (HDPE) of the weather shield is modelled as the weather shield in the simulation model. The HDPE weather shield completely covers the UFTP including the impact limiter, analogous to the illustrations shown in Figures A-1-6 and A-1-7.

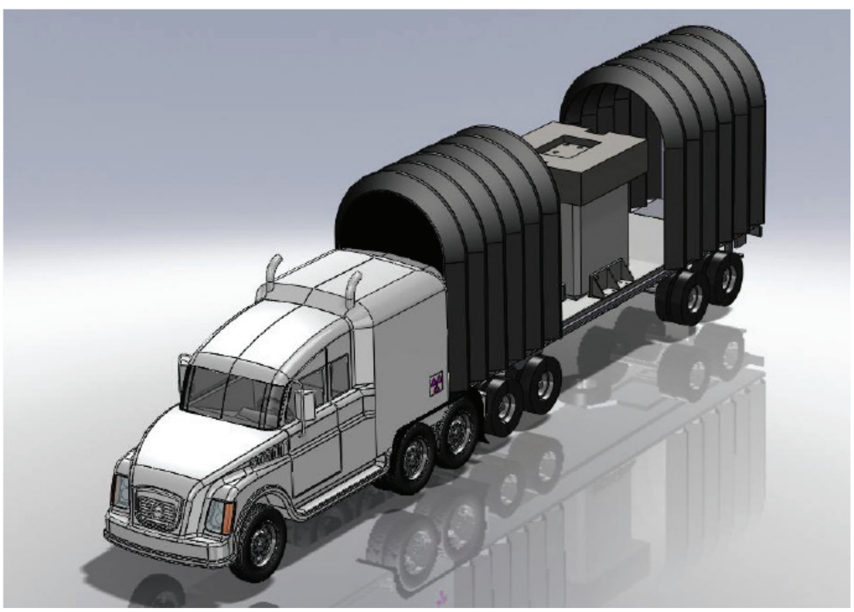

Figure A-1-6
Transportation Vehicle System with UFTP – Weather Cover Open

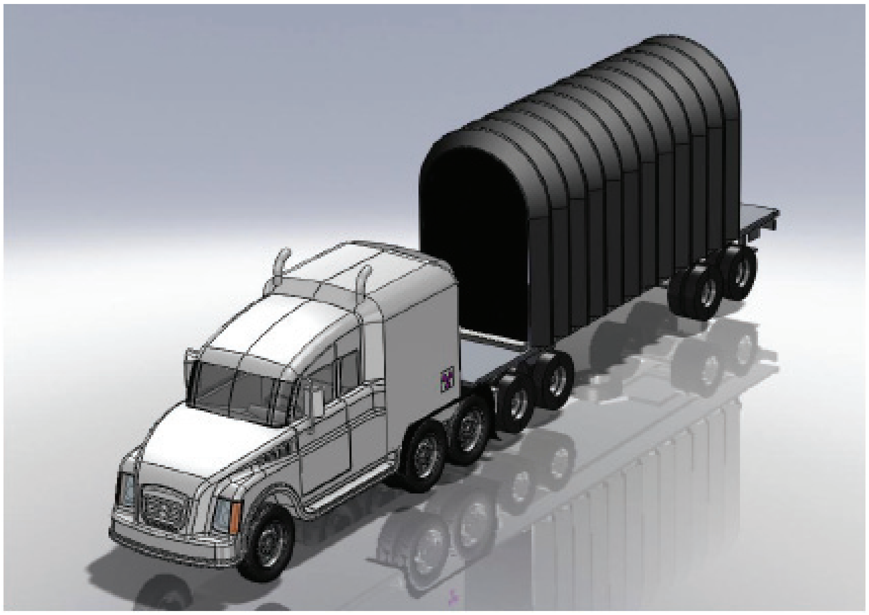

Figure A-1-7
Transportation Vehicle System with UFTP – Weather Cover Closed

The assumptions used in modelling the HDPE cover are as follows:

- HDPE is modeled as a rectangular cap over the UFTP with uniform thickness
- Thickness of the HDPE is 100 mm
- There is a 50 mm all around air gap between Impact Limiter and HDPE
- HDPE density range is 940-970 kg/m^3
- HDPE density used in the model is 955 kg/m^3
- Polyethylene chemical formula is $(CH_2)n$

The simulation model of the impact limiter including the HDPE weather cover is illustrated in Figures A-1-8, A-1-9, and A-1-10. One can see that during road transportation, not only the weather cover (through the HDPE) can provide added neutron shielding; it also acts as a physical barrier to keep personnel at a certain distance from the UFTP surface.

The radioactive source terms used in the modified model are exactly the same as the ones used in the bare UFTP model which are detailed in Appendix A-2.

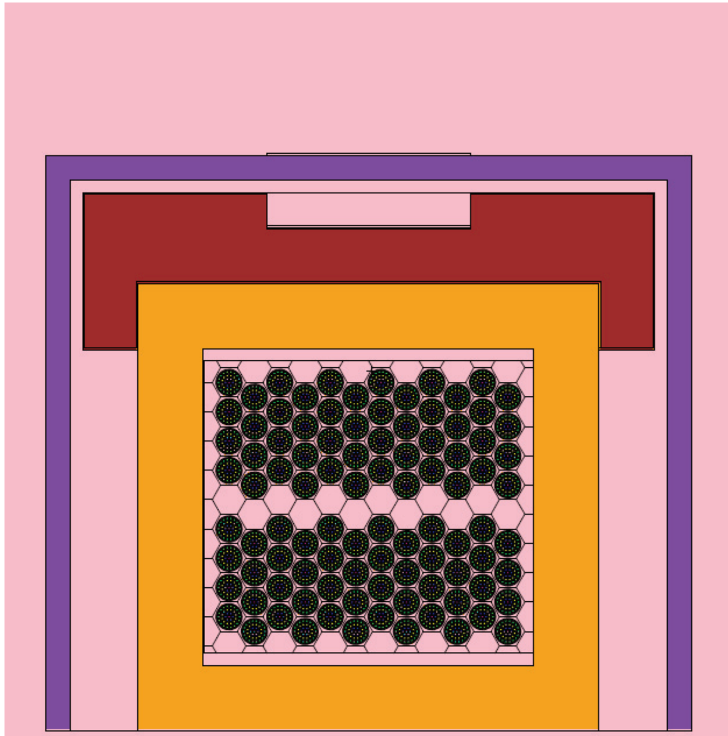

Figure A-1-8
Monte Carlo Model of the UFTP with Impact Limiter and HDPE Cover – End-on View

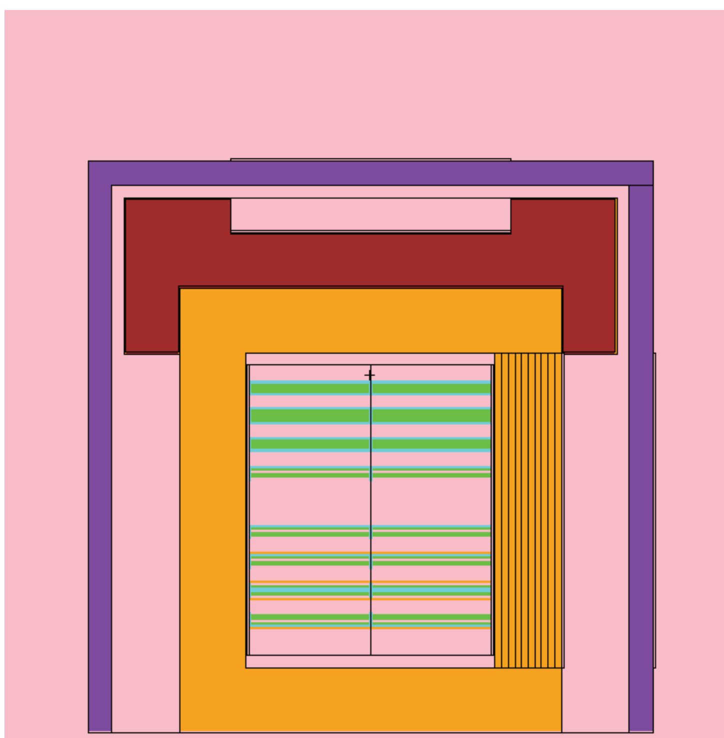

Figure A-1-9
Monte Carlo Model of the UFTP with Impact Limiter and HDPE Cover – Side-on View

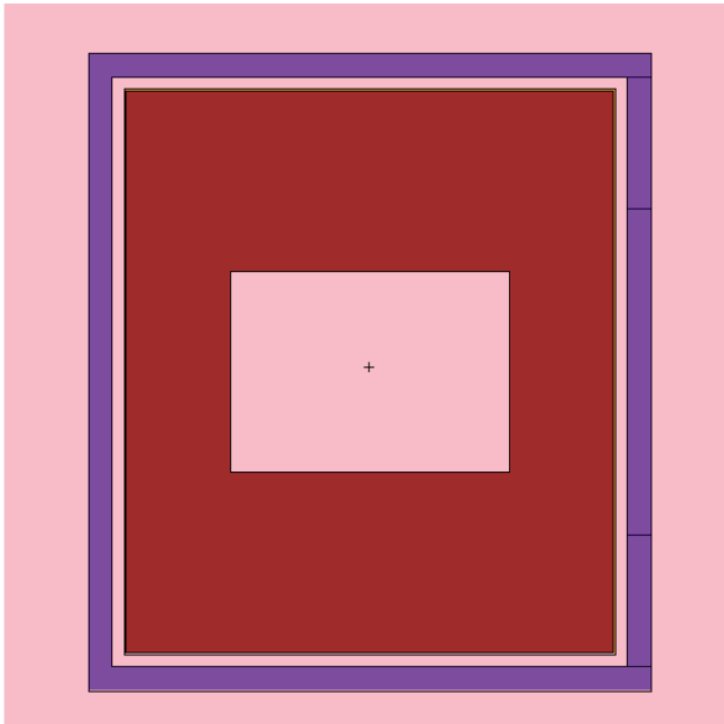

Figure A-1-10
MCNP Rendition of the UFTP, Impact Limiter, and HDPE – Plan View

A-2 Monte Carlo Model Inputs

A-2.1 Particle Release Spectra

The decay photon- and neutron-release spectra were taken from the OHN Report on radionuclide inventories of used nuclear fuel from OPG nuclear generating stations which includes Bruce A and Bruce B [A1]. Two irradiation burnup were considered in this report, i.e. 220 MWh/kgU and 280 MWh/kgU. The particle-release spectra 30 years after reactor shutdown for 2 fuel-bundle burnups are tabulated in Tables A-2-1 and A-2-2.

Table A-2-1: Photon Release Spectra for a 37-element CANDU Natural Uranium fuel bundle after 30 Years Decay for Discharge Burnup of 220 MWh/kgU and 280 MWh/kgU

Photon Release Spectrum at 30-Year Decay, γ/s per bundle		
Mean Energy, MeV	220 MWh/kgU	280 MWh/kgU
0.01	5.76E+12	6.95E+12
0.03	2.57E+12	3.15E+12
0.055	1.35E+12	1.65E+12
0.085	6.27E+11	7.51E+11
0.12	4.45E+11	5.39E+11
0.17	4.07E+11	4.86E+11
0.3	4.19E+11	5.00E+11
0.65	9.62E+12	1.22E+13
1.12	1.15E+11	1.52E+11
1.58	4.06E+09	5.25E+09
2	1.64E+08	1.94E+08
2.4	1.09E+05	1.44E+05
2.8	9.33E+06	1.41E+07
3.25	4.63E+03	1.04E+04
3.75	2.67E+03	6.03E+03
4.25	1.55E+03	3.49E+03
4.75	8.95E+02	2.02E+03
5.5	8.06E+02	1.82E+03

Table A-2-2: Neutron Release Spectra for a 37-element CANDU Natural Uranium fuel bundle after 30 Years Decay for Discharge Burnup of 220 MWh/kgU and 280 MWh/kgU

Neutron Release Spectrum at 30-Year Decay, neutrons/s per bundle		
Mean Energy, MeV	220 MWh/kgU	280 MWh/kgU
13.175	8.419E+00	1.962E+01
11.064	7.352E+01	1.713E+02
9.288	1.135E+02	2.644E+02
7.995	3.602E+02	8.394E+02
6.718	1.089E+03	2.538E+03
5.502	2.173E+03	5.065E+03
4.293	6.378E+03	1.416E+04
3.334	8.358E+03	1.642E+04
2.868	5.197E+03	9.613E+03
2.598	5.626E+03	1.058E+04
2.420	2.323E+03	4.427E+03
2.360	4.598E+02	8.792E+02
2.289	2.678E+03	5.119E+03
2.071	7.605E+03	1.477E+04
1.782	6.616E+03	1.339E+04
1.495	7.688E+03	1.626E+04
1.166	9.283E+03	2.046E+04
0.908	4.451E+03	1.005E+04
0.781	2.079E+03	4.739E+03
0.673	3.657E+03	8.394E+03
0.551	2.914E+03	6.704E+03
0.430	3.345E+03	7.730E+03
0.332	1.738E+03	4.016E+03
0.235	1.404E+01	1.865E+01
0.144	8.872E+00	1.179E+01
0.087	1.365E+00	1.814E+00

The bundle burnup of 220 MWh/kgU and 280 MWh/kgU represents the bounding burnup distribution for average and 95th percentile of all fuel-bundle discharge-burnup in OPG and Bruce Power reactors respectively. And for Darlington and Bruce B reactors, the 280 MWh/kgU represents the higher 99th percentile fuel-bundle discharge-burnup. Hence dose rates calculated using these 2 sets of spectra are considered bounding and conservative.

A-2.2 Photon Release Spectrum

At each burnup, the photon release spectrum of a fuel bundle is a result of 3 separate release contributions, viz. actinides, fission products, and light elements. Actinides and fission products are generated in the fuel due to direct fission yields and from neutron transmutations. For light elements, they represent the activation products of structural materials in the fuel bundle which include the bearing pads, spacer pads, end plates, end caps, and fuel cladding.

Over 98% of photon-energy release at 30 years decay is from fission products. And within fission products, over 90% is coming from ^{137}Cs. The principal contributors for both 220 MWh/kgU and 280 MWh/kgU fuel bundles are fission products which account for 98% of the total photon energy release.

A-2.3 Neutron Release Spectrum

The total neutron-release in a fuel bundle is a result of spontaneous fissions and (α,n) reactions in ^{17}O. Approximately ¾ of the energy is carried off by spontaneous neutrons.

Figure A-2-1 illustrates the neutron release spectra from spontaneous fission and ^{17}O (α,n) reactions. It also illustrates the relative change in spectrum between 220 MWh/kgU and 280 MWh/kgU. Neutron release from (α,n) reactions are higher at lower burnup but the reverse is true for the spontaneous fissions which has a higher release rate at higher burnup.

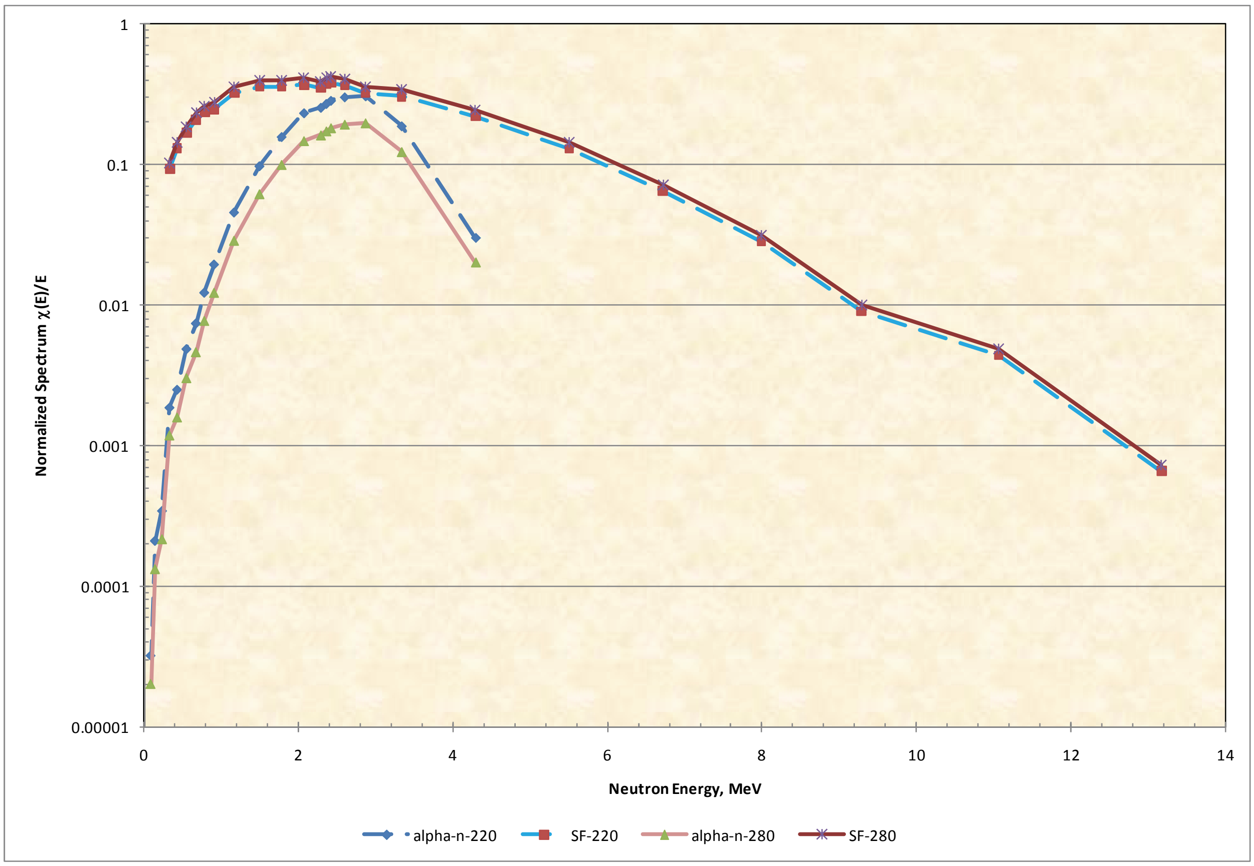

Figure A-2-1: Neutron Release Spectra from Spontaneous Fission and (α,n) Reactions

The integrated neutron release from spontaneous fissions is always higher than that from (α,n) reactions even at the maxima of the (α,n)-release spectrum. The relative decrease in neutrons generated by (α,n) reactions at higher burnup has a small impact to the overall increase in neutron production due to spontaneous fissions.

A-3 Detailed Dose Rate Tables – Monte Carlo Model Outputs

Dose rates listed in all the tables in the Appendix are calculated using the point-detector-tally option in MCNP. This tally option is ideal for obtaining dose rates at distances far away from the radiation source in a low scattering environment like air. However at tally locations where they are close to a scattering medium, the tally results tend to fluctuate significantly because there are significantly more scorings made at the tally locations from scattered particles with high importance and at close distance. The scoring is inversely proportional to the distance between the scoring location and the scatter location; therefore any scoring contributions that are due to scatter events close to the tally location will cause a large variance.

In the following tables, one will find the uncertainties of the gamma contact dose rates are higher than those at other locations. This is because gamma particles go through a lot of scattering events in the steel wall of the UFTP therefore causing large number of scores at close distance to the tally location. This makes the score fluctuates more leading to higher uncertainty. For every contact dose rate in the following tables, there is a footnote showing a near contact dose rate that has been averaged over the entire contact surface of the UFTP. This dose rate is calculated using a different tallying option in MCNP, the volume-averaged-tally option, which provides a much improved precision for the contact dose rates but is only representative to the average UFTP surface and not locally at a point. Because this tally option cannot be applied to dose locations far away from the UFTP, therefore it is only reported at contact with the UFTP.

Table A-3-1a: Dose Rates from UFTP at Receptor Location 1 with 220 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup) Neutron (<±3%)	Gamma (<±7%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.0404	0.0208 ± 36%[5]	1.79×10^{-4}	0.061 ± 12%
0.3 m	0.0245	0.0125	1.10×10^{-4}	0.037
1 m	0.00799 **0.000445**	0.00547	3.70×10^{-5}	0.014 **0.0060**
2 m	0.00275 **0.00017**	0.00224	1.26×10^{-5}	0.0050 **0.0024**
3 m	0.00136 **0.000094**	0.00114	6.10×10^{-6}	0.0025 **0.0012**
10 m	0.00015 **0.000018**	0.000116	5.94×10^{-7}	0.00027 **0.00013**
30 m	0.000018 **0.000003**	0.000012	6.51×10^{-8}	0.000029 **0.000015**

Table A-3-1b: Dose Rates from UFTP at Receptor Location 1 with 280 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup) Neutron (<±3%)	Gamma (±10%)	Capture Gamma (<±5%)	Total (<±5%)
Contact	0.0852	0.0168 ± 24%[6]	3.62×10^{-4}	0.10
0.3 m	0.0510	0.0153	2.29×10^{-4}	0.067
1 m	0.0166 **0.000445**	0.00763	7.77×10^{-5}	0.024 **0.0082**
2 m	0.00575 **0.000169**	0.00310	2.64×10^{-5}	0.0089 **0.0033**
3 m	0.00283 **0.000091**	0.00150	1.28×10^{-6}	0.0043 **0.0016**
10 m	0.000314 **0.000018**	0.000142	1.24×10^{-6}	0.00046 **0.00016**
30 m	0.000037 **0.000003**	0.000014	1.36×10^{-7}	0.000051 **0.000018**

Note: doses from package only are shown in plain text.
doses from package with weather cover are shown in **bold**.

[5] Volume-averaged dose rate at the side-on surface of the UFTP is calculated to be 0.0114 mSv/h with an uncertainty of 4.7%. .

[6] Volume-averaged dose rate at the side-on surface of the UFTP is calculated to be 0.0143 mSv/h with an uncertainty of 4.7%. .

Table A-3-2a: Dose Rates from UFTP at Receptor Location 2 with 220 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
	Neutron (<±3%)	Gamma (<10%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.00228	0.000027 ± 21%	9.68×10^{-6}	0.0023
0.3 m	0.00710	0.00134	3.32×10^{-5}	0.0085
1 m	0.00486 **0.000247**	0.00218	2.26×10^{-5}	0.0071 **0.0025**
2 m	0.00226 **0.000135**	0.00140	1.03×10^{-5}	0.0037 **0.0015**
3 m	0.00123 **0.000085**	0.000886	5.49×10^{-6}	0.0021 **0.00098**
10 m	0.000148 **0.000018**	0.000115	5.88×10^{-7}	0.00026 **0.00013**
30 m	0.000018 **0.000003**	0.000012	6.49×10^{-8}	0.000030 **0.000015**

Table A-3-2b: Dose Rates from UFTP at Receptor Location 2 with 280 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
	Neutron (<±3%)	Gamma (<±5%)	Capture Gamma (<±5%)	Total (<±5%)
Contact	0.00486	0.00004 ± 17%	2.14×10^{-5} ± 16%	0.0049
0.5 m	0.0148	0.00204 ± 12%	6.92×10^{-5}	0.017
1 m	0.0101 **0.000228**	0.00297	4.71×10^{-5}	0.013 **0.0032**
2 m	0.00470 **0.000131**	0.00191	2.15×10^{-5}	0.0066 **0.0021**
3 m	0.00255 **0.000081**	0.00126	1.15×10^{-5}	0.0038 **0.0014**
10 m	0.000308 **0.000018**	0.000146	1.23×10^{-6}	0.00046 **0.00017**
30 m	0.000037 **0.000003**	0.000016	1.36×10^{-7}	0.000052 **0.000019**

Note: doses from package only are shown in plain text.
doses from package with weather cover are shown in **bold**.

Table A-3-3a: Dose Rates from UFTP at Receptor Location 3 with 220 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
	Neutron (<±2%)	Gamma (<±5%)	Capture Gamma (<±5%)	Total (<±6%)
Contact	0.0354	0.0112 ± 23%[7]	1.41×10^{-4}	0.047
0.5 m	0.0214	0.0100	9.69×10^{-5}	0.032
1 m	0.00739	0.00495	3.42×10^{-5}	0.012
2 m	0.00266	0.00210	1.22×10^{-5}	0.0048
3 m	0.00133	0.00115	5.99×10^{-6}	0.0025
10 m	0.00015	0.000116	5.92×10^{-7}	0.00027
30 m	0.000018	0.000012	6.50×10^{-8}	0.000029

Table A-3-3b: Dose Rates from UFTP at Receptor Location 3 with 280 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
	Neutron (<±3%)	Gamma (<±6%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.0753	0.0118 ± 23%[8]	2.94×10^{-4}	0.087
0.5 m	0.0447	0.0129	2.03×10^{-4}	0.058
1 m	0.0154	0.00625	7.16×10^{-5}	0.022
2 m	0.00554	0.00267	2.55×10^{-5}	0.0082
3 m	0.00278	0.00141	1.25×10^{-5}	0.0042
10 m	0.000311	0.00015	1.24×10^{-6}	0.00046
30 m	0.000037	0.000016	1.36×10^{-7}	0.000053

[7] Volume-averaged dose rate at the side-on surface of the UFTP is calculated to be 0.0114 mSv/h with an uncertainty of 4.7%.

[8] Volume-averaged dose rate at the side-on surface of the UFTP is calculated to be 0.0143 mSv/h with an uncertainty of 4.7%.

Table A-3-4a: Dose Rates from UFTP at Receptor Location 4 with 220 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
	Neutron (<±6%)	Gamma (<±9%)	Capture Gamma (<±5%)	Total (<±6%)
Contact	0.00215	0.000024 ± 22%	7.08×10^{-6}	0.0022
0.5 m	0.00627	0.00115	2.94×10^{-5}	0.0075
1 m	0.00454	0.00186	2.10×10^{-5}	0.0064
2 m	0.00219	0.00136	9.98×10^{-6}	0.0036
3 m	0.00121	0.000842	5.40×10^{-6}	0.0021
10 m	0.000148	0.000121	5.86×10^{-7}	0.00027
30 m	0.000018	0.000012	6.50×10^{-8}	0.000030

Table A-3-4b: Dose Rates from UFTP at Receptor Location 4 with 280 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
	Neutron (<±5%)	Gamma (<±4%)	Capture Gamma (<±5%)	Total (<±5%)
Contact	0.00415	0.000036 ± 23%	1.47×10^{-5}	0.0042
0.5 m	0.0131	0.00158 ± 11%	6.15×10^{-5}	0.015
1 m	0.00945	0.00255	4.4×10^{-5}	0.012
2 m	0.00456	0.00176	2.08×10^{-5}	0.0064
3 m	0.00251	0.00112	1.13×10^{-5}	0.0036
10 m	0.000308	0.000145	1.23×10^{-6}	0.00046
30 m	0.000037	0.000016	1.36×10^{-7}	0.000052

Table A-3-5a: Dose Rates from UFTP at Receptor Location 5 with 220 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
Distance from Package	Neutron (<±3%)	Gamma (<±8%)	Capture Gamma (<±5%)	Total (<±6%)
Contact	0.0430	0.00366 ± 32%[9]	1.93 x 10^{-4}	0.047
0.5 m	0.0279	0.00737	1.29 x 10^{-4}	0.035
1 m	0.0099 **0.000539**	0.00407	4.70 x 10^{-5}	0.014 **0.0047**
2 m	0.00352 **0.000206**	0.00205 ± 15%	1.66 x 10^{-5}	0.0056 **0.0023**
3 m	0.00174 **0.000111**	0.000965	8.10 x 10^{-6}	0.0027 **0.0011**
10 m	0.000188 **0.000019**	0.000092	7.95 x 10^{-7}	0.00028 **0.00011**
30 m	0.000021 **0.000003**	0.000009	8.69 x 10^{-8}	0.000030 **0.000013**

Table A-3-5b: Dose Rates from UFTP at Receptor Location 5 with 280 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
Distance from Package	Neutron (<±2%)	Gamma (<±7%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.0885	0.0088 ± 32%[10]	3.99 x 10^{-4}	0.098
0.5 m	0.0579	0.0108	2.69 x 10^{-4}	0.069
1 m	0.0205 **0.000528**	0.00550	9.85 x 10^{-5}	0.026 **0.0061**
2 m	0.00728 **0.000203**	0.00236	3.46 x 10^{-5}	0.0097 **0.0026**
3 m	0.00361 **0.000109**	0.00117	1.69 x 10^{-5}	0.0048 **0.0013**
10 m	0.000388 **0.000019**	0.000117	1.66 x 10^{-6}	0.00051 **0.00014**
30 m	0.000043 **0.000003**	0.000012	1.81 x 10^{-7}	0.000055 **0.000015**

Note: doses from package only are shown in plain text.
doses from package with weather cover are shown in **bold**.

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[9] Volume-averaged dose rate at the end-on surface of the UFTP is calculated to be 0.0079 mSv/h with an uncertainty of 4.8%..

[10] Volume-averaged dose rate at the end-on surface of the UFTP is calculated to be 0.0104 mSv/h with an uncertainty of 6.1%.

Table A-3-6a: Dose Rates from UFTP at Receptor Location 6 with 220 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup) Neutron (<±3%)	Gamma (<±10%)	Capture Gamma (<±9%)	Total (<±4%)
Contact	0.00222	0.000015 ± 34%	9.50×10^{-6} ± 13%	0.0022
0.5 m	0.00720	0.00107	3.43×10^{-5}	0.0083
1 m	0.00537 **0.000238**	0.00174	2.56×10^{-5}	0.0071 **0.0020**
2 m	0.00270 **0.000147**	0.00109	1.27×10^{-5}	0.0038 **0.0013**
3 m	0.00151 **0.000094**	0.000712	7.02×10^{-6}	0.0022 **0.00081**
10 m	0.000185 **0.000019**	0.000098	7.82×10^{-7}	0.00028 **0.00012**
30 m	0.000021 **0.000003**	0.000010	9.49×10^{-8}	0.000031 **0.000013**

Table A-3-6b: Dose Rates from UFTP at Receptor Location 6 with 280 MWh/kgU Burnup

Distance from Package	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup) Neutron (<±5%)	Gamma (<±10%)	Capture Gamma (<±2%)	Total (<±5%)
Contact	0.00470	0.000017 ± 24%	2.24×10^{-5} ± 21%	0.0047
0.5 m	0.0149	0.00110	7.16×10^{-5}	0.016
1 m	0.0111 **0.000243**	0.00189	5.36×10^{-5}	0.013 **0.0023**
2 m	0.00559 **0.000148**	0.00137	2.65×10^{-5}	0.0070 **0.0016**
3 m	0.00314 **0.000095**	0.000869	1.47×10^{-5}	0.0040 **0.00098**
10 m	0.000384 **0.000019**	0.000125	1.66×10^{-6}	0.00051 **0.00015**
30 m	0.000044 **0.000003**	0.000013	1.81×10^{-7}	0.000057 **0.000016**

Note: doses from package only are shown in plain text.
doses from package with weather cover are shown in **bold**.

Table A-3-7a: Dose Rates from UFTP at Receptor Location 7 with 220 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
Distance from Package	Neutron (<±3%)	Gamma (<±6%)	Capture Gamma (<±3%)	Total (<±4%)
Contact	0.0377	0.00564 ± 25%[11]	1.59×10^{-4}	0.044
0.5 m	0.0243	0.00726	1.14×10^{-4}	0.032
1 m	0.00914	0.00367	4.36×10^{-5}	0.013
2 m	0.00340	0.00162	1.60×10^{-5}	0.0050
3 m	0.00171	0.000886	7.97×10^{-6}	0.0026
10 m	0.000187	0.000094	7.94×10^{-7}	0.00028
30 m	0.000021	0.000009	8.68×10^{-8}	0.000031

Table A-3-7b: Dose Rates from UFTP at Receptor Location 7 with 280 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
Distance from Package	Neutron (<±3%)	Gamma (<±7%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.0806	0.0115 ± 43%[12]	3.39×10^{-4}	0.092
0.5 m	0.0504	0.00934	2.37×10^{-4}	0.060
1 m	0.0190	0.00480	9.08×10^{-5}	0.024
2 m	0.00705	0.00213	3.35×10^{-5}	0.0092
3 m	0.00354	0.00115	1.66×10^{-5}	0.0047
10 m	0.000390	0.000127	1.66×10^{-6}	0.00052
30 m	0.000043	0.000012	1.82×10^{-7}	0.000056

[11] Volume-averaged dose rate at the end-on surface of the UFTP is calculated to be 0.0079 mSv/h with an uncertainty of 4.8%.

[12] Volume-averaged dose rate at the end-on surface of the UFTP is calculated to be 0.0104 mSv/h with an uncertainty of 6.1%.

Table A-3-8a: Dose Rates from UFTP at Receptor Location 8 with 220 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 220 MWh/kgU burnup)			
Distance from Package	Neutron (<±6%)	Gamma (<±9%)	Capture Gamma (<±5%)	Total (<64%)
Contact	0.00212	0.000013 ± 23%	6.89×10^{-6}	0.0021
0.5 m	0.00638	0.000825	3.04×10^{-5}	0.0072
1 m	0.00503	0.00149	2.39×10^{-5}	0.0065
2 m	0.00261	0.00104	1.23×10^{-5}	0.0037
3 m	0.00149	0.000672	6.88×10^{-6}	0.0022
10 m	0.000185	0.000095	7.80×10^{-7}	0.00028
30 m	0.000021	0.000010	8.65×10^{-8}	0.000031

Table A-3-8b: Dose Rates from UFTP at Receptor Location 8 with 280 MWh/kgU Burnup

	Dose Rate [mSv/h] (Used Fuel with 280 MWh/kgU burnup)			
Distance from Package	Neutron (<±3%)	Gamma (<±8%)	Capture Gamma (<±5%)	Total (<±4%)
Contact	0.00410	0.000015 ± 34%	1.43×10^{-5}	0.0041
0.5 m	0.0132	0.00111 ± 13%	6.35×10^{-5}	0.014
1 m	0.0104	0.00167	5.00×10^{-5}	0.012
2 m	0.00542	0.00126	2.57×10^{-5}	0.0067
3 m	0.00309	0.000895	1.44×10^{-5}	0.0040
10 m	0.000382	0.000118	1.63×10^{-6}	0.00050
30 m	0.000043	0.000013	1.81×10^{-7}	0.000056

A-4 Shielding Factors and Package Self-shielding

To assess the shield effectiveness of different materials outside the UFTP, one can use the attenuation coefficients of the material and apply it to the calculated dose rates at the appropriate receptor locations assuming radiation is following an exponential drop through the shield.

A-4.1 Neutrons

Majority of the dose rates outside the UFTP is coming from neutrons. The principal energy group of neutrons is below 1 MeV in the resonance region.

To estimate the shield effectiveness of a certain thickness of steel, one can use the neutron removal cross section of iron of 1.98 b/atom [A2] to obtain the mass attenuation coefficient (μ/ρ), where μ equals the product of the atom density and the removal cross section, and ρ is the material density. The attenuation provided by a steel slab of thickness t is simply exp(-μt) multiplied with a buildup factor. For fast neutrons in steel, a buildup factor of 2.5 can be used.

For example, to estimate the fast neutron attenuation through a 27.2 cm thick steel,

$$\mu = 1.98\, x \frac{7.9}{55.8} x\, 0.6 = 0.168\, cm^{-1}$$

where 7.9 is the assumed density of steel, 55.8 is the atomic mass of iron, and 0.6 is the Avogadro's constant. The estimated attenuation is therefore $2.6x10^{-2}$.

As for the shield effectiveness of neutrons going through concrete, one can estimate the drop in dose rates using the same approach but with the concrete macroscopic removal cross section of 0.089 cm^{-1}. Note that this application is not valid for thermal neutron attenuation.

A-4.2 Photons

The mass attenuation coefficients (μ/ρ) for photons are illustrated in Figure A-4-1. Radiation coming out of the UFTP is dominated by photons of energy 1 MeV and below. To estimate the drop in radiation provided by steel, one can apply (μ/ρ) of 0.06 cm^2/g to exp(-μt) to estimate the attenuation factor. Since photons undergo significant number of scattering events when going through the shield, there is a buildup of secondary photons hence decreasing the attenuation factor. To properly estimate the shield effectiveness provided by a steel slab of thickness t, a buildup factor B(t) has to be included, i.e. B(t) x exp(-μt). In general for a steel slab with thickness below 15 cm, one can conservatively set B (< 10 cm) = 10.

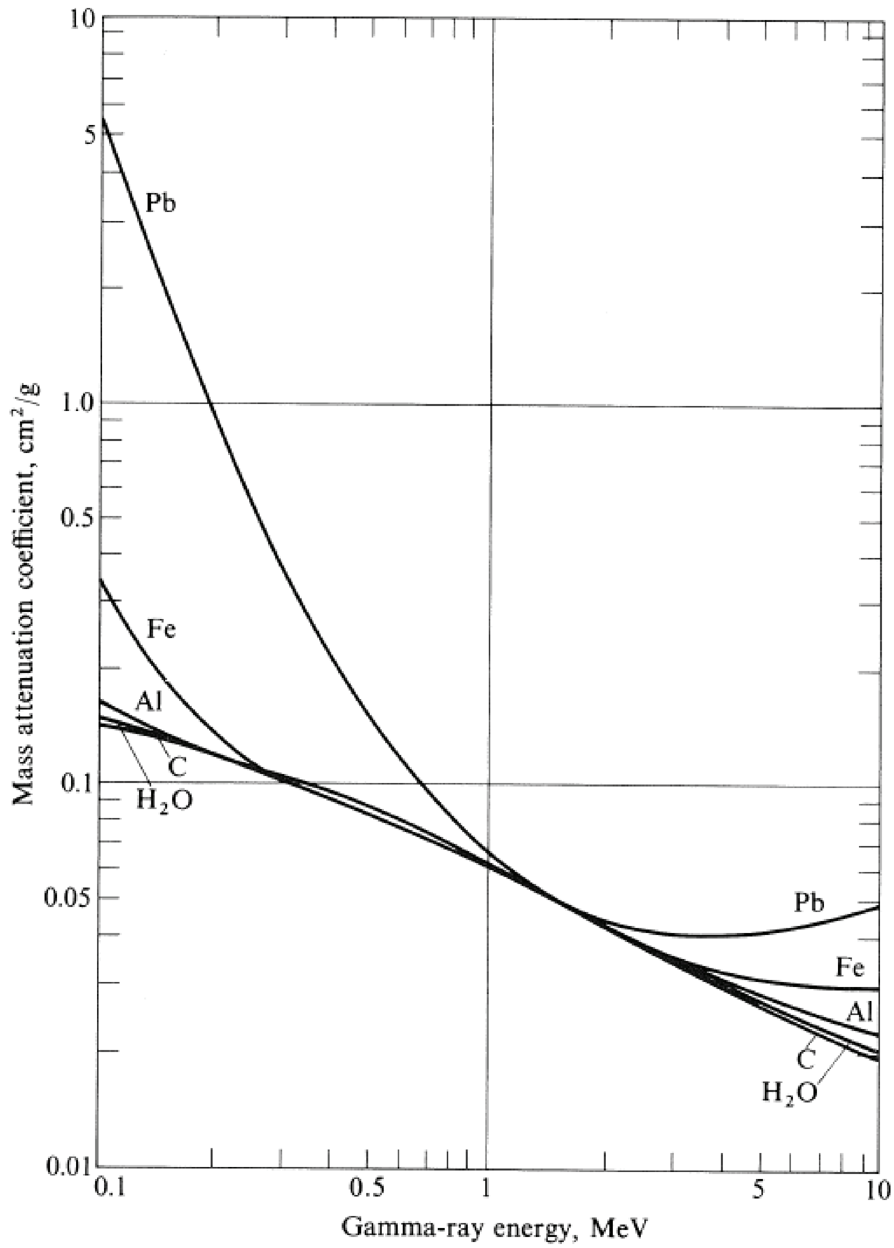

Figure A-4-1
Photon Mass Attenuation Coefficients of Several Elements [A3]

As for concrete, the mass attenuation coefficient is illustrated in Figure A-4-2. For example, photons at energy 1 MeV, (μ/ρ) is 0.06 cm^2/g. And for concrete slab under 30 cm of thickness, a buildup factor of 10 can be conservatively applied when calculating the drop in dose rates outside the UFTP due to a certain concrete thickness.

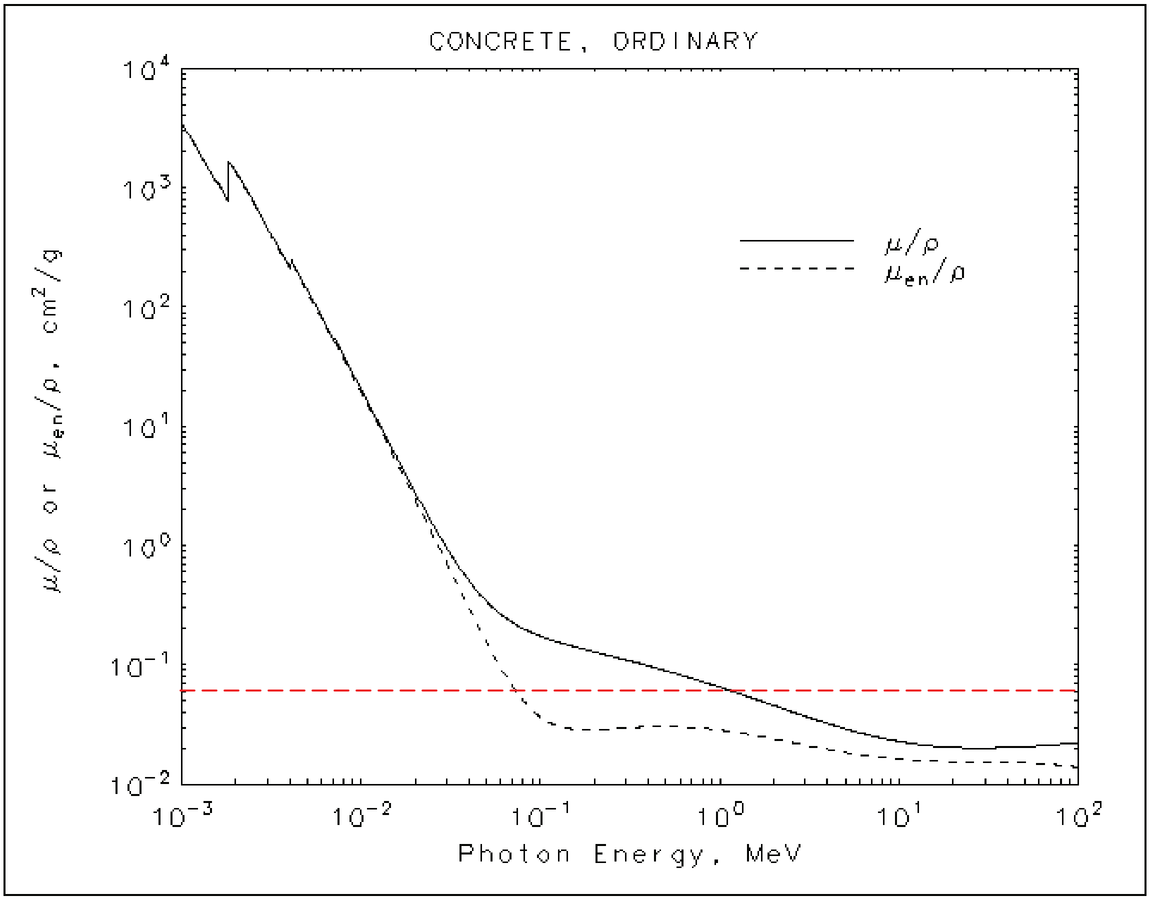

Figure A-4-2
Photon Mass Attenuation Coefficients in Concrete [A-10]

A-5 Rationale for Transportation Exposure Scenarios

The analysis in this report is generic, based on the assumptions described in this Appendix and does not describe either a specific route or geographic location. These assumptions were used to calculate a potential dose to an individual using the described distance and time parameters from a shipment operating under normal conditions. The information will assist NWMO to assess the general level of exposure to the public from the UFTP operating under normal transportation conditions and to highlight opportunities where NWMO can incorporate ALARA principles to further reduce the exposure to the public.

The analysis is not intended to provide an exhaustive analysis of any specific set of scenarios. Scenarios can be defined that could result in different dose levels than those presented in this report. However, a detailed assessment of transportation impact scenarios is premature until such factors as repository location, transportation mode and routes, population densities, transport package design, etc. are known. When these factors are established NWMO will prepare a corridor analysis to include dose assessments for those routes.

A-5.1 Road Transport Scenarios

Section 3.1.1.1 Individual along the Transport Route

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from external surface of cask	30 m is a conservative estimate of the right-of-way along a highway (A4). 30 m is assumed in the Ontario Hydro assessment (A5) and is the default setback distance in the RADTRAN code. (Note: RADTRAN 5 computer code is internationally considered the standard for transportation risk assessment modeling for radioactive materials.)
Exposure Time:	
Time for a tractor-trailer travelling at 24 km/h carrying 1 UFTC to pass by	24 km/h is the average vehicle speed assumed for transportation scenarios in reference A4.
Exposure Frequency:	
Individual is exposed to 620 shipments per year	620 shipments per year is the annual transport rate required to match the used fuel processing rate at the repository.

Section 3.1.1.2 a) Public Sharing Transport Route – Individual alongside UFTP

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 10 m from UFTP	10 m was chosen as the average distance between the individual and the UFTP during the entire length of the traffic jam. This is consistent with similar dose studies (A6). During the traffic jam, instances where the distance between the individual and the UFTP could be as close as 3 m (A3) could occur, however some relative motion between vehicles in the traffic jam was assumed.
Exposure Time:	
1 hour	The typical commute time by car in Canada's urban centres is approximately 30 minutes (A7). A gridlock scenario was assumed to be twice as long as the typical total time for a commute. Studies of similar scenarios (A6) have also estimated exposure time to be 30 minutes.
Exposure Frequency:	
Twice per year	Regulatory Guide G-208 *Transportation Security Plans for Category I, II or III Nuclear Material* stipulates that fixed schedules should be avoided. Variation in transport schedules and routes will tend to reduce the likelihood of repetitive exposure to a given individual. Therefore, 2 exposures over a one year period were assumed. Available sources (A6) suggest that exposure frequency to this type of scenario is a single event.

Section 3.1.1.2 b) Public Sharing Transport Route – Individual in front / behind UFTP

Assumptions:	Rationale:
Exposure Distance:	
Individual is located in vehicle 23 m in front or behind the UFTP on a trailer.	A typical semi trailer is approximately 16 m long. The UFTP is assumed to be positioned centrally on the trailer (external cask surface approximately 8 m from the end). Travelling at an average speed of 24 km/h (A4), the vehicle behind the trailer is assumed to be 2 seconds (13 m or approximately two car lengths) behind the trailer. The individual in the vehicle is assumed to be located approximately 2 m from the front of the vehicle. This results in an estimated exposure distance of 23 m.
Exposure Time:	
1 hour	The typical commute time by car in Canada's urban centres is approximately 30 minutes (A7). A gridlock scenario was assumed to be twice as long as the typical total time for a commute.
Exposure Frequency:	
6 times per year	Regulatory Guide G-208 *Transportation Security Plans for Category I, II or III Nuclear Material* stipulates that fixed schedules should be avoided. Variation in transport schedules and routes will tend to reduce the likelihood of repetitive exposure to a given individual. Therefore, 6 exposures over a one year period were assumed. Available sources (A6) suggest that exposure frequency for this type of scenario is a single event.

Section 3.1.1.3 Public near Rest Stop

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 10 m from UFTP on tractor-trailer	Studies of truck stop activities (A8) concluded that the average exposure distance in a suburban setting is 23 m and in a rural setting is 46 m. The default distance used in RADTRAN code is 30 m. 10 m was used as the estimate based on professional judgment and is within the above referenced cases.
Exposure Time:	
30 minutes	A total stop duration of 30 minutes is consistent with truck stop activity studies (A8).
Exposure Frequency:	
Twice per year	Regulatory Guide G-208 Transportation Security Plans for Category I, II or III Nuclear Material stipulates that fixed schedules should be avoided. Variation in transport schedules and routes will tend to reduce the likelihood of repetitive exposure to a given individual. 2 exposures per year were assumed, based on professional judgment.

Section 3.1.1.4 Public near Unplanned Stop

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from UFTP on tractor-trailer	An unplanned stop is assumed to be a mechanical breakdown, flat tire, or minor fender bender etc. where there is no impact on the UFTP. In this case, a 30 m perimeter would be established around the vehicle by transport, security and/or emergency response personnel (or in front and behind the vehicle if on the side of an active roadway). It is also conceivable that a residence is located 30 m away. 30 m is also the default distance for stops in RADTRAN.
Exposure Time:	
10 hours	In rural areas, it could conceivably take a lengthy period of time to obtain replacement parts, a replacement tractor or trailer and to bring in a portable crane. The 10 hour duration for such a stop was chosen based on professional judgment as to the time necessary to address the causes described and to return the vehicle to safe operation.
Exposure Frequency:	
Once per year	The vehicle fleet will be subject to a high level of inspections and maintenance. Industry experience of transportation programs of similar size has shown that the likelihood of unplanned stops is low. Available sources (A6) suggest that exposure frequency to this type of scenario in the same location as being a single event.

A-5.2 Rail Transport Scenarios

Section 3.1.2.1 Individual along the Transport Route

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from external surface of cask	30 m is a conservative estimate of the right-of-way along a rail line (A4). 30 m is assumed in the Ontario Hydro assessment (A5) and is the default setback distance in the RADTRAN code. Therefore, this distance was chosen for this analysis.
Exposure Time:	
Time for 1 rail shipment of 10 UFTCs travelling at 24 km/h to pass by	24 km/h is the average vehicle speed assumed for transportation scenarios in reference A4.
Exposure Frequency:	
Individual is exposed to 62 shipments per year	62 shipments per year is the annual transport rate required to match the used fuel processing rate at the repository.

Section 3.1.2.2 Public Sharing Transport Route – Individual alongside UFTP

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from external surface of cask	30 m is a conservative estimate of the distance between a rail line and a parallel running roadway. 30 m is used in the Ontario Hydro assessment (A5) and is the default setback distance in the RADTRAN code.
Exposure Time:	
Travelling alongside train at same speed as the train for 10 minutes	In various parts of the country, roads and rail lines run parallel to each other. Typically, trains and vehicles travelling alongside each other will not be travelling at the same rate of speed. A total exposure time of 10 minutes is consistent with rail activity studies (A6).
Exposure Frequency:	
Twice per year	Regulatory Guide G-208 *Transportation Security Plans for Category I, II or III Nuclear Material* stipulates that fixed schedules should be avoided. Variation in transport schedules will tend to reduce the likelihood of repetitive exposure to a given individual. Therefore, 2 exposures per year were assumed. Available sources (A6) suggest that exposure frequency to this type of scenario is a single event.

Section 3.1.2.3 Public near Signal Stop

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from external surface of cask	30 m is a conservative estimate of the distance between a rail line and a parallel running roadway. 30 m is used in the Ontario Hydro assessment (A5) and is the default setback distance in the RADTRAN code.
Exposure Time:	
Train stops at signal stop for 10 minutes	A total stop duration of 10 minutes is consistent with rail stop activity studies (A9).
Exposure Frequency:	
Individual is exposed to 62 shipments per year	62 shipments per year is the annual transport rate required to match the used fuel processing rate at the repository.

Section 3.1.2.4 Public near Unplanned Stop

Assumptions:	Rationale:
Exposure Distance:	
Individual is located 30 m from the train carrying UFTPs	An unplanned stop is assumed to be a mechanical breakdown or minor incident on the rail line where there is no impact on the UFTPs being transported by the train. In this case, it is assumed that a 30 m perimeter be set up around the train by rail, security and/or emergency response personnel. It is conceivable that a residence could be located 30 m away from the rail line. 30 m is the default distance for stops in RADTRAN.
Exposure Time:	
10 hours	In rural areas, it could conceivably take a lengthy period of time to obtain replacement parts or even a replacement engine or rail car, and/or a portable crane. The 10 hour duration for such a stop was chosen as a conservative estimate based on professional judgment for the time necessary to address the cause and return the shipment to safe operation.
Exposure Frequency:	
Once per year	The railcar fleet will be subject to a high level of inspection and maintenance reducing the likelihood of this type of scenario. Available sources (A6) suggest that exposure frequency to this type of scenario in the same location is a single event.

A-6 References

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APPENDIX B CALCULATION OF DOSE TO INDIVIDUAL ALONG TRANSPORT ROUTE

CONTENTS

Page

This appendix provides an estimated exposure of a person standing at a fixed distance (l) normal to the transport-truck path going at a constant speed (v). The assumptions used in deriving the exposure are listed as follows.

1. The transport truck will not change its travelling speed.
2. The person is stationary and facing perpendicular to the transport-truck travel-path during the exposure period.
3. The person is at least 5 m away from the transport-truck travel-path, i.e. $L \geq 5$ m.
4. The person is located sufficiently far away from the UFTP therefore a point-source approximation can be applied, i.e. using the $1/r^2$ approximation to estimate dose rates at distance farther than 5 m.
5. The person is on the same elevation as the UFTP; no credit is given to the risen height of the truck bed.
6. There is no intervening structure between the person and the truck therefore no material attenuation other than geometric attenuation.

B-1 Integrated Dose at the Receptor

Assuming the person standing at P is at a fixed normal distance l from the travel path of the transport truck (see Figure B-1), the dose rate at location P, defined as d(P), is inversely proportional to the distance between the UFTP and the receptor, i.e. distance r.

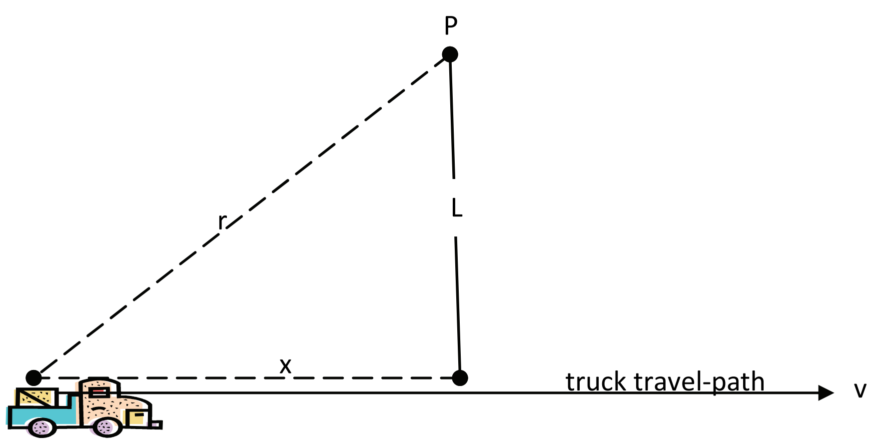

Figure B-1: Schematic of the Receptor Location P to the Travel Path

If we define S to be the end-on dose rate at h metres from the UFTP, then d(r) at distance r can be formulated as follows.

$$d(r) = \frac{S \times h^2}{r^2} \qquad \text{(B1)}$$

or

$$d(r) = \frac{S \times h^2}{(x^2 + L^2)} \qquad \text{(B2)}$$

where x is the distance of the UFTP to the plane that is perpendicular to travel-path. Distance x at time t for a truck travelling at constant speed v is defined as

$$x = vt \qquad \text{(B3)}$$

Therefore the total dose D at the receptor location P due to an approaching transport truck carrying 1 UFTP can be calculated by integrating equation B2.

$$D = \int_{-\infty}^{\infty} \frac{S \times h^2}{(vt)^2 + L^2} dt \qquad \text{(B4)}$$

Since

$$dx = vdt \qquad \text{(B5)}$$

Therefore equation A4 becomes

$$D = \int_{-\infty}^{\infty} \frac{S \times h^2}{x^2 + L^2} \frac{dx}{v} \qquad \text{(B6)}$$

Completing the integration, the total dose D for a person standing at location P due to a non-stopping transport truck carrying one UFTP travelling at constant speed v at a non-varying distance L normal of the travel path is calculated by the following equation.

$$D = \frac{\pi S h^2}{3600 vL} \qquad \text{(B7)}$$

The units for v and L are in m/s and m respectively. The units for D and S are in μSv and μSv/h respectively.

B-2 Dose-Rate Profile at the Receptor

Dose-rate profiles at the receptor location for different truck speeds are illustrated in Figure B-2. The profiles indicate the change in dose rates for the period in which the truck approaches the receptor from -200 m to the truck departs from the receptor at +200 m. As one would expect, the faster the truck travels, the shorter the exposure period hence a lower integrated dose D.

Presenting the information in a different format, Table B-1 shows the variation of the dose exposure as a function of truck speed and the separation distance L. Note that all doses are calculated with no credit given to any intervening structures between the receptor and the truck.

As there are 10 UFTPs on a rail shipment, the dose to an individual from a single shipment would be 10 times that of a road shipment for a particular distance between the individual and the path (rail line in the case of rail transport) as shown in Table B-2.

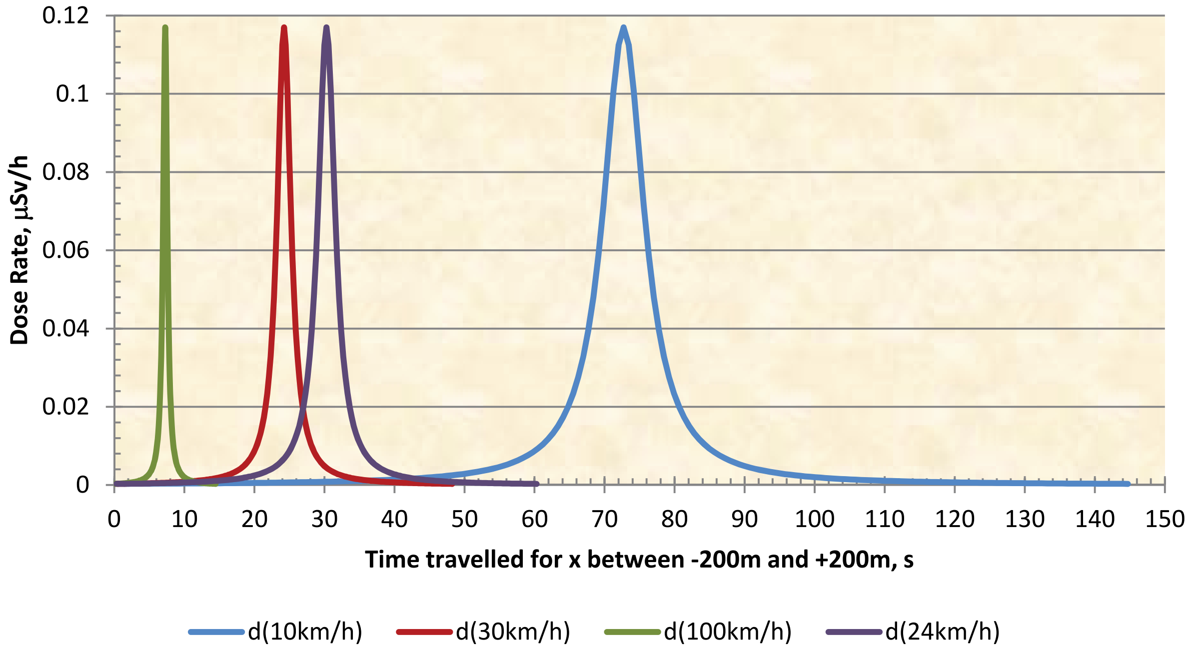

Figure B-2: Dose Rate at Receptor P over the Period the Truck Travels from -200 m to +200 m [13]

Table B-1: Integrated Dose at different Truck Speeds and Separating Distances

Distance L from Receptor to Truck Path, m	Integrated Dose D at Receptor Location P, mSv						
	5 km/h	10 km/h	24 km/h	30 km/h	50 km/h	80 km/h	100 km/h
3 m	2.0E-06	1.0E-06	4.2E-07	3.4E-07	2.0E-07	1.3E-07	1.0E-07
10 m	7.0E-07	3.5E-07	1.5E-07	1.2E-07	7.0E-08	4.4E-08	3.5E-08
30 m	2.5E-07	1.2E-07	5.1E-08	4.1E-08	2.5E-08	1.5E-08	1.2E-08
50 m	1.5E-07	7.4E-08	3.1E-08	2.5E-08	1.5E-08	9.2E-09	7.4E-09
100 m	7.4E-08	3.7E-08	1.5E-08	1.2E-08	7.4E-09	4.6E-09	3.7E-09

[13] The dose-rate profiles are calculated based on the end-on dose rates at 30 m from the UFTP surface with a Weather Cover

Table B-2: Integrated Dose at different Railcar Speeds and Separating Distances

Distance L from Receptor to Rail Line, m	Integrated Dose D at Receptor Location P, mSv						
	5 km/h	10 km/h	24 km/h	30 km/h	50 km/h	80 km/h	100 km/h
3 m	2.0E-05	1.0E-05	4.2E-06	3.4E-06	2.0E-06	1.3E-06	1.0E-06
10 m	7.0E-06	3.5E-06	1.5E-06	1.2E-06	7.0E-07	4.4E-07	3.5E-07
30 m	2.5E-06	1.2E-06	5.1E-07	4.1E-07	2.5E-07	1.5E-07	1.2E-07
50 m	1.5E-06	7.4E-07	3.1E-07	2.5E-07	1.5E-07	9.2E-08	7.4E-08
100 m	7.4E-07	3.7E-07	1.5E-07	1.2E-07	7.4E-08	4.6E-08	3.7E-08

APPENDIX C ASSESSMENT OF POTENTIAL DOSES AND FREQUENCIES FOR SPECIFIED ACCIDENT CONDITIONS

CONTENTS

C-1 Assessment of Potential Doses

In order to determine risks for releases corresponding to the hypothetical IAEA accident conditions scenario [C1], it is necessary to evaluate container damage, radioactivity emissions under the specified conditions and the likelihood of such as a stylized scenario taking place in real conditions.

The cask has been designed to maintain its integrity under both normal transportation and accident conditions as defined in the IAEA Transport Regulations [C1]. The UFTP design is such that under the defined conditions leakage will only be possible by permeation through the elastomer seals used between the faces of the cask lid and the cask body and at the vent and drain ports [C2].

The IAEA Transport Regulations specify tests for demonstrating ability of 'Type B' containers (the UFTP falls under this classification) to withstand defined accident conditions of transport. These conditions involve a drop from the height of 9m and exposure to a 30-minute fire at temperature of 800°C as well as a number of other requirements. As these are postulated testing conditions, the frequency of corresponding "accidents" cannot be assessed directly. In order to estimate such frequencies, a correlation has to be established with "real life" accident scenarios corresponding to the same radioactivity release rate.

The basis for determining an assumed release rate under the hypothetical conditions is the previous testing and assessment work on the cask, conducted by Ontario Hydro in 1980s and 90s [C2, C3]. Internationally, extensive testing on Type B containers was conducted by Sandia National Laboratories in the US and by the Central Electricity Generating Board in the UK, which includes both hypothetical testing conditions and severe accident scenarios, e.g. [C4]. This work demonstrated that certified cask designs sustain only very minor damage in testing conditions.

Impact and thermal response tests were carried out on an empty half-scale model of the UFTP cask, including a 9m drop and a one hour fire at 800°C temperature [C2]. No leakage was observed; and the main seal and the drain plug seal remained elastic at elevated temperatures. The drain cover plate seal remained intact in the seal groove but broke when it was lifted.

For the tested conditions corresponding to the IAEA Transport Regulations, although failure of the cask seal was not expected, it was assumed that there will be a leakage path available, for gases only, via permeation through the seal material, resulting in slow leakage of the gaseous radionuclides Krypton-85 (Kr-85) and Hydrogen-3 (H-3) [C2]. The same assumption was used as the basis for the current risk assessment for a hypothetical accident scenario. The UFTP reference inventory used in this assessment also includes radionuclides of elements that will or may be gaseous under the accident conditions considered, however due to their relative radioactivities and dose coefficients[14], the dose contribution from these radionuclides would be several orders of magnitude less than from Kr-85 and H-3.

It was furthermore assumed that in order for this level of damage to occur, the cask would have to sustain an impact at a speed of over 50 kilometres per hour (corresponding to a free fall from a 9 metre height) against a hard rock surface in combination with a 30-minute fully engulfing, optically dense fire.

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[14] A dose coefficient is the dose that a person would receive from exposure through a specific exposure pathway from unit radioactivity of a radionuclide.

The Kr-85 and H-3 release rates for a UFTP containing used fuel of burnup 280 MWh/kgU aged 30 years out-of-reactor were calculated as follows.

Radioisotope concentrations were obtained from reference [C5] and used to determine the total Kr-85 and H-3 radioactivities in a UFTP:

Used fuel of burnup 280 MWh/kgU aged 30 years	Value	Units
U per bundle	19.2	kg
# of bundles	192	
Total U mass in UFTP	3686.4	kg
Zr per bundle	2.2	kg
Total Zr mass in UFTP	422.4	kg
Fuel Fission Product H-3 concentration	1.25E+09	Bq/kgU
Fuel Fission Product H-3 activity per UFTP	4.62E+12	Bq
Fuel Fission Product Kr-85 concentration	1.57E+10	Bq/kgU
Fuel Fission Product Kr-85 activity per UFTP	5.80E+13	Bq
Fuel Light Element Impurity H-3 concentration	2.00E+09	Bq/kgU
Fuel Light Element Impurity H-3 activity per UFTP	7.38E+12	Bq
Fuel Light Element Impurity Kr-85 concentration	1.59E+02	Bq/kgU
Fuel Light Element Impurity Kr-85 activity per UFTP	5.84E+05	Bq
Zircaloy Light Element Impurity H-3 concentration	3.10E+08	Bq/kgZr
Zircaloy Light Element Impurity H-3 activity per UFTP	1.31E+11	Bq
Zircaloy Light Element Impurity Kr-85 concentration	2.63E+01	Bq/kgZr
Zircaloy Light Element Impurity Kr-85 activity per UFTP	1.11E+04	Bq
Total H-3 activity per UFTP	**1.21E+13**	**Bq**
Total Kr-85 activity per UFTP	**5.80E+13**	**Bq**

Section 3.2 of Annex 6A of the UFTP Safety Analysis Report [C2] details the methodology used to calculate release rates. Updated free inventory fractions and grain boundary fractions for Kr-85 and H-3 were obtained from references [C5] and [C6] to determine the effective free inventories in the UFTP:

	H-3	Kr-85
Free inventory Fraction	0.0863	0.0863
Grain-boundary inventory fraction	0.041	0.041
Effective free inventory fraction (% of total)	12.73%	12.73%
Effective Free Inventory, S (Bq)	**1.55E+12**	**7.39E+12**

The leakage rate L (Ci/s) is given by the equation

$$L = P_k \times \frac{A}{d} \times \frac{S}{V} \times \frac{RT}{v}$$

where [C2]:

P_k = permeation coefficient = 2.60E-06 std cm^3.cm/cm^2.s.atm

A = flow cross-sectional area = 548.76 cm^2

d = flow path length = 1.08 cm

V = UFTP volume = 1.30E+06 cm^3

R = ideal gas constant = 82.05746 cm^3.atm/K.mol

T = temperature of fuel after fire = 414 K

v = volume occupied by 1 mole of gas at 296 K (standard temperature for leakage rate measurement) and one atmosphere (standard pressure) = 2.43E+04 cm^3

Hence the leakage in eight hours (assumed maximum exposure time) and leakage rate per week is as follows:

	H-3	Kr-85
UFTP leakage rate [Bq/s]	2.22E+03	1.05E+04
UFTP leakage in 8 hours [Bq]	6.40E+07	3.08E+08
UFTP leakage per week [TBq/week]	0.00134	0.00635
Allowable leakage under Accident Conditions of Transport (IAEA TS-R-1 para 657) [TBq/week]	40	100

As can be seen, under accident conditions, the calculated leakage rate based on the assumed contents of the UFTP is 0.00134 TBq/week for H-3 and 0.00635 TBq/week for Kr-85; significantly less than the regulatory limits of 40 TBq/week for H-3 and 100TBq/week for Kr-85.

The following assumptions were made in determining doses to members of the public under the specified accident conditions in accordance with the methodology in the CSA Standard N288.2-M91 [C7]:

- Receptor is at ground level for ground level release along plume centre line with lower layer more stable than upper layer;
- Roughness length (z_0) is conservatively assumed to be 0.1 m (corresponding to open grassland, and hence no building wake);
- Pasquill stability category F and wind speed 2 m/s;
- Closest distance of receptor to UFTP is 30 m.

The Atmospheric Dilution Factor (ADF, s/m^3) at the receptor point under these assumptions is given by

$$ADF = \frac{\Psi(x,y,z)}{Q} = \sqrt{\frac{2}{\pi}} \times \frac{1}{\sigma_z \ \mu \ x \left(\frac{2\pi}{16}\right)}$$

where

$\Psi(x,y,z)$ = time integrated activity concentration (Bq.s/m^3) at receptor location (x,y,z)

Q = activity released to the atmosphere of the duration of the release (Bq)

μ = wind speed (m/s)

ϰ = distance from source (m)

$$\sigma_z = g(x)\,F(z_0, x)$$

$$g(x) = \frac{a_1 x^{b_1}}{1 + a_2 x^{b_2}}$$

when $z_0 \leq 0.1$ m
$$F(z_0, x) = \ln\left(\frac{c_1 x^{d_1}}{1 + c_2 x^{d_2}}\right)$$

For $z_0 = 0.1$ m, the following values apply:

$a_1 = 0.0638$, $a_2 = 1.36E\text{-}03$, $b_1 = 0.783$, $b_2 = 0.672$, $c_1 = 2.72$, $c_2 = 0$, $d_1 = 0$, $d_2 = 0$

Dose from a specific exposure route for radionuclide a to receptor at location (*x,y,z*)
= $\Psi(x,y,z)$ x DC_a

where

DC_a = dose coefficient for radionuclide a through a specific exposure route.

H-3 external dose coefficient for a semi-infinite cloud = 3.31E-19 Sv per Bq.s.m^{-3} [C8]

Kr-85 external dose coefficient for a semi-infinite cloud = 1.19E-16 Sv per Bq.s.m^{-3} [C8]

H-3 inhalation dose coefficient (includes skin absorption factor of 2) = 1.10E-14 [C7]

As the cloud is not actually semi-infinite, Figure 8.2 of the CSA Standard provides a finite cloud correction for the external dose coefficients for a semi-infinite cloud based on the height of the release (ground level in this case) and the vertical standard deviation (σ_z). Note: as the figure does not go below σ_z ~1.5, for σ_z values below this a correction factor of 0.05 was conservatively assumed.

For receptor distances from the UFTP greater than 30 m, the UFTP will act as a point source for direct external dose. Hence direct external doses at distances greater than 30 m are calculated using the inverse square law.

The results are shown in Table C-1 and Figure C-1 for road transport accidents and Table C-2 and Figure C-2 for rail transport accidents. For rail transport, the same methodology as for road transport was used, with the exception that it is assumed that two UFTPs are affected by the accident, and hence the potential gaseous radioactive material release inventories are doubled.

The total accident dose in both road and rail transport is dominated by direct external exposure from the UFTP; doses from releases are negligible in comparison.

Table C-1: Potential Doses from Specified Road Transport Accident Conditions

Distance from UFTP to Receptor (m)	30	40	50	75	100	300	500	800
g(x)	0.90290	1.12783	1.33969	1.82968	2.28019	5.22328	7.60783	10.66985
$F(z_0,x)$	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063
σ_z (m)	0.90347	1.12855	1.34054	1.83084	2.28163	5.22658	7.61264	10.67659
ADF (s/m^3)	3.75E-02	2.25E-02	1.52E-02	7.40E-03	4.45E-03	6.48E-04	2.67E-04	1.19E-04
Finite cloud correction	0.05	0.05	0.05	0.062	0.071	0.16	0.22	0.28
H-3 ψ(x,y,z) ($Bq.s.m^{-3}$)	2.40E+06	1.44E+06	9.70E+05	4.73E+05	2.85E+05	4.14E+04	1.71E+04	7.61E+03
Kr-85 ψ(x,y,z) ($Bq.s.m^{-3}$)	1.13E+07	6.80E+06	4.58E+06	2.24E+06	1.35E+06	1.96E+05	8.07E+04	3.60E+04
Dose from H-3 (mSv)	2.64E-05	1.58E-05	1.07E-05	5.21E-06	3.13E-06	4.56E-07	1.88E-07	8.37E-08
Dose from Kr-85 (mSv)	6.74E-08	4.05E-08	2.73E-08	1.65E-08	1.14E-08	3.73E-09	2.11E-09	1.20E-09
External Dose from material remaining in UFTP (mSv)	4.56E-04	2.57E-04	1.64E-04	7.30E-05	4.10E-05	4.56E-06	1.64E-06	6.41E-07
Total Dose (mSv)	**4.8E-04**	**2.7E-04**	**1.8E-04**	**7.8E-05**	**4.4E-05**	**5.0E-06**	**1.8E-06**	**7.3E-07**

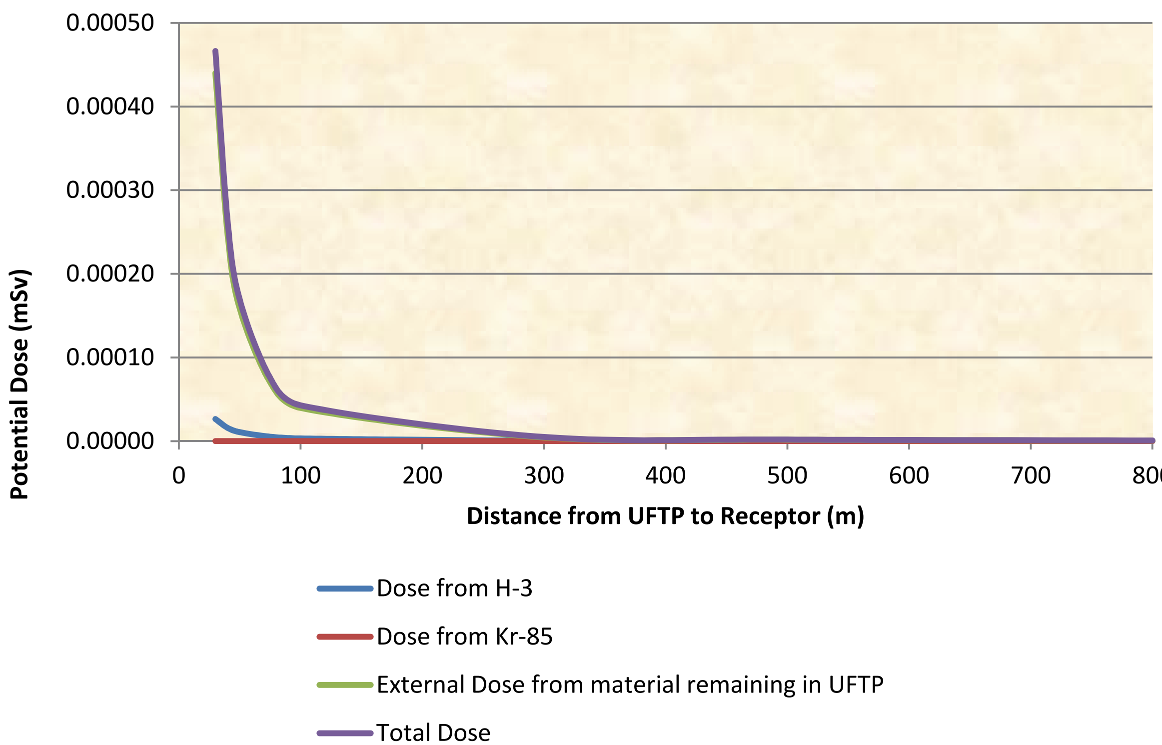

Figure C-1: Potential Doses from Specified Road Transport Accident Conditions

Table C-2: Potential Doses from Specified Rail Transport Accident Conditions

Distance from UFTP to Receptor (m)	30	40	50	75	100	300	500	800
g(x)	0.90290	1.12783	1.33969	1.82968	2.28019	5.22328	7.60783	10.66985
F(z_0,x)	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063	1.00063
σ_z (m)	0.90347	1.12855	1.34054	1.83084	2.28163	5.22658	7.61264	10.67659
ADF (s/m^3)	3.75E-02	2.25E-02	1.52E-02	7.40E-03	4.45E-03	6.48E-04	2.67E-04	1.19E-04
Finite cloud correction	0.05	0.05	0.05	0.062	0.071	0.16	0.22	0.28
H-3 ψ(x,y,z) (Bq.s.m^{-3})	4.80E+06	2.88E+06	1.94E+06	9.47E+05	5.70E+05	8.29E+04	3.41E+04	1.52E+04
Kr-85 ψ(x,y,z) (Bq.s.m^{-3})	2.27E+07	1.36E+07	9.16E+06	4.47E+06	2.69E+06	3.92E+05	1.61E+05	7.19E+04
Dose from H-3 (mSv)	5.27E-05	3.17E-05	2.13E-05	1.04E-05	6.27E-06	9.12E-07	3.76E-07	1.67E-07
Dose from Kr-85 (mSv)	1.35E-07	8.10E-08	5.45E-08	3.30E-08	2.27E-08	7.46E-09	4.22E-09	2.40E-09
External Dose from material remaining in UFTP (mSv)	9.12E-04	5.13E-04	3.28E-04	1.46E-04	8.21E-05	9.12E-06	3.28E-06	1.28E-06
Total Dose (mSv)	**9.7E-04**	**5.5E-04**	**3.5E-04**	**1.6E-04**	**8.8E-05**	**1.0E-05**	**3.7E-06**	**1.5E-06**

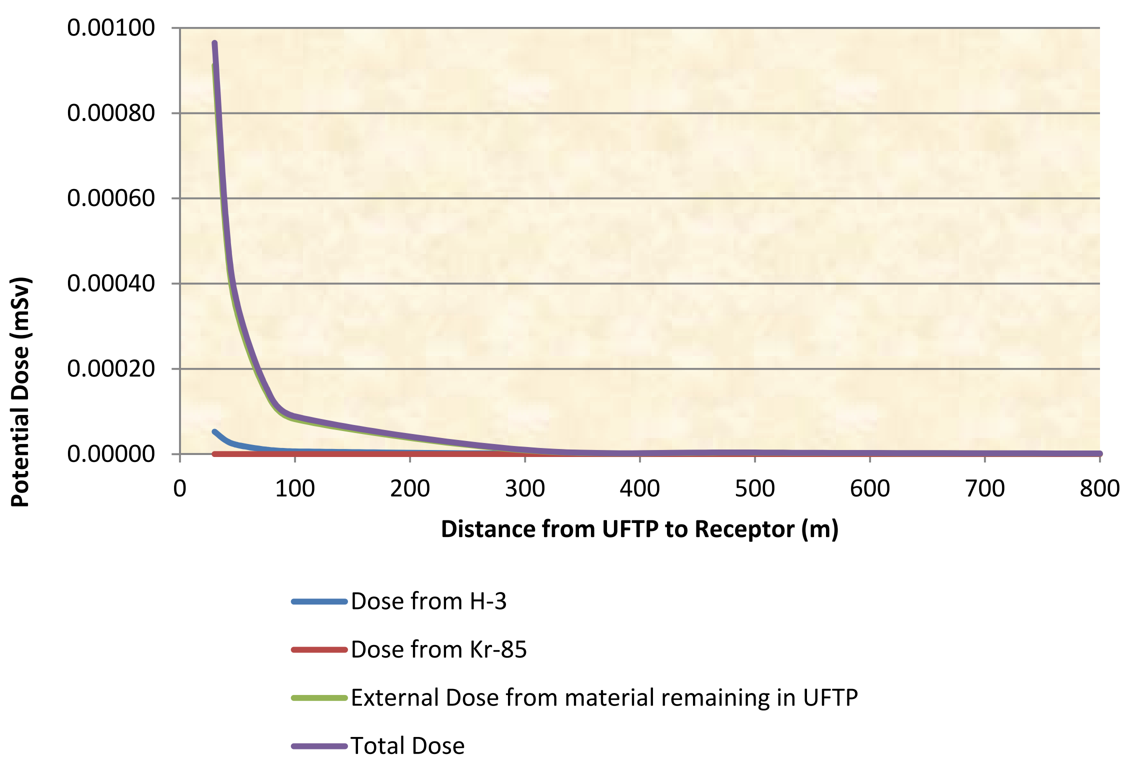

Figure C-2: Potential Doses from Specified Rail Transport Accident Conditions

The potential for loss of shielding from an accident resulting in radiation levels at 1 m from the surface of the UFTP exceeding the IAEA Transport Regulations criterion of 10 mSv/h [C1] was also investigated. The UFTP Safety Analysis Report [C2] concluded through drop tests that the maximum loss of shielding under the conditions specified [C1] would be approximately 75% of that which would be required to exceed the limit, and hence is compliant with the requirement. The reference UFTP inventory used in the UFTP Safety Analysis Report has used fuel aged 10 years out of reactor, whereas the current assessment uses a reference UFTP inventory with used fuel aged 30 years out of reactor. The difference in age of used fuel out of reactor results in lower radiation levels, and hence will also be compliant with the requirement.

C-2 Evaluation of Frequencies

For each mode of transport, annual frequency of accident conditions described in Section C-1 can be estimated as follows:

$$F_{hypothetical} = F_{accident\ rate} \times P_{hard\ rock\ impact} \times P_{severe\ fire}$$

Where $F_{accident\ rate}$ is the frequency of vehicles carrying the UFTP being involved in an accident, $P_{hard\ rock\ impact}$ is the conditional probability that following an accident an container will be subjected to an impact against hard rock at the speed of at least 50 kilometres per hour and $P_{severe\ fire}$ is the conditional probability of the container being immersed in a 30-minute engulfing, optically dense fire.

The basis of the calculations for each mode of transport is described below.

C-2.1 Road Transportation

The following parameters were used in determining annual frequency of a hypothetical accident scenario for road accidents:

- $F_{accident\ rate}$ = 1.72E-06 accidents/vehicle-km. This is the average value for road transport accidents in Ontario on the basis of the Ontario Road Safety Annual Report 2009 [C10].
- $P_{hard\ rock\ impact}$ = 0.0016. This is calculated as combined conditional probability of all sequences of events resulting in impact against hard rock as presented in the Modified Modal Study truck accident event tree [C10], including the following sequences:
  - Collision/On road fixed object/Bridge railing/Hard rock;
  - Non-collision/Off road/Into Slope/Hard rock;
  - Non-collision/Off road/Over Embankment/Hard rock.
- It was conservatively assumed that all impacts occur at speeds at or above 50 kilometres per hour; hence the conditional probability corresponding to this factor is equal to 1.
- $P_{severe\ fire} = P_{fire} \times P_{high\ temperature} \times P_{engulfing}$

P_{fire} is the conditional probability of traffic accidents resulting in a fire. For fires involving transportation of dangerous goods by road, it was shown on the basis of Transport Canada data that P_{fire} = 0.06 [C11].

Electrometric seal begins to leak at temperatures in excess of 350 ^{0}C [C10]. However, higher temperatures are required to cause significant fuel damage, here assumed to be consistent with the hypothetical accident scenario specified in the IAEA Transport Regulations [C1]. In practice, most fires result in temperatures in excess of 750 ^{0}C [C12], only 50 ^{0}C below the postulated condition. Hence $P_{high\ temperature}$ is conservatively taken as 1.

Fires have to be large enough, and be sustained for long enough, to prevent loss of energy from the cask to the atmosphere. This is required to cause internal elevation of temperature resulting in a significant damage to fuel sheath [C10]. In fact, the fire has to be "optically dense", i.e. its dimensions must exceed dimensions of the cask by several meters.

It has been estimated that half of all road fires are engulfing; hence $P_{engulfing}$ = 0.5 [C13].

Thus for transportation by road, $F_{hypothetical}$ = 1.72E-06 x 0.0016 x 1 x 0.06 x 1 x 0.5 = 8.26E-11 accidents/vehicle-km.

C-2.2 Rail Transportation

The following parameters were used in determining annual frequency of a hypothetical accident scenario for rail accidents:

- For Canada, the rail accident rate ($F_{accident\ rate}$) is 8.82E-06 accidents/vehicle-km [C14].
- $P_{hard\ rock\ impact}$ = 0.00036. This is calculated as combined conditional probability of all sequences of events resulting in impact against hard rock as presented in the Modified Modal Study train accident event tree [C10], including the following sequences:
    - All Derailments/Over Bridge/Hard Rock
    - All Derailments/Over Embankment/Hard Rock
    - All Derailments/Into Slope/Hard Rock
- It was conservatively assumed that all impacts occur at speeds at or above 50 kilometres per hour; hence the conditional probability corresponding to this factor is equal to 1.
- A recent US study analyzed statistical data on railway transportation of hazardous material. It was concluded that the frequency of accidents per freight kilometre was 1.8E-06, however only 1.9E-09 for the accidents that resulted in severe fires [C155]. It can be inferred that the conditional probability $P_{severe\ fire}$ is therefore one in a thousand (0.001).

Thus for transportation by rail, $F_{hypothetical}$ = 8.82E-06 x 0.00036 x 1 x 0.001 = 3.2E-12 accidents/vehicle-km.

C-3 References

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C3. "The Disposal of Canada's Nuclear Fuel Waste: Preclosure Assessment of a Conceptual System", Ontario Hydro Nuclear Report OHN N-03784-940010 (UFMED), COG-93-6 (1994).

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C5. "Fractional Inventories in Used Fuel", Ontario Hydro 06819-REP-03700-0042705, July 1998

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