APPENDIX A

DEEP GEOLOGICAL DISPOSAL

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APPENDIX A – DEEP GEOLOGICAL DISPOSAL

A1.0 INTRODUCTION

A1.1 BACKGROUND

This appendix examines the safety of disposal in a deep geological repository (DGR) based on *existing* information. The Canadian concept for deep geological disposal is described in NWMO documents (CTECH 2002; COGEMA 2003)

There have been several assessments of the safety of DGR in Canada over the years (AECL, 1994; Goodwin *et al.* 1994a; Grondin *et al.*, 1994; and Goodwin *et al.* 1996). These assessments have indicated repository design and site characteristics under which a deep geologic repository could be safely sited in the Canadian Shield. International studies have produced similar assessments (NAGRA 2002; SKB 1999; Vieno and Nordman 1999). The present document is largely based on Grondin *et al.,* 1994, and on the so called Third Case Study (TCS), a recent assessment of DGR in the Canadian Shield. It is expected that risk and monitoring aspects of the DGR concept in the TCS would be similar to those presented in other suitable geological media, such as sedimentary rock (NAGRA 2002). The TCS assessment considers a hypothetical repository, which has sufficient capacity to hold all the used fuel from present Canadian nuclear power stations to the end of their planned life. Some of the details of the repository design and site conditions differ somewhat from previous assessments. However, the combined results from all these assessments illustrate the safety of the DGR concept for several combinations of facility design and sites (Gierszewski *et al*., 2004). The present document also uses analysis provided by the Environmental Assessment Panel on Nuclear Fuel Waste Disposal Concept (Seaborn, 1998).

These assessments are useful in providing perspective on the performance of the DGR repository. However, they will have to be expanded and updated as part of a "safety case" if this option is selected (Moshonas *et al*. 2004).

A1.2 OUTLINE

The discussion of the safety of DGR in this section addresses the various stages in the lifetime of the facility. These stages (CTECH 2002) were developed for conceptual design and cost estimating purposes, and are listed as follows:

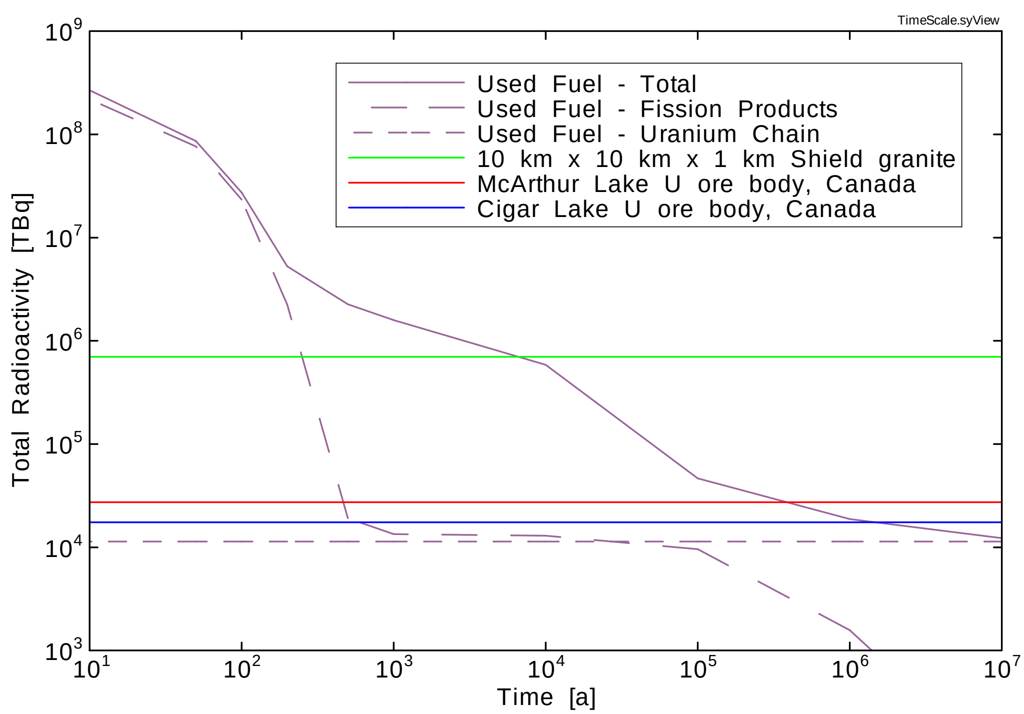
- siting (site screening and site evaluation) (18 years in duration);
- construction (11 years in duration);
- operation (30 years in duration);
- extended monitoring (two periods of undefined duration);
- decommissioning (25 years in duration);
- closure (1 years in duration); and
- post-closure

On this basis, the total duration of the pre-closure phase would be close to 90 years plus the undefined duration of the extended monitoring period. The post-closure period starts beyond this time frame.

The CNSC P-290 policy (CNSC 2004) indicates that the period over which the future impacts of radioactive waste are assessed should include the period over which the maximum impacts are expected. The deep geologic repository is expected to be capable of isolating the used fuel such that the maximum impact would likely occur well beyond 10,000 years. Since 98% of the used fuel is natural uranium, as radionuclides decay, the radioactivity in the repository will eventually become similar to that of uranium ore bodies found in other locations on the Canadian Shield. This occurs on time scales of about one million years (Figure 1.1-1).

Therefore, for the post-closure assessment, future impacts will be assessed to the time of the peak dose rate in general, with a one-million-year time base. It is recognized that estimating impacts becomes increasingly uncertain at long times, but nonetheless this study will consider this time scale in order to illustrate the potential long-term impact.

Figure A1.1-1
Time Scale for Radioactivity Decay in Repository



The gamma-emitting fission products decay within about 1000 years. The remaining fuel radioactivity becomes comparable to that of the granite in the surrounding watershed after about 10,000 years. On time scales of about 1 million years, the residual used fuel radioactivity is dominated by that of the uranium in the fuel (and its decay chain products), a level that is comparable to natural uranium ore bodies (extracted from Gierszewski *et al.* 2004).

A2.0 POTENTIAL RISK FROM SITING

A2.1 POTENTIAL RISK TO THE PUBLIC

Siting activities include office work as well as field research and evaluation. Considering the kinds of activities likely to occur during the siting stage, no significant adverse physical effects on public health and safety are expected. This does not take into account the possibility of stress which some members of a local community might feel. Depending on proximity to a community or individual residence, the most likely potential physical effects would be some noise, traffic and other nuisance effects associated with access road construction, drilling and blasting. However, assuming that a cooperative siting process and reasonable mitigation measures are used, **it is considered unlikely that any public health and safety effects would be significant.** (Grondin *et al.,* 1994)

A2.2 POTENTIAL RISK TO WORKERS

The technical site characterization activities performed during siting would include activities similar to those performed during the geological exploration phase of standard mining projects. They can also be compared to the geotechnical investigations performed prior to the development of large civil structures such as hydro-electric dams, tunnels and underground powerhouses.

According to Grondin *et al*. (1994) some of these activities would be disruptive to the natural environment and could result in some hazards to the workers. Some research personnel may have limited exposure to radiation (from using radioactive logging devices). However, a review of the practices used in mining exploration and hydraulic dam site investigations showed that methods and technologies exist that could be used to mitigate identified effects. Based on this non site-specific analysis, **effects on workers are expected to be minimal** during siting, provided that adequate worker safety measures are taken. (Grondin *et al.* 1994)

A2.3 POTENTIAL RISK TO THE NATURAL ENVIRONMENT

Similar to the discussion for workers, based on a generic analysis, effect on **the environment are expected to be minimal**, provided that adequate environmental protection measures are taken.

A3.0 POTENTIAL RISK FROM THE CONSTRUCTION OF A GEOLOGICAL DISPOSAL CENTRE

A3.1 POTENTIAL RISK TO THE PUBLIC

The construction phase is the period starting when the site and disposal facility design have been approved for construction, and continues until the surface and underground facilities are installed and operational, and initial set of disposal rooms excavated and serviced. The construction stage is expected to last 7 years (Simmons and Baumgartner, 1994).

According to Grondin *et al*. (1994), the impact of the construction of a disposal centre on the **public health and safety would be minimal. The transportation of construction material could result in some insignificant impact.** Specifically, the release of naturally occurring radon and radon progeny to the atmosphere by excavation on-site was estimated to be a small fraction of the natural radon emissions to the atmosphere from surface soils. Therefore, the overall potential effect is expected to be insignificant.

A3.2 POTENTIAL RISK TO WORKERS

Occupational hazards from disposal centre construction would include physical injuries, noise and exposure to dust and fumes from the operation of equipment and blasting. The estimated risks to workers from both surface and underground construction activities are presented in Table A3.2-1 (extracted from Grondin *et al*. 1994 Table 5-7). **The total risk to workers during the construction stage was estimated to be about 0.06 fatalities per year and about 11 lost time injuries for a total workforce of about 1000 persons per year.**

This is considered to be a conservative estimate, based on Ontario industry statistics. It is expected that a disposal facility for used nuclear fuel would achieve a better than industry average worker safety. For example, Ontario Hydro experienced no worker fatality during the 50 million person-hours worked to construct the Darlington Nuclear Generating Station (Zeya 1993a).

A3.3 POTENTIAL RISK TO THE NATURAL ENVIRONMENT

The impacts of construction of a disposal centre on the following environmental factors were considered by Grondin *et al.* (1994) air quality, surface water quality, groundwater quality, soil, land use, forest fires, flora and fauna, ambient noise, non-renewable resources and traffic. Most of the analysis was qualitative in the absence of site-specific characteristics, as it relied upon the conceptual design and generic environmental data. The analysis was based on a review of the construction activities specified in the conceptual engineering study (Simmons and Baumgartner 1994) and a review of the effects of related activities carried out during surface and underground construction projects. It is expected that the construction stage would be (relatively speaking) the most disruptive for the natural environment. A review of conventional practices in surface and underground construction projects showed that **methods and technologies exist that could be used to mitigate negative effects.** *The effect of underground excavation on the water table around the site would need to be investigated further based on site-specific data and mitigated if necessary. The effect of transporting construction material would also be dependent on the state of the local transportation network.*

A4.0 POTENTIAL RISK FROM THE OPERATION OF THE DISPOSAL CENTRE

A4.1 POTENTIAL RISK UNDER ROUTINE OPERATING CONDITIONS

The operation stage involves the receipt, packaging, and disposal of used fuel in the underground facility. This stage is expected to last 41 years.

A4.1.1 Potential Radiological Risk to the Public – Normal Conditions

Even with filtering equipment, routine airborne and waterborne emissions would result from normal operation of the Used Fuel Disposal Centre (UFDC). Radionuclides released from the UFDC may lead to a radiation dose to humans via a number of internal and external pathways.

Russell (1993b) estimated radiological doses for individuals in the population that are expected to receive the highest dose. Specific exposure scenarios, such as exposure of Aboriginal people were also considered. The most exposed individual was assumed to live on a farm at the disposal centre boundary. The location of the farm was assumed to be in the wind direction that gave the largest radionuclide concentrations from airborne emissions. The results of the analysis are shown in Table A4.1.1-1.

Table A4.1.1-1
Maximum Doses to an Adult and an Infant Living on a Farm at the Disposal Facility Boundary (Russell 1993a)

	Maximum Dose (mSv a^{-1})		
	Northern Region	**Central Region**	**Southern Region**
Adult Dose	3.4×10^{-4}	2.2×10^{-4}	2.0×10^{-4}
Infant Dose	5.2×10^{-4}	3.1×10^{-4}	2.9×10^{-4}

These dose estimates are at least three orders of magnitude less than either average natural background radiation (3.0 mSv a^{-1} , Neil 1985, referred to in Grondin 1994) or the CNSC dose limit for a member of the public (1 mSv a^{-1}).

Table A4.1.2-1
List of Occupational Radiological Hazards Associated with Normal Operation of the UFDC (Extracted from Grondin *et al.* 1994)

A) SURFACE ACTIVITIES
Transport Cask Receiving/Shipping - external radiation dose during handling of road and rail casks in the cask receiving/shipping area; - exposure to the ambient radiation field in the cask receiving/shipping area; - contact dose during cask inspection activities.
Transport Cask Storage - external radiation dose and ambient radiation field done during cask handling in the full (containing used fuel) cask laydown area.
Transport Cask Handling - external dose during activities such as guiding the casks to support trolleys, connecting hoses to vent the full casks, linking the overhead crane and impact limiters on full casks; - exposure to the ambient radiation field in the cask handling accessible area; - external dose during cask decontamination.
Used Fuel Handling - no radiological hazard would occur since fuel handling would be done by robot C and remote handling methods.
Used Fuel Temporary Storage - exposure to the ambient radiation field (water provides shielding from direct external radiation from the used fuel); - exposure to airborne contamination from drying used fuel modules.
Used Fuel Immobilization - no radiological hazard would occur since fuel handling would be done remotely by robot C and remote handling methods (including container repair). Containers would be repaired by similar remote means.
Disposal Container Handling - exposure to ambient radiation field when transporting the containers from the headframe area to either the temporary storage area or the waste shaft.
B) UNDERGROUND ACTIVITIES
Transport of Disposal Containers - exposure to ambient radiation field at the bottom of the waste transport shaft; - airborne radiological hazards from the natural background radiation (radon and its daughters) would be negligible since the airborne contamination would be removed by the ventilation system.
Emplacement of Disposal Containers - external radiological hazard to workers from shielded casks and groundshine from boreholes containing disposal containers. - airborne contamination in the event of the premature failure of a container.
Retrieval of Disposal Containers - external radiological hazard to workers from boreholes containing disposal containers, from shielded casks and from contaminated buffer and backfill material.

Table A4.1.2-2
Maximum Dose to Workers (from Grondin *et al.* 1994)

Job Category	Maximum Individual Worker Dose (mSv a^{-1})	Percentage of NEW Limit[1]
Management and Professional	10	50
Engineer / Technical (Operators)	17	85
Trades (Mechanics)	17	85
Support Staff	6	30

[1] Currently 20 mSv a^{-1} .

(ii) Non-Radiological Hazards

The chronic non-radiological hazards from the disposal centre operation were reviewed. They included exposure to dust, noise and emissions from equipment. At this conceptual design stage, it was not possible to quantify these hazards: the extent of airborne pollution would depend largely upon the efficiency of the ventilation system; dust in the rock crushing plant would be inherent, but quantities and concentrations have not been estimated at this stage of assessment; typical levels of noise and vibration from metal stamping machines in the basket and container fabrication area are unknown. In all cases, workers would be required to wear suitable eye, hearing and breathing protection (Simmons and Baumgartner 1994). It is expected that the implementing organization would have a better occupational health and safety record than the industry average because of the establishment of stringent working procedures, the implementation of health and safety programs and less emphasis on production targets.

A4.1.3 Potential Risk to the Natural Environment – Normal Conditions

Potential effects on the natural environment were discussed qualitatively by Grondin *et al*. (1994).

(i) Air Quality

Storage of the sand, gravel and bentonite clay, and the mined rock crushing and transfer operations would be done in enclosed spaces, thus reducing the potential for dust emissions during operations. The only source of dust would be the waste rock area. The application of dust suppression measures would likely be necessary to minimize dust emissions from the waste rock pile. Toxic chemical releases from the fuel were estimated to result in extremely small concentration in the air, of the order of 10^{-15} to 10^{-17} mg.m^{-3}.

(ii) Water Quality

Any effect on the water quality would be associated with operation of the water supply system, site runoff and waste waters discharge. The water treatment provisions and run-off control would prevent degradation of existing water quality. The toxic chemical releases from the fuel were estimated to result in very low concentrations that are insignificant fractions of regulatory and background concentrations. An exception is a radionuclide such as technetium, which is very rare in the environment. However, concentrations of technetium resulting from releases are not expected to lead to any significant impacts on the environment.

(iii) Potential Radiological Risk to Non-Human Biota – Accident Operations

The doses to non-human biota in the three reference environments were similar because the radionuclide concentrations in the three environments were similar. Thus, the detailed dose analysis was restricted to the Northern region.

The estimated annual dose rate to a fish, plant, mammal and bird in the environment near the UFDC was 2.4×10^{-5}, 1.8×10^{-5}, 1.7×10^{-5} and 1.7×10^{-5} mGy/d, respectively. For fish, the critical radionuclides were ^{134}Cs and ^{137}Cs, and the critical pathway was internal exposure. For plants, mammals and birds, the critical radionuclide was ^{90}Sr and the critical pathway was groundshine.

The background dose from natural and fallout sources to non-human aquatic and terrestrial organisms has been estimated to be 6.8×10^{-4} to 1.4×10^{-2} mGy/d (Laratta 1983). Since the estimated annual dose rate to non-human organisms from routine operations was much less than background levels, the impact is expected to be very small.

This exposure levels are also several orders of magnitude less than the doses (so called radiation benchmarks or Expected No Effect Values (ENEVs)) that are known not to have significant ecological effects on biota.

A4.2 POTENTIAL RISKS – ACCIDENT CONDITIONS

A4.2.1 Potential Radiological Risk to the Public – Accident Conditions

(i) Definition of Accident Scenarios

The selection of scenarios that could result in accidental release of radioactivity from the UFDC was based on a systematic review of the used—fuel handling procedures for the UFDC, consideration of accident conditions postulated at existing nuclear facilities, and a review of an accident safety assessment for high level radioactive waste repositories during the conceptual design stage (Jackson *et al.* 1985; Harris *et al*. 1990; Ma and Jardine 1990). When the consequences of an accident scenario were bounded by another accident scenario, the scenario was not fully analyzed.

The accident scenarios that are examined in detail in this assessment include the scissors lift failure where either a road or rail cask is dropped before transfer to the Module Handling Cell (MHC), the overhead carriage failure where a loaded fuel module is dropped on top of another module by the MHC emptying robot, and a failure in the shaft and hoisting facilities where a fuel container (inside a steel transfer cask) is dropped down the shaft (Russell and Villagran 1993). Each of these three events is used to generate two reference accident scenarios, a first set (S1, S3, V1) where correct operation of the ventilation system is assumed and a second set (S2, S4, V2) where loss of filtration of the ventilation exhaust is added to the event sequence.

The reference accident scenarios are summarized in Table A4.2.1-1

Table A4.2.1-1
Accident Scenarios for UFDC Operations

Scenario	Description
S1	**Scissors lift failure**: The open road/rail transportation cask is dropped before transfer of the fuel modules to the Module Handling Cell (MHC).
S2	**Scissors lift and ventilation failure**: Same as S1 but adding a failure in the ventilation system so that the airborne effluent by-passes the High Efficiency Particulate Air (HEPA) filters.
S3	**Overhead carriage failure**: A loaded fuel module is dropped on top of another loaded fuel module in the MHC.
S4	**Overhead carriage and ventilation failure**: Same as S3 but adding a failure in the ventilation system so that the airborne effluent by—passes the HEPA filters.
V1	**Failure in the shaft and hoisting facilities**: A fuel container is dropped down the shaft.
V2	**Failure in the shaft and hoisting facilities with ventilation failure**: Same as V1 but adding a failure in the ventilation system so that the airborne effluent bypasses the HEPA filters. *Facilities and equipment are described in detail in Simmons and Baumgartner (1994).

Protection against the set of external events normally considered in the design of nuclear facilities was also assessed. In general terms, the UFDC would be designed to withstand the most severe natural phenomena expected to occur once in a 100-year period, in a manner that will not result in an unacceptable risk to the public. Two other scenarios initiated by external events, with potentially serious consequences, were also analyzed: criticality due to flooding and vault cave-in.

An assessment of the potential criticality conditions (McCamis 1992) occurring as a result of flooding in the vault concluded that criticality was not possible. Based on the near-field and far-field stability studies (Tsui and Tsai 1994; Golder Associates 1993, respectively), no cave-ins serious enough to result in fuel container damage can reasonably be expected.

Analysis Results for Accident Scenarios

The analysis was performed using the same public safety assessment methodology as that used for accident analysis for licensing nuclear generating stations. The short-term radiological assessment model PSAC (Russell 1993e) was developed to calculate the radiological impact on the public from accidents during operation of the UFDC. Radionuclides released from the UFDC may lead to a radiation dose via a number of routes or pathways.

The maximum individual doses from the identified accident scenarios occurring during operation of the disposal centre are presented in Table A4.2.1-2 (Russell and Villagran 1993). The assessment results indicated that inhalation is the critical pathway during an accidental release of radionuclides from the UFDC. For accident scenario S1, the critical radionuclide was found to be ^{3}H, which accounted for 62% of the total dose. For accident scenario S2, the critical radionuclides were ^{241}Am, ^{241}Pu and ^{239}Pu, which accounted for 89% of the total dose.

Table A4.2.1-2
Maximum Individual Doses (mSv)

Accident Scenario	Adult	Infant
S1	2.3×10^{-4}	2.0×10^{-4}
S2	1.3×10^{-1}	2.0×10^{-1}
S3	7.7×10^{-5}	6.7×10^{-5}
S4	4.4×10^{-2}	6.5×10^{-2}
V1	2.9×10^{-4}	2.5×10^{-4}
V2	1.6×10^{-1}	2.5×10^{-1}

Based on current safety analysis practices, the consequences and probabilities of accidents could be compared to the following regulatory compliance limits (CNSC 1999) currently used for licensing nuclear generating stations (Ontario Hydro 1990a):

Table A4.2.1-3
Accident Classes and Dose Limits

Class of Accident	Dose Limit (mSv) for Whole Body Dose
Class 1: Accidents with a probability $F \geq 10^{-2}$	0.5
Class 2: Accidents with a probability $10^{-2} > f \geq 10^{-3}$	5
Class 3: Accidents with a probability $10^{-3} > f \geq 10^{-4}$	30
Class 4: Accidents with a probability $10^{-4} > f \geq 10^{-5}$	100
Class 5: Accidents with a probability $f < 10^{-5}$	250

The estimated annual frequency and the associated accident event class of the six postulated accident scenarios at the UFDC are shown in the following table. All doses to the critical group (either adults or infants) were found to be a small fraction of the dose limits.

Table A4.2.1-4
Estimated Annual Frequencies and Maximum Doses (to either Adults or Infants)

Accident Scenario	Accident Frequency	Accident Class	Maximum Individual Dose to the Critical Group (mSv)	Fraction of Dose Limit
S1	2.1×10^{-3}	2	2.3×10^{-4}	4.6×10^{-5}
S2	1.6×10^{-4}	3	2.0×10^{-1}	6.7×10^{-3}
S3	2.6×10^{-2}	1	7.7×10^{-5}	1.5×10^{-4}
S4	2.0×10^{-3}	2	6.5×10^{-2}	1.3×10^{-2}
V1	4.0×10^{-3}	2	2.9×10^{-4}	5.8×10^{-5}
V2	**3.0×10^{-4}**	**3**	**2.5×10^{-1}**	8.3×10^{-3}

The estimated doses for each accident scenario can also be compared with the Protective Action Levels (PALs) (Government of Ontario 1986) (see Table A4.2.1-5) to determine which, if any, protective measures would be required.

The worst possible scenario (V2) has a maximum possible infant whole body dose of 0.25 mSv. The lowest action level, corresponding to a ban on food and water consumption, would be triggered at a dose level above 0.5 mSv. Thus, by comparison with the PALs, none of the protective measures would be required even if an accident like this did occur.

Table A4.2.1-5
Protective Action Levels

Protective Action Levels (PALs) Government of Ontario (1986)		
Measure	**Lower Level Effective Dose (mSv)**	**Upper Level Effective Dose (mSv)**
Sheltering	1	10
Evacuation	10	100
Thyroid Blocking	--	--
Banning Food/Water Consumption	0.5	5

A4.2.2 Potential Risk to Workers – Accident Conditions

(i) Radiological Hazards

(a) Surface Facility Accidents

Malfunctions of the equipment in the surface facilities may lead to release of radioactivity (see Table A4.2.1-4. According to Grondin *et al*. 1994, accident scenario S1 involving an unsealed cask should have worse consequences on workers than all the transportation cask handling accidents. Potential radiological accidents in the Module Handling and Used Fuel Packaging Cells (scenario S3) would not cause a high dose because these areas are not accessible to workers during the immobilization process and consequences are not expected to exceed those for accident scenario S1. The dose to worker from this accident scenario (S1) was estimated at 16.5 mSv from the inhalation of volatile radionuclides and particulars.

(b) Underground Accidents

The extremely pessimistic scenario for a radiological accident associated with underground activities involves dropping a full shielded cask (or transfer cask) down the waste shaft onto another full cask (scenario V1 in Table A4.2.1-4). The shielded cask would rupture and the release of radionuclides would follow. The dose to worker from this accident scenario (V1) was estimated at 20.5 mSv from the inhalation of volatile radionuclides and particulars.

The dose to worker for both surface facility and underground accidents is below the 50 mSv limit and below the 100 mSv over 5-yr limit for NEWs. Even if the accident occurs towards the end-of-year, where a NEW was already exposed to 20 mSv/a, the accident dose is less than 30 mSv/a, so the total dose would not exceed 50 mSv/a.

Using the risk coefficient of 4×10^{-2} fatal cancer per Sv for workers (ICRP 1991), the maximum risk of a fatal cancer resulting from an accident at the UFDC, assuming that an accident has occurred and that the worker was there to be exposed, would be 8.2×10^{-4}. Because this is much less than one, no fatal cancer would be expected.

(ii) Non-Radiological Hazards

Operational activities are expected to lead to lost time and potential fatalities from both surface and underground activities. The estimated risks to workers are presented in Table A4.2.2-1 (extracted from Grondin *et al.* 1994 Table 6-32).

The total non-radiological effect on workers from accidents at the disposal centre during the operation stage is 0.25 fatalities and 60 lost-time injuries per year based on average industrial statistics (Zeya 1993a). These numbers are representative of average conditions in the industry, including the mining sector. It is expected that the implementing organization would have a better occupational safety record than the industry average because of the establishment of stringent working procedures, and the implementation of health and safety programs.

Table A4.2.2-1
Estimated Annual Acute Non-Radiological Risks During UFDC Operation Stage

Activity	Labour (Person-Hours a^{-1})	Fatality Rate Per 10^8 Person-Hours	Injury Rate Per 10^8 Person-Hours	Annual Fatalities	Annual Injuries
SURFACE OPERATIONS					
Used Fuel Packaging Plant	170 200	20.1	6.540	0.034	11.1
Basket Fabrication	101 200	21.4	6 190	0.022	6.3
Container Fabrication	132 500	21.6	6 000	0.029	8.0
Utility Services	207 000	29.3	5 940	0.061	12.3
Maintenance Services	116 600	12.2	3 080	0.014	3.6
Protective Services	132 000	3.8	3 970	5×10^{-3}	5.2
Technical Support	33 100	4.6	4 360	1×10^{-3}	1.4
Admin Services	22 100	3.3	110	1×10^{-3}	0**
TOTAL	915 000*	--	--	0.167	47.9
UNDERGROUND OPERATIONS					
EXCAVATION AND BACKFILL					
Shaft Facilities	32 700	30	4 180	0.01	1.4
Underground Ancillary Facilities	7 400	30	4 470	2×10^{-3}	0.3
EMPLACEMENT OPERATIONS					
Room Borehole Prep.	39 800	30	4 700	0.012	1.9
Borehole Emplacement	28 900	30	4 420	9×10^{-3}	1.3
Room Sealing	21 400	30	4 700	6×10^{-3}	1.0
Indirect Support	51 900	30	4 470	0.016	2.3
Capital Equipment	700	30	4 470	0**	0**
Project Indirects	104 900	30	4 470	0.031	4.7
TOTAL	287 600*	--	--	0.086	12.9
TOTAL FOR OPERATION		--	--	0.25 fatalities/yr	60 injuries/yr

* Number may not add up exactly because of rounding.
** Due to rounding.

A5.0 POTENTIAL RISK FROM USED FUEL TRANSPORTATION

Conceptual designs were developed for transportation of used fuel to a centralized facility (COGEMA, 2003). This facility may be a DGR or a CES facility, depending on the option chosen by the federal government after the review of options required by the Nuclear Fuel Waste Act. If continued storage at the current site is chosen, then no transportation system will be required.

The Used Fuel Transportation System (UFTS) will meet all regulatory requirements, and is designed for safe transport. It is designed to operate under an environmental management system based on the ISO 14001 standard (COGEMA, 2003).

A5.1 POTENTIAL RISK UNDER ROUTINE CONDITIONS

A5.1.1 Potential Risk to the Public – Normal Conditions

Under normal conditions of transport, radiological impact on members of the public would be limited to exposure to the low radiation fields around the cask.

Individual doses under normal transportation conditions were calculated (Kempe 1993a) using the models in the code INTERTRAN, sponsored by the International Atomic Energy Agency (IAEA). Doses were calculated for the following potentially exposed groups:

- the general population residing near the transportation route and pedestrians;
- the population near shipments during stops; and
- the population in other vehicles using the same transportation route.

The maximum doses estimated from transporting 250,000 used fuel bundles per year using the three modes were as follows (see Table A5.1.1-1):

Table A5.1.1-1
Maximum Dose to Individuals

Mode	Destination	Dose (mSv a^{-1})	Percentage of CNSC Dose Limit[d]
Road	All	0.09[a]	9
Rail	All	0.0004[b]	.4
Water	All	0.05c	5

[a] Dose to persons present at a truck stop used by the shipments.
[b] Dose to persons living beside the rail link.
[c] Dose to persons following a shipment through a canal (Kempe 1993a).
[d] The CNSC dose limit is 1 mSv a^{-1}.

All individual doses in normal transportation were well below the CNSC limit for members of the public and also well below the dose from natural background radiation, which is 3mSv^{-1} (Neil 1988)

A5.1.2 Potential Risk to Workers – Normal Conditions

(i) Radiological

Specific routine radiological hazards were identified through a systematic analysis of the reference transportation systems, using Ontario Hydro's experience in handling used fuel and experience in the transportation industry. The design of the reference system assessed is not yet refined to minimize worker doses. At the implementation stage, the ALARA (As Low As Reasonably Achievable) design process would be used. Various measures can be identified which would reduce both individual and collective worker doses.

During cask movement, the cab dose was calculated to be 0.00153 mSv h^{-1}. The IAEA guideline (IAEA 1985) of 0.02 mSv h^{-1} maximum is, therefore, met with a comfortable safety margin. Although no specific limits exist for rail and ship crews, dose rate estimates in the rail caboose and in the occupied portions of the tug/barge were well below the 0.02 mSv h^{-1} maximum specified for truck drivers.

The maximum annual individual doses received by members of the transport crews were estimated to be 2.4 mSv a^{-1}, 0.44 mSv a^{-1} and 10 mSv a^{-1} for road, rail and water, respectively. Therefore, radiation doses received by workers during transportation of used fuel were less than the Nuclear Energy Worker (NEW) dose limit of 20 mSv a^{-1}.

For cask handling at the nuclear generating stations, assuming road transport, 3 shifts of 4 workers per shift, and 292 casks shipped from each station per year, the maximum annual individual dose would be approximately 10.6 mSv a^{-1}. This dose is also well below the NEW dose limit of 20 mSv a^{-1}.

(ii) Non-Radiological

For normal transportation, estimates of non-radiological hazards were derived based on experience with similar industries, using equipment of the same size and type. Where quantification was not possible, a qualitative analysis was performed.

The analysis assumed that worker protection measures in accordance to the Ontario Occupational Health and Safety Act (Government of Ontario 1990c) would be implemented to ensure adequate control of noise and exhaust emissions in the working area.

A5.1.3 Potential Risk to the Environment – Normal Conditions

The individual radiological doses to humans were calculated assuming high or 100% occupancy. In addition, since the assessed doses were due entirely to external radiation from the cask, absorbed dose from external radiation from the cask, absorbed dose and dose equivalent were assumed to be, for practical purposes, the same. A maximum dose of 0.09 x 10^{-3} Gy a^{-1}, or about 2.5 x 10^{-4} mGy d^{-1} was estimated for non-human biota. No potential significant ecological effects would be expected at such radiation levels (See Appendix D).

Regarding potential effects on the natural environment (normal conditions), the analysis showed that:

- atmospheric emissions from used fuel transportation should have minimal effects on air quality along the transportation corridors;
- noise and traffic increases would be small enough to be within the normal day-to-day variations of existing transportation traffic; and
- commitment of natural resources to used fuel transportation would be small.

A5.2 POTENTIAL RISK – ACCIDENT CONDITIONS

A5.2.1 Potential Risk to the Public – Accident Conditions

(i) Radiological

The used fuel is not flammable, and only conventional fire hazards would be associated with an accident to the shipment. However, a severe transport accident involving a used fuel shipment could potentially cause radiation doses to members of the public in two ways:

- loss of shielding leading to increased exposure to direct radiation from the used fuel; and
- seal failure and fuel damage leading to escape of airborne radioactive material from the cask.

a) Accident Severity Categories

To examine the radiological impact of hypothetical accidents severe enough to cause a breach of the cask integrity, the range of postulated accident conditions was divided into a number of accident severity categories. The first category consisted of those accidents that were not severe enough to affect the integrity of the cask, and for which the radiological consequences were bounded by the allowable leakage limits imposed by the AECB for the cask. The other categories were chosen to represent a spectrum of accident conditions for which the release from the used fuel transportation cask would vary from minimal up to the most severe credible. The spectrum of possible accidents was broken down into ten categories. The radioactive release in each severity category was characterized in terms of the following:

- the occurrence of seal failure (which might permit escape of gases and fine particulates from the cask);
- the fuel temperature reached (which would affect the release of volatiles from failed fuel, might cause additional fuel failure, and might result in oxidation of failed fuel);
- the fraction of fuel subject to impact rupture; and
- the fraction of fuel subject to creep rupture.

These parameters were in turn related to the impact and thermal environment experienced by the cask. The accident severity categories were, therefore, characterized by the impact and thermal environment experienced by the cask (Grondin *et al.*, 1994). Possible impacts were divided into three ranges: 0 - 50 km h^{-1}, 50 - 75 km h^{-1}, and over 75 km h^{-1}. Note that these speeds represent impact with an unyielding surface although, in reality, objects involved in a collision are not unyielding. This was taken into account in deriving the impact speed with a real target needed to obtain an impact equivalent to a 50 km h^{-1} or 75 km h^{-1} speed of impact with an unyielding target. The thermal environment was characterized by the fire duration, assuming an engulfing fire of 800°C. The possible durations were 0 - 0.5 h, 0.5 - 1 h, 1 - 6 h, and greater than 6 h.

The ten categories were used in the calculation of radioactive releases from the cask and in the estimation of probability of accidents. In the final calculations, the release in Categories 3 and 4, Categories 6 and 7, and in Categories 9 and 10 were found to be the same. In the subsequent calculation of doses due to radioactive releases from the cask, the ten categories were condensed into seven (see Table A5.2.1-1 extracted from Grondin *et al*., 1994).

(b) Accident Probability

A simplified form of fault tree analysis was used to estimate the probability of each severity category, for each mode. This methodology has been commonly used to estimate the probability of rare scenarios where little or no historical data were available for those specific scenarios. The event probabilities (e.g. probability of a collision occurring in a particular speed range) were taken from the literature. Conservative simplifying assumptions were made (e.g. as to orientation of the cask at the time of impact).

The conditional probability of an accident in each severity category (i.e. the probability that an accident occurs in that severity category, given an accident has occurred) was summarized as follows:

Table A5.2.1-1
Fraction of Accidents in Severity Category

Seventy Category	Given an accident, probability of this accident being of a given severity		
	Road	Rail	Water
1	0.99998	0.99988	0.99999
2	10^{-5}	10^{-4}	0
3/4	10^{-7}	10^{-6}	10^{-8}
5	10^{-5}	10^{-5}	10^{-6}
6/7	10^{-8}	10^{-7}	10^{-7}
8	0	10^{-5}	10^{-5}
9/10	0	10^{-7}	10^{-6}

c) Maximum Short-Term Individual Dose

Short-term and long-term dose to the public following a potential transportation accident were estimated. Exposure pathways included in the short-term dose estimate were:

- internal exposure following inhalation of airborne radioactivity;
- external exposure to radiation from radioactivity deposited on the ground (groundshine).

The maximum (worst case) individual dose calculated for severe accident conditions was about 10 to 40 mSv, for an accident frequency of 3 x 10^{-6} per year or less (see Table A5.2.1-2 extracted from Grondin *et al*., 1994). The same radiation dose limits that applied to the safety analysis for disposal centre operation were assumed to apply to transportation accidents. The worst case transportation accident, with an annual frequency of less than 10^{-5}, would fall in class 5, bounded by a limit of 250 mSv. The maximum doses, 10 to 40 mSv for infants, would only be a fraction of this limit.

Table A5.2.1-2
Summary of Maximum Individual Doses Due to Transportation Accidents

Mode of Transportation	Maximum Individual Dose (mSv)			Annual Frequency of Worst Case
	90^{th} Percentile	Worst Case Adult	Worst Case Infant	
Road	3	9	13	3 x 10^{-6}
Rail	30	28	40	4 x 10^{-7}
Water[1]	30	28	40	8 x 10^{-7}

1 The maximum individual dose is given for the water portion of the route. The maximum individual dose for the road or rail portion would be the same as for road or rail transportation alone.

(d) Long-Term Doses

Adult doses from long-term groundshine and re-suspension were compared with the short-term doses. With cleanup, the individual dose would increase by about 60% if long-term pathways were included, but if no cleanup actions were undertaken, the dose could increase by a factor of ten, due to re-suspension. The collective dose would be affected most by inclusion of the long-term pathways, because of the effect of cesium deposition from the air during elevated releases.

Exposure via the foodchain was not included in the main calculations, because control of food supplies would be exercised, and would be the main factor affecting exposure. Calculations (Kempe 1993a) indicated that, for an accident in Severity Category 2, the foodchain dose at 100 m, without intervention (i.e. cleanup), might be a factor of 10 or so more than that for inhalation, or about twice the dose for inhalation and groundshine together. This dose is in the range (>0.5 mSv; Government of Ontario 1984) at which intervention might be considered, but, given the conservatism in the calculation, it is unlikely the intervention would be required.

(ii) Non-Radiological

Non-radiological accident consequences, such as material damage to vehicles, personal injury and, in extreme cases, loss of life were examined.
The expected number of traffic accidents per year on the reference routes was calculated based on reported accident rates for general traffic. The number of these accidents that could statistically involve a used fuel transportation vehicle and their consequences were also estimated.

The consequences of a used fuel transportation accident for all three modes of transport are as follows:

Table A5.2.1-3
Consequences of Used Fuel Transportation Accidents

Consequences of Used Fuel Transportation accidents		Number of consequences per year		
	Location	**Southern**	**Central**	**Northern**
ROAD				
- Material damage only	Rural	0.53	1.13	2.38
	Suburban	0.02	0.03	0.07
	Urban	0.01	0.01	0.01
- Personal injury	Rural	0.27	0.56	1.20
	Suburban	0.01	0.02	0.04
	Urban	0.01	0.01	0.01
- Loss of life (including drivers)	Rural	0.005	0.01	0.02
	Suburban	0.0002	0.0003	0.0007
	Urban	0.0001	0.0001	0.0001

Table A5.2.1-3 (Cont'd)
Consequences of Used Fuel Transportation Accidents

RAIL				
- Personal injury	Rural Suburban Urban	0.35 0.014 0.02	0.14 0.007 0.014	0.02 0.007 0.007
- Loss of life	Rural Suburban Urban	0.11 0.004 0.006	0.04 0.002 0.004	0.06 0.002 0.002
WATER-ROAD				
- Personal injury	Open Water Channel/River Road-Rural Road-Suburban	- - - -	0.0002 0.0004 0.32 -	0.0002 0.0004 0.24 0.03
- Loss of life	Open Water Channel/River Road-Rural Road-Suburban	- - - -	0.0004 0.0008 0.006 -	0.0004 0.0008 0.004 0.0005
WATER-RAIL				
- Personal injury	Open Water Channel/River Road-Rural	- - -	0.0002 0.0004 0.05	0.0002 0.0004 0.6
- Loss of life	Open Water Channel/River Road-Rural	- - -	0.0004 0.0008 0.015	0.0004 0.0008 0.18

Traffic accidents could interrupt the normal road, rail and water flow of traffic and disrupt the surrounding land and water uses. The establishment of an emergency response plan, required under the Transportation of Dangerous Goods Act, should minimize impacts.

A5.2.2 Potential Risk to Workers – Accident Conditions

(i) Radiological

The analysis used the same accident severity categories as those in the public radiological safety analysis presented above.

The potential pathways to worker exposure were:

1. inhalation of radioactive material in the plume;
2. inhalation of re-suspended radioactive materials;
3. external radiation from ground deposits (groundshine); and
4. direct external radiation from radioactive material remaining in the cask.

For the transportation crew, pathways 2, 3 and 4 were insignificant compared to 1 due to the short length of time over which the crew would be exposed and the small amount of ground deposits anticipated within 50 m of the accident site. In addition, no loss of cask shielding is expected; therefore, the contribution from 4 was equal to the chronic dose rate.

The maximum acute dose to an individual worker resulting from a cask release accident based on the accident severity scheme developed for the public safety analysis is presented in Table A5.2.2-1.

The worst credible accident could result in a dose of about 190 mSv. This dose would not result in any acute (or non-stochastic) effects. The probability of such an accident is extremely low.

Table A5.2.2-1
Maximum Acute Radiation Dose to a Worker
for each Model and Accident Severity Category (in mSv)

Accident Severity Category[1]	Mode	
	Road and Water-Road	Rail and Water_Rail
1	0	0
2	64	190
3/4	64	190
5	0	0
6/7	65	190
8	0	0
9/10	64	190

1 See Table A5.2.1-1 for fraction of accidents in each severity category and Grondin *et al*. (1994) for the annual probability of a release accident.

(ii) <u>Non-Radiological</u>

It is assumed that cask handling procedures would comply with the requirements of the Occupational Health and Safety Act Regulations 213/91 and 854 on the safe operation of cranes, and would follow the guidelines of the Construction Safety Association (1975). It was also assumed that working conditions for the driving crew would comply with the Ministry of Labour regulations.

Estimates of the non-radiological risks were based on adjusted fatality data obtained from the Workers Compensation Board (Social Data Research 1986) and on labour requirements for each activity. It was anticipated that the fatality rates in the used fuel transportation activities would be lower than the industrial rates because of the extensive training, safety procedures and standards that would be applied to the system operation. The potential non-radiological hazards would be associated with cask handling (i.e. dropping of cask, cask maintenance), and cask transport (i.e. normal traffic accidents, floundering, capsizing, explosions, fires and cargo-related accidents), and would also include miscellaneous hazards such as falling, machine and tool injuries, and on-site vehicle/personnel collisions. Over the 41 years of disposal operation, the maximum number of worker fatalities resulting from used fuel transportation is estimated as less than 2 (associated with transportation by road to the Northern region.

A5.2.3 Potential Risk to the Environment – Accident Conditions

Contents of diesel tanks or radiator water could be spilled as a result of impact. The diesel tank could also catch fire. Given that these hazards would be of the same nature as for standard transportation activities, and the small amount of hazardous material available for release, there should be minimal impacts on the environment. The operation of an emergency response plan should also minimize the adverse impacts of used fuel traffic accidents on the environment.

Doses to non-human biota under transportation risk accident scenarios were discussed by Kempe (1995). Annual doses to non-human biota from Kempe (1995) are summarized in Table A5.2.3-1. According to this analysis, the doses to representative non-human biota are smaller than natural background, except for fish. The dose to fish from Cs is about 2.7 mGy/a, which is less than the dose at which subtle chronic effects might be observed (IAEA 1992).

Table A5.2.3-1
Annual Doses to Representative Non-Human Biota for an Initial Contamination Level of 10^4 Bq/m^2 (extracted from Kempe, 1995)

Target organism and radionuclide	Air pathways Initial Dose (mGy)	Air pathways Ongoing (at 0 a) (mGy/d)	Water pathways Ongoing (at 0 a) (mGy/d)
Plant			
- actinides	10	2.7e-6	2.7e-2
- Cs	1	2.7e-4	2.7e-2
Mammal			
- Actinides	1e-3	2.7e-6	2.7e-4
- Cs	1	2.7e-3	2.7e-2
Bird			
- Actinides	1e-3	2.7e-6	2.7e-4
- Cs	1	2.7e-3	2.7e-2
Fish			
- Actinides			2.7e-1
- Cs			2.7

A6.0 POTENTIAL RISK OF FACILITY DECOMMISSIONING, EXTENDED MONITORING AND CLOSURE

A6.1 PRE-DECOMMISSIONING MONITORING

At the end of the disposal facility operation, performance and environmental monitoring of undefined duration may take place to provide sufficient assurance of the disposal vault's performance and continued environmental protection to be able to proceed to vault closure. No new effects from the extended monitoring activities would be expected.

A6.2 CONTAINER RETRIEVAL

During the extended monitoring period or during operation, container retrieval may be required (e.g. to demonstrate performance or for safeguards verification). Retrieval procedures were developed as part of the reference design. If retrieval is necessary, environmental protection would be ensured through proper waste water and solid waste management procedures during the buffer cutting and retrieval operations. Occupational safety would be ensured through the use of shielding rings, skirts, decks and housings necessary to minimize radiation exposure to equipment operators. The air in the room would also be filtered to remove particulates that might be present in quantities large enough to be a risk to the operators.

A6.3 DECOMMISSIONING

The decommissioning stage of the UFDC life-cycle would begin after the waste emplacement operations have been completed, sufficient performance monitoring data have been collected to support the application for approval to decommission and seal, and the decommissioning plans have been approved by the appropriate regulatory authorities. The decommissioning plans would outline the specific decontamination, vault sealing, dismantling, demolition, waste removal, and site restoration and marking activities, their durations and their likely effects. Decommissioning would end when the vault has been sealed, and all surface facilities have been decontaminated and removed.

A6.3.1 Public Safety

The reference design assumed that the criteria used in the decommissioning of the Gentilly I reactor would apply to decommissioning of the UFDC. These criteria should make the site surface suitable for unrestricted public access after decommissioning. The emissions of radionuclides from the facility during decommissioning are expected to be small compared to emissions during the operating stage, since the primary source of radioactivity (the used fuel) would have been completely disposed of. Dismantling activities, which could expose activated product sources, would not create sources of the same order of magnitude as those from the operating UFDC. The radiological exposure of members of the public during this stage is expected to be a small fraction of the regulatory limit or exposure from natural background radiation, and the exposure resulting from decommissioning and closure is expected to be even smaller.

A6.3.2 Occupational Safety

Non-radiological occupational hazards during decommissioning would be similar to hazards encountered in any large demolition project, such as airborne pollutants (dust and exhaust emissions from engines), noise and vibration. Provided that procedures were in accordance with regulatory requirements on conventional hazards in the Occupational Health and Safety Act of Ontario (Government of Ontario 1990c), non-radiological effects would be minimized.

The decommissioning risk from non-radiological sources would be less than 1 fatality and about 81 lost time injuries over the decommissioning period.

The average dose per worker was calculated to be 0.1 to 0.2 mSv over a 2-year decontamination period, which is well below the CNSC criteria for NEWs.

A6.3.3 Natural Environment

Although no experience in the decommissioning of a used fuel disposal facility exists, considerable experience has been gained within the nuclear industry in all aspects of nuclear facility decommissioning. In the U.S., highly radioactive fuel reprocessing facilities have been decontaminated and a few have been partially converted to other uses.

Potential effects of decommissioning activities include the following:

- fugitive dust emissions could arise during the demolition of site buildings, and the use of heavy equipment;
- demolition activities could change the site topography and, if not properly managed, could increase site run-off leading to sedimentation of nearby water bodies;
- demolition of the water intake and discharge structures could disturb aquatic life near the shore by increasing water turbidity and sediment concentrations;
- waste water from decontamination activities could affect water quality; and
- local wildlife could be disturbed by the increased traffic and noise from blasting and other demolition activities.

Possible mitigation measures were identified which would minimize effects of decommissioning activities on the natural environment.

In general, the potential effects of decommissioning would likely be less than those during construction or operation. According to the reference UFDC design, radioactive waste from decommissioning of the used fuel disposal facility would be shipped off site to an existing licensed disposal facility for low and intermediate level radioactive wastes (Simmons and Baumgartner 1994). It is inappropriate at this stage to speculate about possible uses of the used fuel disposal site after decommissioning and closure.

A6.4 VAULT CLOSURE

Closure would involve the removal of instruments from surface boreholes used for extended monitoring and the sealing of these boreholes. The objective of closure would be to return the site to a state such that safety does not depend on institutional controls. The closure stage could begin either immediately after the decommissioning stage or after a further monitoring stage. The closure stage would end when all monitoring boreholes, that could compromise long-term safety if left unsealed, were sealed. Effects of closure on public safety and the natural environment are expected to be much less than those during the construction, operation, and decommissioning stages.

A7.0 POTENTIAL RISK FROM DEEP GEOLOGICAL DISPOSAL IN THE POST-CLOSURE PHASE

A7.1 INTRODUCTION

Three Canadian post closure safety assessments have been completed for hypothetical repositories located in Canadian Shield rock – the Environmental Impact Statement (EIS) study (AECL 1994, Goodwn *et al*. 1994a), the Second Case Study (Goodwin *et al*. 1996) and the Third Case Study (Gierszewski *et al*. 2004; Garisto *et al*. 2004). Also, studies have been published for similar repository concepts in other countries, notably Sweden (SKB 1999), Finland (Vieno and Nordman 1999), Japan (JNC 2000), USA (BSC 2001) and Switzerland (NAGRA 2002).

The 1994 EIS study considered titanium alloy containers with 72-fuel-bundle capacity placed vertically into boreholes along the vault rooms, and assumed the repository was located in sparsely-fractured granitic rock with very low permeability. The Second Case Study considered 72-bundle copper containers placed horizontally within the vault rooms, and assumed the repository was located in granitic rock with substantially higher permeability.

The following discussion is extracted from the Third Case Study (Gierszewski *et al*. 2004), except where indicated otherwise.

The Third Case Study (TCS) evaluates an updated repository design and site assumptions. The repository has sufficient capacity to hold about 3.6 million CANDU used fuel bundles, an amount corresponding to all the used fuel from present Canadian nuclear power stations to the end of their planned life, the TCS uses a copper and steel container with 324-bundle capacity, placed horizontally within the vault rooms. The site is also different from that assumed in the previous Canadian studies. The geologic setting is hypothetical, but believed to be representative of potential sites that could exist within the Canadian Shield. The TCS assumes the repository is located in granitic rock that is characterized by an intermediate permeability and a geostatistically-generated discrete fracture network.

The TCS assesses several scenarios: a Base Scenario, a Defective Containers Scenario and a Human Intrusion Scenario.

The so called "Base Scenario" of the TCS (Gierszewski *et al*. 2004) assumes that the various components of the repository perform as expected. In particular, it assumes that the wastes are emplaced in containers that have been carefully engineered, as part of the repository system, to remain intact over timeframes in excess of 100,000 years. In this scenario, no radionuclides are released from the containers for hundreds of thousands of years (McMurry *et al.,* 2003). The used fuel radioactivity decays until there is only the residual radioactivity of natural uranium and its decay chain. When, at some point, in the far future, the containers eventually rupture, the radiological risk would be very low – equivalent to living near a deep natural uranium ore body.

However, although the repository will be built (if this is the selected option) to design specifications, the TCS assumes that some containers are emplaced with defects that penetrate the copper shell, due to, for example, undetected fabrication flaws or installation damage. These defects lead to early development of a pathway for radionuclides release into the groundwater around the repository. According to Maak *et al*., (2001), based on experience with the manufacturing of other high-quality nuclear-grade components, the number of containers that would have full penetration defects in the copper shell and escape detection is expected to be very small. Using the best estimate from Maak *et al*., 2001, statistically, 2 of the approximately 11,200 containers in the repository would have full penetration defects at the time of emplacement. Similar assumptions were made in past assessments.

The Defective Containers Scenario is described in detail by Garisto *et al*. 2004. This scenario takes into account the occurrence of a small number of containers with undetected manufacturing or installation defects. This scenario will be used in this document to discuss risks in the post closure phase under normal conditions.

An assessment of hypothetical human intrusion scenarios was described by Gierszewski *et al.* 2004. This assessment will be used in this document to discuss risks under non-routine conditions.

A7.2 POTENTIAL RISK FROM DGR IN THE POST CLOSURE PHASE – NORMAL CONDITIONS, INCLUDING THE POSSIBILITY OF UNDETECTED DEFECTIVE CONTAINERS

A7.2.1 Potential Risk to the Public

The assessment of risk to the public in the post closure phase is based on the Defective Container Scenario (Garisto *et al*., 2004). This scenario considers a case in which some containers fail early because of the presence of undetected manufacturing or installation flaws.

It should be noted that a small penetration in the copper shell (i.e., small enough to be undetected by the two inspection methods) would not lead to immediate radionuclide releases. It would take some time for groundwater to saturate the emplacement room, enter the container, reach the fuel bundles, and cause failure of the cladding and contact the used fuel.

After the used fuel is contacted by groundwater, a pathway exists for radionuclides to be released from the container and to move through and with the groundwater toward the surface biosphere. The focus of the analysis for the Defective Container Scenario is, therefore, on radionuclide release from the used fuel, transport of radionuclides through the container defect, vault, geosphere and to the surface, and on radionuclide transport within the local biosphere, leading to radiological doses to humans living in the vicinity of the repository.

The geosphere is assumed to have characteristics which are typical of the Canadian Shield.

The surface biosphere near the repository site has the characteristics of the Shield region of central Canada. The properties of the biosphere are assumed to be constant. Treatment of climate change is not included in this scenario. However, normal present-day variation of climate and other biosphere parameters is considered in probabilistic and sensitivity studies.

Another important feature of the biosphere is the people who would be affected by the repository. Following international practice, the concept of a critical group is used. The critical group is defined as a hypothetical group of people, with similar characteristics, that is expected to receive the largest annual doses from radionuclides released from the repository.

Specifically, for the TCS, it is assumed that members of the critical group spend all their lives in the area near the surface discharge locations and obtain all their food, water, fuel and building materials from the contaminated zone. The food includes plants grown in a garden, domesticated animals and fish. This lifestyle is referred to as "self-sufficient farmer".

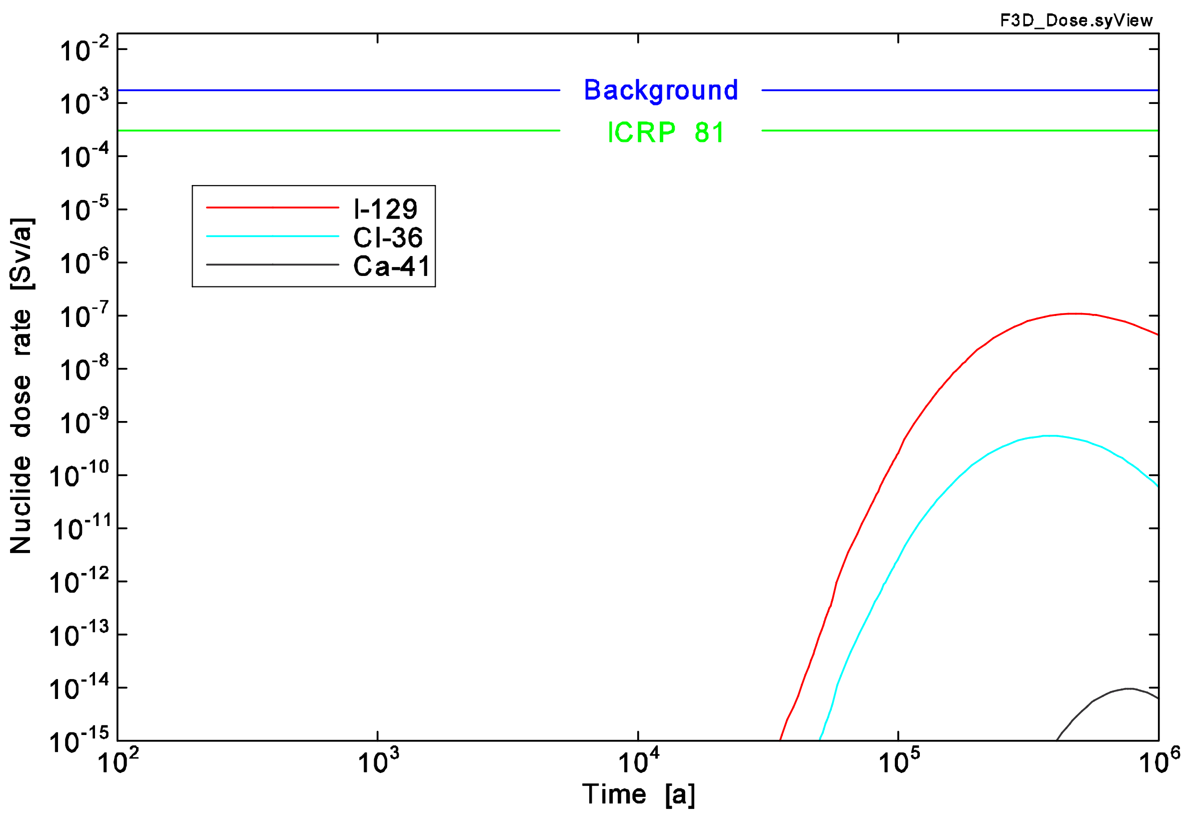
Other lifestyles could also be considered, and for a given site local lifestyles would be explicitly considered. However, the self-sufficient farmer has been found in previous studies to be a good indicator of risk for a range of plausible lifestyles (IAEA 2003, Zach *et al.* 1996), and is the only one evaluated in the Third Case Study.

The analysis included a Reference Case and several Sensitivity Cases.

For the Reference Case calculations, two failed containers were assumed to occur in the emplacement room nearest the east edge of the repository (i.e., nearest the fastest groundwater pathway to the surface). Groundwater was conservatively assumed to contact the used fuel at 100 years.

Figure A7.2-1 shows the results of the Reference Case calculations. It shows that the main contributor to dose was I-129, followed by Cl-36.

Figure A7.2-1
Dose rate impact for the Reference Case (Case 2 geosphere, self-sufficient farmer critical group, 738 m^3/a well). Dose rates from the U-238 and Np-237 chains are less than 10^{-15} Sv/a
Extracted from Garisto *et al.*, 2004



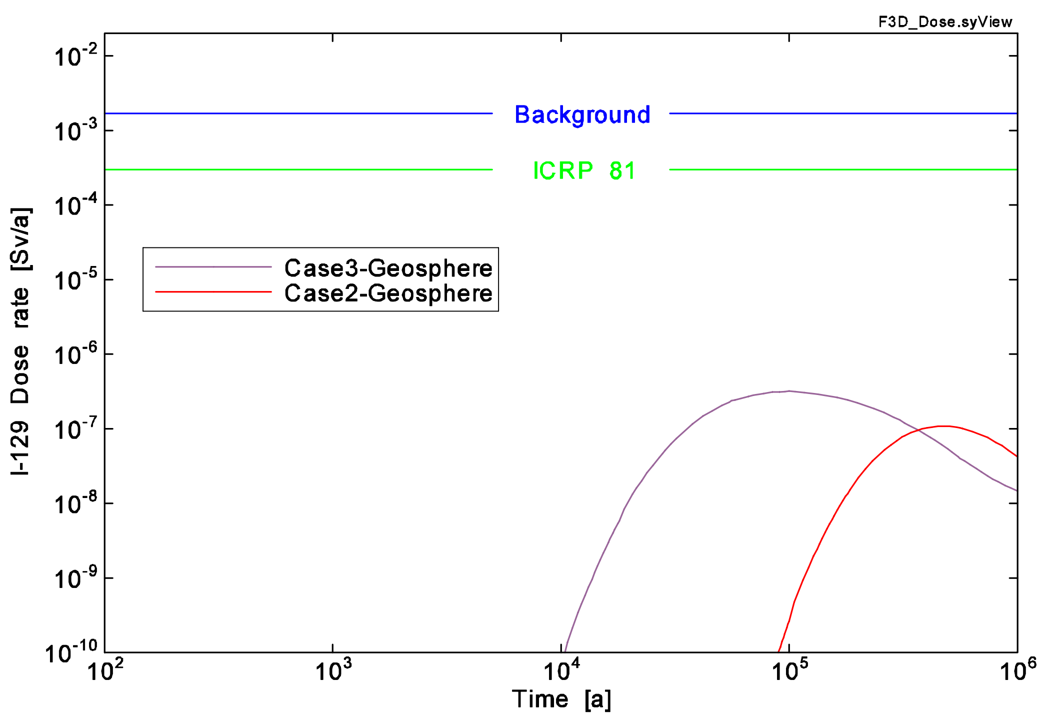
All actinide dose rates were less than 1e-15 Sv/a over the million year time frame.

The peak dose rate of about 10^{-7} Sv/a is well below the average Canadian natural background dose rate and the ICRP 81 recommended dose rate constraint. Although the TCS assumed that the defective containers were emplaced with defects and that groundwater contacted the used fuel 100 years later, the calculated time of peak dose rate is almost 500,000 years after disposal. This is due to the retention and delay characteristics of the engineered barriers and the geosphere. Only very long-lived radionuclides or their progeny can reach the surface. As the results show, it is specifically I-129 and Cl-36 that might do so because of their long life and because they are quite mobile, with low sorption by vault and geosphere materials.

Figure A7.2-2 shows the results of a Sensitivity Case, which used a Case 3 geosphere. (The Reference Case used a Case 2 Geosphere). The Case 3 geosphere has a relatively high permeability. For this higher permeability rock, the time of the peak dose occurs earlier, but the peak dose rate from I-129 is only about a factor of 3 higher and still much less than the ICRP 81 dose rate constraint.

Figure A7.2-2

Effect of geosphere permeability on I-129 dose rates to a self-sufficient farmer critical group. In the Case 3 geosphere (high permeability), the peak dose rate occurs much sooner and is about a factor of 3 higher (extracted from Gierszewski *et al.*, 2004).



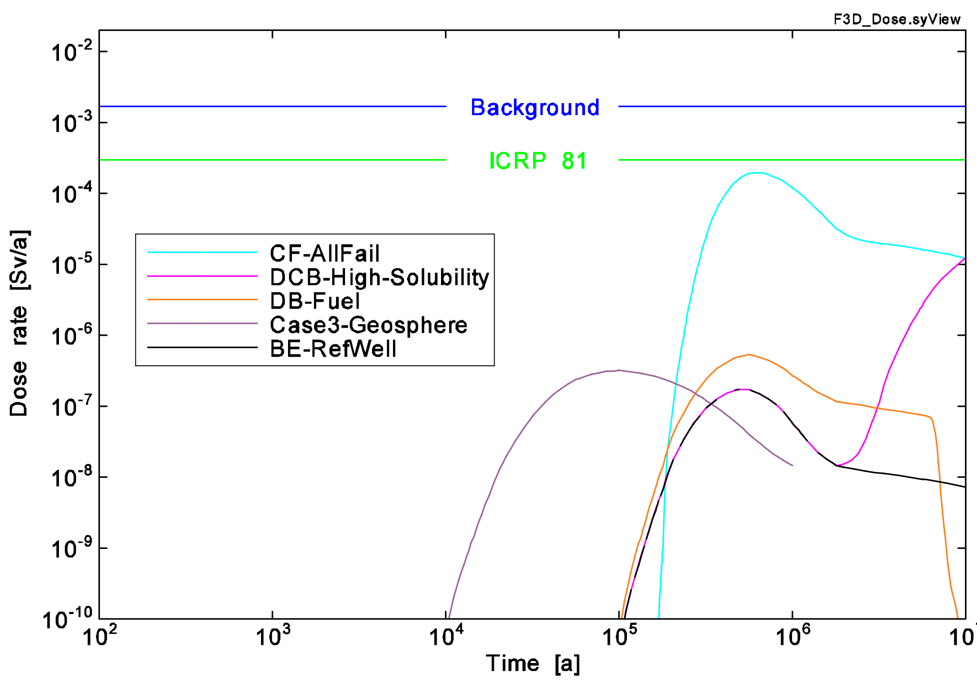
Additional Sensitivity Analysis considered the following cases:

- Best-estimate parameter values, with a self-sufficient farming household and well-water supply (this is the Reference Case);
- Best-estimate parameter values, but with either a lake water supply or a large well demand;
- Defective barrier cases, in which various barriers are assumed to be less effective than designed (e.g., fuel dissolution rate is higher than expected);
- High solubility case, in which there is no radionuclide precipitation in the container;
- Low sorption case, in which radionuclides are not sorbed in the geosphere;
- Simultaneous failure of all containers at the minimum design lifetime of 100,000 years.

The results of these cases are described in the TCS Defective Container Scenario report (2004). Those cases that had the highest peak dose rates are shown in Figure A7.2-3. The calculated dose rates are generally well below the ICRP 81 recommended dose constraint. The one exception, in which the calculated peak dose rate approaches the ICRP 81 dose rate constraint, is for the unlikely case of simultaneous failure of all containers.

Figure A7.2-3

Dose rate results from selected sensitivity study cases. The cases shown had the largest increase in the peak total dose rate relative to the Reference Case (BE-RefWell). They include a 10-fold increase in the UO_2 dissolution rate (DB-Fuel); no precipitation for all elements (DCB-High-Solubility); all containers fail at 100,000 years (CF-AllFail); and a high permeability geosphere (Case3-Geosphere). Extracted from Gierszewski *et al.*, 2004



The Reference Case and Sensitivity Analysis Cases addressed best estimate and specific "what if" scenarios, respectively. In these cases, the model input parameters are set to specific values which depend on the case. However, many of the parameters are uncertain or have a natural degree of variability. This means that they are more generally characterized by a range or distribution of values, as described in the TCS Reference Data and Codes report (Garisto *et al.* 2004a). This uncertainty in the input parameter values leads to an uncertainty in the consequences. A particular concern is the possibility and likelihood of large dose consequences that might occur under some combination of parameter values.

In order to systematically account for parameter uncertainty, the SYVAC3-CC4 system model was used in probabilistic mode. The probabilistic assessment uses results from 45,000 simulations; each selected using a simple random sampling (Monte Carlo) approach. That is, for each simulation, a value for every parameter is sampled randomly from its probability density function (taking into account that some parameters are correlated). The sampled values are then used in CC4 to estimate the resulting dose rate. Each of these thousands of simulations produces a different estimate of impact. Together they produce a distribution of the impact that directly reflects the underlying uncertainty in the parameters.

The thousands of results can then be examined to determine, for example, which combinations of parameter values lead to large (or small) calculated dose rates. One important statistical result is the average dose rate, an estimate of the expected dose rate impact. Another result of interest is the 95th percentile dose rate, which is a measure of high dose rate cases. In particular, this is the dose rate which is larger than 95% of all the calculated dose rates.

Finally, we are interested in identifying the parameters that contribute the most to the uncertainty in the calculated dose rates. This sensitivity analysis is carried out using a sampling approach, called iterated fractional factorial Latin hypercube design, which is particularly effective when dealing with large numbers of parameters (Saltelli *et al*. 1995).

For all these simulations, the nuclides used were as identified in a radionuclide screening assessment, and include fission products and two actinide chains (italicized nuclides were modelled via secular equilibrium):

C-14, Ca-41, Cl-36, I-129, Se-79, and Tc-99
Pu-241 → Am-241 → Np-237 → *Pa-233* → U-233 → Th-229 → *Ra-225* → *Ac-225*
U-238 → *Th-234* → U-234 → Th-230 → Ra-226 → *Rn-222* → *Pb-210* → *Bi-210* → *Po-210*

The key assumption of the Defective Container Scenario is that some containers have a defect at the time of emplacement. The probability of such a defect in any given container is considered to be random, ranging from 10^{-4} to 10^{-3} per container, with a best-estimate defect rate of $2x10^{-4}$

per container. Figure A7.2-4 shows the distribution of container failures from the 45,000 random simulations. The results indicate that there are most likely 2 failed containers (the peak in the profile), and that on average there are 3.5 failed containers.

The maximum number of containers failing in any one of these 45,000 simulations is 22, with 9 of those failures in the vault Sector 1, the vault sector for which the transport path to the surface has the shortest transit time and leads directly to the well.

Figure A7.2-5 shows the histogram of calculated peak total dose rates (the maximum dose rate summed over all nuclides, regardless of the time at which the maximum occurred) to the self-sufficient farmer critical group that used a well. The average peak total dose rate is 3.6×10^{-7} Sv/a. For comparison, the peak total dose rate was 1.7×10^{-7} Sv/a for the Reference Case. These values are in good agreement because the dose rates are dominated by I-129, and there is relatively low uncertainty in I-129 behaviour.

Figure A7.2-4
Distribution of the number of container failures from 45,000 randomly sampled simulations. The most likely number is two failed containers. Extracted from Garisto *et al.*, 2004

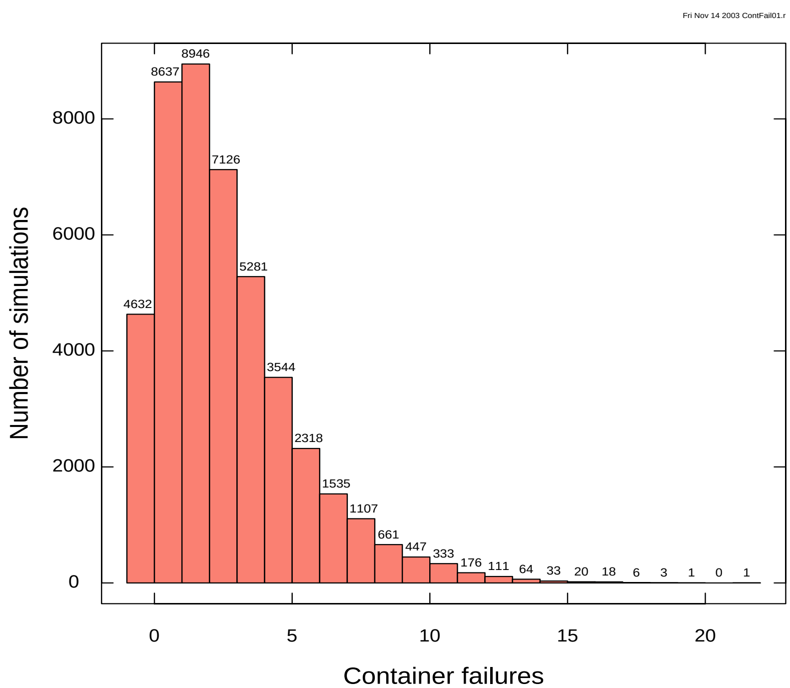
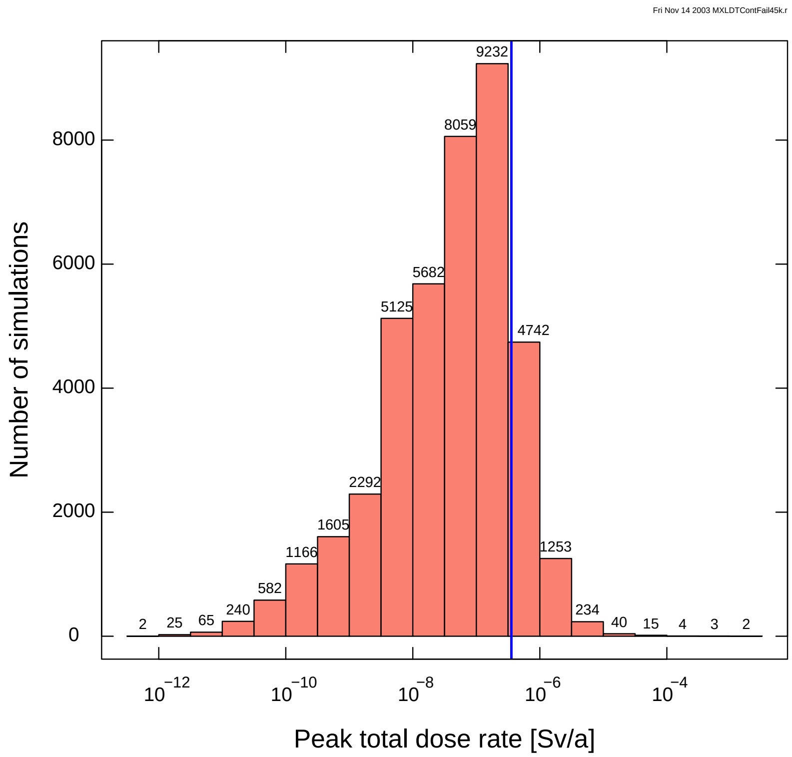


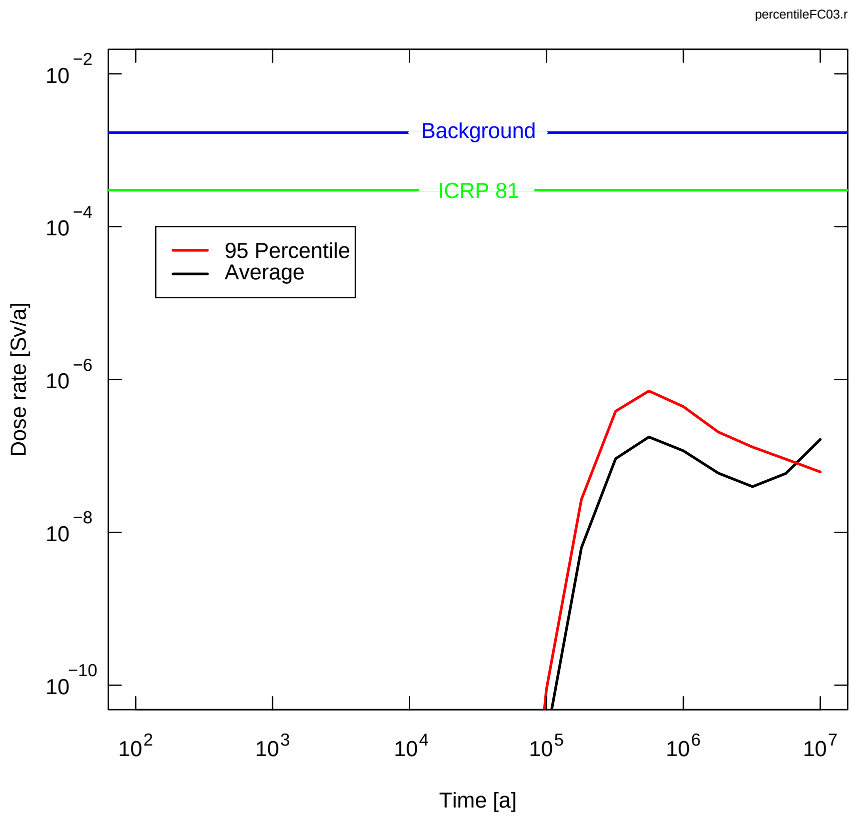
Figure A7.2-5
Distribution of peak total dose rates for simulations with container failures. The vertical blue line is the average peak dose rate of $3.6x10^{-7}$ Sv/a. Extracted from Garisto *et al.*, 2004



The results for the individual nuclides indicated that I-129 has the largest average peak dose rate, but that Rn-222 has the largest maximum peak dose rate.

Figure A7.2-6 summarizes the probabilistic results in terms of the dose rate variation with time. This figure shows that the 95^{th} percentile dose rate is at least 400-times lower than the ICRP 81 dose rate constraint (ICRP 2000), while the average dose rates is about 700-times lower. The average dose rate is greater than the 95^{th} percentile value at long times because, at these times, the average is skewed by a few simulations with high calculated Rn-222 dose rates.

Figure A7.2-6
Average and 95th percentile total dose rates from the probabilistic simulations. Extracted from Gierszewski 2004



Figures A7.2-7 and A7.2-8 show the individual nuclide contributions to the calculated dose rate to the critical group from the fission product and the U-238 decay chain nuclides. In general, the fission product contributions dominate up to 1,000,000 years. Contributions from the actinide chain nuclides do not appear until after 500,000 years. I-129 and Rn-222 are the largest contributors to the average dose rate from the fission products and actinides, respectively.

Figure A7.2-7
Calculated average dose rates to the critical group from fission product radionuclides. Extracted from Gierszewski *et al.*, 2004

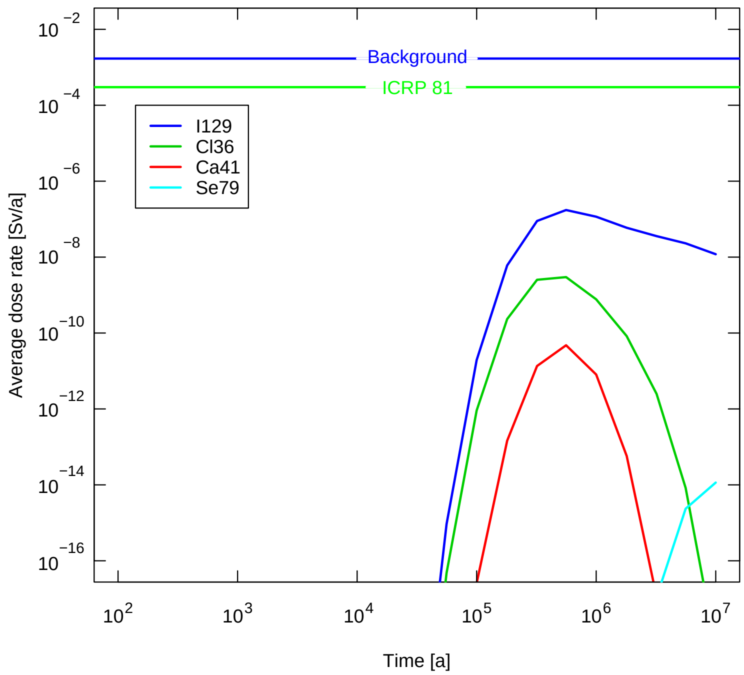
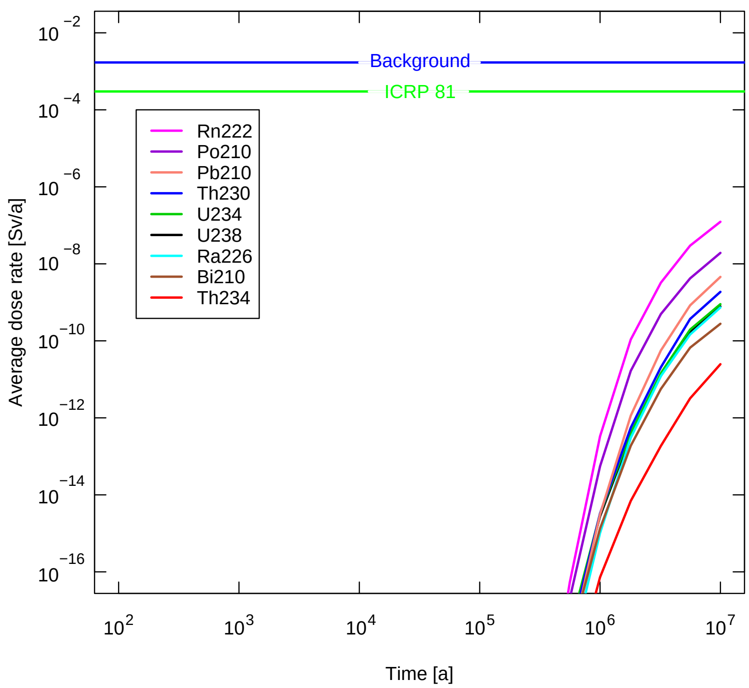


Figure A7.2-8
Calculated average dose rates to the critical group from the natural U-238 decay chain nuclides present in the used fuel. Extracted from Gierszewski *et al.*, 2004



Additional sensitivity analyses of the CC4 system model showed that the main contributors to the uncertainty in the total dose rate are the parameters related to the container failure rate. Secondary factors are the number of people in the critical group and the size of the container defect. Note that this sensitivity analysis is based on the Reference Case geosphere model, and includes uncertainties in vault, geosphere transport and biosphere parameters. It does not include uncertainties in the bulk rock permeability or fracture location. The effect of bulk rock permeability was analysed separately (see Figure A7.2-8).

Analysis of the high dose rate cases from the probabilistic results indicates that many of them are associated with very high uranium solubilities. In these cases, there is significant release of U-238 into the geosphere, with the eventual release of Rn-222 into the biosphere. Uranium solubility is calculated internally in SYVAC3-CC4 using a thermodynamic model and an input porewater chemistry. Under some input parameter combinations, this model yields unrealistic solubilities (much larger than 0.1 mol/m^3). It is likely that this model is very conservative, and results in a larger Rn-222 dose at long times than would be expected.

The calculated peak dose impacts occur at such long times in the future that the results are only indicative of the potential impact. Nonetheless, the overall conclusion is that the deep geologic repository design and hypothetical site considered in the Third Case Study provides effective isolation and retention of radionuclides in the Defective Container Scenario. Almost all the radionuclides decay within the repository and adjacent geosphere before reaching the surface biosphere (see Figure A7.2-9). Only a few radionuclides eventually reach the surface biosphere and their dose impact is much lower than the natural background dose rate and the internationally recommended ICRP 81 dose rate constraint.

Figure A7.2-9

Mass distribution of long-lived radionuclides at one million years, for the Reference Case. Almost all of these radionuclides are retained in the containers or decay - only a very small fraction reaches the biosphere. Extracted from Garisto *et al.*, 2004

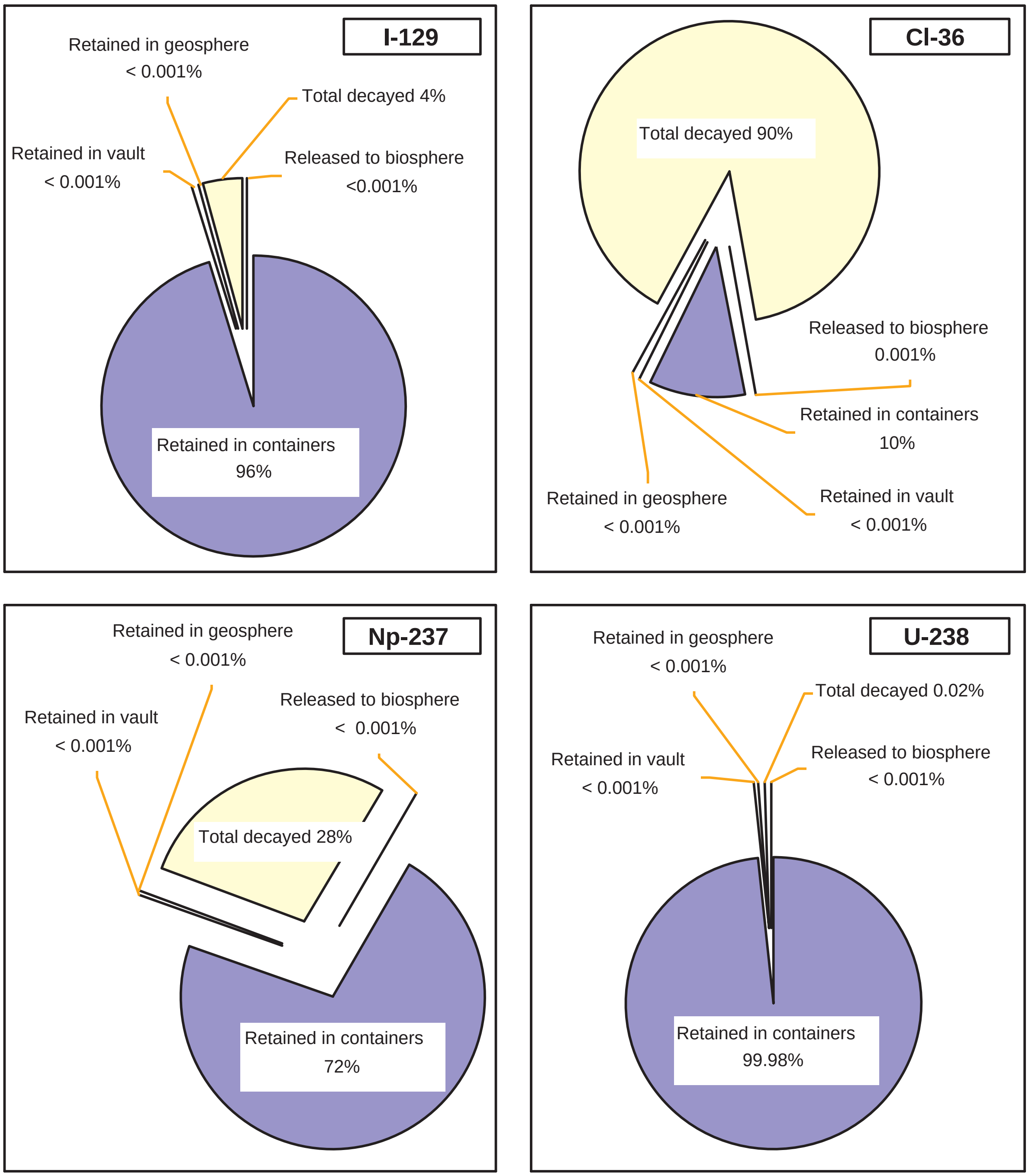
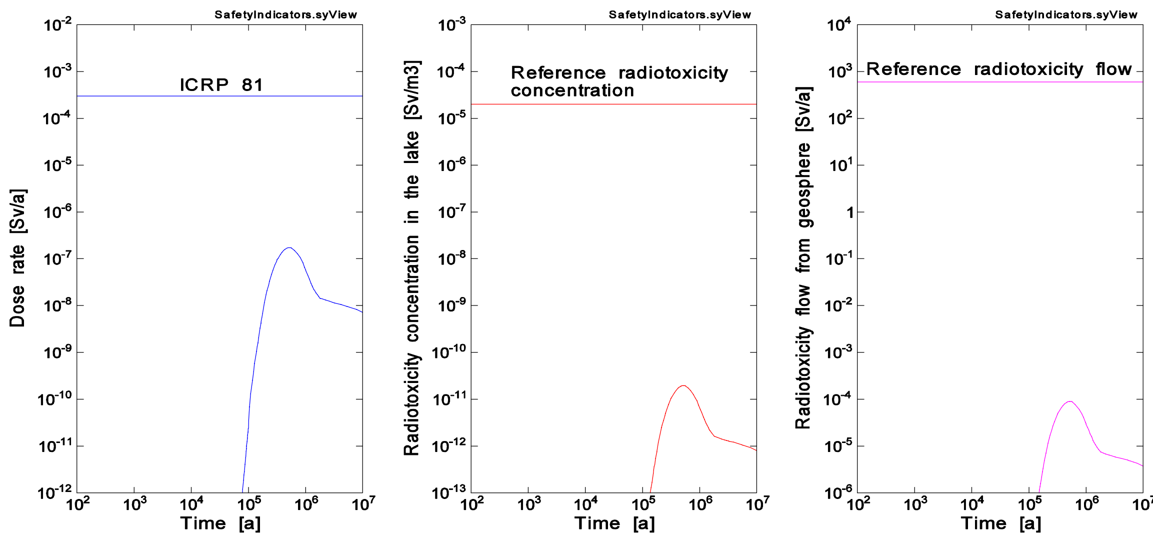


Figure A7.2-10
Comparison of three safety indicators. Reference values correspond to the ICRP 81 dose constraint and to natural radionuclide concentrations or flows in the Canadian Shield. Extracted from Gierszewski *et al.*, 2004



A7.3 POTENTIAL RISK TO NON-HUMAN BIOTA

Goodwin *et al*. 1994 assessed post-closure dose to non-human biota. The assessment focused on ^{14}C and ^{129}I, the only radionuclides that could potentially have environmental increments above a natural baseline. The estimated doses from these radionuclides to 10^5 years are many orders of magnitude smaller than the total annual dose from natural sources of radiation to non-human biota.

A7.4 POTENTIAL RISK FROM DGR IN THE POST CLOSURE PHASE – INADVERTENT HUMAN INTRUSION SCENARIO

The repository is designed with a series of engineered and natural barriers so as to prevent or delay the release of radionuclides after repository closure without further human actions. The Human Intrusion Scenario considers the possibility that future humans may bypass these barriers.

It is expected that a record of the site will be kept through both normal institutional records and societal memory, and by a durable surface marker at the site. If people in the future deliberately choose to enter the repository (e.g., to retrieve the containers), then they are assumed to take responsibility for their own actions, including protection of themselves and of future users of the site. If the markers or records become lost or misunderstood, then any intrusion into the repository would be "inadvertent" in that the people would not be aware that it contained a hazardous material and might not take precautions. The analysis in this document is extracted from Gierszewski *et al*., 2004 and considers the case of inadvertent intrusion.

With respect to human intrusion, ICRP 81 (ICRP 2000) takes the view that human intrusion exposure scenarios should be treated separately from natural process exposure scenarios, and further recommends that the possibility of intrusion events be reduced under certain conditions.

Consistent with ICRP 81, the Third Case Study repository design minimizes the possibility of inadvertent intrusion by adopting the following characteristics:

- a deep repository location, deeper than the range of interest for a water supply well (water from vault level would be too saline);
- a site with no mineral or other known economic potential; and
- the use of records and markers to preserve institutional memory for as long as practical.

The recent Swedish SR 97 assessment (SKB 1999) considered both a societal factor analysis and a technical analysis of human intrusion. The societal factor analysis constructed a framework to consider what types of future societal conditions might be consistent with inadvertent intrusion into a deep repository. One conclusion was that it was difficult to imagine inadvertent intrusion, given continuous development of society and knowledge; however, in the long run, it was not possible to rule it out.

The SR 97 technical analysis identified a number of human actions that could affect a deep repository. From their review of these actions, and taking into account the features of the deep repository design, they concluded that only drilling into the rock was likely to result in breach of the containers, while at the same time being inadvertent, technically possible, practically feasible and plausible. Other assessments of deep geologic repositories have also considered human intrusion scenarios. Table A7.4-1 lists some examples.

Table A7.4-1
Human Intrusion Pathways Considered in Recent Safety Assessments of Used Fuel Repositories

Assessment	Scenario/Exposure cases considered
EIS, Canada, 1996	• Drill crew • Drill core examination technician • Construction worker on contaminated soil from drilling slurry • Resident on contaminated soil from drilling slurry*
SR 97, Sweden, 1999	• Drill crew + core examiner * • Resident on contaminated soil from groundwater via open borehole into waste
H-12, Japan, 2000	• Drill crew + core examiner
STUK, Finland, 2001 (regulatory guidance)	• Deep water well located at site • Core drilling hitting a container
EPA, USA, 2001	• Resident exposed as result of open borehole into waste creating pathway for waste to reach aquifer
NAGRA, Switzerland, 2002	• Resident exposed as result of open borehole into waste creating pathway for waste to reach aquifer * • Deep well located near repository

* Most limiting exposure case.

The deep well scenario is unlikely because deep groundwaters are saline, so their use as water for drinking or irrigation would entail water analysis and treatment, which would increase the probability that contamination would be detected and action taken. Also, specific analysis of the open borehole scenario in the EIS study found that it resulted in only a small increase in the calculated peak dose rate relative to the Defective Container Scenario (Goodwin *et al*. 1994a).

Therefore, in order to consider a qualitatively different exposure route that could have a higher dose impact, the TCS focused on scenarios where the borehole intercepts the container and brings used fuel debris to the surface in the form of drilling slurry and a core example.

As an estimate of the potential exposure from inadvertent human intrusion into the repository, the TCS considers four potential critical groups:

- the drill crew, exposed to contaminated drill slurry spread on the surface around the drill rig,
- a laboratory technician, exposed while examining a drill core section containing used fuel,
- a construction worker (e.g., building a house), exposed while working in soil that was contaminated by drill slurry, and
- a resident at the site, exposed by living near, and growing a garden in, soil contaminated by drill slurry.

The drill crew and core technician are assumed to be exposed before the debris is recognized as hazardous and do not take any precautions (e.g., do not wear face masks). The construction worker and resident are exposed after the drilling occurred, assuming that contaminated drilling slurry was left at the site and not cleaned up.

The model used to estimate dose impacts largely follows Wuschke (1996). The dose impact is evaluated for the four critical groups identified above. The used fuel is assumed to be 30 years old at the time of emplacement and all calculations start at the time of emplacement. The repository is closed and sealed after 100 years of operation and monitoring, so inadvertent intrusion cannot occur until sometime afterwards.

The estimated doses and probabilities of intrusion are shown in Tables A7.4-2 and A7.4-3.

Acute doses are received by the hypothetical drill crew and core examination technician, who are exposed to the used fuel essentially immediately upon intrusion, and the construction worker who is on the site for a few months, e.g., while building a road or a house, and is also conservatively assumed to be on-site shortly after the drilling intrusion occurs. For example, if intrusion occurs at 300 years after the fuel is initially emplaced in the repository (200 years after closure), the doses are estimated at 200, 50 and 4 mSv for the core technician, construction worker and drill crew, respectively. Note also that this is the committed dose over the person's lifetime as a result of the one-time exposure. The potential dose would be lower at later intrusion times due to radioactive decay. These can be compared with the maximum 1-year occupational dose limit for a nuclear energy worker of 50 mSv.

For the resident, the exposure could occur long after the used fuel was inadvertently brought to surface, assuming that the site was not remediated in the interim. The calculated doses are highest for residents living on the (contaminated) site just after intrusion, and then decrease with time due to decay of the radionuclides on the surface (as in Figure 8.3) and also due to natural leaching processes that remove radionuclides from the surface soil layer. In this case, the exposure occurs continuously over the year, so is expressed as a chronic dose rate. For the case of intrusion at 300 years and the resident living on the site immediately afterwards, the calculated dose rate is 20 mSv/a.

In all four Human Intrusion critical groups, the dose is dominated by actinide nuclides through the dust inhalation pathway after the first 300 years. (This is in contrast to the Defective Container Scenario, in which actinides are slow to dissolve, sorb strongly in the vault and geosphere, and generally do not reach the surface.) More specifically, the dose tends to be dominated by Sr-90 and Am-241 for the first 100 to 1000 years, by Pu-240 and Pu-239 for 10^3 to 10^5 years, and the Np-237 and U-238 chains for longer times.

These calculated doses are intended to be indicative only. They are dependent upon the assumptions regarding the amount of fuel debris brought to surface, the suspension of this debris into airborne dust, the exposure duration, and the extent to which the dust is filtered, inhaled and converted into a dose equivalent (dose conversion factors assume a certain dust size profile and also a chemical species).

Table A7.4-2
Dose Estimates for Core Technician and Drill Crew[1]

	Core technician		Drill crew	
Time after Emplacement	**Dose, mSv**	**Probability per year**	**Dose, mSv**	**Probability per year**
110 years	300	$5x10^{-10}$	50	$3x10^{-9}$
300 years	200	$6x10^{-9}$	4	$3x10^{-8}$
1,000 years	140	$4x10^{-8}$	2	$2x10^{-7}$
10,000 years	50	$4x10^{-7}$	0.9	$2x10^{-6}$
100,000 years	2	$4x10^{-7}$	0.08	$2x10^{-6}$
1,000,000 years	0.3	$4x10^{-7}$	0.05	$2x10^{-6}$

[1] Exposure occurs at the time of intrusion

Table A7.4-3
Dose Estimates for Resident and Construction Worker

	Resident		Construction worker	
Time after emplacement	**Max dose rate, mSv/a [1]**	**Probability per year [2]**	**Max dose, mSv [1]**	**Probability per year [2]**
110 years	500	$8x10^{-12}$	160	$2x10^{-13}$
300 years	20	$1x10^{-9}$	50	$2x10^{-11}$
1,000 years	8	$3x10^{-8}$	30	$6x10^{-10}$
10,000 years	3	$6x10^{-6}$	10	$1x10^{-7}$
100,000 years	0.5	$6x10^{-5}$	0.6	$1x10^{-6}$
1,000,000 years	0.4	$6x10^{-5}$	0.2	$1x10^{-6}$

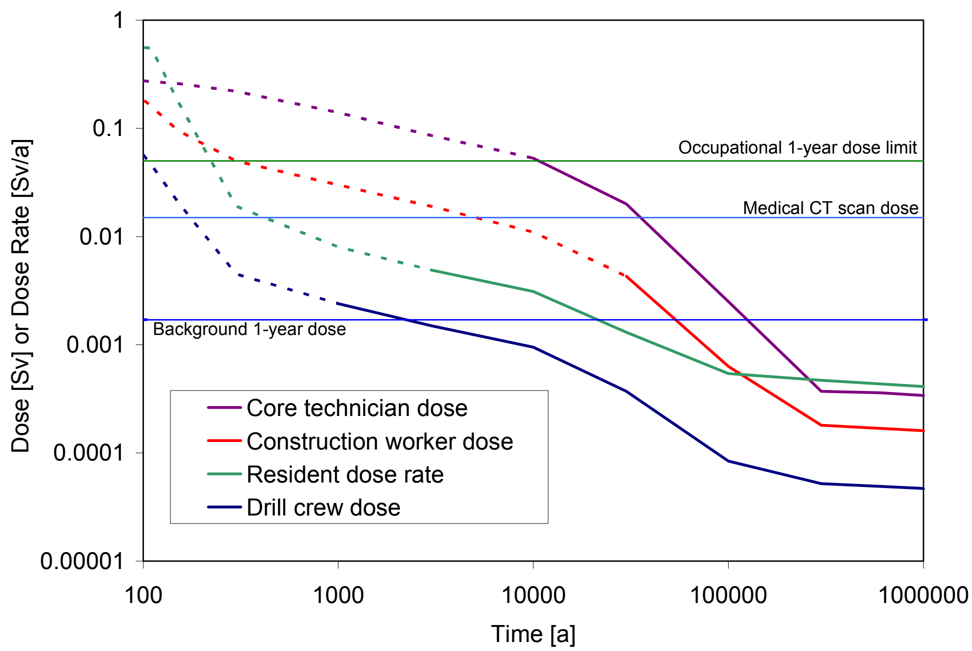
[1] Max dose assumes resident and construction worker are on site shortly after the borehole intrusion; potential dose is less if they come on site later.

[2] Probability of any exposure in given year, including possibility that intrusion occurred within previous 100,000 years and contaminated soil was left on surface.

The drill crew and core technician exposure probabilities increase with time as the effectiveness of records and markers at the site fades, eventually reaching a steady value dependent upon the deep drilling rate. The construction worker and resident exposure probabilities increase with time because of the loss of records and markers, and also because the probability of a previous intrusion increases with time. Their exposure probability reaches a maximum value dependent on the rate at which glaciation cycles periodically remove surface contamination.

Figure A8.3 shows the dose (or dose rate) variation with time for the various critical groups. The overall estimated exposure probability is less than one-in-a-million per year for the first 1000 years (dashed line portion of the figure).

Figure A7.4 - 1
Calculated inadvertent exposures as a result of a borehole drilled into a container. The core technician, construction worker and drill crew receive a one-time (acute) dose, while the resident receives a chronic dose rate. Dashed portions of each line have estimated probabilities smaller than one-in-a-million (see Tables A8.4 and A8.5)



The results of the analysis show that it is possible for some people to be exposed to relatively high doses (0.1 to 1 Sv) if intrusion occurs within about 1,000 years following emplacement. However, the probability of this is very low, less than one-in-a-million.

At a probability of around one-in-a-million, the maximum calculated dose or dose rate is about 50 mSv (Figure A7.4-1). This is an appreciable dose, but corresponds to the 1-year occupational dose limit for nuclear energy workers, for example. The potential dose continues to decrease with time as the used fuel decays. At times greater than one million years, the residual hazard from human intrusion would be due to the natural uranium content of the used fuel (and its decay chain), and the consequences of intrusion into the repository are similar to that of inadvertently intercepting natural uranium ores bodies.

In all these cases, the number of people who could receive these doses is small, and restricted to people who are intimately in contact with the used fuel debris. The highest dose was calculated for the core examination technician, because the technician is assumed to be exposed to undiluted used fuel dust as a result of cutting or grinding the core samples in preparation for their examination. The technician dose would drop significantly if they were to take even simple precautions such as wearing a dust mask.

In the TCS analysis, the calculated dose rates to the resident critical group are high enough that the ICRP 81 recommendations for human intrusion scenarios indicate that it is worth taking actions to reduce the likelihood of exposure (Section 2.2). In fact, such actions are embodied within the deep geologic repository concept, with its emphasis on deep disposal in Canadian Shield rock at a site with no mineral or economic potential and with saline (non-potable) water. This is reflected in the low estimated probabilities.

As mentioned earlier, the numerical analysis in this report is largely based on Grondin *et al.*, 1994 and the recent Third Case Study. This analysis illustrates the safety of DGR based on representative case studies. The conclusions on the safety of DGR are further supported by comprehensive international assessments (see Section A1.1).

Nevertheless, the analysis still has to be expanded, documented in further detail and updated as part of a "safety case" if this option is implemented.

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