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PICKERING WASTE MANAGEMENT FACILITY SAFETY ASSESSMENT SUMMARY REPORT

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Pickering Waste Management Facility Safety Assessment Summary Report

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Prepared By: Kristina Gillin Feb 8, 2017
K. Gillin — Date
Senior Consultant
Candesco Division of
Kinectrics Inc.

Reviewed By: Erin Middaugh Feb. 8 2017
E. Middaugh — Date
Assistant Technical
Engineer/Officer
Nuclear Safety &
Technology Department

Verified By: Shuyin Huang Feb. 8 2017
S. Huang — Date
Analyst
Candesco Division of
Kinectrics Inc.

Concurred By: R. Asadi Feb 08, 2017
R. Asadi — Date
Section Manager
Nuclear Safety &
Technology Department

Approved By: Feb. 8, 2017
N. Fajurally — Date
Manager
Decommissioning &
Health Physics
Candesco Division of
Kinectrics Inc.

Approved By: Feb 09, 2017
H. Seid — Date
Manager
Nuclear Safety &
Technology Department

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Revision Summary

Revision Number	Date	Comments
R000	2017-01-27	Initial issue.

1.0 INTRODUCTION

The Pickering Waste Management Facility (PWMF) is licensed by the Canadian Nuclear Safety Commission (CNSC) under Section 24(2) of the *Nuclear Safety and Control Act*. It is a Class IB nuclear facility, as defined in the *Class I Nuclear Facilities Regulations*, to provide for the safe handling, management and interim storage of intermediate-level waste from the refurbishment of Pickering Nuclear Generating Station (NGS) Unit 1-4 and used fuel produced by all eight Pickering NGS units.

1.1 Objective and Scope

This report presents a summary of the safety assessments prepared by Ontario Power Generation (OPG) for the storage of intermediate-level waste and the processing and storage of used fuel from the Pickering NGS at the PWMF.

Safety assessment is a systematic evaluation of the potential hazards associated with the conduct of a proposed activity or facility and considers the effectiveness of preventative measures and strategies in reducing the effects of such hazards. It evaluates the risk and consequences of normal and accident conditions, to ensure that the facility does not pose an unacceptable risk to workers or the public. The results of the safety assessments are used in the development of the operating limits and conditions for a facility. Safety assessments of structures, systems, components or facilities are carried out to determine the impact on workers and the public. Safety assessments are presented in a facility safety report, which also provides an overview of the facility design and operations.

To assess the overall safety of the operation of the PWMF buildings and structures, deterministic safety analyses are used. Computational tools are used for the dose consequence calculations when required. Bounding (worst-case) accident scenarios are conservatively identified, and the results of off-site dose consequence calculations are then compared against the regulatory dose limits. The dose limits and targets for the PWMF safety assessment for normal operating conditions and accident scenarios are given in Section 1.2.

1.2 Dose Limits and Targets

1.2.1 Normal Operating Conditions

General radiological protection requirements are established in OPG's governing documents on radiation protection.

Doses resulting from PWMF operations will be within the CNSC regulatory dose limits and kept As Low As Reasonably Achievable (ALARA). The regulatory dose limits for the public and nuclear energy workers are shown in Table 1.

OPG has based the PWMF radiation dose rate targets on the public dose limits in the Regulations promulgated under the *Nuclear Safety and Control Act*, which came into force on May 31, 2000.

Table 1: Canadian Nuclear Safety Commission Effective Dose Limits

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

The dose rate targets for PWMF operations, derived from Table 1 for a non-nuclear energy worker and a member of the general public, are as follows:

(a) ≤ 0.5 μSv/h at the station fence and at the perimeter fence surrounding Dry Storage Container (DSC) Storage Building #3 on a quarterly average basis, based on the CNSC dose rate limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-nuclear energy workers.

(b) ≤ 10 μSv/year at the Pickering NGS site property boundary, based on year-round occupancy. This dose rate target is 1% of the CNSC dose rate limit of 1 mSv/year for a member of the public.

1.2.2 Accident Scenarios

The radiological doses from radionuclide releases and direct radiation – either to members of the public at the site property boundary or to workers, following a credible accident scenario during the entire lifetime of PWMF operations – will not exceed the dose limits given in Table 1.

2.0 OVERVIEW OF THE PICKERING WASTE MANAGEMENT FACILITY

Used fuel dry processing and storage at the PWMF began in 1996, in an area on the Pickering Nuclear site, which in turn is located in the Regional Municipality of Durham, Ontario. The PWMF is dedicated to the processing and interim storage of used fuel discharged from the Pickering NGS units. In addition, the PWMF provides safe interim storage for retube components received from the Pickering NGS Unit 1-4 refurbishment operations from 1984 to 1992.

The PWMF is composed of two separate sites (Phase I and Phase II), as shown in Figure 1. The PWMF Phase I site is located within the Pickering NGS protected area, southeast of Unit 8 and adjacent to the east side of the station security fence. The PWMF Phase II site is located approximately 500 m northeast of the PWMF Phase I site, east of the Pickering NGS powerhouse. PWMF Phase II is located within its own security-protected area on the Pickering Nuclear site. The Retube Components Storage (RCS) area, which is part of the PWMF Phase I site, has been operational since 1984.

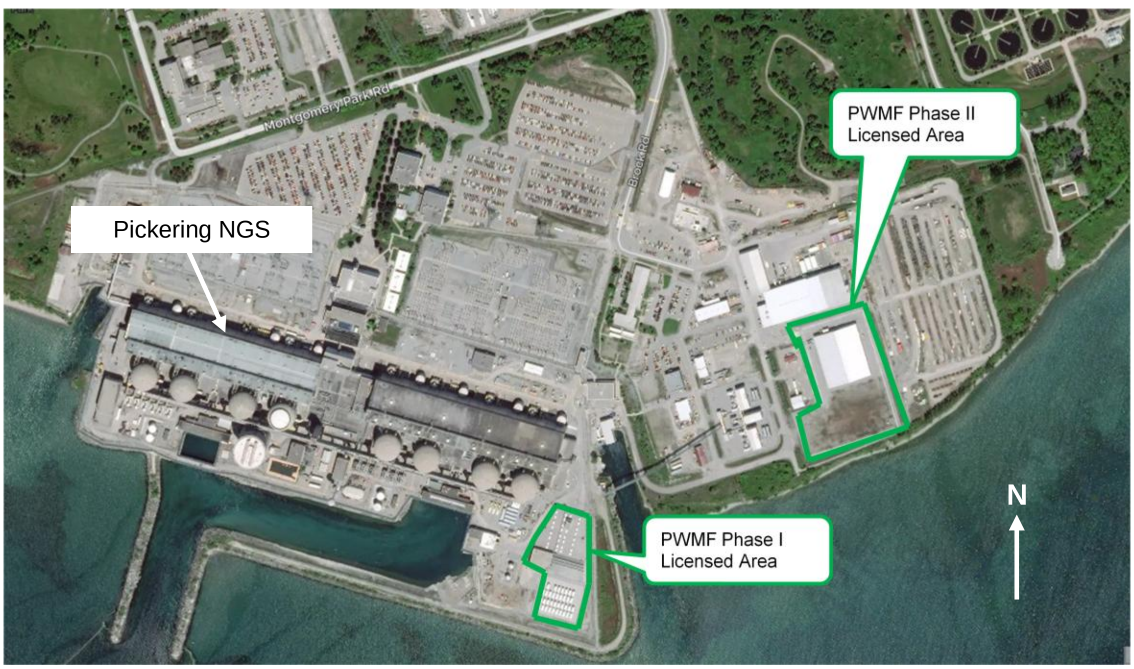

Figure 1: Aerial View of PWMF on the Pickering Nuclear Site

The PWMF Phase I site (shown in Figure 2) consists of a DSC Processing Building, DSC Storage Buildings #1 and #2, and the RCS area. This phase was constructed in two stages, as follows:

- Stage 1 became operational in 1996 and contains the DSC Processing Building and DSC Storage Building #1. DSC Storage Building #1 has a nominal design capacity of up to 185 DSCs.

- Stage 2 became operational in 2001 and consists of DSC Storage Building #2, which has a nominal design capacity of up to 469 DSCs.

1. Retube Components Storage Area 2. DSC Processing Building 3. DSC Storage Building #1
4. DSC Storage Building #2 5. Pickering NGS

Figure 2: PWMF Phase I Site

The PWMF Phase II site (shown in Figure 3) consists of DSC Storage Building #3, which has a nominal design capacity of 500 DSCs. DSC Storage Building #3 was placed into service in 2009.

A summary of the buildings developed at PWMF for used fuel processing and storage is provided in Table 2.

Table 2: Chronology of Development for Used Fuel at PWMF

Building	Number	Capacity	In-Service Date
DSC Processing Building			1996
DSC Storage Building	#1	185 DSCs (nominal)	1996
	#2	469 DSCs (nominal)	2001
	#3	500 DSCs (nominal)	2009

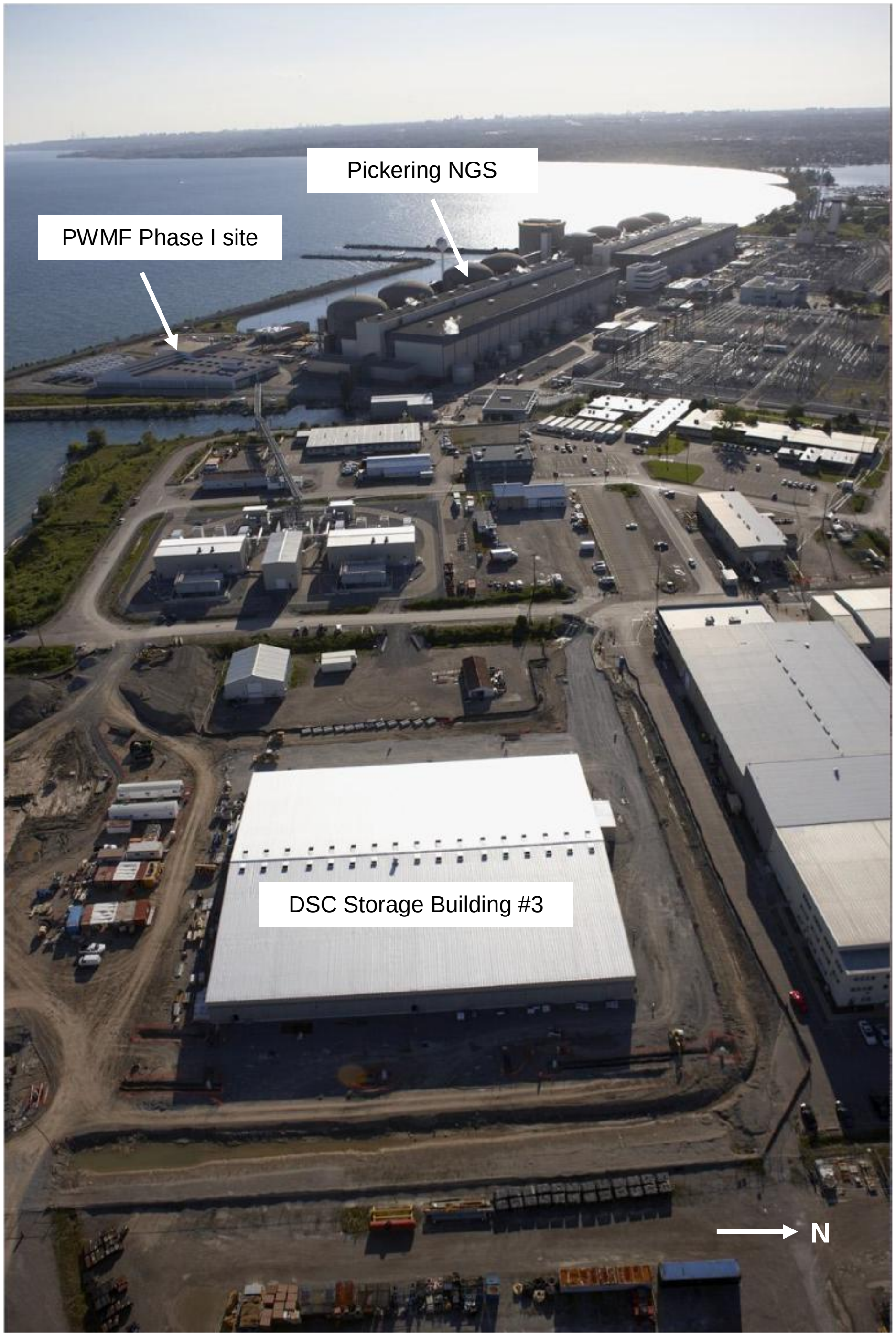

Figure 3: PWMF Phase II Site

3.0 SAFETY ASSESSMENT OF RETUBE COMPONENTS STORAGE AREA

3.1 Description of the Retube Components Storage Area

The RCS area within the PWMF provides interim storage of irradiated reactor components in Dry Storage Modules (DSMs). The DSMs are stored outdoors in a fenced and access-controlled area, situated south of the PWMF Phase I DSC Storage Buildings, see Figure 2. The dose rates from the RCS area are monitored by thermoluminescent dosimeters placed at four locations on the perimeter fence.

The RCS area is paved and further covered with a rubber-membrane coating to provide a non-permeable and maintenance-free surface. A drainage system is provided to direct the runoff water from the storage area to the Pickering NGS Unit 5-8 outfall, with catch basins permitting periodic sampling of the water.

The irradiated reactor components in the DSMs consist of pressure tubes, end fittings, shield plugs and miscellaneous identified components that were removed during the retube of the Pickering NGS Unit 1-4 between 1984 and 1992. Reactor components become radioactive during their residence in the reactor core, due to neutron activation and deposited contamination. The irradiated reactor components were loaded into specifically designed and shielded DSMs for interim storage at the PWMF. In all, 36 DSMs are stored in the RCS area. Of these, 16 contain retube waste from Unit 1/2, 18 contain waste from Unit 3/4 and two are empty.

Radionuclide inventories inside the DSMs have been steadily decreasing due to radioactive decay. With the exception of periodic inspection, monitoring and maintenance, there have been no operational activities in the RCS area since 1993.

Dry Storage Modules

The DSMs are cylindrical casks made of 0.57 m thick reinforced heavy concrete and carbon steel inner and outer liners, see Figure 4. Each DSM is 3.3 m in diameter and has an overall length of 7.6 m. The DSMs are designed to be leak resistant, employing welded construction. To prevent deterioration and minimize maintenance, the 6.4 mm thick outer carbon steel shell is coated with two coats of ceramic elastomeric paint.

A bolted, gasketed shield door is used to seal the fill port on each DSM. Each DSM also has an inspection port provided high up at the back end and a sampling port located below the loading port. Both the inspection and sampling ports are closed with shielding plugs that have self-sealing pipe threads.

Saddle supports (two steel pedestals), each about 1 m high and built on a 1.8 m × 5.5 m × 0.6 m thick foundation slab, support each end of a DSM. The slabs in the RCS area are situated on compacted, crushed stone overlying fill composed of coarse to fine sand and gravel.

Each DSM can hold about 90 pressure tubes or various combinations of pressure tubes, end fittings and miscellaneous components.

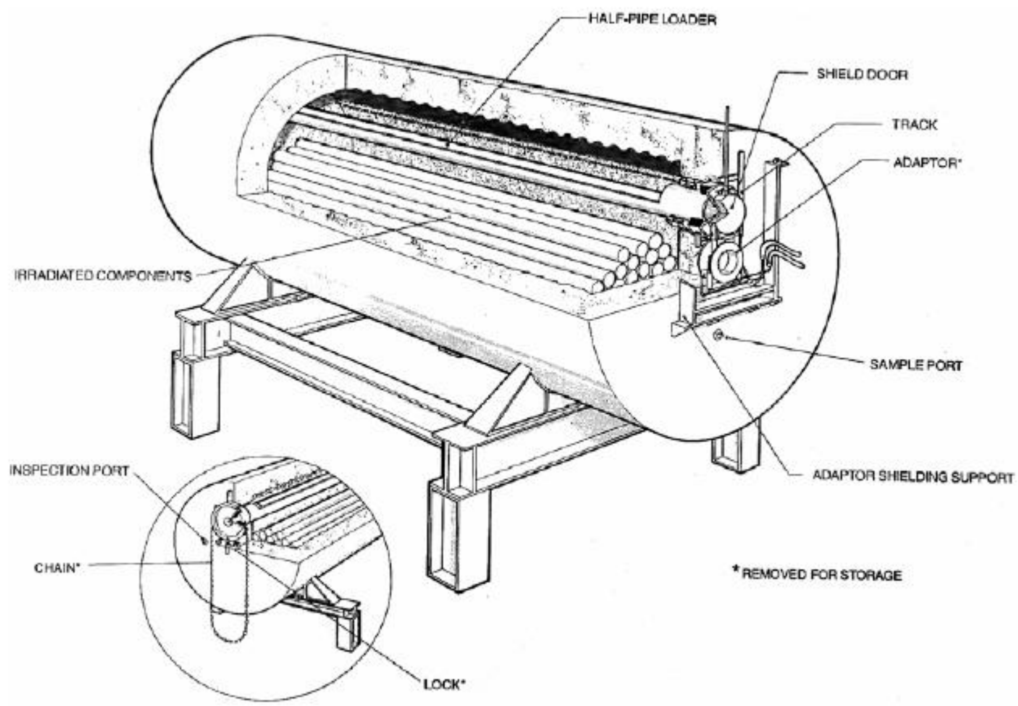

Figure 4: Dry Storage Module

3.2 Radionuclide Inventory Potentially Available for Release

Only a small fraction of the small quantity of loose radioactive material in a DSM could become available for release from the module. The pressure tubes, shield plugs and end fittings were packaged in cans, and some end fittings in bags, before being stored in the DSMs. These cans and bags keep the loose contamination from mixing in the free atmosphere of the DSMs.

After each DSM was loaded, the loading penetrations and the sampling ports were sealed. Loose contamination, which may have reached the DSM atmosphere from the components packaging, is expected to be contained within the module.

Conservative estimates of the inventory of loose radioactivity in the DSMs are given below. These estimates are for inventories at the time of initial loading. Short-lived radionuclides have significantly decayed since then.

- **Carbon-14:** During the course of retubing, it was discovered that some reactor systems had become contaminated with C-14, produced in the annulus gas as a result of the reaction ^{14}N (n,p) ^{14}C. There has been C-14 contamination of the Unit 1/2 components stored in DSMs. For Unit 3/4 retubing, oxygen was added to the annulus gas system to oxidize the C-14. Therefore, only a relatively small amount of C-14 is expected in the DSMs loaded with components from Unit 3/4 retubing. At this time, any remaining C-14 is expected to be present predominantly in particulate form (C-14-p).

  C-14 measurements on pressure tubes and end fittings for Unit 1/2 have indicated a maximum of 5.6×10^3 Bq/cm^2. Using a surface area of 2 m^2 for each pressure tube, for 90 pressure tubes in a DSM, this gives a total inventory of 10^{10} Bq per DSM for loose C-14 activity.

- **Tritium:** Some tritium is embedded in the pressure tubes removed from the Pickering reactors. This tritium is in the form of zirconium tritide, which is a very stable compound requiring temperatures of several hundred degrees Celsius for its dissociation. Vacuum drying of components is effective in removing residual tritiated water. Therefore, no tritium emissions are expected.

- **Loose Crud:** Data on loose crud[1] were obtained for Pickering Unit 1, 2 and 3. These data indicate a maximum initial activity of 2.6×10^7 Bq Co-60 per m^2 for zirconium alloy surfaces and 6.7×10^7 Bq Co-60 per m^2 for shield plugs.

  For a DSM containing 90 pressure tubes, with shield plugs, and using surface areas of 2 m^2 per pressure tube and 0.3 m^2 per shield plug, this gives a total initial inventory of 3.6×10^9 Bq for 180 shield plugs in a DSM and 4.7×10^9 Bq for 90 pressure tubes per DSM. The total initial inventory of loose crud in a DSM was estimated at 8.3×10^9 Bq.

- **Activated Swarf[2]:** It was assumed that, during removal of the pressure tubes from the reactor, an amount of metal dust corresponding to a cut width of 0.1 mm was formed, per cut. This is an overestimation, as no dust formation was observed during cutting of zirconium alloys. The amount of dust activity, per DSM containing 90 pressure tubes, was estimated at 1.4×10^{11} Bq per DSM. Short-lived radionuclides have considerably decayed since removal from the reactor.

[1] Defined as crud that can be removed ultrasonically.
[2] Swarf is fine chips or filings of stone, metal or other material produced by a machining operation.

3.3 Normal Operations

Exposure to radiation during normal RCS operation is primarily from direct radiation fields. Emissions of radioactive contamination during normal operation were taken into account for public dose assessment, but the potential doses due to such emissions are negligible.

Gamma radiation dose rates under normal operating conditions are discussed in Section 3.3.1. The potential for dose due to chronic radioactive emissions is discussed in Section 3.3.2.

3.3.1 Dose Rates

Operations inside the RCS area consist of periodic inspection, monitoring and maintenance of the DSMs and the enclosed RCS area. Dose rates due to normal operation of the RCS are reported below, based on direct radiation fields only.

Estimated dose rates, including predicted dose rates for up to 50 years from removal from the reactor (i.e., to about 2037), are set out below, followed by presentation of the results of dose rate monitoring at the RCS area. These dose rate measurements are representative of more than a decade of decay time. Dose rates continue to decrease with increasing storage time.

Estimated Dose Rates from Dry Storage Modules

The estimated direct gamma dose rates from a DSM are given in Figure 5 (for DSMs containing Unit 1/2 components) and Figure 6 (for DSMs containing Unit 3/4 components). Dose rates are provided for various distances, as a function of time after removal from the reactor. The dose rates were calculated, using the point kernel shielding code, MicroShield 8.03, based on the activation product inventory of each component. A worst-case payload, i.e., the DSM contents giving the highest dose rate, was used.

The DSM storage configuration ensures that the dose rate is ≤ 0.5 µSv/h at the storage area perimeter fence.

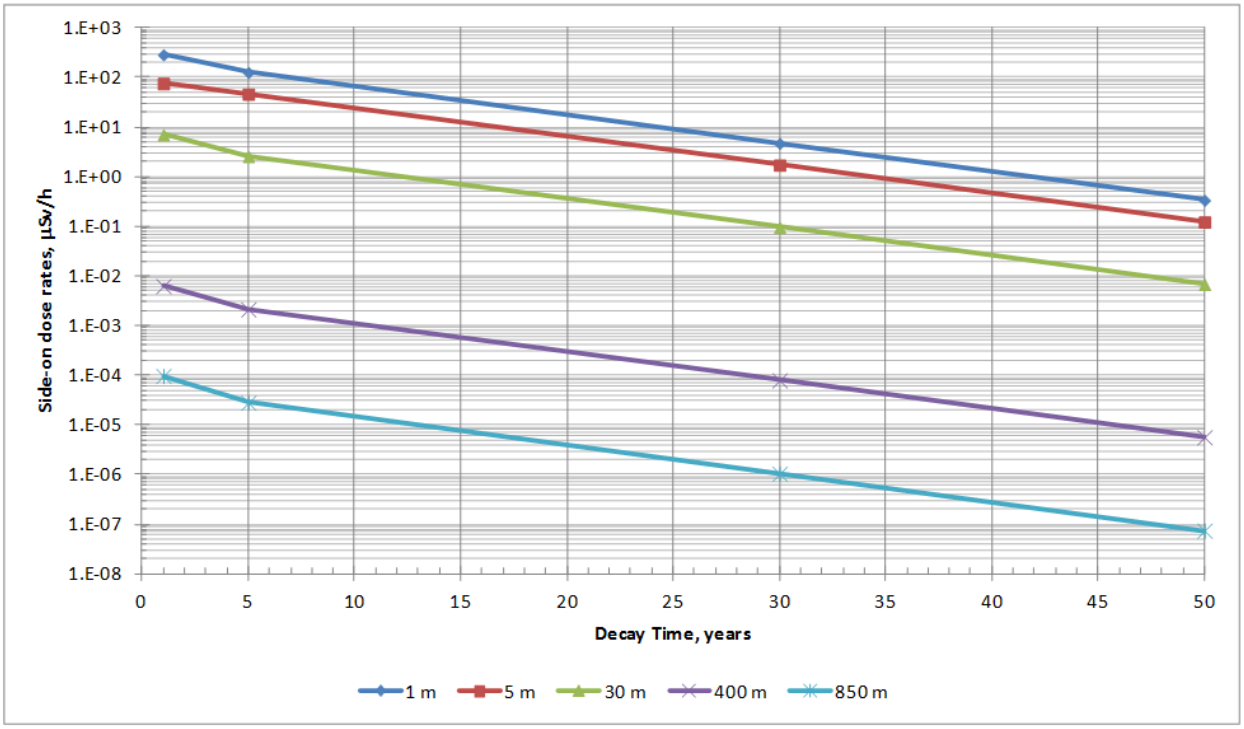

Figure 5: Estimated Dose Rates from a DSM Containing Retube Components from Unit 1/2

Notes for Figure 5:

1. Each DSM is assumed to contain 120 pressure tubes and 240 shield plugs. This assumption corresponds to the maximum activity that a DSM might have.
2. Radiation fields from DSMs containing end fittings are very similar to those from modules containing pressure tubes.
3. These dose rates were calculated using conservative early estimates of the activation product inventories.
4. The earliest time after reactor shutdown that a DSM was initially loaded was 18 months.

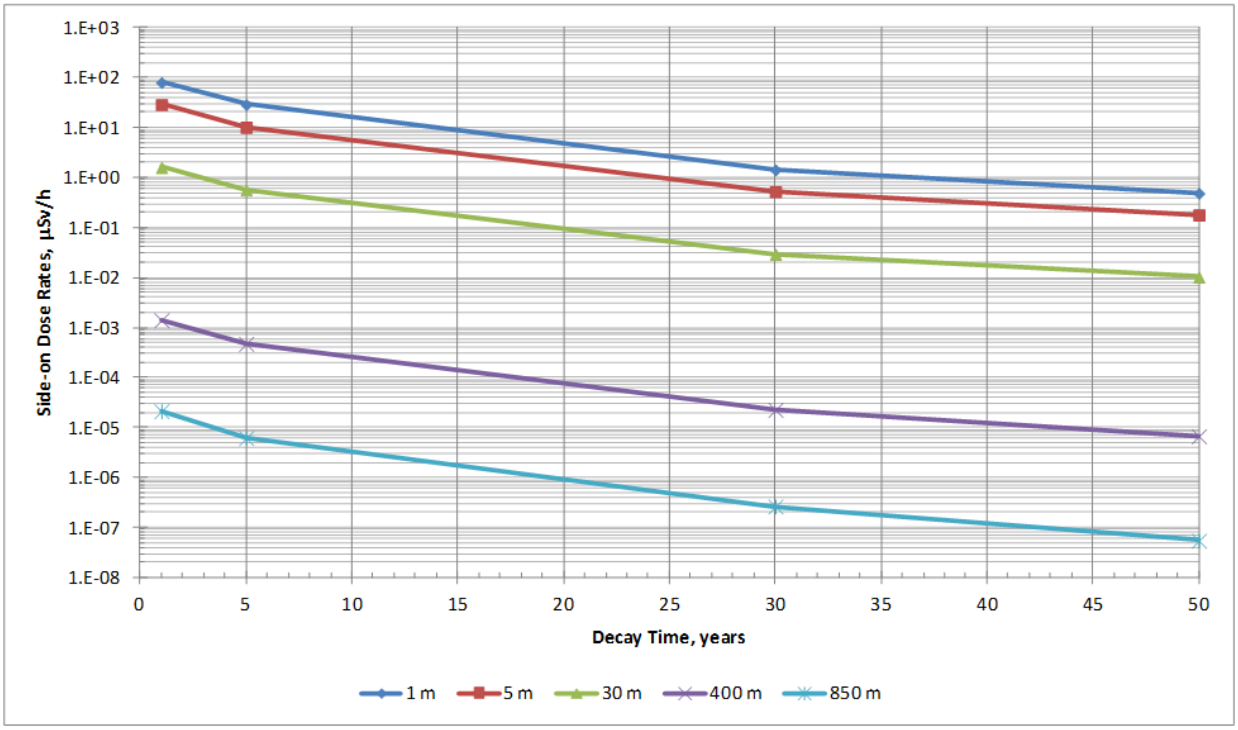

Figure 6: Estimated Dose Rates from a DSM Containing Retube Components from Unit 3/4

Notes for Figure 6:

1. Each DSM is assumed to contain about 90 pressure tubes and about 180 shield plugs.
2. The quoted radiation fields are calculated for a point off the side of the DSM where the shield plugs would be positioned in the cavity.
3. Radiation fields from DSMs containing end fittings are much lower than those from modules containing pressure tubes.
4. The earliest time after reactor shutdown that a DSM was initially loaded was about 8 months.

Results from Dry Storage Module Dose Rate Monitoring

In general, dose rates observed from loaded DSMs are below calculated values for the appropriate decay time (substantially so for DSMs containing only end fittings, as anticipated). However, higher dose rates have been measured in localized spots on contact with some DSMs.

These localized increases in dose rate near certain DSMs do not significantly affect overall dose rates at a distance. The localized dose rates do not pose an occupational hazard for workers at ground level, as warning signs are placed on DSMs and the planning process for work inside the RCS area perimeter fence includes the consideration of radiation monitoring requirements. General gamma radiation dose rates inside the RCS perimeter fence are less than 10 µSv/h. The RCS perimeter fence and the RCS area is a restricted area, not accessible to unauthorized persons.

Dose Rates from the Retube Components Storage Area

Measured Dose Rates

As part of the RCS area monitoring program, gamma dose rates are surveyed quarterly at the RCS perimeter fence using hand held monitors. As part of the environmental gamma monitoring program, the readings from the thermoluminescent dosimeters placed at the fence at each side of the RCS area are also collected quarterly.

In both cases, gamma dose rates have not exceeded 0.5 μSv/h at the RCS perimeter fence (i.e., inside the station protected area). Based on the 2012 thermoluminescent dosimeter survey monitoring results, the average dose rates are 0.1 μSv/h at the south, east and west fences and 0.3 μSv/h at the north fence.

Estimated Dose Rates

Dose rate assessments at the fence and at the site property boundary are summarized in this section. Comparison with the measured values at the fence provides an indication of the conservatism of the calculated values.

The south and north fences of the RCS area were chosen to perform the assessment, due to the layout of the facility. Although the DSMs at the south fence hold Unit 1/2 components, the dose rate estimates given in Figure 6 for Unit 3/4 were used. It has been shown by comparison with measurements that the dose rates calculated for Unit 3/4 are more realistic than those calculated for Unit 1/2.

A calculation factor of 1.4 was used to estimate the dose contribution at 5 m from a row of DSMs versus a single DSM. This factor was based on DSC analysis performed for a single DSC and at 5 m from the row of DSCs, as the concrete shielding thickness is similar for the two container types. The estimated dose rates from the rows of DSMs are approximately 0.35 μSv/h and 0.85 μSv/h at the south and north side fences, respectively.

The above estimates are based on the following considerations:

- The DSMs at the south and north rows closest to the fence hold only half of the maximum permissible load, and the estimated dose rates are based on fully loaded DSMs.
- The density of the heavy concrete used in the DSMs was as high as 3.8 Mg/m^3; however, the nominal value used in the calculations was 3.5 Mg/m^3.

Considering the above factors, the estimated dose rates of 0.35 μSv/h at the south fence and 0.85 μSv/h at the north fence are about a factor of 3 higher than the respective dose rate of 0.1 μSv/h and 0.3 μSv/h measured in 2012.

The annual dose rates expected due to the RCS at the property boundary were calculated. The dose rate at the 850 m site property boundary was estimated to be 0.03 μSv/year based on full-year occupancy. The dose rate at the 420 m in-lake exclusion zone boundary was estimated to be 0.3 μSv/year assuming 1,000 hours per year occupancy (conservative assumption for boaters and fishermen). These dose rates are well below the CNSC dose rate limit of 1 mSv/year and are also within the lower PWMF dose rate target of 10 μSv/year at the site property boundary (see Section 1.2).

3.3.2 Chronic Radioactive Emissions

Methodology for Dose Calculations

For dose estimates due to chronic airborne releases, the dose to members of the public at the site property boundary, taken to be 850 m away, was calculated assuming a ground level release.

The long-term atmospheric dilution factor (ADF) for Pickering NGS was used to calculate radionuclide concentrations at the site property boundary for each of the release scenarios considered in this analysis. The releases were assumed to occur at ground level under adverse weather conditions. The value of the ADF under these conditions is 2.2×10^{-6} s/m^3 (see Appendix A).

Breathing rates utilized for the calculations are 22.2 m^3/day (2.57×10^{-4} m^3/s) for an adult and 5.16 m^3/day (5.97×10^{-5} m^3/s) for an infant (HC99). The dose coefficients (DCs) for $^{14}CO_2$ inhalation are 1.2×10^{-11} Sv/Bq for an adult and 3.8×10^{-11} Sv/Bq for an infant.

The dose due to internal exposure via inhalation is given by:

$$Dose_{int} = R \times P_{01} \times BR \times DC_i$$

where,

R = released activity (Bq)
P_{01} = atmospheric dilution factor (s/m^3)
DC_i = internal dose coefficients (Sv/Bq)
BR = breathing rate (m^3/s)

Potential Airborne Emissions

Conservative estimates were made for emissions of particulate contamination. However, routine contamination survey monitoring conducted since the RCS area became operational has demonstrated no detectable loose contamination on DSM surfaces.

The absence of any evidence of loose contamination escaping from the DSMs is due to the following mitigating factors:

- Packaging of retube components inside bags or cans.
- The gasketed seal of the DSMs.
- The passive nature of storage in the DSMs (i.e., there is no reason for contamination to be resuspended inside the DSM or driven out of the module).

C-14-p may, over time, oxidize and be emitted as $^{14}CO_2$. It was assumed that conversion would take place and that the $^{14}CO_2$ would be emitted, at a rate of 10% per year or about 0.2% per week. This conservative estimate assumes that moisture may be present in the DSM atmosphere.

Based on conservative assumptions, the volume of gas emitted from one DSM is postulated to be 12 m^3/week (although, as mentioned previously, such a high level of air exchange between the DSM inner cavity and the environment is not considered likely). The DSM cavity volume is 16 m^3.

Only the 16 DSMs containing Unit 1/2 components were considered. Since C-14 is expected to be in stable form after more than a decade of storage, these predictions present an extremely conservative assessment of potential dose from emissions.

Based on these assumptions, the postulated emission rate of C-14 as $^{14}CO_2$ is:

$$1.0 \times 10^{10} \text{ Bq/DSM} \cdot 0.2\%\text{/week} \cdot 16 \text{ DSMs} = 3.2 \times 10^{8} \text{ Bq/week}$$

The chronic off-site dose consequences resulting from this postulated scenario, for a member of the public at the site property boundary using the long-term ADF, are estimated to be 2×10^{-6} µSv/week or 10^{-4} µSv/year for an adult or infant. This represents 10^{-5} % of the public dose limit.

Potential Liquid Emissions

No surface contamination of the DSMs has been found that would lead to contamination of water that drains from the site. Routine monitoring of surface water in the RCS area basins confirms that contamination levels are generally below the minimum detectable activity of 14 Bq/L.

The DSMs were dry loaded and are designed to prevent in-leakage of water from snow or rain. Also, welded steel liners form part of the structure. No radioactive liquid releases are, therefore, expected to occur from the modules. Provision for sampling of water that might potentially accumulate inside the modules in extreme conditions is, however, included as part of the design.

Since the RCS area is paved and further protected by a rubber-membrane top coat, there is a controlled pathway for runoff via the drainage system; therefore, no contamination of the subsurface water is expected.

3.4 Malfunctions and Accident Assessments

3.4.1 Abnormal Events

Earthquakes

The safety function of the DSMs is to contain irradiated retube components such that no radioactivity is released during normal conditions or as a consequence of design basis events. Therefore, to support the storage of DSMs at the PWMF until the time when the Pickering NGS is decommissioned, the seismic capacity of the complete storage structure (DSM and support pedestal/foundation) was assessed using the Pickering NGS Unit 5-8 Design Basis Earthquake ground motion as the design basis event. It was found that the required safety function of the DSMs was met.

It has been concluded that there are no radiological implications for DSM storage in the PWMF due to a seismic design basis event.

Flooding

According to the Water Survey of Canada, the maximum recorded water level for Lake Ontario from 1915 to 1988 was 75.7 m above sea level. The RCS area is located at an elevation of 77.4 m. The storage pads for the DSMs are 1 m high. Also, the saddle supports for the DSMs are 1 m high. This means that the DSMs are at a height of 3.7 m above the maximum recorded water level. An embankment protects the RCS area from wave action in case of a storm. The probability of flood water coming in contact with a module is, therefore, negligible. In the unlikely event of flooding, however, the steel shells and the reinforced concrete walls of the module would provide adequate barriers to gross water penetration or damage by wave action.

Fires and Explosions

Two fuel oil tanks located in a dyked area approximately 50-60 m west of the PWMF Phase I site are the only potential fire and explosion hazard near the site. A detailed assessment of this hazard has been carried out using the following postulated scenario.

It was assumed that the entire inventory of fuel oil in the storage tanks was involved in a fire. The fire was assumed to spread to both tanks instantaneously and was allowed to burn until the inventory of flammable materials was exhausted. In order to assess the maximum credible potential hazard to the DSMs, it was assumed that the initiating event caused the oil tanks to rupture and that all the fuel draining into the dyked area was involved in the fire. It was judged that a fire that engulfed the entire dyked area would result in flames of maximum intensity and would, therefore, pose the greatest hazard.

The effects of a localized fire in the storage tanks, without rupture and spillage of the contents, has also been assessed. Actuation of the air foam fire protection system was not included in the assessment. Also, the effectiveness of the fire-fighting squads in controlling the duration of the fire was not taken into account.

It was concluded that, for winds ranging from 10 to 80 km/h blowing in the direction of the modules from the storage tanks, the maximum surface temperature of the nearest module would be 76°C. The fuel oil tanks are, therefore, not considered to pose an unacceptable risk to the safety of the DSMs.

The RCS area is protected from the lake by an embankment. Also, boats are not permitted to dock on the shore near the PWMF. The shipping lanes for lake traffic are many kilometres away from the site. Therefore, fires or explosions that may occur on a boat or a ship in the lake do not pose a safety risk to the DSMs at the PWMF.

Aircraft Crash

A large aircraft strike at the Pickering Nuclear site is not a credible event. The probable frequency of a large aircraft crash has been estimated to be 5.4×10^{-7} per year for Pickering NGS. The probability of an aircraft strike varies linearly with the target area. The target area of the PWMF Phase I site is an order of magnitude smaller compared to Pickering NGS. As a result, the estimated crash frequency is reduced. At less than 10^{-7} per year, such an event is considered incredible.

Extreme Climatic Conditions

<u>Thunderstorms</u>

Lightning would be the only consequence of a thunderstorm that could pose concern for the RCS area, from the safety viewpoint. The effect of a lightning strike would be to increase the temperature of the affected DSM. This might result in an increased release of $^{14}CO_2$ from the DSM, as discussed in Section 3.3.2.

<u>Tornadoes</u>

The effect of Design Basis Tornado-generated missiles on the DSMs, and the potential for overturning by wind loading, was assessed.

Analysis of steel and concrete structures shows that the steel shell could be penetrated, but that the concrete would stop the missile.

The DSMs are both massive and rigid, and thereby unaffected by tornadoes, as the mass counteracts the uplift forces. The DSM supports resist the resulting combination of bending and compressive loadings, without exceeding Canadian Institute of Steel Construction material limits.

3.4.2 Contingency Plans

A DSM failure is very unlikely during its proposed 50-year storage life due to its robust design and the following protective measures:

(a) Deterioration of the DSMs is prevented by regular inspection and maintenance. All DSMs were recoated with a ceramic elastomeric paint in 2009.

(b) If a deficiency were to be discovered during inspection, the affected area would be repaired. Corroded parts would be cleaned and repainted, voids or cracks in the DSM structure would be filled with grout, and exterior steel work would be repaired by welding on new steel plate.

In the unlikely event that the integrity of a DSM was compromised, the most appropriate remedial actions would be taken, including radiation shielding and contamination control. The options for management of the irradiated components would then be evaluated.

Two spare DSMs have been reserved for possible contingency use.

3.5 Potential Off-Site Consequences

In reviewing potential hazards, no credible events have been identified that would lead to failure of the DSMs, i.e., loss of shielding or gross loss of containment of the radioactive contents. Safety is ensured by:

- The nature of the bulk of the radioactivity, which is fixed in the components, and sealed in bags or cans.
- The small number of penetrations.
- The robust, reinforced-concrete and steel-lined structure of the DSMs.

Section 3.4.2 describes the actions available to mitigate the consequences of observed deterioration of, or damage to, the DSMs.

An estimated potential release of $^{14}CO_2$ from one DSM in extreme conditions (such as impact on an inspection or loading port by a light aircraft, or a lightning strike) was calculated using the radionuclide inventory potentially available for release. As reported in Section 3.2, 10^{10} Bq of C-14 was assumed to be present in a DSM. For the purpose of this assessment, the entire C-14 inventory was assumed to be released as $^{14}CO_2$. The short-term ADF for the calculation was 5.2×10^{-5} s/m^3 (see Appendix A).

The off-site dose consequences resulting from this postulated scenario, for a member of the public at the site property boundary, were estimated to be 1.6×10^{-3} µSv/event for an adult and 1.2×10^{-3} µSv/event for an infant. This represents 1.6×10^{-4} % of the public dose limit.

3.6 Long-Term Performance

On-Going Inspection and Maintenance Program

The DSM has been engineered and designed for minimum upkeep over its design life. Periodic inspection and maintenance are performed to determine and mitigate aging effects over the design life. Maintenance is expected to be minimal, consisting of periodically renewing painted surfaces to prevent corrosion of exterior surfaces. Visual examinations of the modules are conducted to inspect for signs of weathering, corrosion or other structural defects that may arise with time. If corrosion is observed on a DSM, the affected area is cleaned and recoated with the specified touchup paint or repaired, as needed.

Surface corrosion of the outer steel liner is not expected to lead to radiological releases, due to the defence-in-depth design of the DSM.

The aging management plan that is in place includes detailed annual visual inspections of the DSMs.

4.0 SAFETY ASSESSMENT OF USED FUEL DRY STORAGE AREAS

4.1 Description of the Used Fuel Dry Storage Areas

The PWMF provides safe interim storage for the used fuel discharged from the Pickering NGS units and cooled for a period of time in the irradiated fuel bays. As of the end of 2015, 809 DSCs were safely stored in the DSC Storage Buildings at the PWMF.

4.1.1 Dry Storage Containers

A DSC is a free-standing, reinforced-concrete container with an inner steel liner and an outer steel shell (Figure 7), for the storage and on-site transfer of used Canada Deuterium Uranium (CANDU) fuel. It is made of two sub-assemblies: a lid and a base. The base provides the storage space for the used fuel.

The DSC MKII model constitutes the reference container design for the PWMF. The DSC is a double-shell rectangular container, with exterior dimensions of 2.121 m x 2.419 m x 3.557 m in height (including the lid) and an inside cavity of 1.046 m x 1.322 m x 2.520 m. The nominal thickness of each carbon-steel shell is 13 mm. The DSC walls consist of 520 mm (nominal thickness) concrete placed between the inner liner and the outer shell. The reinforced high-density concrete provides radiation shielding and structural strength, while maintaining adequate used fuel decay heat dissipation. The concrete has a density in the range of 3.5 to 3.7 Mg/m^3 and a compressive strength of at least 40 MPa. The maximum total mass (including the lid of 11 Mg) is approximately 60 Mg when empty and approximately 70 Mg when loaded with four used fuel storage modules (384 used fuel bundles).

All welds that form this containment system and all welds attaching items to the containment system are classified as “Nuclear Welds.” Helium is used as the inert cover gas in the DSC cavity to protect the fuel bundles from potential oxidation reactions and to facilitate leak testing of the containment boundary.

The DSC is designed with the provision for installing safeguards seals. Two separate U-shaped 25.4 mm outer diameter stainless-steel tubes are embedded in the DSC walls and floor in the plane of the outer reinforcing grid. These tubes are placed so that each tube runs across the centre of opposite container walls. Two similar tubes are embedded in the DSC lid and run diagonally across the lid. The configuration of the safeguards tubes is shown in Figure 7. These tubes are used for attaching two different types of International Atomic Energy Agency (IAEA) seals.

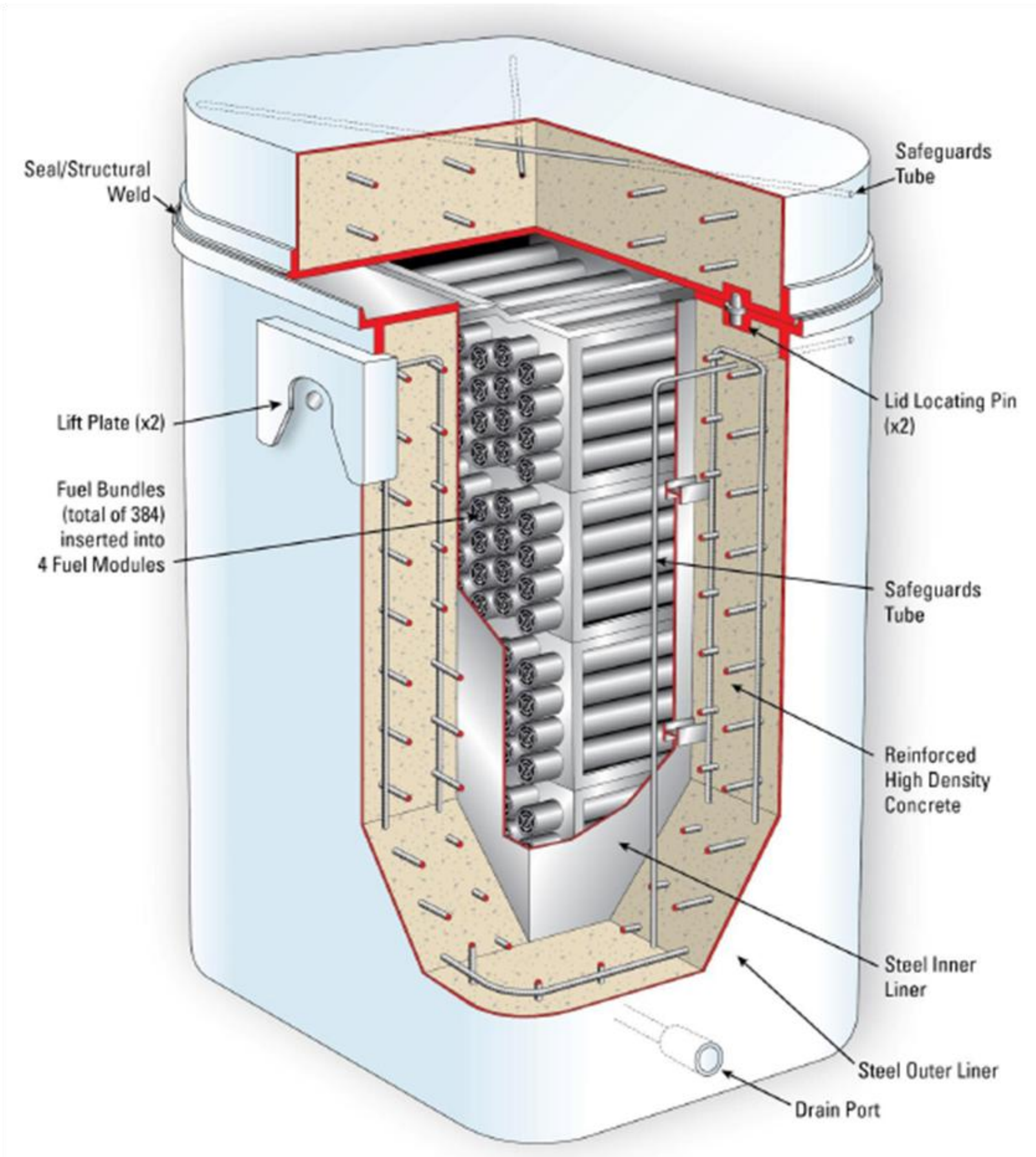

Figure 7: Dry Storage Container

4.1.2 Used Fuel Dry Storage Processing

The processing of a DSC begins with the preparation of new DSCs at the DSC Processing Building and ends with the storage of loaded, hermetically sealed DSCs in the DSC Storage Buildings. The steps are summarized in Figure 8.

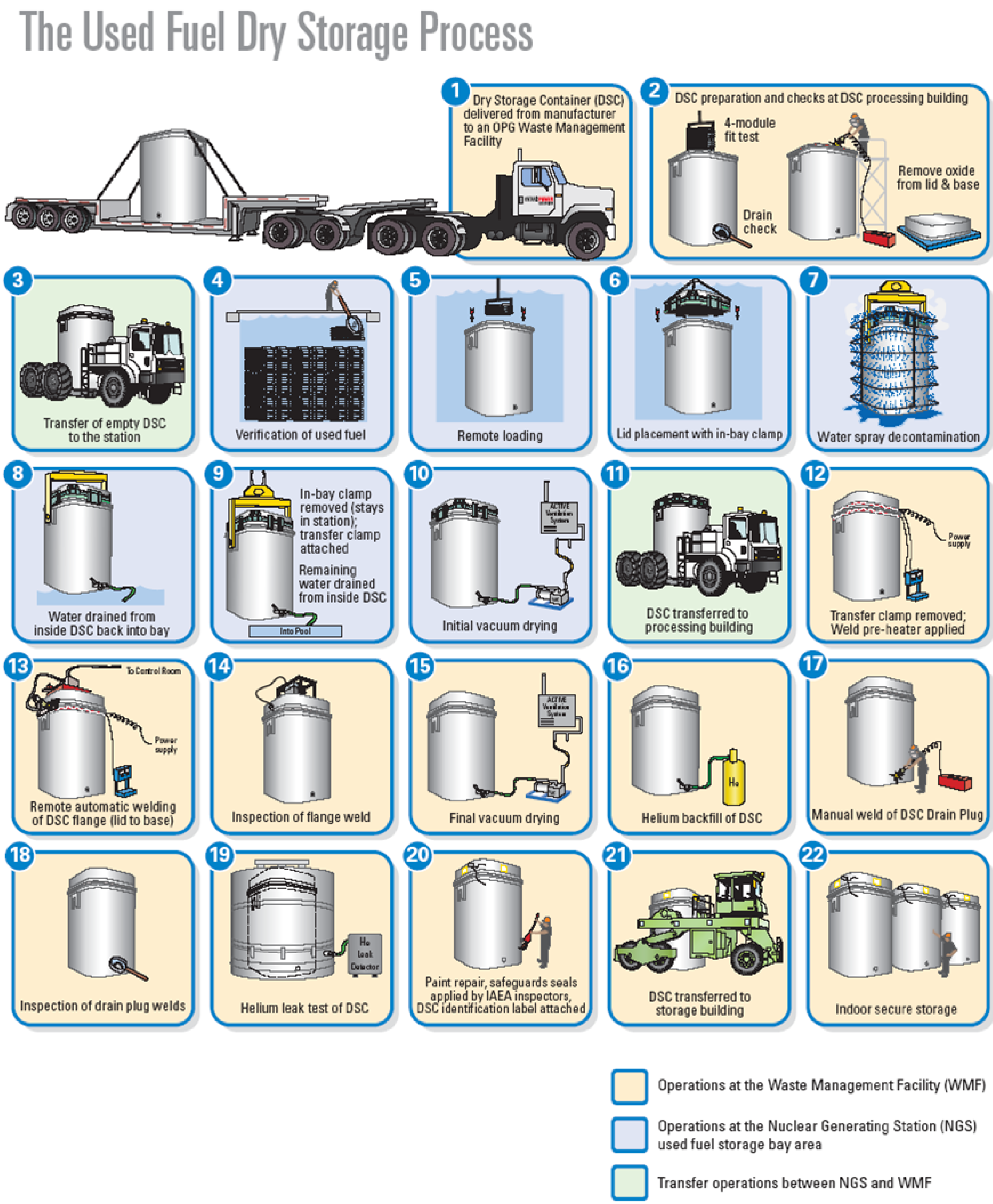

Figure 8: Used Fuel Dry Storage Process

Steps 1-3: Preparing and Transferring Empty DSCs

New, empty DSCs are received at the PWMF Phase I site from the manufacturers. The DSCs are then prepared and transferred to the Pickering NGS for subsequent loading of used fuel.

The DSC Transporter is used to transfer both new (empty) and loaded DSCs between PWMF and Pickering NGS. The on-site transfer of DSCs is described further in Section 4.1.3.

Steps 4-10: Loading a DSC at Pickering NGS

The processes of loading, decontaminating, draining and initial drying are completed at Pickering NGS under the Power Reactor Operating Licence. At the Pickering NGS, fuel bundles are loaded under water into storage modules. After a storage module has been loaded, it is transferred under water to a DSC. Each DSC is designed to hold four storage modules, for a total capacity of 384 bundles per loaded DSC.

While the loaded DSC is still submerged in water in the loading bay, the in-bay clamp is used to secure the DSC lid to the container. The DSC is lifted out of the water, drained and then the DSC exterior is decontaminated. The in-bay clamp is replaced with the transfer clamp, and the DSC interior cavity is vacuum-dried in preparation for on-site transfer to the PWMF.

Prior to leaving the Pickering NGS, the DSC is surveyed and the entire exterior surface of the loaded DSC and its components is decontaminated, including lid flange, drain housings and the transfer clamp, to ensure that there is no detectable loose contamination as per OPG’s Waste Acceptance Criteria.

Step 11: DSC Transfer between Pickering NGS and the DSC Processing Building at the PWMF

The DSC Transporter picks up a loaded DSC from the Pickering NGS after confirmation that it meets OPG’s Waste Acceptance Criteria. Both the vehicle and the DSC are monitored for contamination and decontaminated, as required, before leaving the station.

The transporter with a loaded DSC then leaves the station and travels along the Pickering NGS site roads to the PWMF Phase I site for further processing, in accordance with security and safeguards requirements for on-site transfer. The maximum lift height required during loading or unloading of a DSC is about 0.60 m, which is well within the safety envelope of 2.4 m. The vehicle is always operated by a trained vehicle operator.

Steps 12-20: Processing a DSC at PWMF

The loaded DSC is transferred on Pickering NGS site roads to the PWMF Phase I site, where it is off-loaded at the DSC Processing Building for further processing, as follows:

- <u>Receiving a Loaded DSC (Step 12)</u>: After the loaded DSC is received at the DSC Processing Building, movement of the DSC within the DSC Processing Building is performed using the workshop overhead crane and lifting beam.

- <u>DSC Lid Seal Welding (Step 13)</u>: The DSC is moved to a welding station where the DSC drain port transfer plug, transfer clamp and seal are removed and the weld pre-heater is installed. The pre-heater is used to heat the DSC weld flange to a prescribed temperature. The weld between the lid and base of the DSC is performed with 10 consecutive passes of the semi-automatic welder. At the conclusion of lid welding, the weld machine is removed and the DSC is allowed to cool.

- <u>Welding Inspection (Step 14)</u>: The Phased Array Ultrasonic Testing system is used for the inspection of the DSC lid-to-base seal weld. The scanner is mounted on the DSC base’s top flange and is held in place by three magnetic wheels. A loading ramp is used to minimize the force required by the operator when engaging and disengaging the scanner. The inspection covers 100% of the weld, as well as the Heat Affected Zone.

- <u>Final Vacuum Drying, Helium Backfill and Drain Port Seal Welding (Steps 15-18)</u>: After successful completion of the weld inspection, the DSC is lifted into another work station for final vacuum drying and helium backfilling. The lifting beam is removed and the vacuum drying/helium backfilling system connected. Following helium backfill, the drain port is welded and inspected via visual and dye penetrant techniques.

- <u>Helium Leak Testing (Step 19)</u>: Helium leak testing is carried out using a vacuum chamber (bell jar). The lid of the bell jar is removed and the seal-welded DSC is lifted into the lower half of the bell jar. The bell jar lid is craned over the DSC and sealed onto the base of the bell jar. Using the vacuum skid, air is first removed from the bell jar and then the helium leak detector is activated. If a leak is detected, the vacuum equipment is removed and remedial work is carried out. A follow-up leak test is then performed. After completion of the lid weld inspection, partially processed DSCs may be transferred inside the DSC Processing Building and temporarily stored for up to one year from the time of loading.

- <u>Paint Touch Up and Safeguards Seals (Step 20)</u>: Areas affected by the welding are cleaned and painted. Touch-up paint is also applied to scrapes or scuffs on the DSC that may have resulted from handling. Painting is typically carried out in the paint bays. Documentation and identification labelling are completed, and permanent safeguards seals are installed by IAEA inspectors in a designated IAEA surveillance area.

Steps 21-22: DSC Placement and Storage

The DSC is moved, using the DSC Transporter, to a location in a storage building (Figure 9). In the storage building, the DSC Transporter unloads the DSC in a designated storage location.

Figure 9: Storage of DSCs

4.1.3 On-Site Transfer of DSCs

DSC Transporters/Transfer Vehicles

The OPG DSC Transporters/Transfer Vehicles[3] are specially designed multi-wheeled vehicles for the transfer of loaded DSCs from the Pickering station's irradiated fuel bays to the DSC Processing Building and transfer of processed DSCs from the DSC Processing Building to storage (Figure 10). The DSC Transporters/Transfer Vehicles are self-powered by a diesel engine. The DSC is carried at a low lift height (about 20 cm) during transfer. The tires on the DSC Transporters/Transfer Vehicles will not deflate if punctured.

[3] In other sections of this report, DSC Transporters/Transfer Vehicles are, for convenience, referred to as DSC Transporters only.

When travelling with a DSC, the DSC Transporters/Transfer Vehicles operate at low speed and have a short stopping distance. When travelling at minimal speeds (e.g., when transferring DSCs within the DSC Processing and Storage Buildings), stopping is essentially instantaneous. The DSC Transporters/Transfer Vehicles are capable of forward and reverse motion and have a tight turning radius. Vehicle lighting is provided for night-time operation, if necessary.

PWMF may use either of two types of DSC Transporters/Transfer Vehicles. Each type has a different manufacturer: Liftking (shown in Figure 10) or MacLeans.

The Liftking and MacLeans do not require the assistance of a crane when picking up or positioning a DSC. The DSC is lifted and transferred via lifting trunnions mounted on the upper frame of these two machines. Locking arrangements prevent the DSC from being inadvertently lowered to the ground upon hydraulic failure.

The vehicle control systems limit the maximum speed of each type of vehicle (4 km/h for the Liftking and 12 km/h for the MacLeans).

Figure 10: Liftking DSC Transporter

Transfer Clamp

A transfer clamp is used to securely attach the lid to the DSC base during on-site transfer of a loaded DSC between the Pickering NGS irradiated fuel bays and the PWMF. The transfer clamp prevents the lid and base from separating under credible accident scenarios during the transfer of loaded DSCs between the station and the DSC Processing Building, and during DSC handling and storage inside the DSC Processing Building prior to seal-welding the DSC lid to the DSC base.

4.1.4 DSC Storage Buildings for Used Fuel

Each DSC Storage Building is designed and constructed to provide for the safe storage of DSCs. Each storage building is a single-storey, commercial-type, pre-engineered or precast concrete structure with a concrete slab-on-grade floor. The floors are constructed for long service with minimal maintenance, to retain surface alignment and provide a hard, smooth and durable surface. Floors are sloped to provide drainage to floor drains. Building walls consist of precast concrete panels to provide effective radiation shielding. The walls above the concrete panels consist of metal panels. A combination of wall louvres and roof turbines are installed to assist passive ventilation. The building provides weather protection for DSCs in storage. The DSC Storage Buildings are designed to the *National Building Code of Canada* and the *National Fire Code of Canada*.

The building roof has provisions for drainage of rainwater and melted snow. Access to the roof is by the use of an outside, all-weather permanent stairway. The building is grounded to protect against lightning.

4.2 Normal Operations

4.2.1 Radioactive Emissions and Contamination

Under normal operating conditions, no airborne emissions are expected from loaded DSCs during transfer from the station to the DSC Processing Building. Airborne releases are also unlikely to arise under normal operating conditions during storage of seal-welded DSCs. There is a small potential for airborne emissions as a result of DSC processing operations, such as welding and vacuum drying.

Surface contamination on DSC exterior surfaces is effectively controlled through preventative measures and decontamination at the station irradiated fuel bays. Nevertheless, small quantities of fixed surface contamination may become airborne during welding operations.

PWMF experience demonstrates that particulate emissions in exhaust from DSC processing operations are typically below the minimum detectable activity.

Since the potential for chronic emissions is very low, the assessment presented below is considered an upper bound for any possible chronic emissions during normal operating conditions.

The potential emissions under normal operating conditions have been evaluated. Since each DSC has the capacity to hold 384 used fuel bundles, and assuming that the PWMF processes about 70 DSCs per year, it is postulated that a total of 280 fuel elements (one fuel element in 1% of the fuel bundles is assumed to be damaged, or four elements per DSC) fail during one year under normal operating conditions (a very conservative scenario). The chronic off-site dose consequences from this scenario for a member of the public at the Pickering NGS site property boundary are estimated to be 1.02×10^{-3} µSv/year for an adult and 7.88×10^{-4} µSv/year for an infant. Both dose rates are approximately 10^{-4} % of the CNSC regulatory dose limit of 1 mSv/year (see Table 1). A nuclear energy worker working in the DSC Processing Building is postulated to receive a consequent dose of 0.172 mSv/year. Given the conservative assumptions used, these values are an upper-bound estimate for airborne emissions.

As the DSC is fully drained and vacuum dried after loading at the station irradiated fuel bays, and the elastomeric seal and the drain plug are present during transfer, there will be no liquid emissions from the DSC during on-site transfer to the PWMF.

The exterior surfaces of DSCs are decontaminated prior to their transfer from the irradiated fuel bays to the PWMF. Spot decontamination operations, which may be carried out in the DSC Processing Building, are not expected to generate liquids. No liquid will be present inside DSCs during dry storage in the DSC Storage Buildings. Liquids are not normally used in the DSC Storage Buildings.

No loose contamination is permitted (or expected) either on the exterior surfaces of DSCs or on accessible surfaces such as floors in the DSC Processing and Storage Buildings. This is confirmed through routine contamination monitoring.

Since no liquids are present in the DSC and loose contamination is not permitted on DSC or facility surfaces, no contaminated liquid effluents are expected from PWMF operations.

4.2.2 Radiation Fields

DSC Dose Rates

Using the radionuclide inventory data and taking into consideration the shielding provided by the steel re-bar used to reinforce the heavy concrete in the container, the radiation fields for a fully-loaded DSC have been calculated for 10-year-cooled Pickering used fuel. Table 3 shows the calculated gamma radiation dose rates for different distances from the top, side, front and bottom surfaces of a DSC of the original design, fully-loaded with 10-year-cooled Pickering reference fuel bundles, as defined within the original shielding analysis. Figure 11 shows the calculated gamma radiation dose rates as a function of the distance from the top, side, front and bottom surfaces of a DSC of the modified Long Module design, fully-loaded with 10-year-cooled Pickering reference fuel bundles.

Table 3: Calculated Dose Rates from a DSC of the Original Design

Distance from DSC	Position	Dose Rate (μSv/h)
Contact	Side*	67
	Front*	67
	Top	82
	Bottom	171
1 m	Side*	48
	Front*	56
	Top	50
	Bottom	108
2 m	Side*	30
	Front*	38
	Top	26
	Bottom	55

* The label ‘Front’ corresponds to the wider face of the DSC, and ‘Side’ indicates the narrower face.

Previous analysis showed that, due to the heavy concrete used as shielding material in the container, the contribution of neutrons to dose rate is negligible compared to that of gamma radiation. Neutron dose rate contributions, therefore, were not calculated.

Calculated dose rate estimates have been demonstrated to be conservative compared with actual DSC dose rates measured during used fuel dry storage operations. For DSCs loaded with 10-year-cooled or older used fuel, measured contact dose rates to date are about 9 to 13 μSv/h. This is compared with estimates of 67 μSv/h contact dose rates for 10-year-cooled fuel (at the DSC side or front) as set out in Table 3. At a 1 m distance, measured dose rates are about 5 to 7 μSv/h, compared with the calculated dose rate estimates of 48 and 56 μSv/h in Table 3.

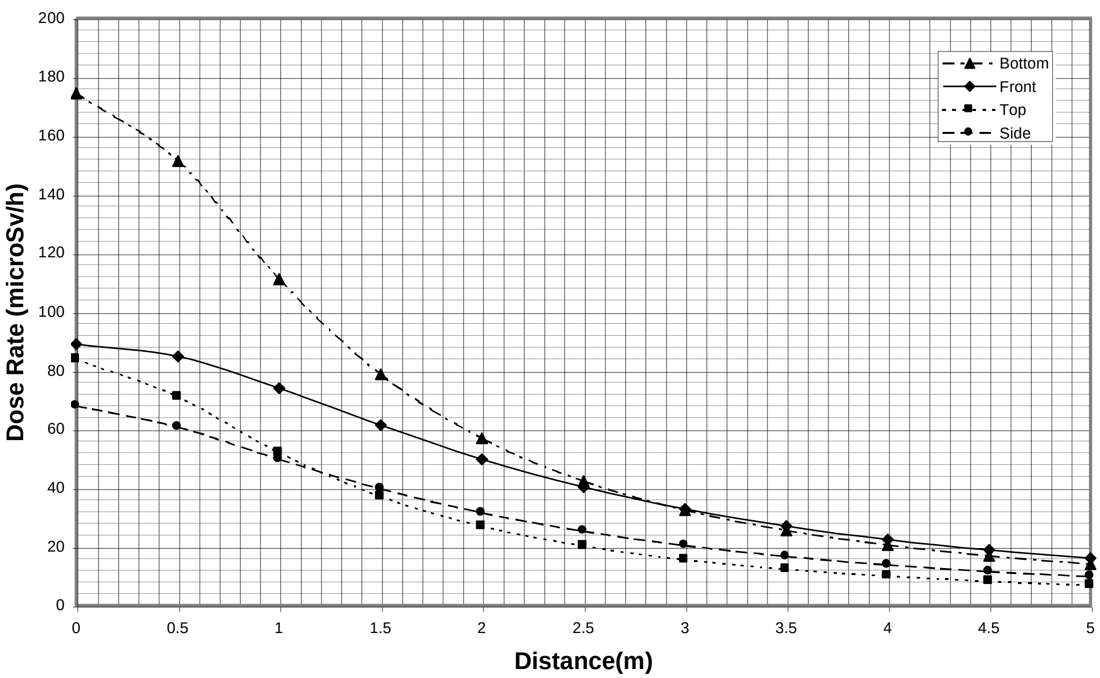

Figure 11: Calculated Dose Rates from a DSC Containing 10-Year-Cooled Used Fuel

<u>Notes for Figure 11</u>:

1. The dose rates have been calculated for a DSC of the modified Long Module design.
2. ‘Front’ corresponds to the wider face of the DSC.
3. The average concrete density of the DSC is 3.57 g/cc (taking into account the steel re-bar).

Dose Rates Inside the DSC Storage Buildings

A shielding analysis was conducted to calculate the dose rates inside the PWMF DSC Storage Buildings from DSCs loaded with 10-year-cooled used fuel. This assumption yields a conservative estimate for occupational dose rates inside the DSC Storage Buildings since it is based on the minimum age of used fuel that can be stored at the PWMF. Operating experience has shown that the calculated dose rates for the assumed fuel age were conservative and that actual dose rates are lower.

Figure 12 shows the calculated dose rates across the 2 m width of the east-west corridors. The maximum dose rate in-between rows was estimated at approximately 160 μSv/h (including roof-shine), almost uniformly across the narrow east-west corridors. Figure 13 shows the calculated dose rates across the 10 m width of the north-south corridor. The dose rate along the sides of the main north-south corridor was estimated to be about 80 μSv/h (including roof-shine). Measured dose rates are well below these calculated values.

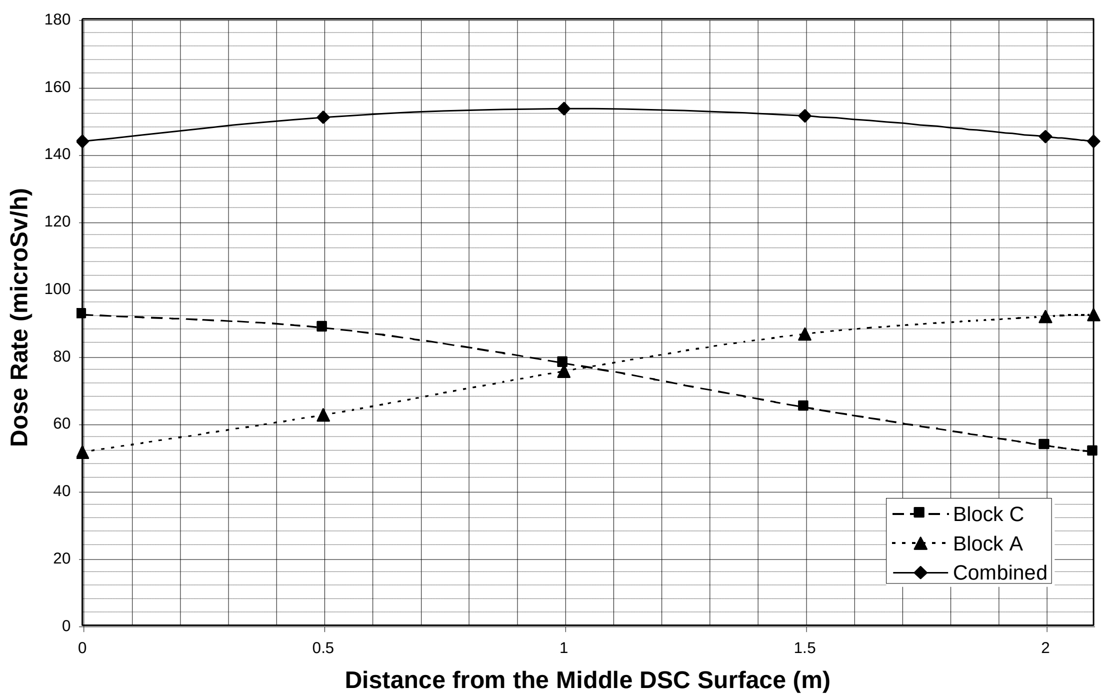

Figure 12: Calculated Dose Rates Inside a DSC Storage Building – East-West Corridor

Dose rates inside the PWMF have been demonstrated to be acceptably low; most working areas are normally at or near ambient background radiation levels. Higher dose rates are normally found when walking between stored DSCs; however, workers employ survey monitoring. Also, appropriate dosimetry is required in areas that have potentially elevated dose rates. Low occupancy rates are expected for all DSC Storage Buildings.

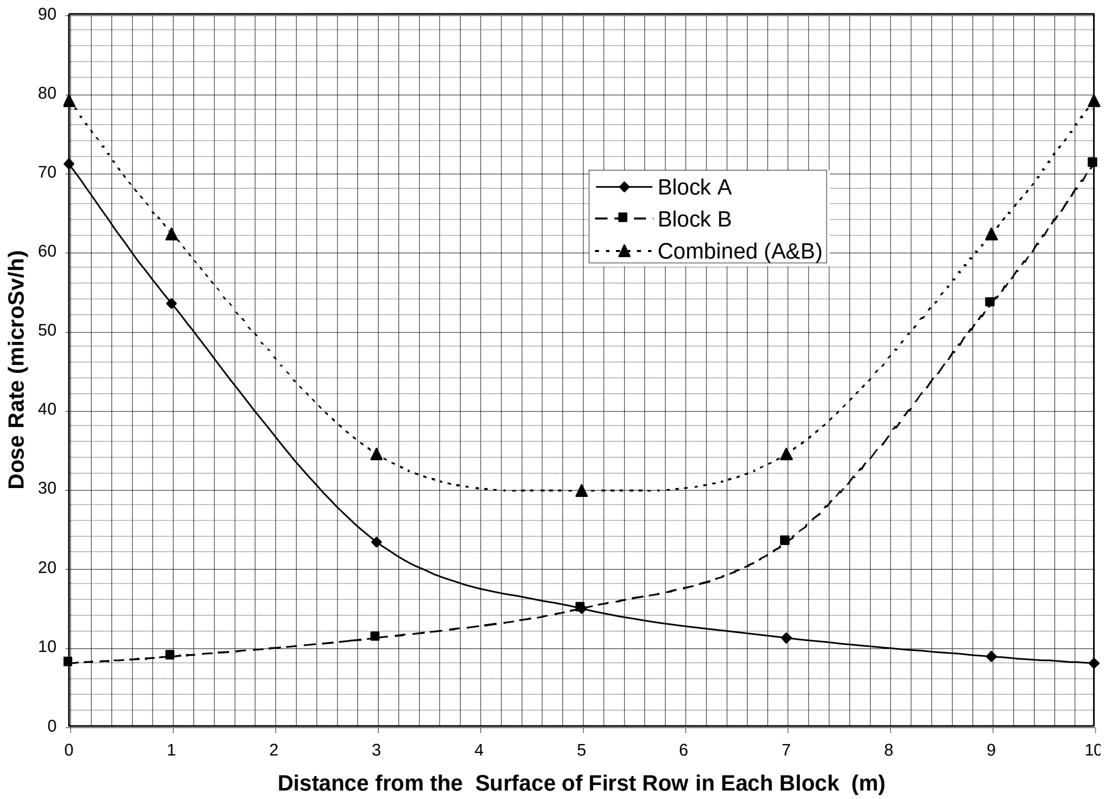

Figure 13: Calculated Dose Rates Inside a DSC Storage Building – North-South Corridor

Dose Rates Outside the DSC Storage Buildings

The design assumptions used in the calculations have resulted in conservative dose rate estimates. The dose rates are provided below at varying distances for the PWMF Phase I and II sites.

PWMF Phase I Site

A dose rate of about 0.5 μSv/h is estimated at a 5 m distance from the PWMF Phase I storage area wall. At the inland site property boundary, 850 m east of the building wall, the estimated dose rate is 6×10^{-6} μSv/h. This is equivalent to an annual dose of 0.05 μSv, assuming full occupancy at the fence. At the eastern lakeside exclusion zone boundary (420 m distance from the PWMF Phase I storage areas), the estimated dose rate is 6×10^{-4} μSv/h, or 0.6 μSv/year based on 1,000 hours' occupancy (conservative assumption for boaters and fishermen).

These results indicate that the PWMF dose rate targets of ≤ 0.5 μSv/h at the station security fence and at the facility perimeter fence on a quarterly average basis and ≤ 10 μSv/year at the site property boundary, as set out in Section 1.2, are met during used fuel dry storage operations.

PWMF Phase II Site

Figure 14 shows calculated dose rates versus distance from the east or west wall of PWMF Phase II DSC Storage Buildings[4]. Figure 15 shows calculated dose rates from the north or south wall. The shielding analysis assumed 0.3 m concrete shielding walls in the DSC Storage Buildings.

The calculated dose rates at the perimeter fence, approximately 15 m from the north or south walls of the DSC Storage Buildings, are 0.25 μSv/h and 0.18 μSv/h, respectively. The corresponding calculated dose rate at a similar distance from the east or west walls is lower at 0.15 μSv/h, due to the DSC design and the planned orientation of DSCs in storage. These dose rates are well within the criterion of ≤ 0.5 μSv/h established for limited occupancy (i.e., up to 2,000 hours per year) by non-nuclear energy worker personnel at the perimeter fence.

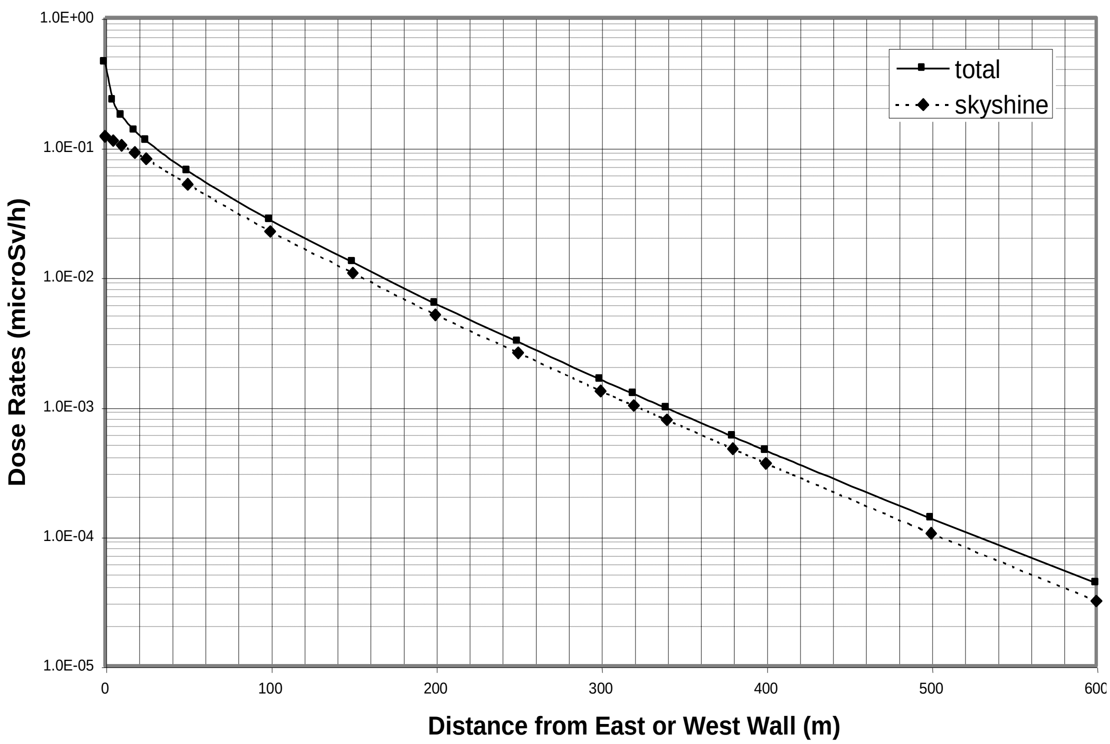

Figure 14: Calculated Dose Rates from Phase II DSC Storage Buildings – East/West Walls

The concrete panels on the north side of DSC Storage Building #3 have been extended in height to provide increased shielding to ensure that dose rates throughout the Training and Mock-up Building are below the dose rate limit of 0.5 μSv/h (see Section 1.2). The dose rate at the Training and Mock-up Building was estimated to be about 0.3 μSv/h.

[4] In this assessment, it has been assumed that the PWMF Phase II site has been expanded with one more DSC Storage Building with the same capacity as DSC Storage Building #3.

The calculated gamma radiation levels from the PWMF Phase II site provide a dose rate of about 1.1×10^{-3} μSv/h, or about 9.6 μSv/year at the Pickering NGS east property boundary (coinciding with the Pickering NGS exclusion zone boundary). The dose estimate is based on year-round occupancy and includes both direct and skyshine contributions. This meets the PWMF dose rate target of ≤ 10 μSv/year established for full occupancy at the Pickering NGS site property boundary and is expected to be indistinguishable from the variations in natural background radiation levels at this location.

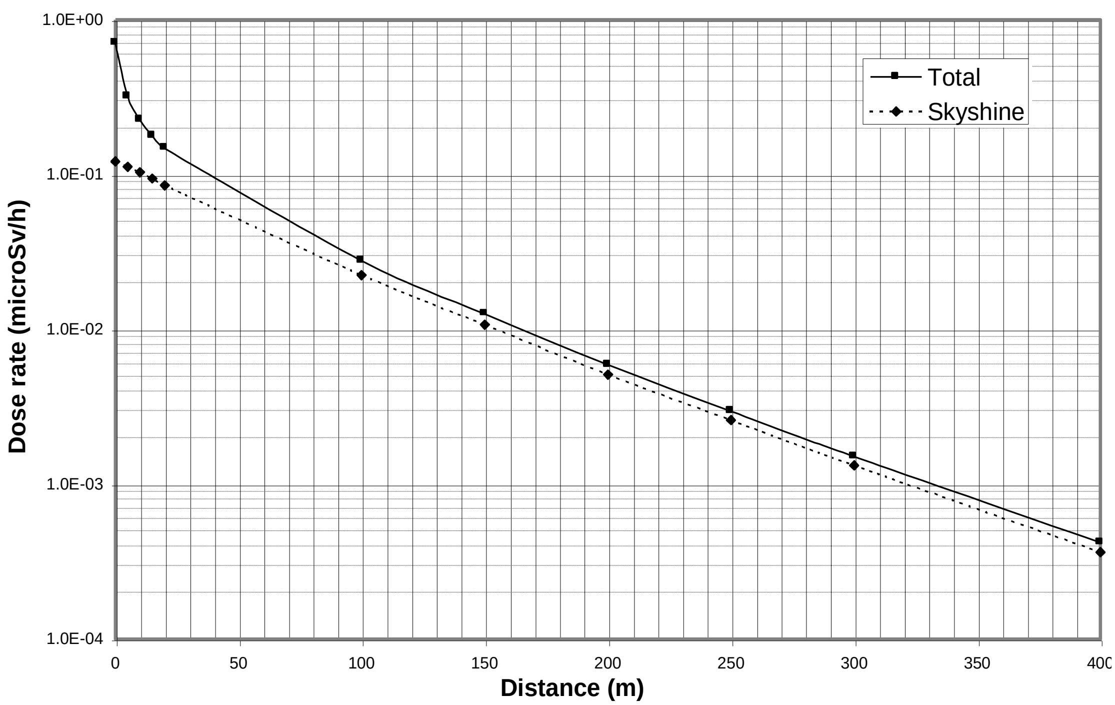

Figure 15: Calculated Dose Rates from Phase II DSC Storage Buildings – North/South Walls

At the lakeside exclusion zone boundary (about 340 m southeast over Lake Ontario at the closest location), the estimated dose rate is 9.7×10^{-4} μSv/h, or about 1 μSv/year based on 1,000 hours' occupancy (a conservative assumption for boaters and fishermen).

4.3 Malfunctions and Accident Assessments

The operation of the PWMF may be affected by abnormal or credible accident conditions. This section provides a summary of assessment of the potential impacts of postulated events both inside and external to the PWMF.

Given the very distinctive stages of the Pickering used fuel dry storage process, the assessment of malfunctions and accidents was divided into the following main stages of the out-of-station used fuel dry storage operations:

(a) On-site transfer operations.

(b) Operations inside the DSC Processing Building.

(c) DSC storage.

For each stage of the used fuel dry storage operations, release of radiation due to fuel sheath failure can occur due to physical damage and/or failure of the systems and components used during used fuel dry storage operations.

Each event was screened to establish if it could result in any radiological impact to the public, the workers and the environment. Design provisions and procedural measures that could prevent the event or mitigate its consequences were also considered.

Release of radionuclides from a seal-welded DSC is not expected, even under abnormal operating conditions, because of the robustness of the DSC and fuel bundle design.

However, to assess the overall safety of the used fuel dry storage operations at PWMF, safety assessments presume that credible accidents will result in the failure of multiple barriers and release of radioactive material. Bounding (worst-case) accident scenarios are conservatively identified, even if they are unlikely to occur, and the results of off-site dose consequence calculations are then compared against the regulatory dose limits.

4.3.1 During DSC On-Site Transfer

As described in Section 4.1.3, the DSC Transporter is used to transfer loaded DSCs from the Pickering NGS irradiated fuel bays to the DSC Processing Building. It is also used to transfer seal-welded DSCs from the PWMF Phase I site to the PWMF Phase II site. The DSC Transporter provides its own motive power and DSC lifting capability via its diesel engine.

The DSC on-site transfer safety assessment has taken into account postulated malfunctions and accidents that could potentially affect the on-site transfer of a loaded DSC.

Table 4 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., with a frequency of occurrence that is $>10^{-7}$ events per year) during DSC on-site transfer.

The bounding dose consequences during this stage of the dry storage process are associated with the drop of a DSC during on-site transfer between the PWMF Phase I and Phase II sites. Although fuel sheath failure is not expected to result from a DSC drop from the low lift height of the DSC Transporter, the drop of a DSC during on-site transfer was conservatively assumed to result in 30% failure of the fuel elements inside a DSC.

Consequently, the free inventory from the failed fuel elements is assumed to be released, making this accident scenario the bounding accident scenario during on-site transfer operations.

Table 4: Postulated Malfunctions or Accidents during DSC On-Site Transfer

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (μSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
DSC Transporter Failure	Credible	3.44×10^{-4}	2.66×10^{-4}	2.46×10^{-3}
DSC Drop During Transfer Between PWMF Phase I and Phase II Sites	Credible	3.33	2.58	1.98
Fire	Credible	$<3.44 \times 10^{-4}$	$<2.66 \times 10^{-4}$	$<2.46 \times 10^{-3}$
Criticality	Incredible*	—	—	—
Adverse Road Conditions	Credible	$<3.44 \times 10^{-4}$	$<2.66 \times 10^{-4}$	$<2.46 \times 10^{-3}$
Earthquake	Credible	$<3.44 \times 10^{-4}$	$<2.66 \times 10^{-4}$	$<2.46 \times 10^{-3}$
Tornado	Incredible*	—	—	—
Thunderstorm	Credible	$<3.44 \times 10^{-4}$	$<2.66 \times 10^{-4}$	$<2.46 \times 10^{-3}$
Flood	Incredible*	—	—	—
Explosion Along Transfer Route	Credible	0	0	0
Turbine Missile Strike	Incredible*	—	—	—
Aircraft Crash	Incredible*	—	—	—

* An incredible event has an expected frequency of occurrence that is less than 10^{-7} per year.

The free inventory of tritium and krypton-85 in the damaged fuel bundles is assumed to be released into the DSC cavity. Ignoring that DSCs being transferred from the PWMF Phase I site to the Phase II site are already seal-welded, it is assumed that these radionuclides are released at once into the environment.

Assuming that this event occurs at the PWMF Phase II site, the total dose to the public was calculated to be 3.33 µSv (0.00333 mSv) for an adult and 2.58 µSv (0.00258 mSv) for an infant at the Pickering site boundary. These are both well below the public dose limit of 1 mSv (see Table 1). The dose to a nuclear energy worker would be 1.98 mSv, which is below the target for accident conditions for a general nuclear energy worker (see Table 1).

4.3.2 During DSC Processing

The processes and systems taken into account for this assessment encompass those at the DSC Processing Building once the DSC Transporter arrives at the PWMF with a loaded DSC and before the DSC is taken to storage, as described in Section 4.1.2.

Table 5 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., with a frequency of occurrence that is $>10^{-7}$ events per year) during DSC processing.

The bounding dose consequences during this stage of the dry storage process are associated with the event in which the DSC drops during handling. Conservatively, it was assumed that, as a result of this event, 30% of the fuel elements inside the DSC are damaged and the free inventory from the failed fuel elements is released from the PWMF into the environment.

The free inventory of tritium and krypton-85 in the damaged fuel bundles is assumed to be released into the DSC cavity. The barrier provided by the transfer clamp seal is ignored, and these radionuclides are assumed to be released at once into the environment. The total dose to the public due to this event was assessed to be 2.78×10^{-1} µSv (0.000278 mSv) for an adult and 2.15×10^{-1} µSv (0.000215 mSv) for an infant at the Pickering site boundary. These are both well below the public dose limit of 1 mSv (see Table 1). The occupational dose to a nuclear energy worker in close proximity to the dropped DSC was assessed to be 1.98 mSv, which is below the target for accident conditions for a general nuclear energy worker (see Table 1).

Table 5: Postulated Malfunctions or Accidents during DSC Processing

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (µSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Drop of a DSC During Handling	Credible	2.78×10^{-1}	2.15×10^{-1}	1.98
Equipment Drop onto a DSC	Credible	$<2.78 \times 10^{-1}$	$<2.15 \times 10^{-1}$	<1.98
DSC Collision During Craning	Credible	$<2.78 \times 10^{-1}$	$<2.15 \times 10^{-1}$	<1.98
DSC Transporter Collision with a Loaded DSC or Another Transporter	Credible	$<2.78 \times 10^{-1}$	$<2.15 \times 10^{-1}$	<1.98
Equipment Collision with a Loaded DSC During Craning	Credible	$<2.78 \times 10^{-1}$	$<2.15 \times 10^{-1}$	<1.98
Criticality	Incredible*	—	—	—
DSC Processing Building Fire	Credible	0	0	0
Earthquake	Credible	$<2.78 \times 10^{-1}$	$<2.15 \times 10^{-1}$	<1.98
Tornado	Incredible*	—	—	—
Thunderstorm	Credible	0	0	0
Flood	Incredible*	—	—	—
Turbine Missile Strike	Incredible*	—	—	—
Aircraft Crash	Incredible*	—	—	—

* An incredible event has an expected frequency of occurrence that is less than 10^{-7} per year.

4.3.3 During DSC Storage

Once the DSC processing is completed, the DSC Transporter moves the DSC from the DSC Processing Building to one of the DSC Storage Buildings for storage.

Table 6 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., with a frequency of occurrence that is $>10^{-7}$ events per year) during the storage process.

The bounding dose consequences during this stage of the used fuel dry storage process are associated with the event in which DSC seal-welds fail during storage. During storage, both the fuel sheath and the DSC seal-weld must fail for a release of radionuclides to occur. Used fuel having a known damaged or defective sheath is not loaded into a DSC. Failure of the sheath is not expected to occur during the operating life of the storage facility. Nevertheless, to make a conservative assessment of the consequences of a release during storage, it is postulated that 10% of the DSCs in the full PWMF experience a lid seal-weld failure over a 1-year period, resulting in 66 DSCs with failed seal-welds at the PWMF Phase I site and 100 DSCs with failed seal-welds at the PWMF Phase II site[5].

It is conservatively assumed that 1% of the fuel bundles in the DSCs are defective with one defective fuel element per bundle, i.e., 1% of 384 bundles (four bundles) having one defective fuel element in each (four fuel elements per DSC).

At the PWMF Phase I site, this scenario postulates that 264 failed fuel elements release their free inventory into the environment over the course of a year. The total dose to the public due to this chronic release event was assessed to be 9.61×10^{-4} µSv (0.000000961 mSv) for an adult and 7.43×10^{-4} µSv (0.000000743 mSv) for an infant at the Pickering NGS site boundary.

At the PWMF Phase II site, this scenario postulates that 400 failed fuel elements release their free inventory into the environment over the course of a year. The total dose to the public due to this event was assessed to be 1.30×10^{-2} µSv (0.0000130 mSv) for an adult and 1.01×10^{-2} µSv (0.0000101 mSv) for an infant at the Pickering NGS site boundary.

The total public dose due to this event from the combined DSCs in storage at both the PWMF Phase I and Phase II sites was assessed to be 1.40×10^{-2} µSv (0.0000140 mSv) for an adult and 1.08×10^{-2} µSv (0.0000108 mSv) for an infant at the Pickering NGS site boundary.

The calculated doses for all scenarios above are well below the public dose limit of 1 mSv (see Table 1).

[5] In this assessment, it has been assumed that the PWMF Phase II site has been expanded with one more DSC Storage Building with the same capacity as DSC Storage Building #3.

Table 6: Postulated Malfunctions or Accidents during DSC Storage

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (μSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Seal Weld Failure During Storage	Credible	1.40×10^{-2}	1.08×10^{-2}	—
DSC Drop During Transfer to Storage	Credible	0	0	0
DSC Transporter Collision with a DSC or Another Transporter	Credible	0	0	0
Criticality	Incredible*	—	—	—
DSC Storage Building Fire	Credible	0	0	0
Earthquake	Credible	0	0	0
Tornado	Credible	0	0	0
Thunderstorm	Credible	0	0	0
Flood	Incredible*	—	—	—
Hazardous Material Building Explosion	Credible	0	0	0
Turbine Missile Strike	Incredible*	—	—	—
Aircraft Crash	Incredible*	—	—	—

* An incredible event has an expected frequency of occurrence that is less than 10^{-7} per year.

5.0 CONCLUSION

Retube Components Storage Area

The DSMs are designed and constructed such that dose rate targets at exterior surfaces of the structures, at the facility fence and at site boundaries are achieved. Dose rates in the RCS area are routinely monitored and shown to be within PWMF targets, well below regulatory limits.

Potential exposures from the RCS area under accident conditions have been reviewed and no credible events that would lead to a failure of the DSMs have been identified. Conservative estimates of worst-case doses from extreme conditions, such as a lightning strike or impact of a small aircraft, are well below the regulatory limits.

Used Fuel Dry Storage Areas

The radiological safety assessment for the used fuel dry storage process at the PWMF has addressed worker and public doses under both normal and credible accident conditions.

Based on PWMF operating experience, the environmental releases from the PWMF under normal operating conditions are expected to remain within regulatory limits. The safety assessment has concluded that the doses to members of the general public arising from radioactive releases are well below the public dose limits established by the CNSC. Doses to members of the general public, from direct radiation at the Pickering NGS site property boundary, are also well below the CNSC public dose limits.

The DSC has been assessed to withstand a range of external accident conditions, including fires, tornadoes, earthquakes and thunderstorms. No significant off-site or occupational dose consequences are expected to result from these events.

6.0 ACRONYMS

ADF	Atmospheric Dilution Factor
ALARA	As Low As Reasonably Achievable
C-14-p	C-14 Particulate
CANDU	Canada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
DC	Dose Coefficient
DSC	Dry Storage Container
DSM	Dry Storage Module
IAEA	International Atomic Energy Agency
NGS	Nuclear Generating Station
OPG	Ontario Power Generation
PWMF	Pickering Waste Management Facility
RCS	Retube Components Storage

7.0 REFERENCES

CSA91 Canadian Standards Association, 1991, "Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors", CAN/CSA-N288.2-M91, April.

HC99 Health Canada, 1999, "Recommendations on Dose Coefficients for Assessing Doses from Accidental Radionuclide Releases to the Environment", March.

Appendix A: Atmospheric Dilution Factors

ADFs are used to provide estimates of the amount of dilution for a contaminant released into the atmosphere, between the emission point and a receptor location. The ADFs are calculated either from the measured ambient concentrations and the source emission rates or from the measured hourly values of wind direction, wind speed and standard deviation of wind direction.

Pollutant concentrations directly downwind from a source decrease with increased sampling time because of increased meander of wind directions. Thus, the derived ADFs follow this same pattern for constant emission rates. The calculated ADFs are based on five years of meteorological data measurements and are grouped into three averaging sampling periods:

Short Term — For the first 1-hour period, defined as the 90th percentile value of the cumulative frequency distribution of the calculated ADFs.

Prolonged Term — Between 1-hour and 24-hour period, defined as the 90th percentile value of cumulative frequency distribution of the calculated ADFs for the worst wind sector, based on 24 consecutive hours of meteorological measurements.

Long Term — For more than 24-hour period, takes into account joint frequency of wind speed, stability class and wind direction over the time period of interest.

The ADFs are calculated using the model suggested in Canadian Standards Association (CSA) standard CAN/CSA N288.2 (CSA91).

For both the PWMF Phase I and Phase II sites, ADFs are calculated for the site boundary. Conservative assumptions are made for all calculation parameters, including local surface roughness and thermal inversion boundary layer conditions.

The calculated ADFs applicable to the PWMF Phase I site, located 850 m from the Pickering NGS east property boundary, are shown in Table 7, as are the calculated ADFs applicable to the PWMF Phase II site, located 330 m from the Pickering NGS east property boundary.

Table 7: Atmospheric Dilution Factors at PWMF Phase I and Phase II Sites

Period	Phase I Site	Phase II Site
Short Term	5.2×10^{-5} s/m^3	6.24×10^{-4} s/m^3
Prolonged Term	7.5×10^{-6} s/m^3	6.05×10^{-5} s/m^3
Long Term	2.2×10^{-6} s/m^3	1.97×10^{-5} s/m^3