Request for Confidentiality

of Material Submitted in Relation to CMD # 24-H102

In the matter of: **OPG's Application to Change the Licensing Basis for the Pickering Waste Management Facility**

This request has been prepared in Canada, in the province of **Ontario** in the matter of **OPG's Application to Change the Licensing Basis for the Pickering Waste Management Facility** scheduled for consideration in a **Hearing in writing,** scheduled for **June 30, 2024.**

I, **Kapil Aggarwal**, of **889 Brock Rd. Pickering, ON, L1W 3J2,** am an authorized representative of **Pickering Waste Management Facility**. I understand that:

- documents and information ("the material") provided to the Canadian Nuclear Safety Commission ("the Commission") as part of a public proceeding may be made publicly available;
- the material is considered confidential only if it is prescribed information under the *Nuclear Safety and Control Act* (NSCA), as defined in section 21 of the *General Nuclear Safety and Control Regulations*, or if the Commission takes measures to protect the information; and
- regardless of any request for confidentiality or approval of same, the material may be disclosed if the Commission is required by law to disclose it (for example, after a request under *Access to Information Act*).

I hereby request that the Commission take measures to protection the following information, pursuant to rule 12 of the *Canadian Nuclear Safety Commission Rules of Procedure*:

Note: Where the request for confidentiality applies only to part of the submission, the portions to be deemed confidential must be clearly identified to distinguish them from any content that is non-sensitive.

Table 1: Material to be Deemed Confidential

	Item Name	Portion(s) to be Deemed Confidential
1.	92896-SR-01320-10002-R006, "*Pickering Waste Management Facility – Safety Report*"	☐ Entire content ✓ Redacted content as shown
2.	CD# 92896-CORR-00531-01487, "*CNSC Staff's Prior Written Notification of Document Changes: 92896-SR-01320-10002, Nuclear Sustainability Services – Pickering Waste Management Facility Safety Report, R007*"	☐ Entire content ✓ Redacted content as shown
3.	Enclosure 1 and Enclosure 2 of CD# 92896-CORR-00531-01430, "*OPG Response to CNSC Staff Comments on OPG's Proposal to Store Minimum 6- Year Old Cooled Used Fuel at the Pickering Waste Management Facility*"	✓ Entire content ☐ Redacted content as shown

This request is made pursuant to the following paragraph(s) of rule 12 of the *CNSC Rules of Procedure*:

- **Commission Rules of Procedure: Confidentiality, Section 12 (1) b.**

Further,

1. The above-noted material should be protected for the following reasons:

   The supporting technical and safety assessments, contained and referenced in these submissions, are deemed confidential information of a financial, commercial, scientific, and

technical nature, that is treated consistently as confidential, and the vendor partners affected have not consented to public disclosure.

2. I attest that the above-noted material is not available through any public sources.
3. I have included a **summary** or **redacted** version of the material that provides adequate detail to satisfy the public interest in public hearings and disclosure of evidence.
4. I understand that if this request is not approved by the Commission, I may withdraw the associated material within five business days of receiving written notice of the Commission's decision from the Commission Registrar (except as noted in items 5 and 6, below).
5. Notwithstanding item 4, above, I understand that if submission of the material is required pursuant to reporting requirements under the NSCA or the regulations under the NSCA, or pursuant to a licence issued under the NSCA, or if the material is specifically requested by the Commission, it may **not** be withdrawn.
6. I understand that upon receipt of this request, the Commission Registrar will treat the material that is subject to this request as confidential unless and until the Commission makes a ruling to deny this request.

Attachments:

The attachments provided under CD# 92896-CORR-00531-01510 include a redacted version or summary of the documents listed in Table 1 as shown below:

- 92896-SR-01320-10002-R006, "*Pickering Waste Management Facility – Safety Report*" – **Redacted (Attachment 1)**
- CD# 92896-CORR-00531-01487, "*CNSC Staff's Prior Written Notification of Document Changes: 92896-SR-01320-10002, Nuclear Sustainability Services – Pickering Waste Management Facility Safety Report, R007"* – **Redacted (Attachment 2)**
- Enclosure 1 and Enclosure 2 of CD# 92896-CORR-00531-01430, "*OPG Response to CNSC Staff Comments on OPG's Proposal to Store Minimum 6- Year Old Cooled Used Fuel at the Pickering Waste Management Facility*" – **Summary (Attachment 3)**

Authorized signature:

Kapil Aggarwal

Kapil Aggarwal, VP, Nuclear Sustainability Services

2024/05/21
Date

ATTACHMENT 1

CD# 92896-CORR-00531-01510 P

Redacted Pickering Waste Management Facility - Safety Report (92896-SR-01320-10002-R006)

ONTARIOPOWER GENERATION

Safety Report

OPG Proprietary	
Document Number: 92896-SR-01320-10002	
Sheet Number: N/A	Revision: R006

Title:
PICKERING WASTE MANAGEMENT FACILITY - SAFETY REPORT

Unless otherwise stated, all information redacted in this document is considered prescribed information under the General Nuclear Safety and Control Regulations

PICKERING WASTE MANAGEMENT FACILITY - SAFETY REPORT

92896-SR-01320-10002-R006

November 2018

OPG Proprietary

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Revision Summary

Revision Number	Date	Comments
R006	2018-10-31	**General** • Revised cover page to include staff changes • Addition of information related to safety-related systems throughout as per W-LIST-03673-00001 • Revised event screening criteria throughout to be greater than or equal to 10^{-6} events per year to be consistent with CSA N292-014 • Section numbering change to reflect the addition of Section 6, Retube Components Storage Safety Assessment • Decay heat for the reference fuel bundle has been retained from the R05 version of the PWMF SR. Sections 3.2.4, 3.3.2.1, 3.3.2.2, B.2.4, B.2.5, B.6.1.3 and Figure B-4 have retained the previous decay heat value of 5.8 W/bundle.
		Chapter 1 • Figure 1-1 updated with a more recent image • Section 1.2.1, par. 5: Updated number of DSCs stored at PWMF Phase I and II sites by December 2017 • Section 1.6.1: Administrative dose target for the public at the site boundary revised to 100 µSv/year. Technical justification included. • Section 1.7: Section deleted to be consistent with WWMF and DWMF Safety Reports
		Chapter 2 • Section 2.4, par. 2: Replaced OPG13a with OPG17c • Section 2.5.1: Temperature data updated for the most recent 5 year period of 2012-2016 • Section 2.5.2: Precipitation data updated for the most recent 5 year period of 2012-2016 • Section 2.5.3: Wind data updated for the most recent 5 year period of 2012-2016 • Section 2.5.4: Updated in accordance with CSA N288.2-14 • Section 2.5.5: Updated to reflect wind data for the most recent 5 year period of 2012-2016 • Section 2.5.6 (R005): Atmospheric Dilution Factor removed as dose calculations now follow the requirements given in CSA N288.2-14. • Section 2.6.5 moved to Section 2.8.1.3 since groundwater is not a subcomponent of the Aquatic Environment. For consistency with DWMF and WWMF Safety Reports, the section was moved to the geophysical environment. • Section 2.6.2: last paragraph deleted since it is dated and misleading. • Section 2.7: Section 2.7.1 and Section 2.7.2 merged and headings removed as a result of text updates. • Section 2.10: Section removed for consistency with the WWMF and DWMF Safety Reports. • Section 2.11 (R005): Title revised to “Indigenous Interests”.
		Chapter 3

		• Section 3.2: Added footnote to clarify that a minimum of 10 years of cooling can include residence time in fuel channels during GSS followed by IFB storage • Section 3.3.1: Added reference to OPG17a • Section 3.3.1.2 (R005): Content merged with Section 3.3.1 for consistency • Section 3.3.2.7: Updated the DSC lift height to be consistent with DWMF and WWMF • Section 3.4.3.1: Added safety related information for the storage buildings • Section 3.4.3.2: Added safety related information for the storage buildings • Section 3.4.4.3.5: Updated the DSC lift height to 1.5 m to be consistent with DWMF and WWMF • Section 3.4.5.3: Updated acronyms for NBCC and NFCC throughout. • Section 3.4.5.4: Added information on Class IV power supplying the public address system • Removed Figure 3-5 • Section 3.6.4: Added safety related information for the transfer clamp • Section 3.8, par. 4: Updated number of DSCs safety stored at the end of 2017
		Chapter 4 • Section 4.1.1: Added safety related information for the DSC • Section 4.3.1: Updated in accordance with N288.1-14 • Section 4.3.1: Added reference CSA14a • Section 4.3.2: Calculated dose rates from a single DSC have decreased from R005 due to a change in the analysis methodology, which previously used a point-kernel code (QAD-CGGP-A) for estimating direct contributions to the dose rate and the code Microskyshine to predict the air-scattered contribution. The calculations have been updated in R006 to use the code MCNP for all dose rate calculations. • Removed Figure 4-4 • Removed Figure 4-5 • Section 4.4.1, par. 4: Added reference CSA14b • Section 4.4.1, par. 4: Frequency of occurrence updated to 10^{-6} • Section 4.4.1: Added footnote on frequency of occurrence change for consistency with CSA N292.0-14 “General Principles for the Management of Radioactive Waste and Irradiated Fuel” • Section 4.4.1: Updated dose consequence for adult, finance and worker • Section 4.4.2: Updated to state that dose consequences to worker and public are bounded by DSC drop during on-site transfer • Section 4.5.1: Revised title to read “Radiation Protection Program” • Section 4.5.2: individual and collective dose updated to reflect 2007-2016.
		Chapter 5 • Safety assessment content moved to Section 6.0 and Appendix C. • Section 5.1.3.1: Added safety related information on DSMs
		Chapter 6 Updated safety assessment and accident analysis related to RCS from Chapter 5 to 6. Changes include: • CSA N288.1-14 and CSA N288.2-14 used in performing safety assessment • ADDAM code used to calculate potential dose to individual members of the public

		• IMPACT code used to analyze public dose for a chronic release of radioactivity from DSMs
		Chapter 7 • Section 7.1: Occupational Radiological Safety Management Program replaced with Radiation Protection Program throughout • Section 7.2: Heading title revised to reflect Radiation Protection Program • Section 7.2: ORSMP replaced with RP program throughout • Section 7.2.1: Content of section revised to reflect N-PROG-RA-0013 • Section 7.2.5: The routine radiological monitoring information was removed and is documented in 92896-INS-09071-00002 • Section 7.3.2: Section revised to add additional details on source of liquid effluent • Section 7.4: OPG03b replaced with OPG03a • Section 7.4.2.2: OPG03e replaced with OPG03d • Section 7.4.2.3: Section added for Derived Release Limits for the PWMF facility • Section 7.4.2.4: Section revised to include two subsections, 7.4.2.4.1 Radiation Protection Action Levels and 7.4.2.4.2 Environmental Action Levels • Section 7.5.4: NBC and NFC replaced with NBCC and NFCC • Section 7.6: Section removed for redundancy and is covered by Section 7.5.2
		Chapter 8 • Section 8.0: Revised section for consistency with DWMF and WWMF Safety Report
		Chapter 9 • Section 9.0: Minor text updates to reflect organizational changes • Section 9.2: Title revised to “First Nations and Métis Relations”
		Chapter 10 • Revised to reflect most recent PWMF PDP throughout • Section 10.1.1: Reference OPG12b removed and replaced with OPG17b
		Chapter 11 • Section 11.1.1.2: Dose data updated to reflect MCNP calculations • Section 11.1.1.2: Reference OPG13a removed and replaced with OPG17c • Section 11.1.1.2: Reference OPG03c changed to OPG03b • Section 11.1.2.2: The routine radiological monitoring information was removed and is documented in 92896-INS-09071-00002 • Section 11.2: Revised for consistency
		Chapter 12 • Dose rate data updated
		Chapter 13 • Reformatted references • Removed reference CSA91 and replaced with CSA14a • Added references CSA14b, CSA14c, CSA14d • Removed references AECL91, HC99, OH79, OH96, OPG01, OPG03a • Removed reference OPG12b and replaced with OPG17b • Replaced OPG13b with OPG13 • Added references OPG11a, OPG17a, OPG17b, OPG17c, OPG18

		Chapter 14 • Updated to reflect changes in the report text
		Chapter 15 • Updated NK30-D0A-10200-0001 R036
		Appendix B • Title revised to Out-of-Station Safety Assessment for Used Fuel Dry Storage • Section B.1.0: Reference B-CNSC00 replaced with B-CNSC17 • Section B.2.1: Added footnote to clarify that a minimum of 10 years of cooling can include residence time in fuel channels during GSS following by IFB storage • Section B.2.1.2: Reference B-OPG13 replaced with Reference B-OPG13b • Section B.2.1.2: Revised section to include justification for selection of reference burnup • Section B.2.2: Reference B-ORNL95 removed and replaced with B-ORNL11 • Section B.2.2: Table B-4 added to include light element radionuclide inventories from irradiated reference fuel bundle • Section B.2.3: Reference B-OPG03a removed • Section B.2.5: Updated description of sources of Carbon-14 in irradiated fuel bundle • Section B.2.7: Carbon-14 added to list of volatile radionuclides available for release • Section B.2.7: Updated Bq per bundle for tritium, krypton-85 and carbon-14 • Section B.2.8 (R005): Section removed since the methodology used in N288.2-14 does not use ADFs • Section B.2.8: Section numbering revised throughout to reflect removal of ADF section • Section B.2.8: Reference B-CSA14c and B-ICRP02 added and B-HC99 and B-ICRP95 removed • Section B.2.9: Reference B-ICRP12, B-ICRP95 added • Section B.2.9: Reference B-AECL83 and B-CSA91 removed • Section B.2.10.3: Storage module dimension changed to [redacted] and empty module mass changed to [redacted] • Section B.2.11: The normal operations safety assessment of SB3 was revised from the R005 revision. This section was updated to reflect the updated information, which includes revised ages for the DSCs and revised aisle spacing. Reference B-OPG03a was removed since it was not applicable to the revised safety assessment of the storage buildings. • Section B.3.0: References B-OPG00b, B-OPG03a, B-ORNL95, B-GAULD95, B-ORNL92a, B-ORNL97, B-ORNL92b and B-GROVE87 removed • Section B.3.0: References B-ORNL11, B-OPG18, B-LANL11 and B-OPG17a added • Section B.4.2: Event frequency changed to 10^{-6} to reflect N292.0-14, “General Principles for the Management of Radioactive Waste and Irradiated Fuel” • Section B.4.4: Dose consequence methodology updated to reflect N288.1-14 and N288.2-14 • Section B.4.4.1: Public dose consequence methodology updated to reflect N288.1-14 and N288.2-14 • Section B.4.4.1, par. 1: IMPACT code reference added (B-COG15) • Section B.4.4.1, par. 2: ADDAM code reference added (B-COG11)

Freedom of Information and Protection of Privacy Act (FIPPA) S. 18 and Access to Information and Privacy (ATIP) S.13.

		• Section B.4.4.1, par. 4: Dose consequence for adult and infant calculated for airborne emissions from PWMF following bounding scenario are listed • Section B.4.4.2: Content of this section has been revised to be consistent with the WWMF Safety Report (R004). The worker dose calculation has been updated to include revised release assumptions, updated skin absorption factor for tritium and updated dose conversion coefficients • Section B.6.1.1: Content revised to be consistent with DWMF and WWMF Safety Report • Section B.6.1.3: The 20 degree celsius thermal gradient during storage was revised to 54 degrees celsius to reflect a more recent structural integrity assessment that demonstrated that no through wall cracking would occur during the 50 years of planned storage. Reference B-OPG14a and B-OPG14b were added to the section • Section B.6.2.2: Updated to reflect bounding dose consequences • Section B.6.2.5.5: Flood information updated to include Review Level Conditions (RLC) for Probable Maximum Precipitation (PMP) at PWMF • Section B.6.2.5.6: Updated to reflect recent hazard event screening • Section B.6.2.5.8: Updated to reflect Phase I and Phase II • Section B.6.2.5.9: New section titled “Toxic Gas Release – Chlorine Originated from Ajax Water Treatment Plant” • Section B.6.2.5.10: New section titled “Soil Failures/Slope Instability” • Section B.7.1.2: Dose data updated for the most exposed age group and location • Section B.7.2.1: Failure probabilities revised and reference B-NRC03 added • Section B.7.2.2: Failure probabilities revised • Section B.7.2.7: Updated to reflect recent PWMF fire hazards assessment in reference B-OPG17c • Section B.7.2.8.1: Reference B-OPG13a added • Section B.7.2.8.6: Aircraft impact frequency updated • Section B.7.2.8.7: New sections titled “Release of Oxidizing, Toxic, Corrosive Gases and Liquids Stored in the Processing Building” • Section B.7.2.8.7: Reference B-CAN87 added • Section B.8.1.1.1: Dose rate information updated for Phase I used fuel storage buildings • Section B.8.1.1.2: Dose rate information updated for Phase II used fuel storage buildings • Section B.8.1.1.3: Dose rate information updated for Storage Building 2 and Storage Building 3 • Section B.8.2.1: Updated to reflect updated safety assessment • Section B.8.2.5: Updated to reflect updated fire hazards assessment • Section B.8.2.6.1: Reference B-OPG05 replaced with B-OPG11 • Section B.8.2.6.5: Updated to reflect updated safety assessment • Section B.8.2.6.6: New section added titled “Toxic Materials Stored in Storage Building 3” • Section B.8.2.6.7: Turbine missile strike content updated to reflect updated safety assessment • Section B.8.2.6.8: Updated to reflect updated safety assessment • Section B.10.0: References B-AECL83, B-CNSC00, B-CSA91, B-GAULD95, B-GOD82, B-GROVE87, B-NRC80, B-OH79, B-OH96, B-OPG00b, B-OPG03a, B-OPG03b, B-OPG05, B-ORNL92a, B-ORNL92b, B-ORNL95 and B-ORNL97 removed.

		• Section B.10.0: References B-CAN87, B-CNSC17, B-COG11, B-COG15, B-CSA14a, B-CSA14b, B-CSA14c, B-ICRP02, B-ICRP12, B-LANL11, B-NRC03, B-OPG11, B-OPG13a, B-OPG17a, B-OPG17b, B-OPG17c, B-ORNL11 and B-OPG19 added • Section B.11.0: Updated to reflect changes in the report text • Table B-1: Updated to reflect updated safety assessment • Table B-2: Updated to reflect updated safety assessment • Table B-3: Updated to reflect updated safety assessment • Table B-4: New table titled “Pickering Waste Management Facility Reference Fuel Bundle Light Elements Inventory (10-year-cooled fuel)” • Table B-5: Gamma spectrum updated to reflect updated safety assessment • Table B-5 (R005): Atmospheric Dilution Factors table removed to reflect updated N288.2-14 methodology • Table B-6: Updated to reflect updated safety assessment • Table B-7: Updated to reflect updated safety assessment • Table B-8: Updated to reflect updated safety assessment • Table B-9: Updated to reflect updated safety assessment • Figure B-8: Updated to reflect updated safety assessment • Figure B-9: Updated to reflect updated safety assessment • Figure B-10: Updated to reflect updated safety assessment
		Appendix C New appendix included for the safety assessment related to RCS. The following is captured in the appendix: • Source inventory for Retube components is moved from Section 5.0 (R005) to Appendix C (R006) • Detailed information on hazard event screening • Updated normal and accident dose consequences to reflect PWMF safety assessment update
		Appendix D Dry Storage Module Area Utilization Summary (Appendix C (R005)), moved to Appendix D to accommodate the RCS safety assessment.

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1.0 INTRODUCTION

1.1 Purpose

This report provides the information to support the Pickering Waste Management Facility (PWMF) Operating License, as required by the Nuclear Safety and Control Act (NSCA) and associated Regulations. The PWMF consists of the Used Fuel Dry Storage (UFDS) area for the storage of used fuel in Dry Storage Containers (DSCs), the Retube Component Storage (RCS) area, and land that has been reserved for future expansion.

1.2 Overview of the Pickering Waste Management Facility

Ontario Power Generation (OPG) currently operates the PWMF. The PWMF is composed of 2 sites. The PWMF Phase I site is located within the Pickering Nuclear Generating Station (NGS) protected area, southeast of Pickering NGS Unit 8, adjacent to the east side of the station security fence. The PWMF Phase II site is located approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”.

The PWMF Phase I site consists of the following sub-facilities: UFDS for interim storage of Pickering used fuel in DSCs; and RCS for interim storage of Pickering NGS A irradiated reactor components in Dry Storage Modules (DSMs).

The PWMF Phase II site contains a security kiosk, DSC Storage Building 3 and the site for additional DSC storage.

A photograph depicting the layout of the PWMF is shown in Figure 1-1.

1.2.1 Used Fuel Dry Storage at the Pickering Waste Management Facility

When fully constructed, the PWMF is expected to provide sufficient capacity to store the used fuel from the Pickering reactors until the end of the service life of the stations. UFDS at the PWMF has been planned in two Phases.

The PWMF Phase I site was constructed in two stages.

- Stage 1 became operational in January 1996 and contains a DSC processing building and DSC Storage Building 1. DSC Storage Building 1 has a design capacity of up to 185 DSCs.
- Stage 2 became operational in 2001 and consists of DSC Storage Building 2. DSC Storage Building 2 has a design capacity of up to 469 DSCs.

The PWMF Phase II has also been planned in two stages. Construction of DSC Storage Building 3 was completed in 2009. Construction of DSC Storage Building 4 will follow, depending on the expected rate of used fuel generation, as authorized for construction by the current licence. The site for the PWMF Phase II is within the Pickering NGS property site, approximately 500 m north-east of the existing PWMF Phase I facility, immediately east of the Hazardous Materials Building.

The Environmental Assessment (EA) for the PWMF DSC Storage Buildings 3 and 4 was completed and Construction Approval was given on December 23, 2004, by the Canadian Nuclear Safety Commission (CNSC).

As of December, 2017, 421 loaded DSCs have been placed in dry storage at the PWMF Phase I site and 480 loaded DSCs have been transferred to storage at the PWMF Phase II site. Thus, a total of 901 loaded DSCs have been placed in dry storage at the PWMF.

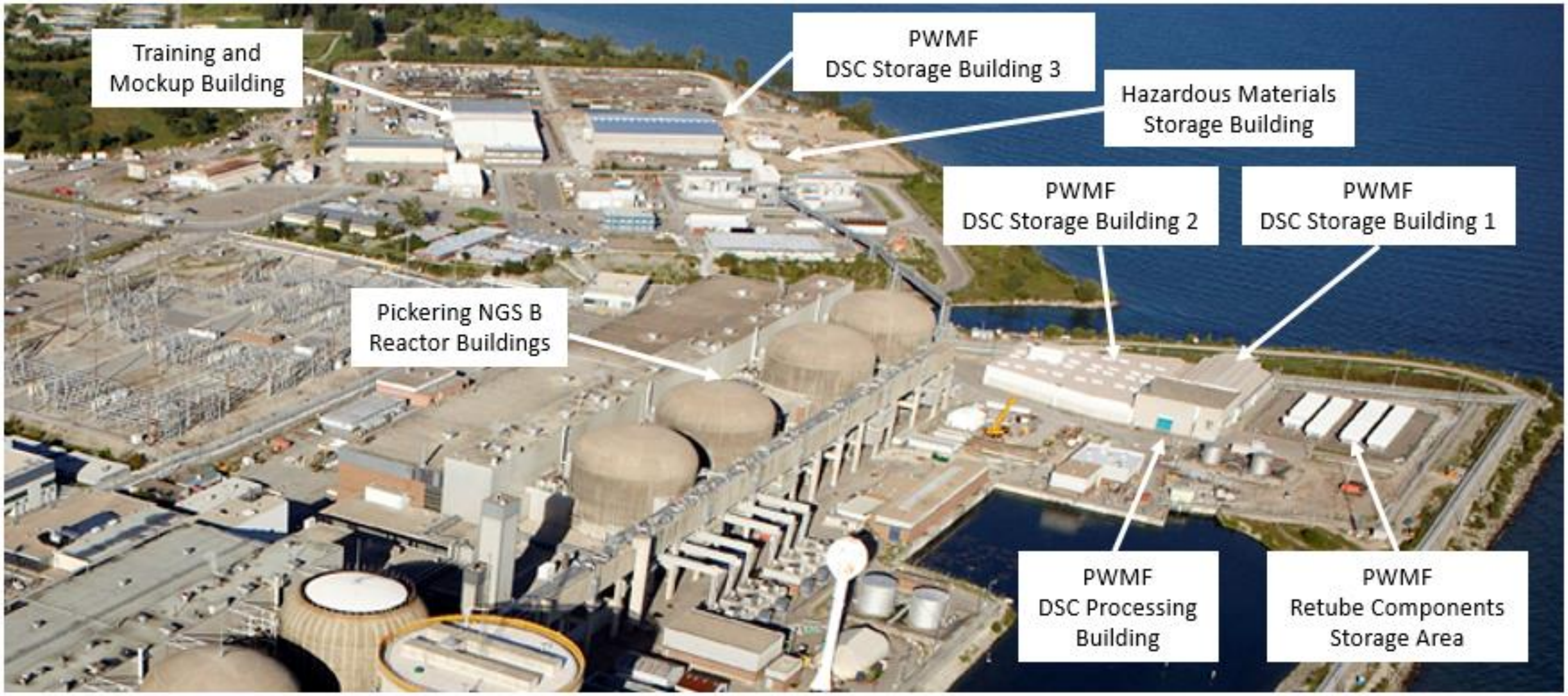

Figure 1-1: Pickering Waste Management Facility Layout and Surrounding Buildings

1.2.2 Dry Storage of Used Fuel

Since 1996, used fuel that has been cooled for at least 10 years in the Pickering Irradiated Fuel Bays (IFBs) has been routinely transferred into DSCs for dry storage. The DSC is a rectangular-section container made of a double carbon-steel shell filled with reinforced high density concrete. The DSC has been designed for above ground storage and transportability. Each DSC has a storage capacity of 384 used fuel bundles. The used fuel in the seal-welded DSCs is stored in an inert helium atmosphere. The design life of UFDS systems and components is at least 50 years.

Dry storage of used fuel, initially cooled in IFBs for several years after discharge from the reactor, is used worldwide as the preferred method for safe and economical storage of used fuel. In addition to UFDS at the PWMF, there are other dry storage facilities in operation in Canada. There are facilities located at Atomic Energy of Canada Limited's (AECL's) Douglas Point Generating Station on the Bruce Nuclear Power Development (BNPD) site, Hydro Quebec's Gentilly Nuclear Power Plant, and New Brunswick Power's Point Lepreau NGS. OPG has UFDS at its Western Waste Management Facility (WWMF) on the Bruce site receiving used fuel from Bruce Power since 2003. OPG's Darlington Waste Management Facility (DWMF) at the Darlington Nuclear Generating Station (DNGS) site became operational in early 2008. The purpose of the DWMF is to store DSCs with used fuel from the DNGS reactors.

1.2.3 Retube Components Storage at the Pickering Waste Management Facility

The RCS at the PWMF provides interim storage of irradiated reactor components in DSMs. These components were removed during the retube of the Pickering NGS A reactors in the period of 1984 to 1992. With the exception of periodic inspection, monitoring, and maintenance of DSMs and the RCS area, there have been no operational activities for RCS since 1993.

1.2.4 Storage of Retube Waste in Dry Storage Modules

DSMs contain irradiated fuel channel components from the Pickering NGS A retubing operation. The DSMs are stored outdoors in the RCS area, situated south of the PWMF Phase I UFDS buildings, as shown in Figure 1-1.

The irradiated components, consisting of pressure tubes, end fittings, shield plugs, and miscellaneous identified components, are stored in 34 DSMs. Two empty DSMs are stored in the RCS area for contingency purposes.

1.3 Scope of the Safety Report

The Safety Report describes the PWMF and its operations, including the on-site transfer of DSCs between the station IFBs and the PWMF, the processing of DSCs at the PWMF, DSC storage at the PWMF, DSC transfer between the PWMF Phase I and Phase II sites, and the storage of retube waste in DSMs. Within the scope of the safety assessment, the report discusses used fuel and retube waste characteristics, long-term performance of the DSCs and DSMs, and the potential public and occupational dose consequences and environmental effects of facility operations under normal, postulated abnormal and credible[1] accident conditions.

A site description is presented, encompassing the atmospheric, aquatic, terrestrial, geophysical and social environments, as well as the use of the land. The report gives an overview of safety and environmental monitoring programs and the nuclear management system applicable to the PWMF.

The Safety Report aspects of the loading, decontamination, and vacuum drying of DSCs at the station IFBs are covered in the Pickering NGS Safety Reports and the off-site transportation of DSCs will be covered in a Safety Report by the Nuclear Waste Management Organization (NWMO) in the future.

Public views regarding the storage of used fuel and components from retubing operations have been considered from the early stages of the PWMF. Public communication activities relevant to the PWMF are outlined in this report. The report includes an overview of the preliminary decommissioning planning for the PWMF.

1.4 Waste Management Strategy

1.4.1 Used Fuel Storage Strategy

OPG is committed to safe interim storage of its used nuclear fuel on its reactor sites until a long-term management approach becomes available. Alternative long-term management options for used fuel have been investigated by the NWMO in compliance with the federal Nuclear Fuel Waste Act. In 2007, the federal government accepted the recommendation of the NWMO for long term storage of used fuel.

The intended purpose of UFDS at the PWMF is to store used fuel from Pickering NGS reactors only. DSCs are capable of being transported off-site although this is not currently a planned or licensed activity.

1.4.2 Retube Waste Storage Strategy

The DSMs provide safe interim storage for the irradiated retube components.

There are no liquid emissions and no significant airborne emissions from the DSMs. OPG plans to safely store retube reactor components in the DSMs at the Pickering site until the Pickering NGS reactors are decommissioned; allowing the radioactivity levels to decay on site is considered a safe option for the interim management of this waste.

[1] An accident scenario is considered “credible” if its probability of occurrence is deemed to be 10^{-6} per year or higher.

1.5 Performance Objectives

Objectives of the PWMF are:

- To provide safe interim storage of the used fuel from Pickering NGS reactors in DSCs until all the used fuel is transported to an alternative long-term used fuel storage or disposal facility; and

- To provide safe storage of the retube reactor components for Pickering NGS A in DSMs until they are transported to a disposal facility.

OPG is committed to managing radioactive waste in an environmentally, socially, and financially responsible way, to ensure the protection of workers, the public, and the environment, and to ensure full compliance with regulatory and licensing requirements.

Radioactive waste and used fuel stored at the PWMF shall be retrievable.

1.6 Performance Criteria

During normal PWMF operations and under abnormal and credible accident conditions, the dose consequences to the public and the workers will be within the criteria described in the following sections.

1.6.1 Normal Operations

General radiological protection requirements are established in OPG's governing documents on Radiation Protection (RP). Any radiation dose resulting from PWMF operations will be within the regulatory dose limits and kept As Low As Reasonably Achievable (ALARA). The CNSC regulatory dose limits for the public and Nuclear Energy Workers (NEWs) are shown in Table 1-1.

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

Table 1-1: Canadian Nuclear Safety Commission Effective Dose Limits

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

The dose/dose rate targets for PWMF operations, derived from Table 1-1 for a member of the general public, are as follows:

(a) ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.

(b) ≤ 100 μSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.

The administrative dose target is an internal target used to ensure that the facility is designed to protect the public during normal operations. The OPG target is applied at the site boundary, which is located closest to the PWMF on the east side as shown in Figure 2-1. This location is a transient walking trail and is not occupied by the representative person of the most exposed public population. There is a fence that runs along the site boundary and overlaps with the PNGS exclusion zone on the east side of the property.

With the future expansion of the PWMF Phase II site to include SB4/5/6, construction of used fuel dry storage buildings will be closer to the site boundary. The previous administrative target of 10 μSv/year was implemented early in the PWMF Phase I development and is now overly restrictive for expansion activities. An administrative dose target of 100 μSv/year has been adopted and will only be applied during the design of future Storage Buildings.

Future used fuel storage buildings would be over-designed if the previous 10 μSv/year dose target was kept. Furthermore, this would not result in a substantial dose reduction at the site boundary for an actual public population.

Doses to the public from OPG nuclear waste facilities will continue to be well below the CNSC annual regulatory dose limit of 1000 μSv/year.

1.6.2 Potential Abnormal Operating Conditions and Credible Accident Conditions

The radiological doses from radionuclide releases and direct radiation, either to members of the public at the Pickering NGS site boundary or to workers, following abnormal operating conditions or a credible accident are not expected to exceed the annual dose limits given in Table 1-1.

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2.0 SITE DESCRIPTION

2.1 Site Location

The PWMF Phase I site is located within the Pickering NGS protected area at the southeast corner of the Pickering site, as shown in Figure 2-1. Figure 2-1 also indicates the location of the PWMF Phase II site containing DSC Storage Building 3 and the site for DSC Storage Building 4 (shown as PWMF II), approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”.

Pickering NGS is located in the City of Pickering within the Regional Municipality of Durham. The Pickering NGS property is on the north shore of Lake Ontario, 32 km east-northeast of downtown Toronto and 21 km southwest of the City of Oshawa. The two major watercourses closest to the site are Duffins Creek, 2.2 km to the east and the Rouge River, 4 km to the west.

The Pickering NGS property is approximately 240 ha in size with a continuous landscaped buffer paralleling all adjacent municipal roads. The property is fenced and access is restricted and controlled by OPG. There is a 914 m exclusion zone around Pickering NGS. This exclusion zone restricts the type of land uses that can occur within its confines. The exclusion zone is predominately owned by OPG. These lands are primarily used for industrial purposes related to power generation. Two public outdoor recreation parks, Alex Robertson Park and Kinsmen Park, are located approximately 600 m northwest of Pickering NGS A, on lands leased to the City of Pickering and outside the Pickering NGS site boundary fence.

2.2 Site Accessibility

The Pickering site is well serviced by road. Two major highways, Highway 401 and Highway 2, and the main Canadian National (CN) rail line run in the east-west direction, at a closest distance of 2.8 km to the site. The site is accessible from either highway via Brock Road.

Vehicular access to the PWMF is through the north side paved road in front of the Pickering NGS site, leading to the facility through an existing road.

Pedestrian access to the PWMF Phase I site is via a paved road on the east side of Pickering NGS.

2.3 On-Site Facilities

Emergency, medical aid and fire prevention facilities for the PWMF Phase I site are provided by Pickering NGS. A 12 inch (30 cm) buried ring header loops around the reactor buildings and powerhouse, providing water for fire protection purposes.

Emergency, medical aid and fire prevention facilities for the PWMF Phase II site are provided by Pickering Fire Services and Durham Emergency Medical Services with backup support provided by the Pickering NGS Emergency Response Team (ERT).

OPG maintains its own full-time fire crews. It also has a standing agreement with the City of Pickering's fire department. Together, these ensure a quick response to all emergencies at the Pickering site.

OPG has developed and maintains a comprehensive on-site and off-site emergency response plan, referred to as the Consolidated Nuclear Emergency Plan (CNEP). Response teams are trained and equipped to respond to fires, as well as emergencies involving the non-routine release of radioactivity.

The CNEP is a consolidated plan that applies to both the Pickering and Darlington nuclear sites.

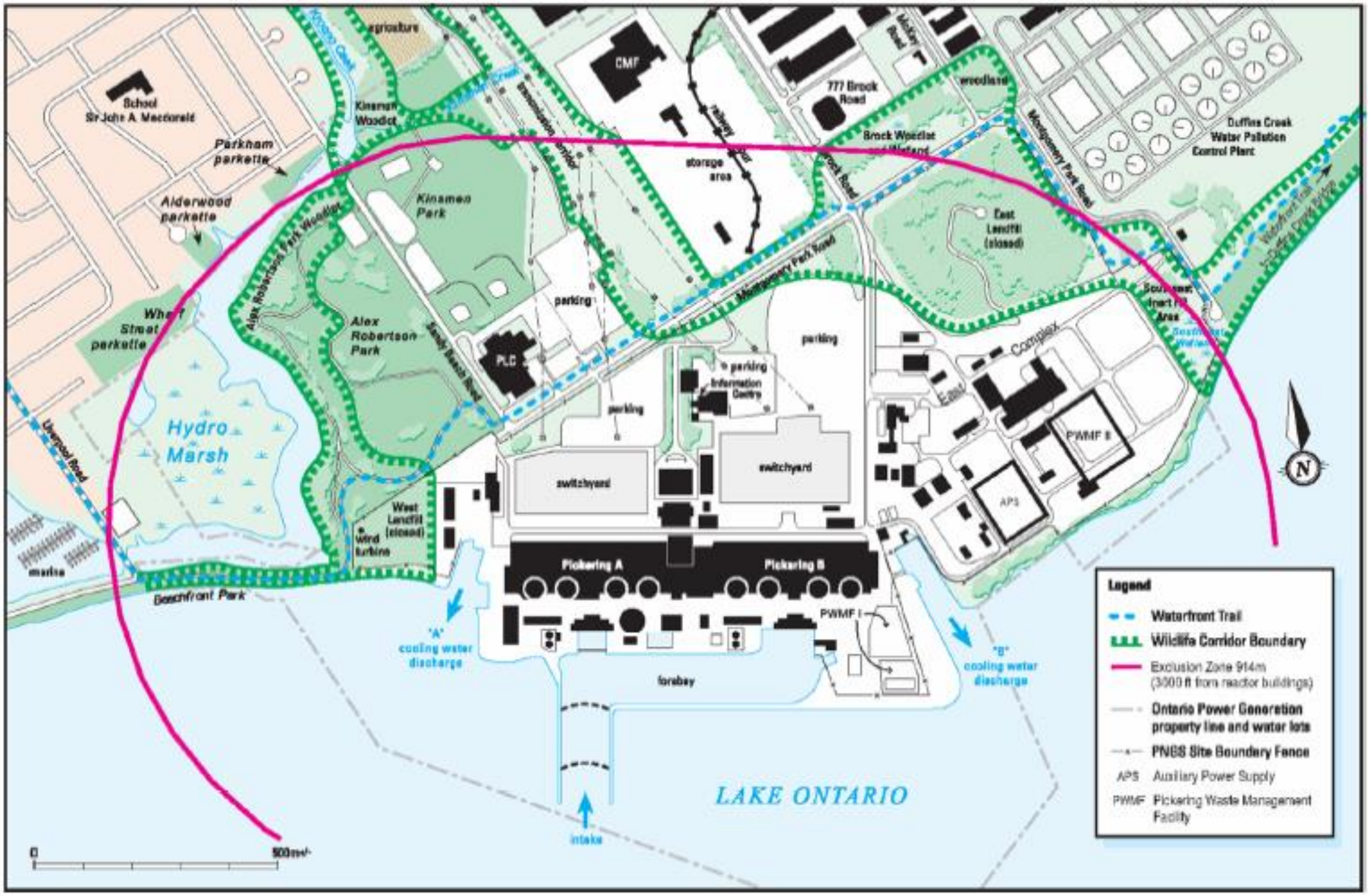

Figure 2-1: Pickering Nuclear Generating Station Site Layout Showing the Pickering Waste Management Facility Sites for Phase I and Phase II Relative to the Pickering Nuclear Generating Station and the Exclusion Zone Boundary

2.4 Environmental Studies

OPG has been conducting environmental studies at the Pickering NGS site since 1969. Studies have focused on environmental effects of station operations, routine radiological environmental monitoring, and EAs.

Environmental monitoring program reports that include Pickering NGS and the PWMF are prepared annually (OPG17b). Environmental radiation monitoring at the PWMF is also conducted and the results are reported to the CNSC on a quarterly basis.

The Environmental Risk Assessment Report for Pickering Nuclear (OPG18) evaluated the risk to relevant human and ecological receptors resulting from exposure to contaminants and physical stressors related to the Pickering Nuclear site and its operations. The report also recommended potential monitoring or assessment as needed based on the results of the Environmental Risk Assessment. Overall, the report concluded that the Pickering Nuclear site is operating in a manner that is protective of human health and ecological receptors residing in the surrounding area.

The Pickering NGS B Refurbishment EA Report (OPG07a) summarizes much of the historical information, and results of an extensive study of the site environment at local and regional levels. Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report (OPG03b) identified no significant residual adverse environmental effects of the PWMF Phase II project, which involved the site preparation, construction, operation and maintenance of two additional storage buildings at the PWMF site. Screening EA (OH98a) prepared for the Stage II expansion document environmental information for the site.

2.5 Atmospheric Environment

The meteorology in the vicinity of Pickering NGS is affected by meso-scale/synoptic factors consisting of the general circulation of air masses and the effects of the Great Lakes, and micro-scale factors that include off-shore/on-shore winds (due to temperature difference between land and lake surfaces), terrain and topography. OPG has been gathering on-site meteorological data at Pickering NGS since the 1970s.

2.5.1 Temperature

The mean temperature measured at the Pickering NGS site's meteorological tower is presented in Table 2-1. Mean daily temperatures fall below freezing in January and February. The coldest recorded daily temperature measurement was -25.6°C, and the warmest recorded daily temperature measurement was 35.8°C during this period.

Table 2-1: Mean Temperature Information for the Pickering Nuclear Generating Station Site[2]

Month	Based on measurements taken between the years 2012 and 2016		
	Maximum Daily (°C)	**Minimum Daily (°C)**	**Mean Daily (°C)**
January	11.1	-19.6	-3.9
February	6.9	-19.8	-4.9
March	14.4	-14.4	0.5
April	19.6	-4.8	5.8
May	26.9	2.3	13.7
June	34.5	8.6	18.5
July	36.7	12.1	21.9
August	33.0	9.9	21.0
September	27.3	5.6	17.1
October	20.1	-2.6	10.8
November	14.1	-8.3	4.4
December	10.0	-11.3	0.3
Annual Average	**21.2**	**-3.5**	**8.8**

2.5.2 Precipitation

The Pickering site weather tower does not record the precipitation amount and fog occurrence. The Frenchman's Bay Meteorological Station is the closest weather station to the Pickering NGS site and represents the best record of precipitation at the Pickering NGS between the years 1971 and 2000 (EC06). Prior to 2004, precipitation data was taken from the Frenchman's Bay Meteorological Station. Post 2004, precipitation data was taken from Oshawa, which is 21 km from the Pickering site. The precipitation data is presented in Table 2-2.

Measurable precipitation occurred an average of approximately 112 days per year. Precipitation is quite consistent throughout the year with slightly more precipitation in the second half of the year. The annual precipitation is approximately 684 mm.

2 Measurements were taken at 10 m elevation at the Pickering NGS site Meteorological Tower. Minimum and maximum temperatures were measured on the coldest and warmest days of that month.

Table 2-2: Precipitation Normals at Frenchman's Bay (1971-2000) and Oshawa (2012-2016)

Month	Frenchman's Bay (1971-2000)		Oshawa (2012-2016)	
	Precipitation (mm), monthly average	Precipitation (mm), daily maximum	Precipitation (mm), monthly average	Precipitation (mm), daily maximum
January	62.7	45.6	44.3	26.8
February	48.7	39.3	40.7	19.3
March	67.0	48.5	37.6	27.7
April	76.0	40.8	69.2	35.8
May	80.3	61.4	38.5	32.8
June	76.9	74.7	91.6	43.8
July	73.2	84.8	61.1	29.3
August	82.7	71.6	64.5	28.4
September	83.6	71.4	77.3	45.0
October	70.8	61.6	88.9	58.8
November	81.8	48.0	24.9	17.0
December	75.4	42.7	44.8	17.6
Annual	**879**		**684**	

2.5.3 Wind

Table 2-3 contains the wind direction and wind speed joint frequencies measured at the 10 m elevation Pickering NGS Meteorological Tower. The data was collected between the years 2012 and 2016, and was measured at a height of 10 m. Wind speeds are grouped into sectors according to compass direction (the direction from which the wind is blowing). Figure 2-2 presents wind roses derived from this data. From this figure, the prevailing winds in the vicinity of the Pickering NGS are shown to occur most commonly (greater than 35 percent of the time) from the northwesterly quarter; winds from the southwest and northeast occur 20 percent to 32 percent of the time, while winds from the southeast are least frequent. Between the years 2012 and 2016, the average measured wind speed at the Meteorological Tower was approximately 2.6 m/s (9.4 km/h). Periods with little or no air movement, or "calms" (<1 km/h), were reported 8 percent of the time.

The meteorology in the vicinity of the plant is affected by the proximity of the station to the lake. This is commonly referred to as the lake effect, which causes lake breezes. Lake breezes result from temperature differences between land and water. An important characteristic of lake breezes is the formation of the Thermal Inversion Boundary Layer (TIBL).

In the spring and summer, when the skies are clear and the geostrophic winds are light, a strong temperature gradient exists between the air over the land and the air over Lake Ontario. This gradient begins forming in the morning as the land is heated at a higher rate relative to the water due to solar radiation. As a result the warm air over land rises and is replaced by the cooler lake air, thus producing a lake breeze. At night this situation is reversed, resulting in a land breeze. Lake breezes are usually stronger than land breezes. In the fall and winter, the lake is generally warmer than the land, resulting in more frequent land breezes.

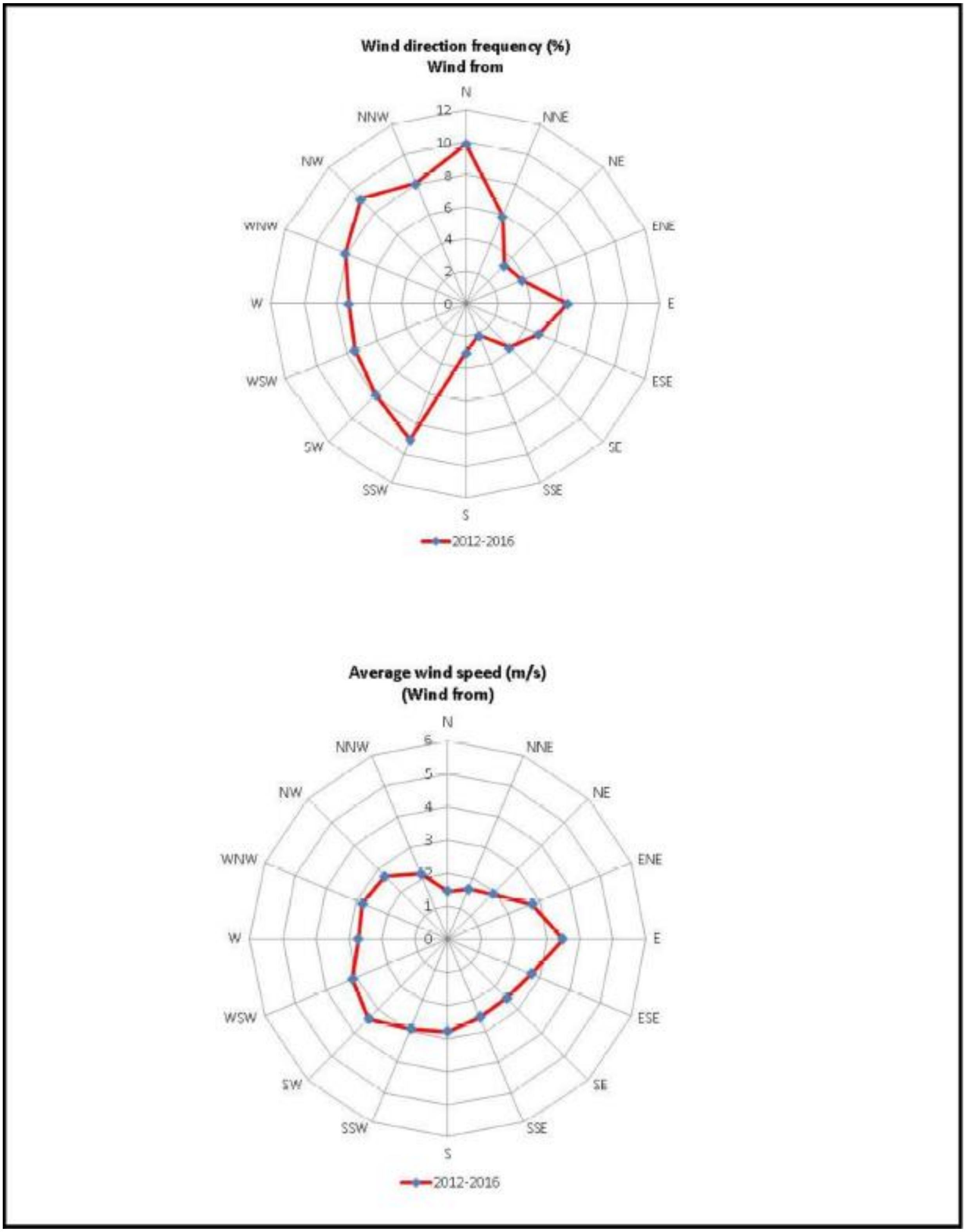

Figure 2-2: Wind Rose for the Pickering Site 10 m Tower

Table 2-3: Wind Direction and Wind Speed Joint Frequencies (%) between the years 2012 and 2016 at the 10 m elevation Pickering Meteorological Tower

Direction Wind Blowing		Wind Speed (m/sec)						
From	To	<2	2-3	3-4	4-5	5-6	>6	Total
N	S	7.87	0.91	0.60	0.30	0.13	0.08	9.88
NNE	SSW	4.21	0.74	0.61	0.21	0.09	0.04	5.90
NE	SW	1.89	0.77	0.43	0.16	0.09	0.01	3.34
ENE	WSW	1.29	0.76	0.88	0.47	0.26	0.11	3.77
E	W	1.44	0.95	1.39	1.28	0.64	0.57	6.27
ESE	WNW	1.59	1.18	1.07	0.63	0.23	0.15	4.86
SE	NW	1.22	1.28	0.81	0.33	0.13	0.03	3.80
SSE	NNW	0.73	0.68	0.39	0.17	0.09	0.06	2.11
S	N	0.90	0.88	0.64	0.30	0.16	0.13	3.01
SSW	NNE	2.03	2.58	2.74	1.07	0.37	0.26	9.05
SW	NE	2.13	1.51	1.44	1.08	0.71	1.04	7.90
WSW	ENE	2.04	1.60	1.53	1.08	0.72	0.47	7.45
W	E	2.64	1.58	1.40	0.92	0.51	0.22	7.27
WNW	ESE	2.86	1.69	1.49	1.16	0.54	0.36	8.10
NW	SE	3.53	1.81	1.73	1.29	0.54	0.33	9.22
NNW	SSE	4.44	1.36	1.10	0.73	0.33	0.13	8.09
Total		**40.80**	**20.28**	**18.27**	**11.19**	**5.53**	**3.96**	**100.00**

In warm seasons, due to solar heating, the air over the land is often 10°C (or more) warmer than the air over the water. When cold, stable lake air flows over warmer land, the resulting upward heat flux gives rise to a TIBL. This TIBL grows in depth with distance inland as the stable air is advected over land and adjusts to changes in surface roughness, heat and moisture input. The depth of the TIBL is typically hundreds of metres and extends on the order of 10 km inland before a new equilibrium is reached.

For emission sources near the ground, pollutants emitted into the unstable boundary layer would result in higher than expected ground level concentrations during on-shore flows with a TIBL because the stable layer aloft would limit vertical diffusion.

2.5.4 Atmospheric Stability

Atmospheric stability is a measure of atmospheric turbulence. The turbulent nature of the atmosphere strongly affects the concentration of contaminants downwind of the release point. A highly turbulent atmosphere is referred to as Stability Class A and occurs under warm sunny conditions. A highly stable atmosphere is referred to as Stability Class F and occurs typically at nighttime and under low wind speed conditions. A neutral atmosphere, referred to as Stability Class D, is representative of average turbulence conditions and occurs typically under cloudy, windy conditions. All other things being equal for ground level releases, downwind contaminant concentrations are highest when the atmosphere is highly stable (F stability) and lowest when the atmosphere is highly unstable (A stability).

Using the modified sigma theta with daytime and nighttime correction method given in Canadian Standards Associated (CSA) N288.2-14 (CSA14d), meteorological data collected at the 10 m-elevation Pickering NGS Meteorological Tower between the years 2012 and 2016 produces the atmospheric stability distribution shown in Table 2-4.

Table 2-4: Frequency of Occurrence (%) of Atmospheric Stability Class (2012-2016)

A	B	C	D	E	F
3.12	6.46	17.39	43.13	16.61	13.30

2.5.5 Joint Frequency of Occurrence

The meteorological data used to predict stability class are measured at the 10 m elevation Pickering NGS Meteorological Tower, at a height of 10 m.

2.5.6 Severe Weather

Severe weather events in the region generally include thunderstorms and lightning, ice storms, wind storms, heavy precipitation, and fog. Thunderstorms require low-level, warm, moist air which, when lifted, will release sufficient latent heat to provide the buoyancy necessary to maintain its upward movement in an extremely unstable atmosphere. Thunderstorms produce lightning and, on occasion, tornadoes. In Southern Ontario locations, thunderstorms normally occur 28 to 30 days a year.

Ice storms, including freezing rain and ice pellets, are associated with atmospheric conditions that are characterized by an elevated inversion having a maximum temperature above 0°C, overriding lower subfreezing. Freezing rain occurs in Southern Ontario, on average, 12 to 17 days per year. Ice storms are usually accompanied or followed by precipitation such as snow, wet snow, rain or fog.

Although infrequent, the most common high windstorm is a tornado. Tornadoes are caused by excessive instability and steep lapse rates in the atmosphere. Tornadoes most often occur along squall lines of a tropical cyclone (low pressure centre), in conjunction with cumulonimbus clouds and severe thunderstorms.

In Southern Ontario, a few tornadoes or near-tornadoes are reported every year, but these storms are not as intense, nor do they cause as much damage, as those in the United States south and west of the Great Lakes. The average tornado in Southern Ontario has a diameter of between 150 and 600 m, and typically travels at a speed of 50 to 70 km/h in a southwest to northeast direction. Tornadoes normally touch ground for less than 20 minutes. In the Pickering area, a maximum of one tornado per 10,000 km^2 can be expected annually (OPG11b).

Table 2-5: 10 m-elevation Pickering Nuclear Generating Station Meteorological Tower: Wind Speed and Wind Direction Frequencies by Stability Class (2012-2016)

Wind Blowing From	Percentage (%) Wind Speed Category (m/s)							Percentage (%) Wind Speed Category (m/s)						
	<2	2-3	3-4	4-5	5-6	>6	Total	<2	2-3	3-4	4-5	5-6	>6	Total
	Stability Class: A							Stability Class: B						
N	0.22	0.11	0.00	0.00	0.00	0.00	0.33	0.13	0.11	0.15	0.00	0.00	0.00	0.39
NNE	0.15	0.04	0.00	0.00	0.00	0.00	0.19	0.10	0.05	0.09	0.00	0.00	0.00	0.24
NE	0.10	0.02	0.00	0.00	0.00	0.00	0.12	0.06	0.03	0.01	0.00	0.00	0.00	0.10
ENE	0.09	0.02	0.00	0.00	0.00	0.00	0.11	0.04	0.04	0.02	0.00	0.00	0.00	0.10
E	0.08	0.02	0.00	0.00	0.00	0.00	0.10	0.08	0.03	0.03	0.00	0.00	0.00	0.14
ESE	0.08	0.02	0.00	0.00	0.00	0.00	0.10	0.12	0.09	0.04	0.00	0.00	0.00	0.25
SE	0.12	0.03	0.00	0.00	0.00	0.00	0.15	0.19	0.13	0.04	0.00	0.00	0.00	0.36
SSE	0.10	0.02	0.00	0.00	0.00	0.00	0.12	0.11	0.13	0.04	0.00	0.00	0.00	0.28
S	0.14	0.04	0.00	0.00	0.00	0.00	0.18	0.11	0.14	0.04	0.00	0.00	0.00	0.29
SSW	0.15	0.04	0.00	0.00	0.00	0.00	0.19	0.18	0.18	0.15	0.00	0.00	0.00	0.51
SW	0.16	0.06	0.00	0.00	0.00	0.00	0.22	0.12	0.12	0.11	0.00	0.00	0.00	0.35
WSW	0.09	0.07	0.00	0.00	0.00	0.00	0.16	0.14	0.12	0.16	0.00	0.00	0.00	0.42
W	0.12	0.08	0.00	0.00	0.00	0.00	0.20	0.14	0.20	0.32	0.00	0.00	0.00	0.66
WNW	0.15	0.11	0.00	0.00	0.00	0.00	0.26	0.12	0.18	0.32	0.00	0.00	0.00	0.62
NW	0.17	0.14	0.00	0.00	0.00	0.00	0.31	0.12	0.25	0.63	0.00	0.00	0.00	1.00
NNW	0.19	0.19	0.00	0.00	0.00	0.00	0.38	0.12	0.18	0.43	0.00	0.00	0.00	0.73
Total	**2.11**	**1.01**	**0.00**	**0.00**	**0.00**	**0.00**	**3.12**	**1.88**	**1.98**	**2.58**	**0.00**	**0.00**	**0.00**	**6.44**
	Stability Class: C							Stability Class: D						
N	0.18	0.07	0.10	0.16	0.09	0.00	0.60	2.55	0.23	0.33	0.13	0.04	0.08	3.36
NNE	0.12	0.09	0.15	0.12	0.03	0.00	0.51	1.51	0.43	0.37	0.09	0.06	0.04	2.50
NE	0.08	0.08	0.07	0.04	0.02	0.00	0.29	0.58	0.54	0.35	0.12	0.07	0.01	1.67
ENE	0.09	0.15	0.20	0.11	0.09	0.00	0.64	0.23	0.40	0.66	0.36	0.16	0.11	1.92
E	0.18	0.22	0.44	0.41	0.13	0.00	1.38	0.31	0.52	0.92	0.87	0.51	0.57	3.70
ESE	0.25	0.41	0.43	0.20	0.03	0.00	1.32	0.38	0.55	0.60	0.43	0.20	0.15	2.31
SE	0.26	0.57	0.39	0.10	0.03	0.00	1.35	0.20	0.46	0.38	0.23	0.10	0.03	1.40
SSE	0.11	0.31	0.13	0.02	0.00	0.00	0.57	0.08	0.15	0.22	0.15	0.09	0.06	0.75
S	0.15	0.35	0.27	0.07	0.02	0.00	0.86	0.11	0.28	0.33	0.24	0.13	0.13	1.22
SSW	0.36	0.79	0.79	0.24	0.03	0.00	2.21	0.51	1.30	1.80	0.82	0.34	0.26	5.03
SW	0.22	0.27	0.34	0.17	0.11	0.00	1.11	0.53	0.81	0.98	0.90	0.59	1.04	4.85
WSW	0.11	0.25	0.35	0.36	0.23	0.00	1.30	0.24	0.72	1.01	0.73	0.49	0.47	3.66
W	0.10	0.19	0.29	0.44	0.26	0.00	1.28	0.27	0.60	0.78	0.48	0.25	0.22	2.60
WNW	0.10	0.20	0.32	0.63	0.30	0.00	1.55	0.44	0.72	0.85	0.53	0.24	0.36	3.14
NW	0.09	0.11	0.21	0.75	0.36	0.00	1.52	0.37	0.64	0.89	0.53	0.18	0.33	2.94
NNW	0.07	0.06	0.09	0.47	0.20	0.00	0.89	0.66	0.39	0.55	0.26	0.13	0.13	2.12
Total	**2.47**	**4.12**	**4.57**	**4.29**	**1.93**	**0.00**	**17.38**	**8.97**	**8.74**	**11.02**	**6.87**	**3.58**	**3.99**	**43.17**
	Stability Class: E							Stability Class: F						
N	2.61	0.24	0.02	0.00	0.00	0.00	2.87	2.18	0.15	0.00	0.00	0.00	0.00	2.33
NNE	1.30	0.11	0.00	0.00	0.00	0.00	1.41	1.02	0.02	0.00	0.00	0.00	0.00	1.04
NE	0.53	0.08	0.00	0.00	0.00	0.00	0.61	0.54	0.01	0.00	0.00	0.00	0.00	0.55
ENE	0.39	0.13	0.01	0.00	0.00	0.00	0.53	0.46	0.02	0.00	0.00	0.00	0.00	0.48
E	0.35	0.13	0.00	0.00	0.00	0.00	0.48	0.43	0.03	0.00	0.00	0.00	0.00	0.46
ESE	0.35	0.10	0.00	0.00	0.00	0.00	0.45	0.41	0.03	0.00	0.00	0.00	0.00	0.44
SE	0.19	0.07	0.00	0.00	0.00	0.00	0.26	0.27	0.02	0.00	0.00	0.00	0.00	0.29
SSE	0.11	0.05	0.00	0.00	0.00	0.00	0.16	0.22	0.02	0.00	0.00	0.00	0.00	0.24
S	0.16	0.06	0.00	0.00	0.00	0.00	0.22	0.22	0.02	0.00	0.00	0.00	0.00	0.24
SSW	0.45	0.19	0.01	0.01	0.00	0.00	0.66	0.38	0.08	0.00	0.00	0.00	0.00	0.46
SW	0.57	0.20	0.01	0.00	0.00	0.00	0.78	0.54	0.04	0.00	0.00	0.00	0.00	0.58
WSW	0.81	0.37	0.01	0.00	0.00	0.00	1.19	0.66	0.07	0.00	0.00	0.00	0.00	0.73
W	1.12	0.44	0.00	0.00	0.00	0.00	1.56	0.90	0.07	0.00	0.00	0.00	0.00	0.97
WNW	1.09	0.41	0.00	0.00	0.00	0.00	1.50	0.95	0.08	0.00	0.00	0.00	0.00	1.03
NW	1.36	0.51	0.01	0.00	0.00	0.00	1.88	1.43	0.16	0.00	0.00	0.00	0.00	1.59
NNW	1.69	0.35	0.02	0.00	0.00	0.00	2.06	1.71	0.19	0.00	0.00	0.00	0.00	1.90
Total	**13.08**	**3.44**	**0.09**	**0.01**	**0.00**	**0.00**	**16.62**	**12.32**	**1.01**	**0.00**	**0.00**	**0.00**	**0.00**	**13.33**

2.5.7 Non-Radiological Air Quality

The air quality in the vicinity of the Pickering NGS site is typical of the general air quality in Southern Ontario within the Quebec-Windsor corridor and the Greater Toronto Area (GTA). Around the Pickering area, the Ministry of the Environment and Climate Change (MOECC) operates a few ambient air quality monitoring stations to measure carbon monoxide (CO), sulphur dioxide (SO_2), nitrogen oxides (NO, NO_2, and NO_x, which is the measure of all nitrogen oxides combined), volatile organic compounds (VOCs), and particulate matter (PM) – PM measurements include total suspended particulates (TSP) and all “respirable” particulates (those measuring less than 2.5 microns in diameter, referred to as $PM_{2.5}$).

Examining the MOECC air quality data from the years 2000 to 2004, air quality in the vicinity of the Pickering NGS site was found to be most influenced by NO_2, TSP, and $PM_{2.5}$.

From atmospheric dispersion modeling (OPG07a), air quality at the Pickering NGS was found to be most strongly influenced by local road traffic, Brock Road, Montgomery Park Road, and Liverpool Road. Only a small fraction is influenced by existing emissions from the Pickering site, namely from combustion equipment, on-site vehicular movement, and boiler chemical venting.

The same modeling concluded that predicted concentrations of monitored air quality factors are below applicable air quality criteria, even with background contributions taken into account.

2.5.8 Noise

The noise environment in the vicinity of the Pickering NGS site is typical of an urban setting, being dominated by traffic on local roads such as Brock Road, Montgomery Park Road, and Highway 401 to the distant north. There is noise from a nearby water pollution control plant, Pickering NGS, and other industrial and commercial establishments. Lake Ontario shore winds and waves can also be heard.

Background sound level measurements were recorded in 2011 as part of the Acoustic Assessment Report (OPG11a) in support of the Environmental Complication Approval (ECA) application process for the entire Pickering NGS site. The sound levels recorded at the closest Points of Reception (POR) (northwest residential neighbourhood (Bay Ridges Neighbourhood) and the Duffin Creek Water Pollution Control Plant adjacent to the east property line) were below limits for all times of the day, and PNGS was found in compliance with applicable MOECC noise limits.

The range and distribution of sound levels within the Bay Ridges Neighbourhood, one of the closest residential noise receptors to the northwest of the Pickering site, indicated that the sound environment was characteristic of a “Class 1 Area”, defined by the Ontario Municipal Noise Control by-law NPC-205. This designation generally pertains to an area where the background sound level is dominated by the urban hum, which is mainly traffic noise.

2.6 Aquatic Environment

2.6.1 Drainage

Drainage in the Pickering NGS site is a mix of ephemeral swales, ditches, culverts, and storm sewers – no major watercourses traverse the site. The discharge points are approximately 6 m to 10 m above the Lake Ontario water level. No water body other than a small (0.5 ha) isolated wetland, known as the Southeast Wetland, is located within the Pickering site. This small isolated wetland, which was once farmland and created during the construction of Pickering NGS as a result of landfilling activities, lies in the southeast corner of the Pickering NGS property at the foot of Montgomery Park Road. This wetland receives drainage from the area around the former construction landfill within the Pickering site.

Surface drainage from the Pickering NGS site is enabled by 19 separate storm water drainage basins, or catchments. The PWMF Phase I site is part of a smaller drainage basin with an area of 3.7 ha. The PWMF Phase I site shares this area with the Pickering NGS B standby generators. Greater than 95 percent of this area is considered to be impervious. Runoff, including the PWMF Phase I roof drainage, is directed through the Pickering NGS drainage network into the Pickering NGS B discharge channel. Drainage from the RCS area is directed via catch basins to the Pickering NGS drainage system for discharge to the outfall.

Even though liquid effluents generated inside the PWMF Phase I site are infrequent and small in volume, they are sampled and pumped into the Pickering NGS Active Liquid Waste Management System.

Surface drainage from the PWMF Phase II site in the East Complex area drains to Lake Ontario, with drainage areas of between 0.4 ha and 19.8 ha. Catchments numbered 12 through 16 are affected by the PWMF Phase II site and associated stormwater management.

2.6.2 Fish

More than 90 fish species are known to inhabit Lake Ontario, of which approximately 60 species have been found in impingement, gillnetting and electrofishing studies at the Pickering NGS site and are considered to represent the fish community in its vicinity. As there are no surface water features suitable as fish habitat within the site, the only aquatic habitat and biota are located within Lake Ontario. However, almost all of these species make use of the nearshore waters of the lake for one or more of spawning, rearing, feeding, migration and over-wintering. These nearshore waters include the Rouge River, Frenchman’s Bay, and Duffins Creek; the Pickering B discharge channel is used by smallmouth bass as a spawning area.

Sport fish taken near the Pickering NGS include brown trout, walleye, chinook and coho salmon, rainbow trout, smallmouth bass, white bass and carp. Commercial fishing is not a significant industry in the area.

2.6.3 Lake Water Levels

Using the Marine Environment Data Source, a review of historic water levels at Toronto and Cobourg was undertaken as part of the Pickering A Return to Service EA (OPG00a). Based on the monthly average water levels of Lake Ontario from 1918 to 1998, the annual maximum daily average water levels at Toronto for the period 1908 to 1998 ranged from a low of 74.26 m International Great Lakes Datum (IGLD) (1935) to a high of 75.81 m IGLD (1952). At Cobourg for the period 1956 to 1998, the maximum value ranged from a low of 74.64 m IGLD (1958) to a high of 75.76 m IGLD (1973). Lake Ontario water levels have been regulated since the completion of the St. Lawrence Power Project in 1958.

For the post lake regulation period, portions of the Toronto and Cobourg data (1960 to 1998) were analyzed using a Gumbel Extreme Value analysis to provide a range of maximum daily average level estimates for different return periods. The results are shown in Table 2-6.

Table 2-6: Estimated Daily Maximum Water Levels along the Toronto-Cobourg Corridor

Return Period (years)	Maximum Daily Average Water Level Toronto (m, IGLD)	Maximum Daily Average Water Level Cobourg (m, IGLD)
10	75.5	75.4
50	75.8	75.8
100	76.0	76.0
200	76.1	76.0

The lowest bank height along the shoreline of Pickering NGS is at the southwest corner of the East Complex, which is at 76.7 m IGLD. The PWMF Phase I site is slightly higher at an estimated 77 m IGLD (with Phase II even higher). When compared to the estimated 1 in 200 year water level of 76.1 m IGLD and the observed historic fluctuations, there is a low probability of flooding at the site due to high lake levels.

A review of the deep water wave climate, in conjunction with the existing topography at the site, was undertaken as part of the Pickering A Return to Service EA (OPG00a) to assess the potential for flooding due to wave runup and overtopping. The review indicated that, with nearshore water depths of 2 to 2.5 m, the maximum wave runups will be approximately 2 m on a riprap shore. As the crest of the protection works at Pickering NGS exceeds 77.5 m IGLD for much of their length, the probability of flooding due to wave runup and overtopping is considered low. Periodic wetting of extreme events due to wave spray and splash will occur. However, it is considered unlikely that waves breaking directly over top of the foreshore works will occur. It is also reasonable to assume that the surface drainage system will have adequate capacity to deal with any spray that may occur.

2.6.4 Lake Ice Conditions

Observations of Lake Ontario ice conditions near the Pickering NGS site from 1969 to 1980 included airborne reconnaissance to map the extent of ice cover and ground observations to determine ice conditions along the shore. It was found that during Pickering NGS operations the extent and duration of ice accumulation depends on meteorological conditions and the extent of the thermal plume (OPG07a).

Ice conditions in Lake Ontario were found to vary substantially every year throughout the winters: a band of brash and slush ice was generally observed along the shore throughout the winter season, but the build-up was found to vary considerably from year to year. The minimum build-up of shore ice occurred during the winter of 1974/1975 as a result of mild weather. The maximum accumulation was observed during the 1976/1977 winter when an ice foot approximately 3 m high extended a distance of about 50 m offshore west of the Pickering NGS site.

Lake ice accumulation in the intake channel can block the supply of water to the Condenser Cooling Water (CCW) system, as well as to other service water systems. Large ice jams are rare and only two significant events have been reported since 1971 (OPG00a). In the past, ice jams have primarily affected Pickering NGS B because of the configuration of the forebay. Ice can accumulate at the Pickering NGS B intake because it is at the end of the forebay where the ice can flow no further. Ice booms installed at the intake channel are expected to prevent any ice jams from occurring as shore ice is pulled into the intake channel.

Pickering NGS has adopted procedures for clearing ice blockages, which include lowering flow rates in the CCW (if necessary, the reactors are powered down to reduce the need for cooling water), directing warm water from the outfall into the intake to melt the ice, and installation of ice booms.

2.7 Terrestrial Environment

The Pickering property is visited by birds during the spring and fall migration; a number of species have been identified as breeding on-site, particularly in association with the Hydro Marsh and adjacent Frenchman’s Bay Marsh.

Regarding Species at Risk, Barn Swallow is an annual breeder at the Pickering NGS property including within the Protected Area, one pair of Peregrine Falcons has nested annually on Pickering NGS property since 2014, and one Eastern Wood-Pewee was recorded at the East Landfill in 2017.

No significant breeding bird populations have been identified in association with the PWMF.

2.8 Geophysical Environment

2.8.1 Geology

2.8.1.1 Regional Geology

On a regional basis, fairly thick 15 m to 30 m Pleistocene ground moraines overlie the Ordovician shale and limestone bedrock. The Ordovician sedimentary rocks are essentially flat lying and have a measured thickness of 212 m (OH82). These rocks rest unconformably on the Precambrian bedrock surface.

2.8.1.2 Site-Specific Geology

During the course of subsurface investigations and excavations for Pickering NGS, OPG has added substantially to the geological information of the site area. Further geological information for the PWMF Phase I site was obtained in a study conducted exclusively for this site (OH90). The borehole locations and a sample geotechnical data sheet are given in the subsoil investigation and evaluation report for the site (OH90). The ground surface at the PWMF Phase I site is at an elevation of approximately 3 m above the lake level.

The PWMF Phase I site is covered with a variable fill deposit, composed of coarse to fine sand and gravel, ranging in thickness up to 4.6 m. The state of compaction of this fill is erratic. Caissons were used to support the structure at the PWMF Phase I site. In general, the fill overlies a natural deposit of clayey silt till having stiff to hard consistency. The clayey silt till deposit extends to a depth of 13 m to 15.5 m to rest on a basal till deposit of very dense silt or clayey silt containing pieces of a weathered shale. Zones of hard to very stiff silty clay and dense sand are found sandwiched between the upper clayey silt till and the basal shale-till complex.

Shale bedrock at the PWMF Phase I site exists at depths varying from 16.9 m to about 20 m below present grade (OH90). The surface of this bedrock in the area ranges between elevation 58 and 64 m above sea level, and rises 2 to 3 m above this, inland from the Pickering NGS location. In the offshore area, the bedrock surface slopes off to an elevation of 49.4 m above sea level for a distance of 305 m.

The PWMF Phase II site is underlain by 1 m to 2 m of grading fill and 15 m to 20 m of stiff to hard upper and lower till complex soil deposits overlying shale bedrock. The grading fill for road bases and parking lots was derived from either reworked glacial till from on-site or imported granular soils. This shallow horizon is underlain by a hard, low permeability silty clay to clayey silt till horizon approximately 5 m thick which extends out beneath Lake Ontario. This upper till layer is a barrier that restricts downward seepage of groundwater. The upper till is underlain by a more permeable silty to sandy horizon over a lower till layer above the bedrock surface.

The Lake Ontario shoreline south of the PWMF Phase II site is a relatively undisturbed natural shoreline and bluff, rising an average of 3 to 4 m to the level plateau above. The bluff is relatively stable, composed of silty clay till, and covered with natural vegetation. The beach and foreshore are relatively flat, composed of sand, gravel and boulders associated with the parent tills. The bluff is routinely monitored by OPG for erosion to ensure safety and security of the Pickering NGS facilities.

During several borings and excavations carried out at and around the Pickering NGS site, no evidence was found of any structural weakness in the bedrock foundation. Neither mining activity nor withdrawal of fluids under the site has occurred that may affect the PWMF.

2.8.1.3 Groundwater

Subsurface investigations indicate a lower water-bearing zone of sand, silt and gravel within the lower till complex and local water-bearing lenses in the upper till complex. A relatively minor water-bearing zone occurs in the top part of the bedrock.

Generally on a regional basis, the groundwater level is about 1 m below the ground surface, which slopes gently towards Lake Ontario. The groundwater gradient is relatively flat and towards the lake. Groundwater movement, along with monitoring around Pickering NGS, have been summarized (OPG13).

Groundwater and sump samples collected in 2006 for the Pickering NGS B Refurbishment and Continued Operation EA Study Report demonstrated that groundwater quality exhibits seasonal changes in response to the level of the water table and variation in recharge. Overall, sampling results indicate that current conditions are comparable to historical data where available

2.8.2 Seismicity

2.8.2.1 Regional Seismicity

The western Lake Ontario region lies within the tectonically stable interior of the North American continent, which is characterized by low rates of seismicity (OPG00b). The seismic zoning maps in the National Building Code of Canada (NBCC), for example, place the site in Zone 0 to 1, with Zone 6 corresponding to the most seismically active regions of the country. The region surrounding Pickering NGS experiences low to moderate levels of seismicity. Most events have magnitude (M, called Richter magnitude) less than 5, with rare occurrences of larger events. In general, earthquakes in stable interior regions, such as the Lake Ontario region, occur at depths of 5 to 20 km, on faults formed hundreds of millions of years ago during previous active tectonic episodes. These faults are widespread throughout the crust, and typically have little to no surface expression.

As a guide to levels of seismic activity in the region, the western Lake Ontario region experiences an earthquake of M = 5 or larger about once every 100 years. This estimate of the level of seismicity is well-established. A magnitude 5 is considered a moderate earthquake, which may cause significant damage to poorly built (unreinforced) structures in the epicentre area, but does not generally damage modern well-engineered structures or heavy industrial structures. More damaging earthquakes of M = 6 or greater are considered possible but the rate of occurrence is ten times lower. This means that the likelihood of a large potentially damaging event (of M = 6 or greater) occurring in the local area is less than 1 in 1,000 per year.

Seismic hazards and design issues for Pickering NGS have been summarized (OPG00b). The seismic load assessment of the DSC is described in Section 3.3.2.6.

2.8.2.2 Site-Specific Seismicity

Research which has drawn on geological investigations and geophysical surveys in Lake Ontario, as well as the development of probabilistically based methodologies have provided assessments of the seismic hazard for the area. Improved understanding of eastern North American magnitude attenuation relationships and the characteristics of seismic wave propagation in stable continental regions have aided quantification of the seismic hazard. Uniform hazard spectra, representing current evaluations of regional seismic hazard, are similar to the Pickering NGS B design basis seismic ground response spectra in the dominant response frequency range of the DSMs. The seismic capacity assessment of the DSMs described in Appendix C was based on the Pickering B Design Basis Earthquake (DBE) as the input ground motion.

2.9 Land Use and Social Environment

2.9.1 Land Use

2.9.1.1 Pickering Nuclear Generating Station Site

The Pickering NGS is part of the Brock Industrial Neighbourhood in the City of Pickering, immediately east of the Bay Ridges Neighbourhood, south of Highway 401, west of the Town of Ajax and north of Lake Ontario. The property is located on a 240 ha site bordering Lake Ontario. There is a 914 m non-residence radius, also known as the exclusion zone, around the development. Beyond this limit, structures include recreational, institutional, and park facilities.

The land use surrounding the Pickering NGS site boundary fence is largely urban, including industrial, residential and parkland uses. The notable exceptions are Frenchman’s Bay marsh, Hydro Marsh and Duffins Creek Marsh, which are provincially significant wetlands found in the vicinity of the Pickering NGS property.

The Pickering NGS site has been developed since its inception in the early 1960s. The Pickering property is designated as a utility in the Region of Durham Official Plan. The property is zoned Industrial Zone M2. A small site located between the closed Brock Road right-of-way and the Pickering NGS B thermal discharge bay is zoned Public Service Zone M3.

The City of Pickering Official Plan designates the property as a controlled access area. The entire property is fenced and access to the site is restricted and controlled by OPG. Within the Pickering NGS boundaries, existing land uses consist of buildings, structures, switchyards, and transportation access required to operate and support the stations' functions.

The buildings include the reactor buildings, powerhouses, a variety of low rise office, warehouse, maintenance, and storage facilities as well as structures designed specifically for the technical functions required for generating and transmitting electricity from nuclear power. Other land uses within the Pickering NGS boundaries include landfills and waste management facilities.

The Duffin Creek Water Pollution Control Plant (WPCP) is located on lands immediately east of the Pickering property. The remaining portions of the Brock Industrial neighbourhood contain a variety of employment uses and vacant land.

The Waterfront Trail, an active recreational path paralleling Lake Ontario across the City of Pickering, with connections to municipalities to the east and west, runs adjacent to the Pickering NGS site boundary fence along Montgomery Park Road, on lands leased to the municipality by OPG.

The Bay Ridges, West Shore and Rosebank neighbourhoods west of Pickering NGS contain predominately residential and compatible ancillary uses (e.g., schools), with some employment and commercial uses generally along Bayly Street and Liverpool Road.

The southern boundary of the Pickering NGS property extends as a water lot into Lake Ontario. The lake is used locally for sport fishing, as well as recreational swimming and boating. The lake also provides water supply to the adjacent municipalities, the nearest water supply plant being in Ajax, 4 km to the east.

2.9.1.2 Pickering Waste Management Facility Site

The PWMF is distributed across the Phase I and Phase II sites. The RCS area is at the southern end of the PWMF Phase I site, covering about 0.43 ha. At the northern end of the PWMF Phase I site are the DSC processing building, offices, and DSC Storage Buildings 1 and 2. A berm to the east of the PWMF Phase I site prevents direct discharge of runoff to the Pickering NGS B outfall channel. The PWMF Phase II site provides Storage Building 3 for DSCs. Space on the Phase II site is reserved for future DSC storage requirements.

2.9.2 Population

The Regional Municipality of Durham is one of the largest municipalities in Canada and is also among the fastest growing. Durham Region includes the City of Pickering, the City of Oshawa, the Towns of Ajax and Whitby, the Townships of Brock, Scugog and Uxbridge, and the Municipality of Clarington.

Population growth in Durham Region has been closely linked to development and economic growth in Toronto and surrounding regions (i.e., Peel and York Regions). Buoyed by the strong economy in the GTA, strong population and economic growth is projected for Durham Region and its area municipalities (OPG07a). Detailed population data for Durham Region, including historical, current, and projected, are shown in Table 2-7.

Table 2-7: Historic, Existing, and Projected Population Growth – Durham Region and Area Municipalities (2001-2060)

Region	Area (km²)	2001	2006	2016	Population density 2016 (persons/km²)	2025	2060
Town of Ajax	67.1	76,675	87,890	119,677	1784	131,279	154,665
City of Oshawa	145.7	144,560	151,045	159,458	1094	214,325	331,086
Town of Whitby	146.5	90,875	118,615	128,377	876	193,937	349,694
City of Pickering	231.6	90,595	90,650	91,771	396	171,134	324,274
Municipality of Clarington	611.1	72,605	81,135	92,013	151	150,854	282,294
Uxbridge Township	420.7	18,065	20,580	21,176	50	23,649	29,056
Scugog Township	474.6	21,025	22,890	21,617	46	25,803	30,982
Brock Township	423.7	12,590	12,890	11,642	27	16,655	23,821
Total Regional Municipality of Durham	—	526,990	585,695	645,731	—	927,636	1,525,872

Forecasts indicate that the populations and housing stock of Durham Region and the City of Pickering will continue to grow by an average of approximately 2.5 percent per year over the next 20 years.

Agriculture has historically been an important component of the local and regional economies, environment and social fabric, and represents the most significant land use throughout most of the northern portions of the City of Pickering and Durham Region. Notwithstanding this agricultural heritage, both the City and the Region have developed an industrial platform with major industries in energy, automotive manufacturing, plastics/packaging, pharmaceuticals, aerospace/defense, chemicals/rubber and environmental technologies. In 2001, the regional employment was estimated to be 173,343 and the employment in the City of Pickering was estimated at 30,975. Pickering NGS is a major employer within the Region. Approximately 3,000 persons are employed at the Pickering NGS site. Approximately 70 percent of those employed at the station live within Durham Regions, while 8 percent live within the City of Pickering.

Pickering NGS is one of the City of Pickering’s most distinctive features. It is the most visible component of the Brock Industrial area, where most of the manufacturing industries in the city are located (i.e., south of Highway 401). The areas nearest to Pickering NGS are also characterized by a number of established residential neighbourhoods. At present, new developments are being located in the small remaining undeveloped areas within these neighbourhoods, some providing mixed residential and commercial uses.

As a mature city, residents of Pickering have access to a wide variety of educational, community and recreational facilities and services. The area nearest to Pickering NGS contains several municipal parks and the Waterfront Trail, which are heavily used by the local community for recreation. The lands and ravines associated with Frenchman's Bay, immediately west of the Pickering site boundary fence, provide the greatest concentration of recreational amenities in the City of Pickering. The river mouths, Frenchman's Bay and the shore area in the vicinity of Pickering NGS are also used for recreational fishing and boating. Pickering NGS contributes to these recreational activities, by maintaining approximately 120 hectares of land north of the station and outside the Pickering NGS site boundary fence as natural areas and parklands through lease agreements with the City.

2.9.3 Social Environment

The social environment has been described in detail in the Pickering A Return to Service EA (OPG00a) and is also discussed in the Pickering B Refurbishment EA (OPG07a).

In 1999, OPG commissioned a study (WATSON00) to identify the economic impacts of the Pickering and Darlington Generating Stations on the Region of Durham and its individual municipalities including the host municipalities of Pickering and Clarington. Potential impacts of the stations were modeled for the forecast period of 1999 to 2021. The results of the study indicate that, in Durham Region, the two stations will create 137,000 person-years of employment and will contribute $8.5 billion from expenditures and $171 million in taxes to 2021.

While there is a significant impact of Pickering NGS on the community, PWMF is a small but integral part of the Pickering NGS. The PWMF has only a minimal impact on the social and community environment, mainly due to its limited size and number of employees. However, the PWMF supports the continued operation of Pickering NGS, and thus contributes indirectly to the social economic benefits of Pickering NGS to the neighbouring communities.

2.10 Indigenous Interests

There are seven indigenous communities located within 155 km of the Pickering NGS property. First Nations in particular have a relationship with the lands along the north shore of Lake Ontario (from Toronto east to the Bay of Quinte) and north to Lake Simcoe and Rice Lake as a result of their occupation and traditional use of these lands prior to European settlement.

These indigenous communities are listed here, along with their approximate location:

Alderville First Nation: 20 km southeast of Peterborough on south side of Rice Lake.

Curve Lake First Nation: 15 km north of Peterborough on Buckhorn Lake.

Hiawatha First Nation: 15 km southeast of Peterborough on north side of Rice Lake.

Mississaugas of Scugog Island First Nation: 35 km north of Oshawa on Scugog Island in Lake Scugog.

Chippewas of Georgina Island First Nation: 10 km north of Sutton West on the southern end of Lake Simcoe.

Mohawks of the Bay of Quinte: 20 km northeast of Belleville.

Métis Nation of Ontario, Region 8: Three community councils, which take in an area between Durham and Guelph.

While there are no historic Métis settlements in or near the Pickering NGS property, there are Métis persons residing within the area. For example, the Oshawa and Durham Region Métis council represents Métis people in the Durham region. Overall, Durham has a fast-growing, diverse Indigenous population from across the country.

2.11 Security

As a Class IB Nuclear Facility that is used to handle and store Category II nuclear material, all buildings belonging to the PWMF are located within “protected areas” and are provided with appropriate security and alarm systems, in accordance with the CNSC Nuclear Security Regulations. OPG Nuclear has established a comprehensive and effective security program for the two different “protected areas” belonging to the PWMF. A description of the program, in the form of a Security Report, has been submitted to the CNSC in response to applicable regulations. The Security Report falls under the category of Prescribed Information, as defined by the General Nuclear Safety and Control Regulations, and is protected from public disclosure. The Report is not released to the general public, and is distributed by OPG Nuclear on a strict need-to-know basis.

Access to protected areas is restricted to authorized persons and all entry is controlled by code-access locking devices and monitored by security.

Security systems, staff, and equipment are available to monitor the transfer of DSCs from the station IFBs to the PWMF Phase I site, which is within the Pickering NGS protected area, and from the PWMF Phase I site to the PWMF Phase II site, which is within a separate nearby protected area. The on-site transfer of loaded DSCs is performed in accordance with the Nuclear Security Regulations.

Access to the RCS area is through a locked gate and access requires controlling authority approval.

3.0 USED FUEL DRY STORAGE

3.1 Used Fuel Dry Storage Process Overview

The UFDS process encompasses the facilities, structures, systems, and components (SSCs), and the operations necessary to transfer used fuel from the Pickering NGS IFBs to dry storage at the PWMF. This includes UFDS SSCs and operations inside the Pickering NGS, the PWMF, and en-route between the associated buildings. The in-station UFDS process, DSC handling, and safety assessment at the Pickering IFBs form part of the Pickering NGS licensing basis documentation. The summary below provides a complete overview of the dry storage process.

New, empty DSCs are received from the manufacturer at the PWMF Phase I processing building, where they are prepared and then transferred to the station for subsequent loading. The dry storage process, beginning with the preparation of new DSCs at the processing building and ending with the storage of loaded hermetically sealed DSCs in a DSC storage building, is summarized in Figure 3-1. Details of the modified DSC MKII design are given in Section 3.3.1.

At Pickering NGS, each of the IFBs are filled with demineralized water and contain the used fuel. Irradiated fuel bundles are placed into 96-bundle storage modules. Modules with 10-year or older fuel may be loaded into a DSC. The 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 and drained while being raised, 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. The loaded DSC is transferred on Pickering NGS site roads to the PWMF Phase I site for further processing.

In the DSC processing building, the DSC lid is seal welded and the integrity of the lid seal is inspected by Phased Array Ultrasonic Testing (PAUT). Final vacuum drying and helium backfilling of the DSC cavity is then performed, followed by installation of the drain port plug; the drain port plug is seal welded and inspected. The DSC is then placed in a vacuum chamber and helium leak tested. On successful completion of this leak test, International Atomic Energy Agency (IAEA) seals are applied and touch-up painting is completed. The application of the IAEA seals is completed in either the DSC processing building or in a designated IAEA surveillance area inside DSC storage building #1.

Details of dry storage operations are provided in Section 3.5. Note that many operations have logical pre-cursors, while others could be done in a different sequence to achieve the same results. The important objective is to complete all steps; minor variations to sequence may occur due to staffing and scheduling constraints, or may result from continuous improvement initiatives.

Used fuel is discussed in detail in Section 3.2, and details of the DSC are provided in Section 3.3. The UFDS area and equipment design descriptions are provided in Section 3.4 and the UFDS operations are described in Section 3.5. The systems specific to DSC handling are described in Section 3.6. An overview of safeguards provisions for the UFDS area at the PWMF is provided in Section 3.7. Illustrative photographs are shown in Appendix A.

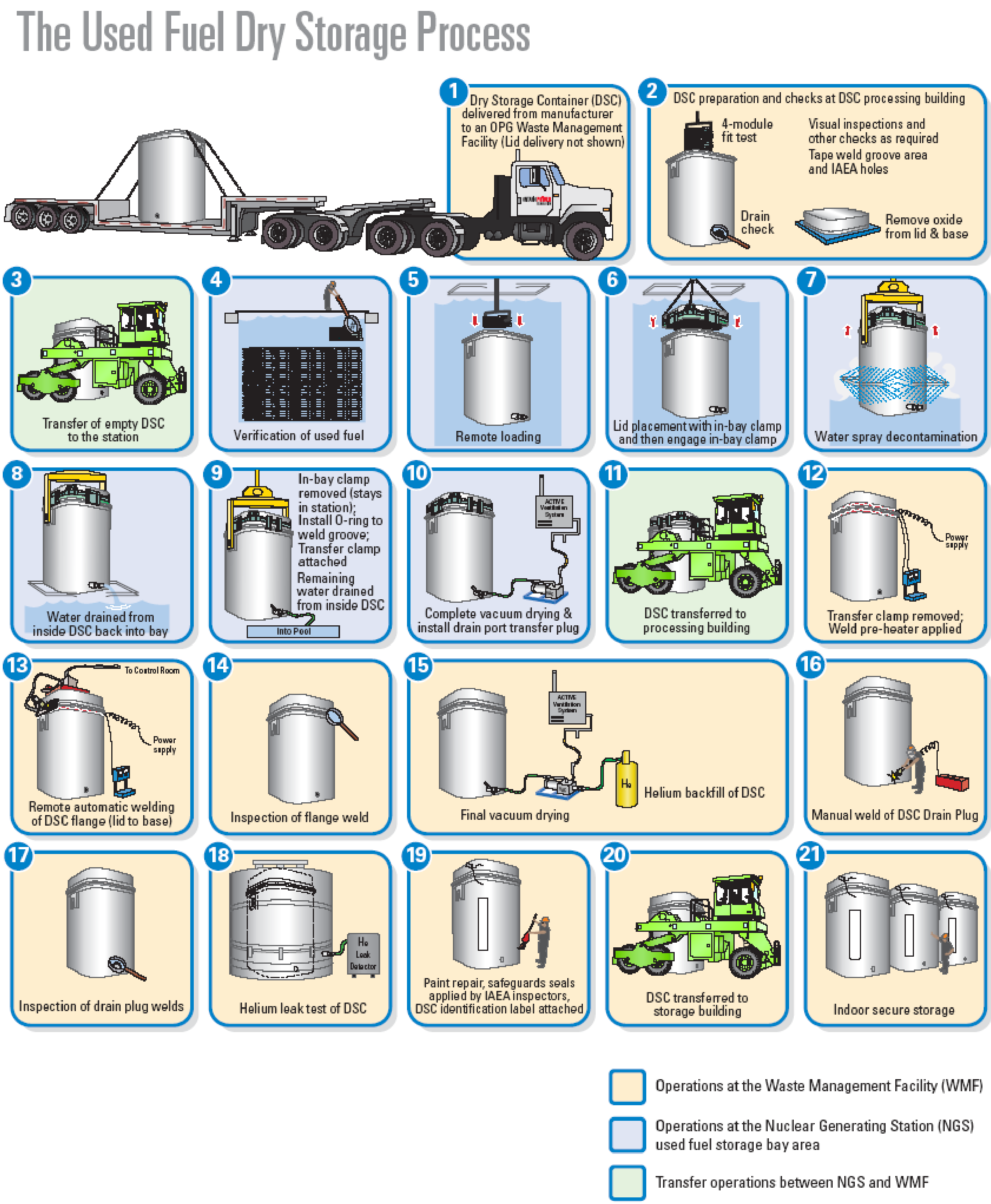

Figure 3-1: Used Fuel Dry Storage Process for Dry Storage Container MKII Design

3.2 Pickering Used Fuel Description

The fuel bundles used at the Pickering NGS reactors are 28-element Canadian Deuterium Uranium (CANDU) type fuel. Approximately 3,000 bundles are discharged annually from each of the reactors at Pickering NGS. After a minimum of 10 years of cooling[3], fuel bundles may be transferred to DSCs for interim dry storage.

3.2.1 28-Element Fuel Bundle

The fuel bundles are assemblies of 28 cylindrical fuel elements, arranged in concentric rings of 16, 8, and 4 elements (see Figure 3-2). Each fuel element contains high-density natural UO_2 pellets in a zirconium-alloy (Zircaloy-4) tube sheath[4]. CANLUB, a commercial graphite coating, is applied to the inner surface of fuel elements to minimize sheath strain during operation. The Zircaloy-4 sheath (hereafter referred to as Zircaloy) contains the alloy elements tin, iron, chromium, and sometimes nickel. In addition, beryllium is used to form a braze alloy to attach the appendages to the fuel sheath (i.e., bearing pads, inter-element spacers).

3.2.2 Reference Used Fuel Bundle

The primary factors that determine the characteristics of the used fuel are physical attributes, power and burnup histories, and decay time. These factors are in turn influenced by fuelling strategies and reactor conditions. Therefore, for the purpose of performing the safety assessment of the PWMF structures, systems, and processes, a reference used fuel bundle has been defined. Table 3-1 presents the characteristics of the Pickering reference used fuel bundle that has been used for the safety assessment (Appendix B) presented in this safety report.

Table 3-1: Pickering Waste Management Facility Reference Used Fuel Bundle Properties

Number of fuel elements	28
Length	495 mm
Mass of UO_2	22.87 kg
Mass of Zircaloy	1.67 kg
Mass of U	20.16 kg
Mass of the bundle	24.54 kg
Reference bundle power	[redacted]
Exit reference burnup	[redacted]
Time after discharge	10 years

[3] A minimum of 10 years of cooling can include residence time in fuel channels during GSS followed by subsequent IFB storage
[4] The fuel sheath is also referred to as clad or cladding. These terms are interchangeable in the nuclear industry.

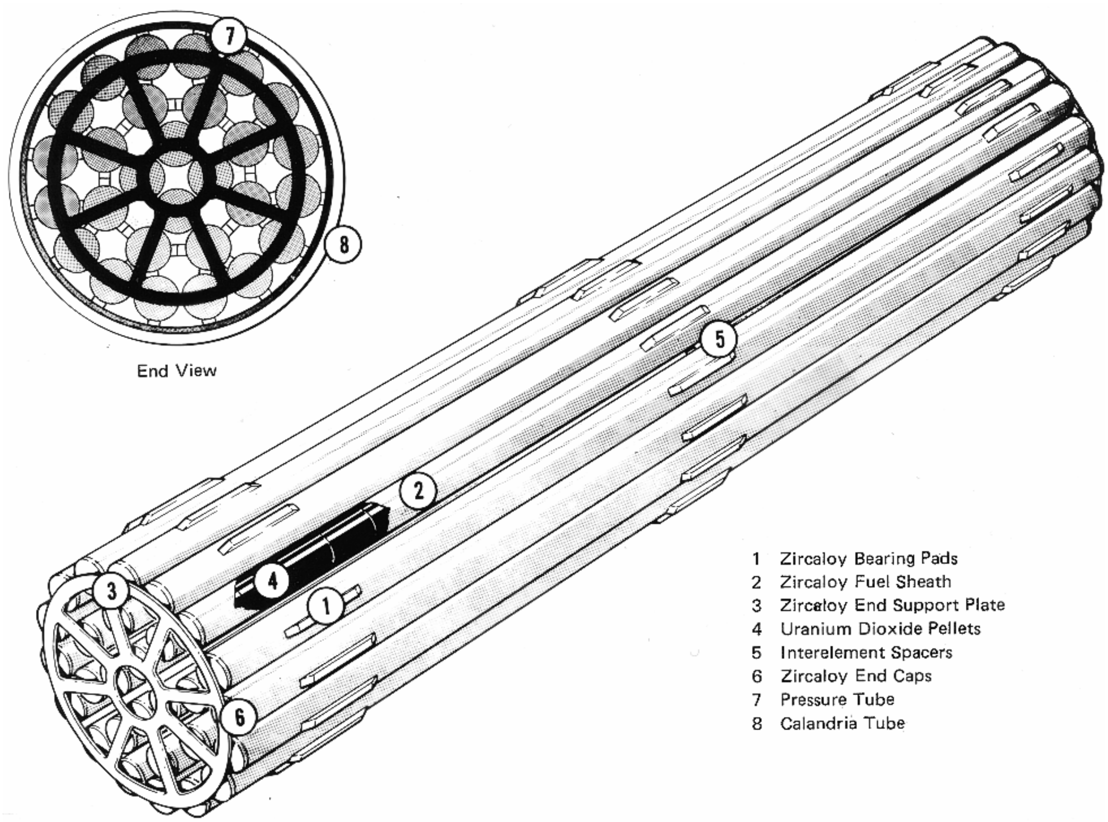

Figure 3-2: Pickering Fuel Bundle

3.2.2.1 Reference Used Fuel Bundle Dimensions

The fuel bundle used at the Pickering NGS reactors is 495 mm in length, has an outer diameter of 100 mm, and has a nominal total bundle mass of 24.6 kg. The complete dimensions of the reference fuel bundle are given in Table 3-1.

3.2.2.2 Reference Used Fuel Bundle Age

The reference fuel age for dry storage of Pickering used fuel is 10 years. In practice, however, the age of fuel loaded in DSCs will generally exceed 10 years, as operational procedures require the loading of the oldest available fuel, when feasible. Therefore, the average age of used fuel bundles in dry storage at the PWMF is greater than 10 years.

3.2.2.3 Reference Used Fuel Bundle Burnup[5]

Given statistical data of the fuel discharged from the Pickering NGS reactors (see Appendix B), [redacted] has been retained as the burnup for the Pickering reference used fuel bundle. Due to fuelling strategies and reactor operating conditions, the distribution of the fuel bundles within the IFBs is such that the

5 Burnup is the fission energy generated per unit mass of heavy element initially in the fuel (unit: MWh/kgU).

average burnup of the fuel loaded into each DSC is expected to be below the chosen reference used fuel bundle burnup.

3.2.2.4 Reference Used Fuel Bundle Power

The total core fission power for a Pickering NGS reactor is 1,744 MW (f). The average bundle power is [redacted] and the average fuel bundle residence time[6] is [redacted] Full Power Days[7].

3.2.3 Reference Used Fuel Bundle Radionuclide Inventories

After irradiated fuel is discharged from the reactor, nuclear fission essentially ceases. The fuel bundle radionuclide inventories and their associated radiation fields and decay heat decrease rapidly after discharge, due to decay of short-lived fission products and actinides with short half-lives.

At the time of discharge from the reactor core, the radionuclide inventory in a bundle is determined by the bundle power history. After 10 years of decay time following discharge from the reactor, the inventory of radionuclides in the fuel depends on the final burnup achieved and not on the power level or irradiation history of the fuel (see Appendix B for more information).

3.2.4 Reference Used Fuel Bundle Decay Heat

The energy produced by radioactive decay is released from the fuel bundle in the form of heat and radiation. The heat produced by the decay of actinides and fission products for the 10-year-old reference fuel bundle is 5.8 W. This value was retained from R05 of the Safety Report.

3.2.5 Chemical and Physical Characteristics of Radionuclides in Reference Used Fuel

The location of radionuclide species in a fuel element depends on their chemical and physical behaviour and where they were produced. The majority of new radionuclides, such as fission products, actinides and heavy elements in 10-year-cooled used fuel, are embedded within the lattice of uranium and oxygen atoms, very close to where they were produced. Activation products that are produced in the zircaloy sheath are primarily trapped by the zirconium alloy and cannot diffuse any significant distance from the site of their formation.

As discussed in Appendix B, at maximum expected sheath temperatures for 10-year-cooled Pickering used fuel (less than 150°C in dry storage in a helium atmosphere), krypton-85, tritium and carbon-14 would be the nuclides released within the DSC cavity should the fuel sheath become damaged. Carbon-14 is included for completeness, however, the dose contribution is significantly lower than tritium or krypton-85.

[6] Fuel Bundle Residence Time is the time for which a fuel bundle resides in the reactor.
[7] Full Power Day is defined as 24 hours of reactor operation at 100 percent of full power.

3.2.6 Deposits on the Exterior Surfaces of Fuel Elements

Corrosion products in the Primary Heat Transport System (PHTS) are present in low concentrations in the reactor coolant and subsequently become radioactive as a result of neutron activation in the reactor core. These are removed from the PHTS coolant by purification systems. However, some of these corrosion products will deposit on the surfaces of fuel bundles.

Experimental studies (see Appendix B) have shown that activated corrosion and fission products that have been deposited on fuel surfaces while in the reactor core, and that have remained adhered under the flow of PHTS coolant and during subsequent storage in the IFB, will be fixed to the outer surfaces of fuel elements and will require either physical abrasion or chemical dissolution to be released.

3.2.7 Defective and Damaged Fuel

When a sheath fails in the reactor core at high temperatures, the free inventory of volatile radionuclides residing in the fuel-sheath gap and other open voids in the fuel is released almost instantaneously upon sheath failure. Leaching of water-soluble radionuclides from the fuel matrix occurs slowly over a longer term, while the fuel element remains submerged in the fuel bay.

Used fuel with visible or known defects affecting the integrity of the fuel bundle is not transferred to DSCs.

3.3 Dry Storage Container

3.3.1 Dry Storage Container Description

The DSC is a free-standing reinforced concrete container, with an inner carbon-steel liner and an outer carbon-steel shell, for the storage, on-site transfer and off-site transportation (with an outer packaging) of used 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 is a safety-related structure because failure of the DSC to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (B-OPG17a).

The DSC has the capacity to store 384 used CANDU fuel bundles in four storage modules; each module has the capacity to hold 96 fuel bundles. The DSC provides the necessary radiation shielding, heat removal path, and containment of radioactive materials.

There are currently three configurations of the DSC in service:

1. The original DSC (also called Standard DSC) design for standard fuel modules. Refer to OPG Drawings 92896-D0H-29642-0001 and 00104-DRAW-79171-10001;

2. The Long Module DSC design with an enlarged interior cavity for long fuel modules and standard fuel modules. Refer to OPG Drawing 00104-DRAW-79171-10024;

3. The current configuration, the Long Module DSC Mark II design, which was evolved from the Long Module DSC design, with the vent port removed, and a smaller drain port. Refer to OPG Drawing 00104-DRAW-79171-10051.

The standard module DSC is a double-shell rectangular container with exterior nominal dimensions of 2.121 m × 2.419 m by 3.557 m in height (including the lid), and an inside cavity nominal dimensions of 1.024 m × 1.322 m × 2.520 m. The nominal thickness of each carbon-steel shell is 13 mm. The DSC walls consist of 523 mm high-density 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 range of 3.5 to 3.7 Mg/m³ and a compressive strength of at least 40 MPa. The total mass (including the lid of 11 Mg) is approximately 60 Mg when empty and approximately 70 Mg when loaded with four modules (384 used fuel bundles).

The outer dimensions of the DSCs remain unchanged for all the three DSC configurations. The long module DSC design was modified slightly from its original standard design: the enlarged inner liner results in a thinner high-density concrete shielding thickness by 11 mm on each side of the longer sides (i.e., those containing the lifting plates). In all other respects, the original standard DSC and the long module DSC designs are identical. The design change was implemented to facilitate storage of fuel bundles that are slightly longer and are used only at the Bruce NGS and at the Darlington NGS and are stored at the WWMF and the DWMF, respectively. PWMF stores only standard length fuel bundles and has adopted the long module DSC design to permit consistency in DSC designs across all three OPG waste management facilities. In 2009, another modified DSC design (long module DSC MKII) was introduced at the PWMF. An illustration of the long module DSC MKII is provided in drawing 00104-DRAW-79171-10051. From a safety perspective there is no difference between the three DSC designs.

The DSC MKII, shown in Figure 3-3 constitutes the reference container design for the PWMF.

In the DSC MKII design the vent port has been eliminated and the size of the drain port has been reduced. The DSC MKII has been modified to take advantage of operating experience and to further reduce the radiological dose to personnel during DSC processing.

The containment system for all three DSC configurations is defined as the inner liner, the to plate of the base, the bottom plate of the lid, the lid locating pin housings, the lid-to-base seal weld, and the drain port. The lid-to-base seal weld is a full penetration groove weld between bottom plate of the lid and the top plate of the base. The weld is designed to secure the lid in place to provide the DSC with the required structural strength, and to complete the containment barrier. The drain port has a stainless steel shielding plug that is seal welded after the DSC is loaded with used fuel.

The DSC containment system is backfilled with helium gas. Helium is used as the inert cover gas in the DSC cavity to protect DSC inner liner and the fuel bundles from potential oxidation reactions and to facilitate leak testing of the containment boundary.

The outer shell is coated with a high performance protective coating system to facilitate decontamination of the DSC following wet-loading operations in the IFB and to provide corrosion protection of the carbon steel.

Lifting plates are designed to safely lift the DSC, with the dedicated lifting beam or the transporter vehicle.

The DSC is designed with 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 3-3. These tubes are used for attaching two different types of IAEA seals.

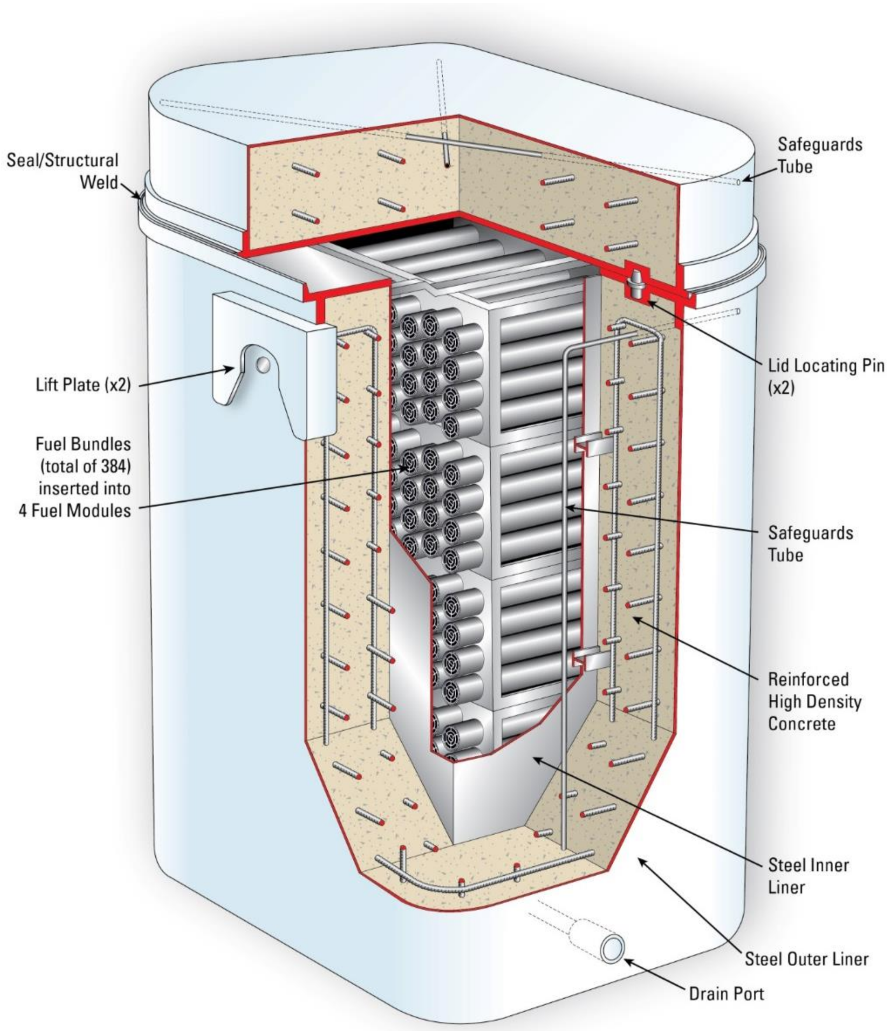

Figure 3-3: Ontario Power Generation Dry Storage Container

3.3.1.1 Used Fuel Storage Module

Used fuel bundles are placed into storage modules. [redacted] Each empty fuel module [redacted] and can hold 96 bundles, two bundles in each of the 48 [redacted] tubes. The storage module design is shown in Figure 3-4. A stack of four modules loaded with used fuel is placed inside the DSC inner cavity.

Freedom of Information and Protection of Privacy Act (FIPPA) S. 18 and Access to Information and Privacy (ATIP) S.13.

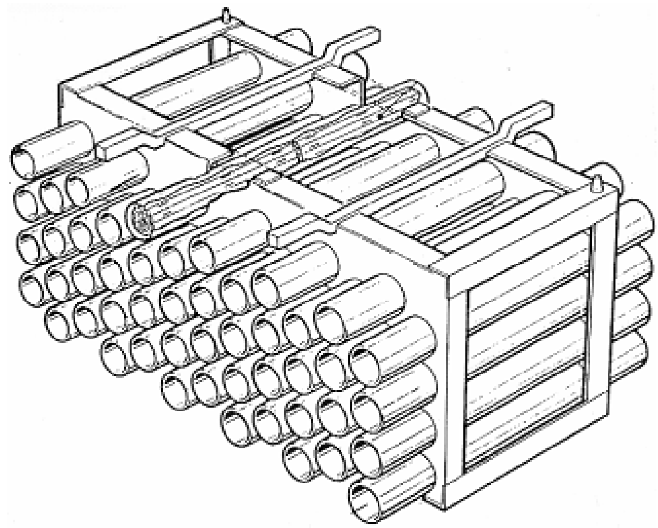

Figure 3-4: Storage Module

3.3.2 Key Dry Storage Container Parameters

Results of investigations performed on the DSC design regarding the integrity and stability of the DSC for different load cases are given below.

3.3.2.1 Decay Heat Removal

A thermal analysis carried out for the DSC (OPG04) assumed that the DSC was loaded with 10-year-old used fuel bundles with a decay heat of 6.4 W/bundle. The analysis demonstrated that the fuel would be adequately cooled. Ten-year cooled Pickering reference used fuel has a lower decay heat of 5.8 W/bundle, therefore the thermal analysis is considered to be conservative with respect to the PWMF conditions (see Appendix B).

3.3.2.2 Dry Storage Container Integrity under Thermal Load

The structural integrity assessment (OPG14b) for DSCs considered fuel bundles with a significantly higher decay heat of 7.4 W, which is conservative with respect to the PWMF conditions. The resulting thermal gradient in the concrete base of the DSC was estimated to be 54°C (OPG14a).

The predicted stresses generated in the concrete by the thermal gradient of 54°C indicate that through wall cracking will not occur and thermal expansion does not compromise the structural integrity of the DSC. For the Pickering reference used fuel bundle with a lower heat load of 5.8 W/bundle, the temperatures are expected to be lower. Therefore, no significant loss of either structural strength or shielding is expected to occur over the DSC design life.

3.3.2.3 Stability under Wind Load

The Design Basis Tornado (DBT) defined for the Darlington nuclear site (OPG12) has a rotational wind speed of 322 km/h, a translational wind speed of 96 km/h, a pressure drop of 9.6 kPa, a rate of pressure drop of 5.6 kPa/s and a radius of maximum rotational wind speed of 46 m. These parameters are considered to be large enough to envelope any credible tornadoes in Southern Ontario.

Environment Canada database between 1980 – 2009 (EC13a) confirms no occurrence of F5 category tornado in Southern Ontario. The F5 category tornado has a wind speed of 420 – 510 km/h (EC13b).

Safety of the DSC against overturning was investigated for a severe wind load simulating a tornado wind speed of 425 km/h (OH92). The wind pressure on the DSC was calculated in accordance with the NBCC.

A safety factor against overturning of greater than four was found for an empty DSC and greater than five for the DSC loaded with used fuel (OH92).

3.3.2.4 Tornado-Generated Missile Impact

The design of the DSC, with respect to overturning, along with transfer clamp and seal weld integrity, has been assessed for the impact of the following tornado generated missiles (AECL03):

(a) Wood plank, 102 mm × 305 mm × 3.7 m, mass 91 kg, velocity 335 km/h (80 percent of total tornado velocity, i.e., rotational plus translational);

(b) Steel pipe, 76 mm diameter, schedule 40, 3 m long, mass 35.4 kg, velocity 168 km/h (40 percent of total tornado velocity);

(c) Steel rod, 25 mm diameter × 914 mm long, mass 3.6 kg, velocity 251 km/h (60 percent of total tornado velocity);

(d) Steel pipe, 152 mm diameter, schedule 40, 4.6 m long, mass 129 kg, velocity 168 km/h (40 percent of total tornado velocity);

(e) Steel pipe, 305 mm diameter, schedule 40, 4.6 m long, mass 337 kg, velocity 168 km/h (40 percent of total tornado velocity);

(f) Utility pole, 343-mm diameter, 10.7 m long, mass 676 kg, velocity 168 km/h (40 percent of total tornado velocity); and

(g) Automobile, frontal area 1.9 m^2, weight 1,800 kg, velocity 84 km/h (20 percent of total tornado velocity.

Analysis shows that the transfer clamp will keep the lid in place and, if welded, the DSC lid seal weld integrity will not be impaired, DSC containment would not be breached and the DSC will not overturn, under the impact of tornado winds or the above postulated missiles (AECL03). Based on these results, it can be concluded that neither tornadoes nor missiles generated by them can cause significant damage to the containers that might affect used fuel during storage or transfer.

3.3.2.5 Effect of Creep and Shrinkage of Concrete

The maximum creep in DSC height at the end of a 50-year period was evaluated to be about 3.22 mm. The maximum creep in wall thickness was calculated to be about 0.67 mm and the maximum expected shrinkage of concrete to be 1.57 mm at the end of 50 years (OH92). These deformations are small enough to ensure that the structural integrity of the DSC will not be compromised.

3.3.2.6 Seismic Load

The DBE is defined as the earthquake which has an estimated probability of occurrence of not more than 0.001 events per year for the particular location. Using the seismic ground response spectra of the Pickering NGS B site, the applicable accelerations corresponding to a postulated DBE are evaluated as 6.25 percent of gravity (*g*) in the horizontal direction and 5 percent *g* in the vertical direction. The horizontal ground acceleration has been corrected for the container vibration frequency. The DSC has a safety factor of five against overturning and approximately two against sliding under these loads (OPG03c). As designed, the DSC can withstand a peak horizontal ground acceleration of 32 percent *g* in any direction without overturning. Overturning of the DSC due to an earthquake is therefore considered incredible. Sliding of an unwelded DSC lid of overturning of the transporter would also not occur under the above loads (OPG03c). Hence the integrity of the DSC is not affected by the seismic loads.

Ground acceleration cannot be directly related to a Richter scale magnitude value since factors such as distance from the earthquake epicentre and site geology must be taken into consideration on a case-by-case basis. However, magnitudes up to six, postulated to occur on the 50 to 80 km from Pickering NGS, have been considered (OPG00b).

3.3.2.7 Impact Load

As designed, the DSC can withstand an impact load of 45,608 kN, which is equivalent to about 65 *g* deceleration. This has been confirmed by the quarter-scale model drop tests in which the model survived intact under a 250-300 *g* deceleration (SMITH91; BOAG93), equivalent to about 62-75 *g* in full scale. In the half scale tests the model survived 230-310 *g* deceleration, equivalent to 115-155 *g* deceleration in full scale (BOAG93).

Within the PWMF, the normal lift height for a loaded unwelded DSC with transfer clamp installed is about 0.20 m. A seal welded DSC is raised about 1.5 m when lifted into the leak detection bell jar, or onto a raised platform for drain plug welding. Provisions have been made to limit the maximum height to which the DSC can be lifted during DSC transfer, processing and storage operations. It has been estimated that if dropped from 2.4 m on to a concrete floor, the deceleration would be equivalent to 40 *g*, well below the design basis of 65 *g* (SMITH91).

3.3.2.8 Immersion Load

The seal-welded DSC is designed to withstand a hydrostatic load of 1,965 kPa (that which would be imparted by 200 m depth of water) for more than one hour without rupture of the containment system.

3.3.2.9 Internal Pressure

The DSC can withstand ±100 kPa(g) internal pressure.

3.4 Pickering Waste Management Facility Description

3.4.1 General

The PWMF Phase I site consists of two stages. The PWMF Phase I Stage I contains a DSC processing building, which also includes workshop, offices, utilities, and DSC Storage Building 1 to accommodate up to 185 DSCs (71,040 bundles); the DSC processing building and office area are attached to DSC Storage Building 1. The PWMF Phase I Stage II contains DSC Storage Building 2 to accommodate up to 469 loaded DSCs (180,096 bundles). DSC Storage Building 2 includes an area for the receiving of new, empty DSCs.

The PWMF Phase II site consists of one DSC storage building, referred to as DSC Storage Building 3. Storage Building 3 became operational in 2009 and can accommodate up to 500 DSC's. The Phase II site has allocated space for future DSC storage. Construction of additional DSC storage space at the PWMF will be staged, as additional storage space is required and as authorized by the current licence.

3.4.2 Layout of the Pickering Waste Management Facility Buildings

3.4.2.1 Pickering Waste Management Facility Phase I Site

The PWMF Phase I site consists of industrial type buildings that are designed and constructed to provide for the safe processing and storage of DSCs. Storage Building 2 shares the north wall of the Storage Building 1 to form a single structure.

The DSC processing building has a floor area of about 830 m² and DSC storage buildings 1 and 2 have floor areas of about 2,070 m² and 4,775 m², respectively. The total floor area occupied by the PWMF Phase I site buildings, including the ground floor office area, is approximately 8,000 m².

The PWMF Phase I floor plans are shown in Section 15.0.

3.4.2.2 Pickering Waste Management Facility Phase II Site

DSC Storage Building 3 is a single story, commercial-type, pre-engineered or precast concrete structure with a concrete slab-on-grade floor. Storage Building 3 can accommodate up to 480 DSCs. The conceptual layout of a DSC storage building showing the placement of 480 DSCs is provided in Section 15.0.

A small kiosk facilitates radiological and security monitoring of personnel at the Phase II site. A small services structure is used to house the Uninterruptible Power Supply (UPS), switchgear, and control cabinets. The backup diesel generator is located outside, immediately west of the services structure.

3.4.3 Structural Description of the Pickering Waste Management Facility Buildings

3.4.3.1 Pickering Waste Management Facility Phase I Site

The floors are designed and constructed for long service with minimal maintenance. Floors in the PWMF are sloped to provide drainage to floor drains.

Building walls consist of 0.2 m thick precast concrete panels from ground level to a 3.7 m height. The walls above the concrete panels consist of metal panels. Wall louvres are installed at upper wall elevations above the DSC height. The precast concrete wall system provides effective radiation shielding, however the storage buildings are not safety related structures as failure of the buildings to perform to their design intent will not result in any doses to workers/public exceeding regulatory limits (OPG17a).

The PWMF has a built-up roof design with provisions for drainage of rainwater and melted snow. Access to the roof is provided by use of an outside, all-weather permanent stairway and a fixed ladder system. The Storage Building 1 roof has three large steps on both sides of the roof trusses to accommodate roof ventilation louvres. Storage Building 2 has roof vents. The buildings provide weather protection for the DSCs in storage.

3.4.3.2 Pickering Waste Management Facility Phase II Site

The floors of the storage building are designed for long service with minimal maintenance. Floors have been designed to slope to provide drainage to floor drains. Building walls consist of precast concrete panels from ground level to a 4.2 m height, with the walls above the concrete panels consisting of metal panels.

Storage Building 3 is equipped with a passive ventilation system of wall louvres and roof vents. It also has provisions for drainage of rainwater and melted snow, as well as permanent access to the roof through an outside, fixed ladder system.

The north side of Storage Building 3 does not include wall louvres. Instead, to provide the adjacent Training and Mock-up Building (TMB) with increased shielding against direct gamma radiation from the DSCs, the concrete shielding panels have been extended in height to approximately 6.5 m high. The TMB is located north of the PWMF Phase II site and can be seen in Figure 1-1.

The storage buildings are not safety related structures as failure of the buildings to perform to their design intent will not result in any doses to workers/public exceeding regulatory limits (OPG17a).

3.4.4 Pickering Waste Management Facility Used Fuel Dry Storage Systems and Areas

3.4.4.1 Office and Utility Area

The PWMF Phase I, Stage I office and utility area has two stories. The ground floor accommodates an electrical room, heating, ventilation, and air conditioning (HVAC) equipment room, tool room, washrooms, a coffee shop, an inventory office, and office space for IAEA staff. Office space, washrooms, and a viewing gallery permitting a view of both the DSC Storage Building 1 and DSC processing building are provided on the second floor. The lobby has an elevator to provide barrier-free access to the second floor.

3.4.4.2 Receiving and Preparation Areas for New (Empty) Dry Storage Containers

New DSCs are received and inspected in the DSC processing building. Additionally, a receiving area in the DSC Storage Building 2 provides added flexibility to receive new, empty DSCs from the manufacturer. Preparation of new DSCs includes visual inspection for physical defects and component fit.

The receiving and preparation area has a 77.1 Mg (85-ton) capacity overhead crane to handle an empty DSC, lid, transfer clamp, and the DSC lifting beam.

3.4.4.3 Workshop

The DSC processing building, also referred to as a workshop, houses the following dedicated systems for DSC processing:

- Lid welding and welding-related systems;
- Welding inspection system;
- Vacuum drying system;
- Helium backfilling system;
- Helium leak detection system; and
- Paint bays.

The preparation of new (empty) DSCs is performed in the processing area. Additionally, as discussed in Section 3.4.4.2, an area for the receipt of new DSCs has also been provided in DSC Storage Building 2. The DSC processing building is not a safety-related structure at the PWMF since it is not credited in the safety assessment as a barrier to the release of radiation.

3.4.4.3.1 Closure Welding and Welding-Related Systems

The following systems are used in closure welding of the DSC lid to the base:

- Weld preheat system;
- Seal welding system; and
- Weld cover-gas system.

Upper and lower pre-heaters are provided to preheat the flange and weld area of the DSC prior to and during the seal welding operation.

The seal welding system is designed to seal weld the DSC lid to the base. This weld is a full penetration, multi-pass groove weld that provides a permanent closure seal between the lid and the base.

The welding system is fitted with two Gas Metal Arc Weld (GMAW) weld heads using an inert shield gas. Each weld head has its own remote camera system and separate monitoring and control console. The weld head assembly is positioned by the overhead crane to rest on the DSC lid during the weld cycle.

The system is remotely operated from the control room. Leading and trailing closed-circuit television cameras provide views of the weld puddle to the operator during the welding process. Data monitors and warning systems are provided to monitor the essential welding parameters and report deviations outside the welding procedure tolerances.

The weld cover gas supply system supplies compressed shielding gas to the seal welding equipment. The system also provides a curtain screen of inert gas that protects the camera assemblies from damage by welding slag. Weld cover gas bottles are kept in the gas bottle storage room. This room is located in the northwest corner of the workshop and is only accessible from the outside.

Seal-welding of the DSC drain port, and weld repairs as necessary for lid weld, are accomplished using a gas tungsten arc welding (GTAW) process.

3.4.4.3.2 Weld Inspection System

A PAUT system is used for the inspection of the DSC lid-to-base seal-weld. The PAUT inspection system includes a scanner, two phased array probes, an equipment cabinet with the ultrasonic electronic motor drive control unit, and acquisition computer, and a storage cabinet with the couplant supply pump, calibration blocks stand, work bench and storage for tools. The weight of the PAUT scanner is approximately 14 kg.

All analysis variables (for example, position, detection, size) can be input into a file that is then utilized by the analysis routine. After completion of the inspection, the scanner system will be disengaged from the DSC and mounted into the storage cabinet.

A minimum clearance of 0.6 m is available to adjacent structures/equipment over the full circumference for application of the inspection scanner. All inspections are completed indoors in an environmentally controlled space at room temperature.

3.4.4.3.3 Vacuum Drying System

The DSC vacuum drying system evacuates and dries the DSC internal cavity through the drain port, after lid-to-base seal welding and weld inspection are complete.

The vacuum drying system filters particulate contamination that might be drawn from the DSC, prior to entering the vacuum pump. A dedicated hose is used in connection with the vacuum system to prevent the spread of contamination to other systems. The vacuum pump is connected to the active ventilation system.

3.4.4.3.4 Helium Backfilling System

The DSC cavity is backfilled with helium gas, after final vacuum drying and before seal welding the drain port. The helium gas facilitates leak detection for the seal welded DSC and creates an inert atmosphere for the stored used fuel.

Helium is piped from a bulk bottled supply, stored in the gas bottle storage room located in the northwest corner of the workshop. Helium is delivered to the DSC through the DSC vacuum system, using the hose dedicated for vacuum and helium backfilling operations.

3.4.4.3.5 Helium Leak Detection System

The helium leak detection system is designed to leak test the final welds on all DSC seal welds, including the lid and drain port welds.

The leak detection system consists of the helium leak detection equipment cart and a vacuum chamber large enough to hold a DSC. The seal welded DSC is placed in the vacuum chamber after raising the container about 1.5 m and lifting it over the side of the vacuum chamber base. The vacuum chamber lid is then lifted over the DSC using the overhead crane.

The vacuum chamber is evacuated by the DSC vacuum drying system and the leak detector vacuum pump, using the hose dedicated for helium leak testing operations.

3.4.4.4 Paint Bay

A paint bay is provided in DSC Storage Building 1 for painting the DSC weld area after the container has been welded and inspected. Weld affected areas are cleaned and painted, and touch-up paint applied to scrapes or scuffs on the DSC that may have resulted from handling.

The paint bay consists of an area with walls, platforms and two electric roll-up doors. During the painting operations, the workshop ventilation and exhaust system maintains air quality in the paint bay.

3.4.4.5 Dry Storage Container Storage Buildings

The DSC storage buildings at the PWMF are provided to facilitate all-weather operation. The DSC storage buildings are not safety-related structures at the PWMF since they are not credited in the safety assessment as a barrier to the release of radiation.

DSCs are stored in a pattern that allows retrieval of any DSC, if needed. The layout of the storage areas permits placement of DSCs using a transporter to achieve the desired storage capacity.

A designated IAEA surveillance area is provided inside the DSC storage building 1 to temporarily store partially processed DSCs. DSC placement is discussed in Section 3.5.3.7.

3.4.5 Pickering Waste Management Facility Building Services

The building services provided for UFDS at the PWMF include the following:

(a) Ventilation;

(b) Drainage;

(c) Fire protection and detection;

(d) Electrical services;

(e) Instrument and service air;

(f) Heating and air conditioning;

(g) Overhead cranes;

(h) Security;

(i) Safeguards; and

(j) Radiological monitoring.

Items (a) to (f) are discussed in this section. The overhead cranes used for UFDS are discussed in Section 3.6.2. Site and facility security provisions have been set out previously in Section 2.11. Safeguards requirements are discussed in Section 3.7, and radiological monitoring is further detailed in Chapter 7.

The building services also include domestic water, sewage, lighting, public address (PA) and telephone systems.

3.4.5.1 Ventilation

The DSC processing building is provided with active ventilation consisting of exhaust fans, radioactive filter assemblies and a discharge stack. This system provides active ventilation hook-up to DSCs for processing operations including vacuum drying. Airborne particulate contamination, if present, would be effectively removed by High Efficiency Particulate Air (HEPA) filters in the active ventilation system.

Localized ventilation exhaust, tied into the active ventilation system, is also provided for the welding station in the workshop. The ventilation exhaust from the DSC processing building also serves the paint bay of DSC Storage Building 1. Pre-filters installed in the paint exhaust hood collect paint aerosols, if present. Make-up air for the paint bay is drawn from DSC Storage Building 1 through a passive air passage in the paint bay wall.

The DSC Storage Buildings use passive ventilation through wall and roof louvres to dissipate decay heat from used fuel in storage to the atmosphere. The wall and roof louvres have been covered to prevent the ingress of snow and rain. The louvres can be uncovered if necessary to reduce ambient temperatures inside the facility.

Further, the louvres for Storage Building 3 are designed to prevent the ingress of rain, snow, and sand. Screens reduce the likelihood of small animals or birds entering buildings through the ventilation system. The roof vents are designed to minimize the retention and build-up of water, snow, or ice.

3.4.5.2 Drainage

Floor drains in DSC Storage Buildings 1 and 2 are connected to the active drainage system. However, these storage buildings are not expected to be contaminated and liquid effluents are not expected to be generated under normal conditions during DSC storage.

Drainage is directed to two underground active liquid sumps and transferred via sump pumps to two holding tanks, each with a 4 m^3 working volume, located in the workshop area.

After monitoring, the contents of the holding tanks are periodically transferred for routine treatment by pumping their contents, via underground piping, to the existing active liquid waste treatment system at Pickering NGS.

There are no provisions for active drainage within DSC Storage Buildings 3. Inactive drainage is provided in the floor for Storage Building 3. The inactive drain system within DSC Storage Building 3 is routed to a collection sump, where it can be monitored, if desired, and then released for discharge into the existing Pickering NGS site sewer system.

3.4.5.3 Fire Protection and Detection

The fire protection and detection systems at the PWMF are designed in accordance with the NBCC and the National Fire Code of Canada (NFCC).

The fire protection and detection system provides fire detection at the PWMF, fire water supply to the fire hose cabinets in the PWMF Phase I facility, and hydrants servicing the outside areas of the PWMF Phase I and Phase II sites.

Fire protection provisions including material usage meet the OPG Fire Protection Requirements. The Storage Buildings are appropriately grounded to protect against lightning.

3.4.5.3.1 Pickering Waste Management Facility Phase I

Fire water for the office, processing, and storage areas is supplied by a ring header located in Pickering NGS south yard. Fire hose cabinets are located at each access door in the workshop and storage buildings. The office area has two fire hose cabinets: one on each level.

The fire protection system in the unheated storage buildings is a dry leg system. When initiated manually, fire system piping is charged with water from the fire protection system in the workshop and office area. The dry leg system is vented to the atmosphere unless filled with water.

Fire detectors are connected to the PWMF fire alarm panel. A manual alarm hand pull station is provided at each exit and connected to the fire alarm panel. Fire alarms associated with the PWMF Phase I site are displayed in the Pickering NGS main control room for dispatch of the Pickering NGS ERT. Upon arrival of the ERT, local panel alarms indicate the required response area.

3.4.5.3.2 Pickering Waste Management Facility Phase II

Fire protection for the PWMF Phase II site is provided by a ring of fire hydrants located outside Storage Building 3. Fire detectors for Storage Building 3 are connected to an addressable local fire alarm panel, which in turn provides trouble and alarm signals to an annunciating alarm panel located in the kiosk. A manual alarm hand pull station is provided at each exit and connected to the fire alarm panel.

Fire alarms associated with the PWMF Phase II site are relayed to the Brock Road Guardhouse for notifying Pickering Fire Services. The initial responsibility for extinguishing fires in the PWMF Phase II rests with the Pickering Fire Services.

3.4.5.4 Electrical Services

The PWMF Phase I site at the PWMF is provided with three power supplies:

(a) Class IV Power
Class IV power is for general building loads, and electrical equipment within the DSC processing and storage buildings.

The Class IV power system supplies normal loads associated with the DSC processing building, such as office and shop area heating and ventilation, domestic and demineralized hot water systems, sewage pumps, active drainage pumps, overhead cranes, welding equipment and the air compressor. These loads can tolerate long-term power interruptions without impairment or have no safety implication following a power failure. The system receives its power from Pickering NGS Unit 8 via a 4.16 kV/600 V transformer.

(b) Class II Power

Class II power is for emergency lighting, IAEA Safeguards cameras and camera lighting, fire protection panel and alarms, and telephone and PA systems. The PWMF is provided with Class II power from the Unit 8 powerhouse. A transformer is used to provide the main source of Class II power to the emergency lighting. Also, a stepped down feed is used as a provision for IAEA Safeguards equipment.

(c) Class I Power

Class I power is provided from Unit 8 switchgear, and is used only to power the critical switchgear loads such as the protective and control circuits.

More essential loads are supplied by more reliable sources of power, to ensure continued electrical service during abnormal conditions.

The PWMF Phase II site is provided with 600V, 120V Class II, III, and IV power distribution systems.

(a) Class IV Power

Class IV power supply is taken from a 4.16kV/600V transformer to service the storage buildings, including the public address system.

(b) Class III Power

Class III power is supplied from a stand alone standby generator to meet the power requirements of the security systems, the exterior building lighting, the kiosk, the fire detection systems and the overhead door.

(c) Class II Power

Class II power is provided for emergency lighting and security monitoring equipment circuits. A suitably sized local battery powered UPS is included within the facility for Class II backup power supply requirements.

3.4.5.5 Instrument and Service Air

The instrument air system provides compressed air at sufficient pressure and flow capacity for the operation of pneumatically operated instrumentation and controls in the DSC processing building. The instrument air is supplied from Pickering NGS B Unit 8. Should normal supply of instrument air from Unit 8 fail, air is supplied to the instrument air receiver from the PWMF service air system. A service air system provides dry air at adequate pressure and flow for pneumatic tooling and general maintenance activities.

3.4.5.6 Heating and Air Conditioning

Office areas and the workshop are heated. The office area is additionally provided with air conditioning. Heating and air conditioning are not provided in DSC storage buildings.

The entrance kiosk to the PWMF Phase II site is provided with heating and air conditioning. The electrical and LAN rooms on the PWMF Phase II site are heated and the LAN room is additionally provided with air conditioning.

3.5 Used Fuel Dry Storage Operations

New DSCs are received from the manufacturer and are inspected and checked for component fit by the PWMF before being sent to the stations for loading.

At the stations, each DSC is wet-loaded with four used fuel storage modules in the fuel bay, decontaminated, drained and vacuum dried, and the transfer clamp and seal are installed to secure and seal the lid during on-site transfer. The loaded DSC is then transported to the DSC processing building using a special-purpose vehicle as described in Section 3.6.1.

At the DSC processing building workshop, the DSC is received, the transfer clamp and seal are removed, and the lid is seal-welded to the DSC body. The lid weld is subsequently inspected for defects. The DSC undergoes final vacuum drying and helium backfilling. The drain port is then welded and the weld is inspected, followed by helium leak testing. Finally, touch-up paint is applied to welded areas and scuffs or scrapes on the DSC exterior and the DSC is placed in a DSC storage building. IAEA Containment/Surveillance (C/S) of the DSC is maintained during the entire operation. Each out-of-station operation is described in the following text.

3.5.1 Preparation of New (Empty) Dry Storage Containers

Preparation of each new DSC consists of the following operations:

(a) Receiving new DSCs delivered by the manufacturer;

(b) Inspecting the DSCs for physical defects;

(c) Checking for component fit (e.g., lid fit, module emplacement, and lifting beam fit);

(d) The DSC drain tool and drain path are checked to ensure they are free of foreign material;

(e) Temporarily holding the DSC (with the lid) in a storage building until ready to be delivered to the station IFB for loading; and

(f) Weld groove inspection and grinding to remove excess rust and to prepare the DSC for processing. This is normally done within a day or two of transfer.

3.5.2 On-Site Transfer

After the DSC is loaded with used fuel at the station IFB and prepared for on-site transfer, a DSC transporter is used to pick up the DSC for on-site transfer. Labels are attached to the DSC identifying its contents, date of loading, and gamma dose rates.

Before the transporter exits from the IFB, both the vehicle and the DSC are monitored for contamination and if needed, they are decontaminated prior to being released into the unzoned area. The vehicle carrying the DSC travels from the IFB to the DSC processing building along the designated transfer route in accordance with security and safeguards requirements for on-site transportation.

3.5.3 Dry Storage Container Handling in the Processing Building

After the loaded DSC is received at the DSC processing building, it is prepared for storage as described below. After completion of lid weld inspection, partially processed DSCs may be transferred inside the DSC processing building and temporarily stored for up to one year from time of loading.

3.5.3.1 Receiving a Loaded Dry Storage Container

The DSC transporter places the DSC on the floor in the DSC processing building. The DSC is lifted from the floor using the overhead crane and lifting beam and moved into one of the welding stations. A loaded DSC is not left unattended unless a transfer clamp is installed and engaged. The clamp must be in place for all DSC movements with fuel on board, unless the lid seal weld has been applied. This includes craning a recently transferred DSC to a weld bay in preparation for processing.

3.5.3.2 Dry Storage Container Lid Seal-Welding

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 installed. The pre-heater is used to heat the DSC weld flange to a prescribed temperature.

The welding equipment is installed on the container and the DSC lid welded to the base by a full-penetration groove weld. This weld is deposited by a mechanized welding system using a GMAW process. At the conclusion of lid welding, the weld machine is removed and the DSC allowed to cool.

3.5.3.3 Lid Weld Inspection

The PAUT system is used for the inspection of the DSC lid-to-base seal weld.

PAUT can be performed in any location in the facility that has suitable AC power and a sufficient platform to allow access to the 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 percent of the weld as well as the heat affected zone (HAZ). The system drives two probes to scan the welds with a single pass, with one probe vertically placed on the vertical side and the other horizontally at the bottom of the DSC Base top flange. The bottom probe's position can be adjusted in the non-motorized axis of the scanner so that it is properly positioned to cover the weld and HAZ area. Inspection results are automatically saved electronically by the data acquisition system in the circumferential direction. All analysis variables are able to be input into a file that is then utilized by the analysis routine. After completion of the inspection, the scanner system is disengaged from the DSC and mounted into the storage cabinet.

3.5.3.4 Final Vacuum Drying, Helium Backfill, and Drain Port Seal-Welding

After successful completion of the weld inspection, the DSC is lifted into position for final vacuum drying and helium backfilling. The lifting beam is removed and the vacuum drying/helium backfilling system connected.

The interior of the container undergoes a final vacuum drying through the DSC drain port using the vacuum pump. Although the container is vacuum dried at the station before transfer to the DSC processing building, this final step removes residual moisture including moisture in the air that may have entered the DSC during processing. Pump discharge is directed to active ventilation. After final vacuum drying, helium is backfilled into the container to a pressure slightly below atmospheric pressure.

The drain port shield plug and tapered pin are inserted to permit the interspace behind the plug to fill with helium. The shield plug and tapered pin are welded and the welds are checked using dye penetrant inspection.

3.5.3.5 Helium Leak Testing

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 vacuum chamber. The vacuum chamber lid is craned over the DSC and sealed onto the base of the vacuum chamber. Using the helium leak detection system, air is first removed from the bell jar and then the helium leak detection system is activated.

This test simultaneously verifies the leak tightness of all DSC welds, including the lid closure weld and the drain seal weld.

3.5.3.6 Decontamination, Paint Touch-Up and Safeguards Seals

Exterior DSC surfaces are checked for loose surface contamination at the time of receipt and decontaminated if needed.

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.

Documentation and identification labelling are completed and permanent safeguards seals are installed in a designated IAEA surveillance area.

3.5.3.7 Dry Storage Container Placement and Storage

The DSC is moved, using a DSC transporter, to a designated storage location at a DSC storage building for storage. In the DSC storage building, the transporter is positioned to unload the DSC and place it on a designated storage location.

If the designated storage location is at the PWMF Phase II site, the DSC transporter will transfer the DSC between the PWMF Phase I and Phase II sites following the designated transfer route.

The container arrangement allows a minimum of 0.6 m spacing between the wider faces of the DSCs and a minimum of 0.2 m spacing between the narrower faces. This spacing between DSC rows is necessary to provide sufficient space for air circulation and cooling and to permit safe access to each DSC for periodic inspection by IAEA safeguards inspectors, radiation monitoring personnel, and maintenance personnel.

3.5.4 Dry Storage Container Records

The arrival and preparation of each new DSC, along with relevant operations carried out at the IFB and the PWMF, are recorded by PWMF Operations.

At the station IFB, a record is made of the identification number of the fuel modules loaded, the age of fuel loaded into the labelled DSC, and the time of loading. A record is also made of the time the vehicle leaves the bay.

3.6 Dry Storage Container Handling Systems Description

3.6.1 Dry Storage Container Transporters

The DSC transporters are specially designed multi-wheeled vehicles for the transfer of DSCs between the station IFBs and the DSC processing building, between the PWMF Phase I and Phase II sites, and for placement and retrieval of the seal welded DSCs inside the DSC storage buildings. The transporters are also used to place DSCs into the paint bay. Appendix A provides a photograph of a transporter carrying a DSC.

The transporters are self-loading and self-powered by a diesel engine and do not require the assistance of a crane when picking up or depositing a DSC. The DSC is lifted and transported via lifting trunnions mounted on the upper frame of the machines. The DSC is carried at a low lift height during transfer (about 0.2 m). Locking arrangements prevent the DSC from being inadvertently lowered to the ground upon hydraulic failure. The tires on the transporters are designed not to deflate if punctured.

When travelling with a DSC, the transporters operate at low speed and have a short stopping distance (within 3 m). When travelling at minimal speeds (e.g., when moving DSCs within the DSC processing and storage buildings), stopping is essentially instantaneous.

The transporters are capable of forward and reverse motion and have a tight turning radius. A radio remote control may be used to operate the transporter either from the cab or remotely. Vehicle lighting is provided for operation on site roads, if necessary.

3.6.2 Overhead Cranes

The PWMF DSC processing building is equipped with a 77.1 Mg (85-ton) overhead crane that is fitted with a 9.1 Mg (10-ton) auxiliary crane.

3.6.3 Dry Storage Container Lifting System

The DSC lifting system consists of lifting plates on the DSC and a lifting beam with trunnions. The lifting beam has been designed for DSC handling and is compatible with the swivel hooks on the processing building overhead cranes and on the crane in the IFB. The lifting beam is designed to engage into the lifting plates attached on the DSC body and not to disengage from the DSC while the beam is under load.

A lifting beam is provided for use in the DSC processing building and in the IFBs. When not in use, the lifting beam is stored on a custom built frame.

3.6.4 Transfer Clamp

The transfer clamp is designed to prevent the lid from separating under credible accident scenarios during transfer of loaded DSCs between the stations and the DSC processing building, and during DSC handling inside the processing building prior to seal welding the DSC lid.

At the station IFB, once the in-bay clamp has been removed, the elastomeric seal is installed in the weld groove and confirmed seated before the transfer clamp is installed. The transfer clamp is used to securely attach the lid to the DSC base during on-site transfer of a loaded DSC between the stations and the PWMF Phase I site. The transfer clamp is used in conjunction with a seal between the lid and base, to permit the cavity of a loaded DSC to be vacuum dried.

The transfer clamp is a safety related structure because the failure of the transfer clamp to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (OPG17a).

3.7 Safeguards

In accordance with IAEA requirements, the Integrated Safeguards approach and the Additional Protocol to the Treaty on the Non-proliferation of Nuclear Weapons (NPT) have been implemented. These include the following elements:

- Provision of information, including:
    - Advanced Information of upcoming planned activities:
        - Operational Programme – annual, quarterly and interim updates;
        - Weekly notifications of scheduled DSC loadings and planned transfers in the following week;
        - Schedule changes to DSC transfers; and
        - Force majeure notifications (e.g. malfunctions of a facility computer systems).
    - Declarations of activities performed, including:
        - Monthly integrated safeguards declarations;
        - Annual physical inventory taking;
        - Annual additional protocol updates;
        - Monthly nuclear material accounting and operational records (e.g. general ledgers);
        - Inventory summary of fissionable and fertile materials which have foreign obligations; and
        - Operational records are made available upon request.
- Inspections including:
    - Unannounced inspections of random DSC transfer-related activities at the discretion of IAEA;
    - Short notice random inspections; and
    - Complimentary access inspections of the facility.
- Design information questionnaire and attachments updates, periodically, as directed by the CNSC and the facility attachment;

- Conducting a physical inventory verification based on a 50 percent selection probability;
- Radiological profiling for each DSC transferred;
- Application of IAEA containment and surveillance measures, including video surveillance in the processing building and storage building, as well as applying dual containment seals on loaded, welded DSCs within the designated IAEA surveillance areas;
- Remote Monitoring by IAEA to access their real-time safeguards data of the facility; and
- Complementary and managed access to the facility as part of the additional protocol to the NPT.

The PWMF management stays current with the IAEA's safeguards requirements and is committed to meeting OPG's safeguards obligations in an efficient and timely manner.

3.8 Pickering Used Fuel Dry Storage Operating Experience

Operating experience has demonstrated that the PWMF can be operated safely and without undue risk to workers, members of the general public, or the environment.

The PWMF has been operating since January 1996. The safety performance of the facility has been excellent during its operation over the entire period.

Dose rates have remained below the regulatory limits. Collective occupational radiation exposures have been less than the predicted exposures by 30 percent or more. Emissions have remained below the regulatory limits. The PWMF routinely operates contamination free.

There have been no public safety events at the facility. In the past five years there have been no MOECC-reportable spills. As of December, 2017, 901 DSCs have been successfully and safely stored in the PWMF.

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4.0 USED FUEL DRY STORAGE SAFETY ASSESSMENT

4.1 Introduction

This chapter provides a summary of the PWMF UFDS radiological safety assessment during DSC on-site transfer, processing, and storage. Details of the UFDS safety assessment methodology, assumptions and results are given in Appendix B.

Estimated gamma radiation dose rates are presented in this section for distances from the PWMF to the Pickering NGS site boundary. Dose rate calculations consider UFDS at the PWMF to contain 1068 DSCs. Although the site has a higher design capacity, a lower number of DSCs was assumed to account for wider spacing requirements of the new DSC transporter design as well as for physical limitations due to electrical panels and roof bracing.

Conservative estimates of public dose rates due to releases resulting from hypothetical failures of an assumed fraction of fuel elements for normal and abnormal operating conditions and credible accident conditions are also presented.

4.1.1 Safety Assessment Approach

The PWMF safety philosophy embodies the defense-in-depth approach to keep radionuclide emissions within levels that are within regulatory limits and are ALARA. This defense-in-depth approach is represented by multiple barriers between the used fuel and the public. Each barrier independently provides a measure of safety toward preventing the release of radioactive materials as follows:

- The uranium dioxide (UO_2) matrix effectively contains the radionuclides present in 10-year-cooled used fuel (either under wet or dry storage conditions), except for the free fractional inventory of tritium (in vapour form), krypton-85 (in gaseous form) and carbon-14 (in gaseous form);

- The fuel sheath additionally contains the free fractional inventory of tritium, krypton-85 and carbon-14 that would otherwise be available for release;

- The seal-welded DSC is considered a safety related system credited in the containment of radioactivity and provides an additional barrier against the release of tritium and krypton-85 in the event of fuel sheath failure; and

- The reinforced concrete construction of the DSC base and lid provides a barrier against radionuclide release and also provides effective shielding for gamma radiation from used fuel.

Conditions relevant to UFDS operations are classified as normal and abnormal operating conditions, or credible accident conditions, as defined below:

(a) Normal operating conditions are routine. UFDS SSCs are expected to remain functional and to experience no unacceptable degradation.

(b) Abnormal operating conditions do not occur routinely, but can potentially occur over the lifetime of the facility. UFDS SSCs are expected to experience abnormal events and conditions without permanent degradation of functional capability (although operations are expected to be suspended or curtailed during abnormal conditions, unless the appropriate compensatory measures are taken).

Potential abnormal operating conditions may include failure of used fuel sheath or a DSC seal weld, operator error, and equipment failure.

(c) Accident conditions are unlikely to occur over the lifetime of the facility; however, the safety implications resulting from accidents may exceed the potential consequences of abnormal operating conditions. In nuclear facilities, design features are employed to preclude the occurrence of potential accidents and limit their consequences.

Accidents can result from events within a facility (e.g., equipment failure resulting in the drop of a loaded DSC), or from events that are external to the facility. External events include hazards relating to human activities (e.g., external fires and explosions or a small aircraft crash), and natural hazards (e.g., earthquakes, tornadoes, thunderstorms including lightning, and floods).

For each failed fuel element (as discussed in Appendix B), it is postulated that 100 percent of the tritium and krypton-85 present in the gap (i.e., the space between the fuel pellet and the fuel element sheath) will be released, along with 10 percent of the tritium and krypton-85 present in the uranium dioxide grain boundary. In addition, 100 percent of the carbon-14 inventory in each failed fuel element is released. Carbon-14 is included for completeness, however, the dose contribution is significantly lower than tritium or krypton-85.

4.1.2 Design and Operating Acceptance Criteria

Under normal operating conditions during storage, PWMF UFDS SSCs are expected to provide reasonable assurance that the used fuel can be stored in DSCs and the DSCs can be retrieved without undue risk to workers, members of the general public, or the environment.

PWMF operations comply with the OPG requirement to keep total radioactive emissions under normal operating conditions within regulatory limits and ALARA.

The PWMF SSCs have been designed to fulfill the following criteria under normal and abnormal operating conditions and credible accident conditions including external hazards:

(a) Comply with CNSC regulatory dose limits (occupational and public) and the ALARA principle;

(b) Maintain subcriticality;

(c) Maintain the integrity of used fuel in dry storage; and

(d) Recover safely from abnormal operating conditions and credible accident conditions.

The safety assessment of normal and abnormal operating conditions, and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions and are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions.

4.2 Acceptance Criteria

The radiation safety requirements under normal operation for the PWMF are the following:

- ≤ 0.5 µSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.
- ≤ 100 µSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years.

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose limit for the public at or beyond the OPG property boundary due to an abnormal event/credible accident shall be 1 mSv.
- The dose target for a worker due to an abnormal event/credible accident shall be 50 mSv.

The target of 50 mSv for abnormal event/accident refers to NEWs. The equivalent target for non-NEWs/members of the public is 1 mSv.

4.3 Radiological Safety Assessment – Normal Operating Conditions

4.3.1 Radioactive Emissions and Contamination

Under normal operating conditions, no airborne emissions are expected from loaded DSCs during transfer from the stations 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. A dedicated active ventilation system is used to deal with any airborne emissions.

Surface contamination on DSC exterior surfaces is effectively controlled through preventative measures and decontamination at the station IFBs. 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 (MDA). The dose resulting from the normal operation emissions have been assessed using the emission survey data from the 2007-2017 PWMF quarterly reports and the methodology based on the CSA N288.1-14 standard (CSA14a). The highest annual individual dose to a member of the public during the evaluated period is calculated to be less than 10^{-3} μSv per year.

In addition to the dose from the recorded emission data, potential emissions under normal operating conditions have been evaluated (see Appendix B). Since each DSC has the capacity to hold 384 fuel bundles and assuming the facility processes about 70 containers per year, it is postulated that a total of 280 fuel elements (four elements per DSC, i.e., one fuel element in 1 percent of the fuel bundles is assumed to be damaged) fail during 1-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 boundary, are estimated to be 2.28×10^{-4} μSv/year. When combined with the dose from the recorded normal operation emissions, the upper bound estimate for dose consequence for airborne emissions is 3.65×10^{-4} μSv/year.

As the DSC is fully drained and vacuum dried after loading at the station IFBs, 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 IFBs 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 DSC 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.

Radioactive airborne emissions and liquid effluents from the PWMF are well within the restrictive administrative targets set for the facility and contribute a negligible fraction of the release limits for the Pickering site.

The dose to workers during normal operating conditions is discussed in Section 4.5.2. The processing building has an active ventilation system and welding is performed remotely, therefore worker dose from welding operations is unlikely. Furthermore, dose to workers is managed under the RP Program (N-PROG-RA-0013), which includes whole body counting to detect potential internal uptakes.

4.3.2 Radiation Fields

4.3.2.1 Dry Storage Container Dose Rates

Using the radionuclide inventory data and taking into consideration the shielding provided by the steel rebar 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 4-1 shows the calculated gamma radiation dose rates for different distances 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 as defined in Chapter 3 of this report. Figure 4-1 shows the calculated gamma radiation dose rates as a function of the distance from the top, side, front, and bottom surfaces of a DSC, fully-loaded with 10-year-cooled Pickering reference fuel

Table 4-1: Calculated Dose Rates from a Dry Storage Container of the Modified Long Module Design, Fully-Loaded with Pickering 10-year-cooled Used Fuel Bundles

Distance from DSC	Position	Dose Rate (μSv/h)
Contact	Side	33.8
	Front[8]	39.7
	Top	26.7
	Bottom	109.6
1 m	Side	16.3
	Front	20.9
	Top	9.5
	Bottom	41.0
2 m	Side	8.4
	Front	11.0
	Top	4.1
	Bottom	16.3

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 UFDS 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 compares with estimates of 40 μSv/h

[8] The label ‘front’ corresponds to the wider face of the DSC and ‘side’ indicates the narrower face.

contact dose rates for 10-year-cooled fuel (at the DSC side or front) as set out in Table 4-1. At a 1 m distance, measured dose rates are about 5 to 7 μSv/h, compared with calculated dose rate estimates of 16 to 21 μSv/h.

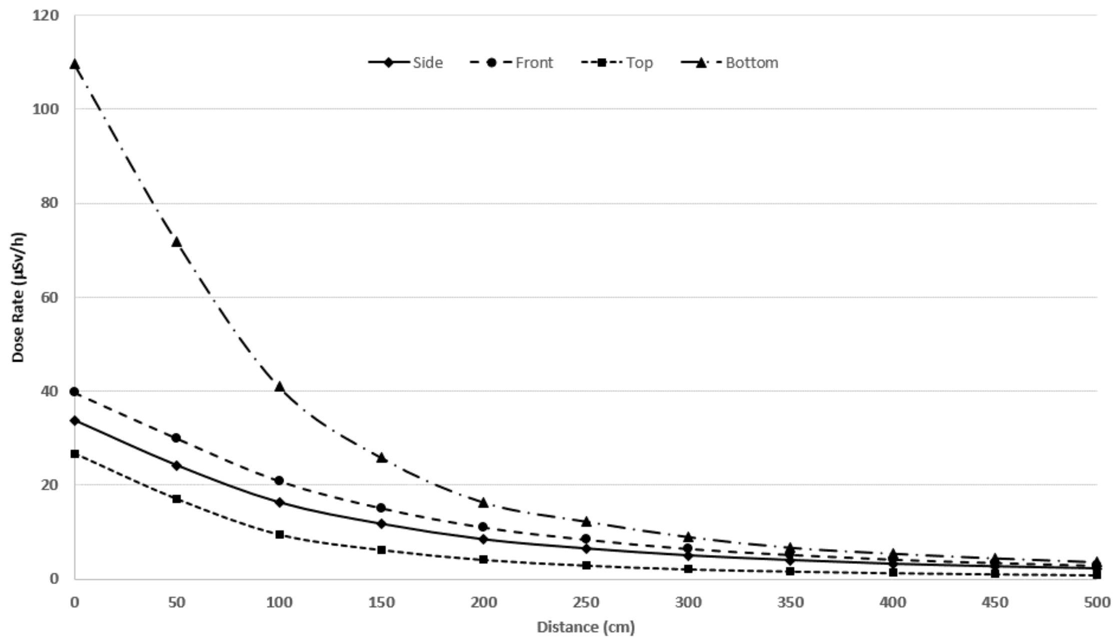

Figure 4-1: Calculated Dose Rate versus Distance from the Surface of a Single Dry Storage Container of the Modified Long Module Design, Fully-Loaded with 10-year-cooled Pickering Reference Fuel Bundles.

4.3.2.2 Dose Rates Inside the Dry Storage Container Storage Buildings

The predicted dose rates from a row of DSCs facing the corridor in the middle of Storage Building 2 loaded with 10-year cooled fuel in storage, are presented in Figure 4-2. The results show that the dose rates on the west side are approximately 30 μSv/h, this drops to 12.5 μSv/h in the middle of the corridor and increases to 28 μSv/h on the east side.

The predicted dose rates from a row of DSCs facing the corridor in the middle of Storage Building 3 loaded with an average of 26 year old fuel in storage, are presented in Figure 4-3. The DSC configuration is symmetric on the east and west side. The results show that the dose rates on the west and east side are approximately 18 μSv/h and this drops to 5 μSv/h in the middle of the corridor.

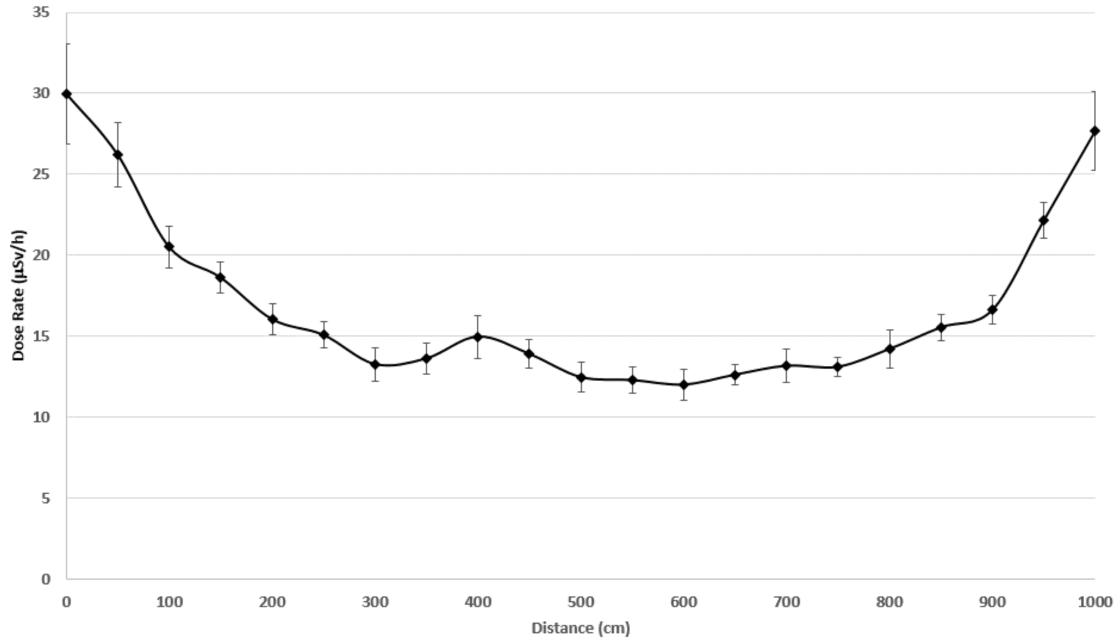

Figure 4-2: Calculated Dose Rates across the width of the North-South Corridor of Dry Storage Container Storage Building 2

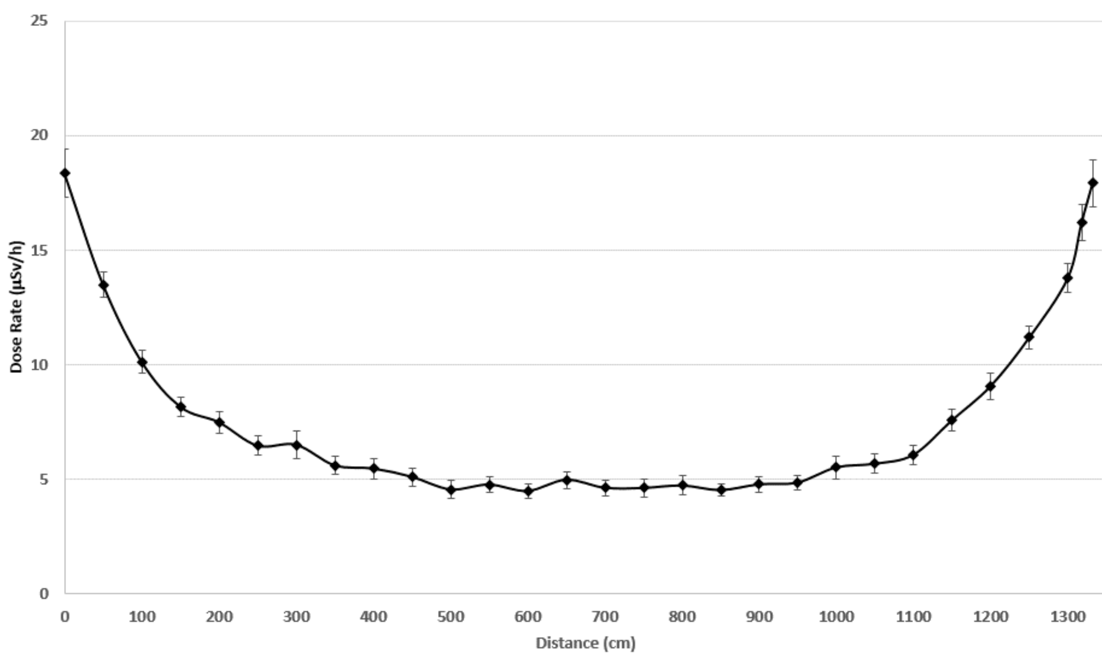

Figure 4-3: Calculated Dose Rates across the width of the North-South Corridor of Dry Storage Container Storage Building 3

4.3.2.3 Dose Rates Outside the Dry Storage Container Storage Buildings

When the PWMF DSC storage buildings 1 to 3 are filled to nominal design capacity, the dose rate at the site boundary is calculated to be 1.04×10^{-3} µSv/h.

This is equivalent to an annual dose of 2.08 μSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 7.23×10^{-4} μSv/h. This is equivalent to an annual dose of 0.72 μSv based on 1,000 hours occupancy; this is a conservative occupancy assumption for boaters and fishermen.

These results indicate that the PWMF dose target of ≤100 μSv/y at the station site boundary as set out in Section 1.6.1, are met during UFDS operations.

4.3.2.3.1 Pickering Waste Management Facility Phase I Site

Dose rates at the perimeter fence east of Phase 1 are calculated to be less than 0.24 μSv/h. The predicted dose rates are less than 50% of the dose rate target of 0.5 μSv/h.

These results indicate that the PWMF dose rate targets of ≤0.5 μSv/h at the station security fence on a quarterly average basis as set out in Section 1.6.1, are met during UFDS operations.

4.3.2.3.2 Pickering Waste Management Facility Phase II Site

The calculated dose rates at the perimeter fence, [redacted] from the DSC Storage Building 3, is 0.29 μSv/h. This dose rate is well within the criterion of ≤0.5 μSv/h established for limited occupancy (i.e., up to 2,000 hours per year) by non-NEW personnel at the perimeter fence. OPG will take every precaution to ensure that administrative targets are met. If measured dose rates outside the perimeter fence should exceed 0.5 μSv/h, warning signs will be posted where needed in areas that are accessible by non-NEWs. The dose consequences for personnel walking or driving on the Pickering site roads (i.e., at further distances and for short time durations) will be well within the public dose limits.

As discussed in Section 3.4.3, 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 TMB are below the dose rate target of 0.5 μSv/h (see Section 1.6.1). The dose rate at the TMB was estimated to be 6.58×10^{-2} μSv/h.

4.4 Radiological Safety Assessment – Malfunctions and Accidents

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 within and external to the PWMF. A full description of this assessment is provided in Appendix B.

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

(a) On-site transfer operations;

(b) Operations inside the DSC processing building; and

(c) Storage.

To assess the overall safety of the UFDS operations at PWMF, safety analyses presume that abnormal operating conditions and credible accidents will result in the failure of multiple barriers and release of radioactive material. The results of off-site dose consequence calculations from the bounding (worst case) scenarios are compared against the regulatory dose limits.

For each stage of UFDS 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 UFDS operations. The safety assessment for the in-station operations is part of the Pickering NGS licensing basis documentation.

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 UFDS operations at PWMF, safety analyses presume that abnormal operating conditions and 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.

4.4.1 Malfunctions and Accidents Assessment for Operations during Dry Storage Container On-Site Transfer

As described in Chapter 3, the DSC transporter is used to transfer loaded DSCs from the Pickering NGS A and B IFBs 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 described in Appendix B has taken into account postulated malfunctions and accidents that could potentially affect the on-site transfer of a loaded DSC.

Table 4-2 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., with a frequency of occurrence that is $\geq 10^{-6}$ events per year (CSA14b))[9] during DSC on-site transfer.

Table 4-2: Postulated Malfunctions or Accidents during Dry Storage Container On-Site Transfer

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Transporter failure	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Transporter Operator Health-Related Emergency	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
DSC drop during on-site transfer from IFB to DSC Processing Building	credible	5.10×10^{-3}	6.14×10^{-3}	5
DSC drop during on-site transfer between the PWMF Phase I and Phase II sites	credible	6.50×10^{-3}	7.91×10^{-3}	5
Fire	incredible[10]	—	—	—
Criticality	incredible	—	—	—
Adverse road conditions	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5
Earthquake	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5
Tornado	incredible	—	—	—
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Explosion along transfer route	incredible	—	—	—
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—
Toxic gas releases – chlorine from Ajax water treatment plant	credible	0	0	0
Soil failure/slope instability	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5

The bounding dose consequences during this stage of the dry storage process are associated with the drop of a DSC during on-site transfer. Although fuel sheath failure is not expected to result from a DSC drop from the low lift height of the

[9] The frequency of occurrence of 10^{-6} is consistent with the value given in CSA N292.0-14, "General Principles for the Management of Radioactive Waste and Irradiated Fuel".
[10] The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.

transporter, the drop of a DSC during on-site transfer was conservatively assumed to result in 100 percent failure of the fuel elements inside a DSC.

Consequently, the free inventory from 10,752 failed fuel elements is assumed to be released into the environment. The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements is assumed to be released into the DSC cavity. As Phase II is closer to the site boundary the assessment conservatively assumes that the drop of an unwelded DSC occurs during transfer from the PWMF Phase I site to the Phase II site. In reality, all DSCs transferred from PWMF Phase I to Phase II are already seal welded.

Assuming that this event occurs at or near the PWMF Phase II site, the total dose to the public was calculated to be 6.50 μSv for an adult and 7.91 μSv for an infant at the Pickering site boundary. The dose to a NEW would be 5.00 mSv.

4.4.2 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Processing

The processes and systems taken into account for this assessment (see Appendix B) encompass those at the DSC processing building once the transporter arrives at the PWMF with a loaded DSC and before the DSC is taken to storage, as described in Chapter 3.

Table 4-3 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible during DSC processing. Full details of this assessment are described in Appendix B.

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 percent of the fuel elements inside the DSC are damaged and the free inventory from 3,226 failed fuel elements is released from the PWMF into the environment.

The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements 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 and worker due to this event is bounded by the DSC drop during transfer discussed in Section 4.4.1.

Table 4-3: Postulated Malfunctions or Accidents during Dry Storage Container Processing

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Drop of a DSC during handling	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Equipment drop onto a DSC	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
DSC collision during craning	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Transporter collision with a loaded DSC or another Transporter	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Equipment collision with a loaded DSC during craning	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Criticality	incredible	—	—	—
DSC Processing building fire	credible	0	0	0
Earthquake	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Tornado	incredible	—	—	—
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—
Release of oxidizing, toxic, corrosive gases and liquids stored in the Processing Building	credible	0	0	0

4.4.3 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Storage

Once the DSC processing is completed, the transporter moves the DSC from the DSC processing building to one of the DSC storage buildings for storage.

Table 4-4 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (that is, with a frequency of occurrence that is $\geq 10^{-6}$ events per year) during the storage process. Details of the assessment for each event are given in Appendix B.

Note that the storage buildings are not safety related structures credited in the containment of radioactive releases (OPG17a).

Table 4-4: Postulated Malfunctions or Accidents during Dry Storage Container Storage

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Seal weld failure during storage	incredible	—	—	—
DSC drop during transfer to storage	credible	$<6.50\times10^{-3}$	$<7.91\times10^{-3}$	<5
Transporter collision with a DSC or another Transporter	credible	$<6.50\times10^{-3}$	$<7.91\times10^{-3}$	<5
Criticality	incredible	—	—	—
DSC storage building fire	credible	0	0	0
Earthquake	credible	$<6.50\times10^{-3}$	$<7.91\times10^{-3}$	<5
Tornado	credible	$<6.50\times10^{-3}$	$<7.91\times10^{-3}$	<5
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Hazardous Material Building explosion	credible	0	0	0
Toxic Material stored in SB3	credible	0	0	0
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—

4.5 Occupational Safety Assessment

4.5.1 Radiation Protection Program

The OPG RP Program applies to all OPG nuclear facilities including the PWMF and is discussed in Section 7.2. ALARA dose targets set under the RP Program are reviewed annually for continuous improvement. The RP Program provides regulatory framework and standards of performance for all the staff and operational activities.

4.5.2 Worker Dose Assessment

In worker dose assessments for operations at PWMF through conservative bounding assumptions, it has been demonstrated that the facility can be operated well within regulatory dose limits.

Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported during the 2007-2017 period are 11.2 person-mSv and 1.6 mSv. Worker doses are submitted to the CNSC as a part of the Pickering Waste Quarterly Operations Reports.

4.6 Summary of Pickering Waste Management Facility Safety Assessment

This radiological safety assessment for the UFDS process at the PWMF has addressed worker and public doses under both normal and abnormal operating conditions, and credible accident conditions.

Based on the 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 boundary, are also well below the CNSC public dose limits.

Occupational doses arising from UFDS operations are managed under the RP Program and are expected to be well below regulatory dose limits.

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

4.7 Long-Term Integrity of Used Fuel and Dry Storage Container

4.7.1 Used Fuel Integrity

Dry storage is a passive mode of storage for used fuel and is the preferred interim option for the long-term management of used fuel at OPG. A requirement of dry storage is that the fuel will not be adversely impacted by the storage conditions to ensure that the fuel remains structurally sound, retrievable and can be safely handled during subsequent steps in its management.

The CANDU fuel bundle is a circular array of fuel elements made of sealed welded Zircaloy tubes known as “cladding” and containing inside the tubes a stack of natural UO_2 fuel pellets. The cladding wall thickness is about 0.42 mm thick allowing the cladding to collapse onto the fuel pellets under the operating pressure of the coolant of about 10 MPa, thus, allowing an efficient thermal contact with the fuel allowing a high heat transfer from the fuel to the coolant. The fuel elements are held together by two endplates resistance welded to the tubes end caps, thus, providing stability and structural strength to the bundle. The fuel elements are 0.5 m in length and 15.2 mm in diameter for the 28-element Pickering bundle and 13.2 mm for the 37-element type bundle. Both bundles are about 10 cm in diameter.

Appendages in the form of bearing pads and inter-element spacers are brazed to the Zircaloy cladding at designated locations to maintain the bundle configuration.

They also allow ease of handling during in-reactor service and subsequent management operations.

Key to the performance of the bundle is its structural integrity as an array of elements and the integrity of the Zircaloy cladding. The cladding acts as the primary barrier for the containment of the fission products generated during the bundle in-reactor service.

To allow the future safe retrieval of the fuel from dry storage, the used fuel bundles need to remain structurally intact and retain sufficient strength to sustain the stresses associated with future handling operations, transportation and disposal. The integrity of the used fuel sheath is also a key requirement for radiological safety. As discussed in Section 4.1.1, the used fuel matrix and the Zircaloy sheath provide a primary barrier to prevent the release of radionuclides. Although the lid seal welded DSC provides containment for any radionuclides released by the fuel, retrieval and other future operations will be simpler and safer if the fuel sheath retains its integrity.

4.7.1.1 Processes that May Affect Used CANDU Fuel Integrity during Dry Storage

During the fuel bundle in-reactor service, the structural materials of the fuel bundle are affected by the bundle irradiation history, the bundle burnup and power rating and the release of radionuclides from the fuel pellets. The fission processes in the fuel result in the formation of radionuclides or fission products in the form of gases, volatiles and solid phase particles that remain contained in the fuel pellets and the fuel elements. Additionally, the high temperatures experienced by the fuel during radiation contribute to thermo-mechanical stresses that may lead to mechanical deformations – albeit nearly imperceptible - of the fuel elements and the fuel assembly.

Since the bundle remains a dynamic entity due to the decay of the radionuclides and the heat power of decay, a number of processes have been postulated that could have an impact on the long-term behaviour of the fuel. The following mechanisms can be listed:

(a) Oxidation of the sheath;

(b) Fast brittle fracture;

(c) Creep rupture;

(d) Stress Corrosion Cracking (SCC) by iodine and metal vapour embrittlement by cesium and cadmium;

(e) Effects of hydrogen/deuterium migration;

(f) Delayed hydride cracking; and

(g) UO_2 oxidation of defected fuel.

The possibility that any of these processes might contribute to the degradation of the fuel has been studied for CANDU fuel since the early 1980's. Results from these studies indicate that for the conditions of storage, these processes are not expected to have an impact on the long-term condition of the fuel.

Early studies by (BYRNE84) and (HUNT81) addressed the potential impact of these postulated mechanisms on the long-term integrity of the fuel. The (BYRNE84) analysis concluded that for fuel dry stored at temperatures below 200°C:

1. Oxidation of the Zircaloy sheath when the fuel is exposed to air was found to be negligible. A 0.04 percent thickness loss of the sheath thickness was predicted for the first 100 years of storage and 0.4 percent in 1000 years.
2. Failure by fast fracture occurs preferentially by plastic collapse rather than brittle fracture.
3. Failure of cladding due to fast fracture by plastic collapse will be insignificant. Typical bundles representative of the CANDU fuel population will only fail if cracks with a depth of 92 percent of the sheath thickness are present.
4. Typical CANDU fuel will be unaffected by stress rupture for at least 106 years.
5. Migration of hydrogen to the endplate/endcap welds is negligible and will not contribute significantly to a decrease in its mechanical strength.

(HUNT81) developed a database for both SCC by iodine and vapour metal embrittlement by cesium and cadmium of the Zircaloy sheath by stressing irradiated fuel sheath rings in atmospheres of these elements contained in glass vials at various concentrations relevant to the expected concentrations in the fuel element gap. The results of those tests indicated that unless a 95 percent through the wall crack is already present in the sheath, an unlikely event, both SCC and vapour metal embrittlement will not be operative at the higher temperatures of storage relevant to dry storage of CANDU fuel.

For intact fuel, creep in the sheath material could potentially be the limiting factor for fuel sheath degradation. However, when used fuel is stored in a helium atmosphere, temperatures of up to 300°C can be considered safe for the planned storage period for intact used fuel in DSCs (PEEHS91). The upper temperature limit ensures that creep strain remains within acceptable limits and the inert gas precludes oxidation processes. The temperature limit of 300°C offers a conservative safety buffer to CANDU fuel stored in the DSCs, which is normally below 150°C. (CANN02) also studied the impact of creep on fuel in dry storage and concluded that for at least 300 years, creep will not be operative.

The robustness of the CANDU fuel bundle as an assembly has also been demonstrated early on by shock and vibration studies (FOREST82) as well as repeated monitoring of fuel performance during transportation which has repeatedly confirmed the robustness of the welds.

To further confirm the integrity of the bundles when stored both under water (wet) and dry conditions, and their ability to be handled safely post storage for transportation and disposal, former Ontario Hydro and AECL initiated a comprehensive program of research and testing in 1977 for the wet storage of used fuel (Wet Storage Program), and, in 1980 for the dry storage of the fuel (Dry Storage Program). Commercial CANDU fuel with burnup and linear power ratings bounding the characteristics of the fuel population in wet storage at the nuclear stations were selected for the programs. Fuel elements of designated bundles were intentionally defected by drilling a pin-hole through the sheath to follow any potential degradation of the UO_2 pellets from oxidation.

The fuel was characterized initially in great detail prior to their storage and withdrawn at regular intervals for re-examination to detect any significant deterioration that could affect their condition in the long term. Since their last examination, the fuel remains available for further testing if required. Main tests used to characterize the fuel and determine if there was any degradation of the zircaloy sheath and the UO_2 pellets included:

- Visual examination;
- Profilometry;
- Metallographic examination;
- Scanning Electron Microscopy, neutron radiography, and x-ray analysis;
- Fission gas analysis;
- Hydrogen and deuterium analysis;
- Ring tensile tests to determine increase susceptibility to SCC from iodine; and
- Torque tests of the end cap/end plate welds to determine any changes in their structural strength.

After over 27 years in storage, results of the Wet Storage program indicated that no significant degradation of the fuel had taken place during its storage. It was concluded that fuel could remain in wet storage for at least 50 years without any significant adverse changes that could compromise its handling or integrity in subsequent steps of fuel management including dry storage, transportation and disposal (FROST84; WASYWICH91).

The Dry Storage Program consisted of three experiments: Controlled Experiment (CEX) CEX-1, CEX-2 and Easily Retrievable Basket (ERB). The fuel was stored in baskets located in designated concrete canisters at the AECL-Whiteshell facilities. The fuel was stored:

(1) In the CEX-1 experiment in air at 150°C;

(2) In the CEX-2 experiment in steam at 150°C; and

(3) In the ERB air at ambient temperatures.

The CEX-2 experiment is of minor relevance to the OPG dry storage system and will not be discussed any further.

The results of the CEX-1 experiment and ERB indicated that for undefected fuel, the storage conditions did not lead to any significant degradation of the fuel. In all instances, the torque tests confirmed the robustness of the end plate/end cap welds and did not detect any significant mechanical changes to the welds.

In the case of the intentionally defected fuel, the bulk of the UO_2 remained intact but some changes to the appearance of the UO_2 matrix were observed in the CEX-1 experiment with limited conversion to U_3O_7 after about 10 years in storage. Bulk oxidation of the UO_2 matrix to U_3O_8 was not observed and there were no diametral changes detected of either the UO_2 matrix or the sheath that could have led to the sheath splitting. In summary, the results of the Dry Storage Program corroborated the assessment studies and provided evidence about the safety of storing CANDU fuel dry.

The conditions of fuel storage in the DSC are more benign than the conditions of fuel storage in the Dry Storage Program since helium, an inert gas, is used as the storage medium. Further, no defected fuel is intentionally stored in the DSCs.

Since both sheath creep and fuel matrix oxidation are temperature-dependent processes, the temperature of the fuel in dry storage is an important factor in the assurance of fuel integrity and safety. The provisions used to maintain used fuel integrity during storage include seal welding of the DSC and the addition of an inert helium atmosphere in the DSC cavity. Low temperatures during DSC processing and storage keep the rate of sheath creep sufficiently slow to prevent sheath rupture over time, and minimize the rate of used fuel oxidation for elements with existing sheath defects and available oxygen.

Further, at OPG facilities, fuel is placed in dry storage after a minimum of 10 years of storage in the IFBs. Analysis and measurements carried out at the PWMF indicate that the maximum fuel cladding temperature is not expected to exceed 150°C in dry storage. At this temperature, UO_2 oxidation is sufficiently slow for residual oxygen in the fuel cavity to be of any concern (FC16). Additionally, the DSC seal weld boundary is helium leak-tested. This ensures that the helium atmosphere is retained in the DSC cavity through the entire storage period and that there is no in-leakage of oxygen into the DSC cavity.

A worst case scenario leading to fuel degradation of a fuel element with an existing sheath defect has been analyzed (OPG99). This case assumed a fuel element in which the ~ 1 cc volume of the gap between the fuel pellets and the cladding was filled with water. Radiolysis of the water content was assessed to yield ~ 0.9 g of oxygen. If this oxygen were to combine with the uranium fuel matrix at low temperatures, the resulting increase in pellet diameter should not result in damage to the fuel sheath.

The above considerations support the conclusion that under normal operating conditions DSCs provide safe and retrievable storage for OPG's used CANDU fuel.

4.7.2 Dry Storage Container Integrity

The DSC is designed to provide a design life of at least 50 years and to meet all shielding and containment integrity requirements over this period. Shielding is provided by the high-density concrete and containment is provided by the steel inner liner, the bottom plate of the lid, the base perimeter flange, the lid locating pin housings, the drain port plug, and all seal welds.

4.7.2.1 High Density Concrete

The reinforced high-density concrete of the DSC has been assessed for potential degradation resulting from elevated temperature, low temperature, radiation fields, presence of water and chemical reactivity. The potential effects are a reduction in mechanical properties (compressive strength, modulus of elasticity), cracking and chemical attack. It has been concluded that the concrete of the DSC will provide at least 50 years of service even when subjected to conservative conditions; for example, higher heat load and thermal gradient than actually experienced during storage.

4.7.2.2 Steel Components and Welds

The steel of the DSC has been assessed for potential degradation resulting from temperature, radiation fields and corrosion. It has been concluded that there is no metallurgical factor or combination of factors which limit the expected service life for the DSC to less than its design life, provided the integrity of the surface coating is maintained. The surface coating is expected to provide at least 50 years of service when stored indoors and adequately maintained.

4.7.2.3 Aging Management Program

OPG has implemented an Integrated Aging Management Program (N-PROG-MP-00008), the purpose of which is to determine the life-limiting characteristics of the critical components and to provide timely detection and mitigation of significant aging effects. The goals of the Integrated Aging Management Program are achieved by establishing a set of programs and activities which ensure performance requirements of all critical equipment are met on an ongoing basis. Aging Management requires life cycle plans, to assess end of life and identify an aging management strategy, and ongoing condition assessments. This program provides assurance that the design service life will be achieved.

4.7.2.4 Inspection and Maintenance Program

The DSCs have been designed to remain maintenance-free for many years. Visible parts of containers are, however, checked for corrosion and paint deterioration on a regular basis. The weld area is specifically inspected for visible signs of degradation. If corrosion is observed on a DSC, the affected area is cleaned up and recoated with the specified touchup paint or repaired as needed.

A limited number of DSC bottoms are inspected annually as per the DSC base inspection plan.

4.7.2.5 Used Fuel Retrievability

OPG recognizes that DSCs may have to be retrieved and the fuel may have to be placed back into the IFB. Therefore, the design of the DSC and processing and storage processes fulfill the requirement that the fuel is safely retrievable from storage.

Retrieval of fuel from a DSC would be a reversal of the loading sequence. The basic steps would be:

(a) Remove drain plug weld by grinding or milling.

(b) Sample interior atmosphere;

(c) Remove lid weld by grinding or milling;

(d) Clamp lid to base;

(e) Return DSC to IFB; and

(f) Unload used fuel from DSC.

The technology is readily available and detailed procedures would be developed when required after securing appropriate approvals. Availability of sufficient space at the IFB would be a pre-requisite to such a transfer.

5.0 RETUBE COMPONENTS STORAGE

5.1 Retube Components Storage Description

The purpose of the RCS area at the PWMF is to provide storage for components removed during retubing of the Pickering NGS A reactors. Reactor components become radioactive during their residence in the core, due to neutron activation and deposited contamination.

The retube components (including pressure tubes, end fittings, garter springs, shield plugs and miscellaneous identified components) have been loaded into specifically designed and shielded Dry Storage Modules (DSMs) for interim storage at the PWMF. The DSMs are large cylindrical casks (see Figure 5-1), made of reinforced heavy concrete and thick carbon steel inner and outer liners. A bolted and gasketed shield door is used to seal the fill port on each DSM. Saddle supports are used to hold each DSM in a horizontal position. The DSMs are stored outdoors in the fenced RCS area.

All four Pickering NGS A reactors were successfully retubed during the Large Scale Fuel Channel Replacement Program (LSFCRP). DSMs containing components from Pickering NGS Unit 1 (P1) and Unit 2 (P2) were loaded inside the station and placed into storage at the RCS area between 1985 and 1988. Unit 3 (P3) and Unit 4 (P4) DSMs were loaded and stored between 1990 and 1993.

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

5.1.1 General Lay-Out of the Retube Components Storage Area

The RCS area is located in the southeast corner of the Pickering NGS site and within the PWMF Phase I area. The physical location of the area is shown in Figures 1-1 and 2-1. The facility is designed to accommodate 38 DSMs. At present it has 16 DSMs from retubing Units 1 and 2 and 18 DSMs from Units 3 and 4, plus two empty DSMs and two vacant storage positions. A chart of DSM contents is provided in Appendix D.

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 supported on compacted crushed stone overlying fill, composed of coarse to fine sand and gravel.

The storage area is paved and further covered with a polymer membrane coating to provide an easy to maintain surface. A drainage system is provided to direct the runoff water from the storage area to the Pickering NGS B outfall, with catch basins permitting periodic sampling of the water.

The dose rates from the storage facility are monitored by Thermoluminescent Dosimeters (TLDs) placed at two locations on the perimeter fence.

5.1.2 Dry Storage Module Description and Performance

5.1.2.1 Dry Storage Module Description

The DSMs (the P1/P2 design is shown in Figure 5-1) are cylindrical casks made from reinforced heavy concrete (nominal density is approximately 3.5 Mg/m^3). A 6.4 mm thick carbon steel liner forms the outer shell. The inner liner, which acts as formwork during construction, is 6.4 mm thick carbon steel for P3/P4 DSMs. For P1/P2 DSMs, a 3.2 mm thick galvanized corrugated pipe forms the inner liner. DSMs are designed to be leak resistant, employing welded construction. The DSM is a safety-related structure because failure of the DSM to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (OPG17a).

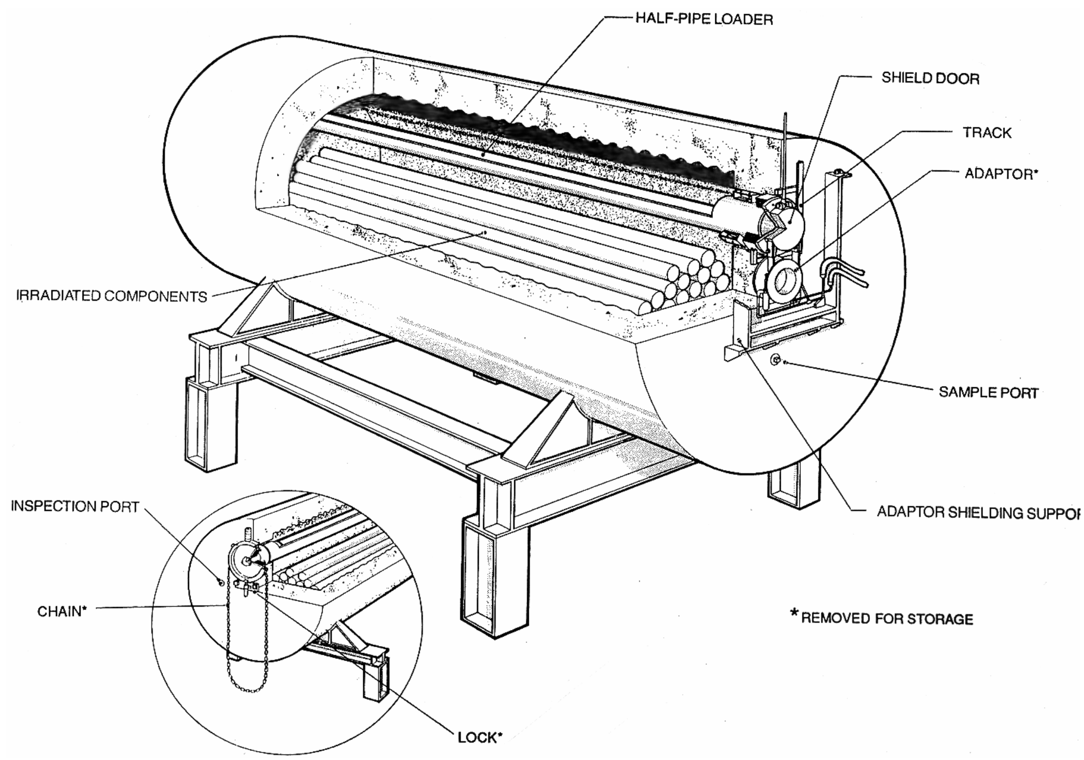

Figure 5-1: Dry Storage Module

In order to provide adequate shielding to meet dose rate requirements outside the facility and to keep worker dose rates ALARA, the perimeter and end walls of each DSM are made of 0.57 m thick heavy concrete.

Each DSM is 3.3 m in diameter and has an overall length of 7.6 m. To prevent deterioration and minimize maintenance, the outer steel shell is coated with two coats of ceramic elastomeric paint.

The DSMs are designed for horizontal loading, with a fill port and a loading mechanism running the full length of the module. Each DSM can hold about

90 pressure tubes or various combinations of pressure tubes, end fittings and miscellaneous components. After the loading operation is complete, the fill port is sealed with a bolted and gasketed shield door. The gasket is made of silicone rubber.

Each DSM additionally 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.

[redacted]

5.1.2.2 Module Integrity

The DSMs are designed for outdoor storage for a minimum of 50 years. The module is designed to withstand the following loads:

(a) Dead load of irradiated components up to 23 tons;

(b) A uniformly distributed wind load of 21 lbs/ft^2 and a uniformly distributed snow load of 44 lbs/ft^2;

(c) The forces due to dropping of irradiated components from the half-pipe unloader into the module; and

(d) The forces caused by the rotation of the module from the vertical to horizontal position when picked up at the centre.

The outer metal shell of the module prevents the ingress of moisture into the heavy concrete and, therefore, protects the heavy concrete from freeze-thaw action. In addition, rapid freeze-thaw cyclical testing of heavy concrete specimens (far more severe than the conditions to which DSMs would be actually exposed) has concluded that the DSMs have satisfactory resistance under such conditions (OH89).

The DSMs are capable of withstanding the thermal stresses arising from the heat load in the module. The heat load due to the radioactive decay of isotopes has been calculated to be less than 1,200 Watts, considering 150 pressure tubes and 300 shield plugs loaded in a module. The thermal stresses due to this heat load on a DSM are negligible.

The silicone rubber gasket used for sealing the DSMs is capable of withstanding an absorbed radiation dose of 1.0×10^5 Gy. Since the gasket is located outside the DSM shielding, the radiation dose over 50 years would be less than 100 Gy.

5.1.2.3 Irradiated Components

The irradiated components stored from Units 1, 2, 3, and 4, consist of pressure tubes, the inboard section of the end fittings, garter springs, shield plugs and miscellaneous identified components. In order to facilitate handling and to reduce the potential spread of radioactive contamination, the shield plugs were pushed into the pressure tubes and the pressure tubes, along with their garter springs, were inserted into waste containers.

The pressure tube waste containers are made of carbon steel and their dimensions are approximately 0.16 m in diameter and 6.2 m in length. The end fittings from P1/P2 were bagged but for P3/P4 these were also put in containers (0.2 m outer diameter) in order to control loose contamination. All the end fittings were cut into two parts, and only the irradiated inboard sections were packaged in containers and stored in the modules. The components are corrosion-resistant. Corrosion of the stored components is, therefore, expected to be negligible over the design life of the module.

During the P1/P2 retube, carbon-14 in particulate form was found to have adhered to the components removed from the reactor. However, specific procedures were in place to remove carbon-14 from the annulus gas system before component removal for the P3/P4 project. With these provisions, there should be less carbon-14 in the P3/P4 storage modules than in the P1/P2 storage modules. This is discussed further in Appendix C. A description of the components stored in the DSMs is given in Table 5-1.

Table 5-1: Irradiated Components for Storage[11]

Quantity per Reactor Unit	Component	Diameter (approximate)	Length (approximate)	Weight (approximate)
390	Pressure Tube	0.114 m	6.1 m	73 kg
780	Shield Plug	0.102 m	0.9 m	41 kg
780[12]	End Fitting (inboard section)	0.172 m	1.8 m	109 kg
780	Garter Springs			

[11] Some components were sent to Chalk River Nuclear Laboratories (CRNL) for research purposes. The actual number of stored components would, therefore be less.

[12] 390 end fittings for P1/P2.

5.1.3 Long-Term Performance

5.1.3.1 On-Going Inspection and Maintenance Program

The DSM has been engineered and designed for minimum upkeep over its design life. Periodic inspection and maintenance is 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 touch-up paint or repaired as needed.

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

An aging management plan is in place, which includes detailed annual visual inspections of DSMs.

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6.0 RETUBE COMPONENT STORAGE SAFETY ASSESSMENT

6.1 Introduction

This chapter provides a summary of the PWMF RCS area radiological safety assessment for the continued storage of DSMs. Conservative estimates of public dose rates due to releases resulting from hypothetic failures of DSMs for normal and abnormal operating conditions, and credible accident conditions are also presented.

6.1.1 Safety Assessment Approach

Under normal operating conditions during storage, DSMs are expected to provide reasonable assurance that the waste can be stored without undue risk to workers, members of the general public, or the environment.

RCS waste operations comply with OPG requirements to keep total radioactive emissions under normal operating conditions below regulatory limits and ALARA.

The safety assessment of normal and abnormal operating conditions and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions that are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions

6.1.2 Acceptance Criteria

The radiation safety requirements under normal operation for PWMF are as follows:

- ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.
- ≤ 100 μSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose target for the public at or beyond the PNGS site boundary due to an abnormal event or credible accident is 1 mSv.
- The dose target for a worker due to an abnormal event/accident is 50 mSv.

The target of 50 mSv for abnormal event/accident refers to NEWs. The equivalent target for non-NEWs/members of the public is 1 mSv.

6.2 Radiological Safety - Normal Operating Conditions

6.2.1 Public Dose

Dose to members of the public from normal operation of the PWMF have been determined based on the latest information on radionuclide emissions, representative group locations, and meteorological data.

The maximum dose rate calculated to an individual member of the public is 1.04×10^{-3} μSv/h at the site boundary. This is equivalent to an annual dose of 2.08 μSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 7.23×10^{-4} μSv/h. This is equivalent to an annual dose of 0.72 μSv based on 1,000 hours occupancy; this is a conservative assumption for boaters and fishermen.

These results indicate that the PWMF administrative dose target of ≤100 μSv/y at the station site boundary as set out in Section 1.6.1 is met during PWMF operations. Details of the assessment can be found in Appendix C.

6.2.2 Worker Dose

The actual worker doses received during normal operation of the PWMF are reported in the PWMF quarterly reports. Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported during the 2007-2017 period are 11.2 person-mSv and 1.6 mSv.

The maximum effective dose to NEWs working at the PWMF is well below the regulatory dose limits for NEWs; 50 mSv in a one-year dosimetry period and 100 mSv in a five-year dosimetry period.

6.2.3 Dose Rates from the Retube Components Storage

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 TLDs 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 2007-2017 TLD survey monitoring results, the maximum dose rates are 0.16 μSv/h at the south fence, 0.11 μSv/h at the east fence, 0.12 μSv/h at the west fence and 0.34 μSv/h at the north fence.

6.3 Radiological Safety - Abnormal Operations

The DSMs located in the RCS area are in interim storage. They are occasionally inspected, however, there is no regular handling of the waste. A detailed screening of external and internal events has demonstrated that there is no credible event that will result in a release of radioactive material. Therefore, there are no radiological consequences to workers or the public from continued storage of waste in the RCS area. Details of the hazard assessment for each malfunction or accident are given in Appendix C.

6.3.1 Potential Off Site Consequences

A DSM failure is very unlikely during its proposed 50-year storage life because of 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.

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7.0 SAFETY AND MONITORING PROGRAMS

7.1 Introduction

OPG has established comprehensive occupational, public, and environmental protection programs, including occupational dose monitoring, radiation and contamination monitoring in the workplace, environmental monitoring programs, and conventional safety programs in support of its nuclear facility operations to assure compliance with the Nuclear Safety and Control Regulations, applicable Provincial Legislation, and OPG requirements. Program administration is conducted on a site-wide basis, and encompasses all nuclear facilities at the Pickering NGS site including the PWMF.

The safety and environmental monitoring program elements applicable to the PWMF include the following:

- Radiation Protection (RP) Program;
- Effluent Monitoring;
- ALARA – Occupational Radiological Risks and Safety Management; and
- Occupational Non-Radiological (Conventional) Environment, Safety and Health Management.

Details of these program elements are provided below.

7.2 Radiation Protection Program

7.2.1 Program Overview

An RP program is in place at the PWMF. The program addresses occupational radiation safety and contamination control. It identifies the operations and materials that have the potential to contribute to occupational dose, and provides guidelines to monitor and minimize occupational dose and reduce the potential for contamination in the facility.

The RP program (N-PROG-RA-0013) is implemented through a series of standards and procedures for the conduct of activities within nuclear sites and with radioactive materials intended to achieve and maintain high standards of RP including the achievement of the following objectives:

(a) Controlling occupational and public exposure:

(1) Keeping individual doses below regulatory limits.

(2) Avoiding unplanned exposures

(3) Keeping individual risk from lifetime radiation exposure to an acceptable level.

(4) Keeping collective doses ALARA, social and economic factors taken into account.

(b) Preventing the uncontrolled release of contamination of radioactive materials from the nuclear sites through the movement of people and materials.

(c) Demonstrating the achievement of (a) and (b) through monitoring.

This program complies with the CNSC requirement that all licensees implement an RP program and establish a quality program that meets the specific Canadian Standards Association (CSA) standards for RP programs.

This program is designed to comply with the RP program requirements of the following acts and regulations as applied to licensed OPG facilities and licensed OPG activities:

- General Nuclear Safety and Control Regulations (SOR/2000-202)
- Radiation Protection Regulations (SOR/2000-203)
- Class II Nuclear Facilities and Prescribed Equipment Regulations (SOR/2000-205)
- Nuclear Substances and Radiation Devices Regulations (SOR/2000-207)
- Occupational Health and Safety Act, R.S.O. 1990, Chapter O.1
- Occupational Health and Safety Act, R.R.O. 1990, Regulation 861, X-Ray Safety
- Radiation Emitting Devices Act, R.S., 1985, c. R-1
- Radiation Emitting Devices Regulations, C.R.C., c. 1370

This program is applicable to OPG Nuclear Facilities such as the Pickering Waste Management Facility, including contract and consulting personnel.

The RP program takes its authority from N-CHAR-AS-0002, Nuclear Management System.

Control and measurement of releases of radioactive materials to the environment through nuclear systems is exercised through N-PROG-OP-0006, Environmental Management.

7.2.2 Occupational Dose Control

The dose expenditures arising from routine PWMF operations are monitored and assessed against dose targets. TLD badges are worn as a minimum external dosimetry requirement by personnel involved in all tasks of PWMF operations. In addition PWMF staff are required to wear Electronic Personal Dosimeters (EPDs) for all radioactive or potentially radioactive work, which are used to monitor dose received and alarm when dose or dose rates reach predetermined levels.

Internal and external occupational radiation exposures are kept ALARA through facility design, procedural controls, and the control of access to the PWMF.

Access to the PWMF is limited to designated personnel and those escorted by qualified personnel. Access to the RCS area is restricted via a locked gate. Only qualified personnel are allowed to access the area for the purpose of monitoring and maintenance of DSMs. Inadvertent or unplanned radiation exposures are avoided by means of clear warning signs within the facility for higher than expected radiation fields, when required. As discussed in Section 7.2.5.1, alarming gamma monitors are provided in the DSC processing building where sudden changes in gamma radiation fields could potentially occur.

7.2.3 Contamination Control

Surface contamination on DSCs is minimized by decontamination of the container surface after loading at the station IFBs. During storage, DSCs and DSMs are monitored for loose contamination as discussed below. Any case of loose contamination found is removed by manually wiping with a cloth, or by wet methods if necessary, taking appropriate measures for containment of contamination at the source and personnel protection. Wet decontamination of storage containers has not been required and is not expected to be required at the PWMF.

7.2.4 Zoning

An important means of contamination control is the division of a facility into zones. These are Zone 1, a clean area which may be considered as the equivalent of a normal public access area, and radiological zones of higher number (i.e., Zones 2 and 3) in which the potential for radioactive contamination or radiation exposure exists.

Zoning for the PWMF Phase I site is as follows: the DSC processing building and DSC Storage Buildings 1 and 2 are designated Zone 2; the storage room for helium bottles is accessible only to the outdoors and is designated an unzoned area. The RCS area is designated as a fenced, restricted, radioactive storage area within the unzoned radioactive work area.

DSC Storage Building 3 is located outside the radiologically zoned Pickering NGS protected area, in a separate nearby protected area within the Pickering NGS property perimeter fence. DSC Storage Building 3 and the outdoor area within the

PWMF Phase II site protected area are designated as a Zone 2 area. The entrance kiosk is designated Zone 1.

Radiological controls are in place in accordance with OPG RP Requirements.

7.2.5 Radiological Hazard Monitoring

RP requirements include area gamma radiation monitoring and routine radiological surveys, as well as surface and airborne contamination monitoring. The main objective of monitoring is the timely detection of changes in radiological hazard levels so that appropriate remedial actions can be taken.

Details of the Routine Radiological Surveys are documented in Pickering Waste Management Facility Routine Radiological Survey Instruction, 92896-INS-09071-00002.

7.2.5.1 Gamma Radiation Monitoring

Continuous alarming, semi-portable gamma monitors are provided in the DSC processing building workshop where sudden changes in gamma radiation fields could potentially occur.

Personnel initiated surveys and periodic monitoring of the PWMF are conducted in accordance with facility-specific procedures. Gamma surveys are performed upon each loaded DSC receipt at the PWMF, and during “hands on” work with a DSC or DSM.

Routine gamma radiation surveys are performed at appropriate points covering the entire sequence of UFDS operations to:

- Monitor for overall changes in radiation levels; and
- Initiate corrective action, if needed, as per approved RP procedures to maintain occupational safety standards.

Gamma radiation monitoring is conducted weekly inside the station fence along the DSC processing building, the DSC storage building walls, and at the RCS perimeter fence. Gamma radiation monitoring of the DSMs is performed twice per year.

7.2.5.2 Contamination Control and Monitoring

Surface contamination checks are performed at set frequencies on DSCs and DSMs in storage as well as in the DSC processing building (including offices, washrooms, and coffee shop), and DSC storage buildings. Loose contamination found in these areas is required to be isolated and cleaned up immediately.

7.2.5.3 Airborne Contamination Monitoring

Airborne contamination is monitored by a Continuous Air Monitor in the welding shop. A portable monitor is also used for weekly monitoring of the air inside the DSC storage buildings.

7.3 Environmental Monitoring

Environmental effects due to radioactive releases from the PWMF are monitored under the Pickering NGS Environmental Monitoring Program (EMP). The EMP is designed to measure environmental radioactivity in the vicinity of Pickering NGS from all site sources. Data from the EMP are used to assess off-site public dose consequences resulting from the operation of nuclear facilities at the Pickering NGS site.

Additionally to the site-wide environmental monitoring program, the following PWMF effluent streams are monitored:

- Active ventilation exhaust;
- Active liquid waste; and
- RCS surface area drainage.

7.3.1 Active Ventilation Exhaust Effluent Monitoring

The active ventilation exhaust from the DSC processing building is monitored for radioactive particulates. A continuous effluent sample is passed through a particulate filter that is replaced and analyzed on a weekly basis. No significant tritium, noble gas or radioiodine emissions are expected from the PWMF, and no fixed monitors are provided for these radioactive species. Although no significant particulate emissions are expected, the exhaust is monitored as a precautionary measure since there is a potential for airborne emissions as a result of DSC processing operations such as welding and vacuum drying.

DSC paint touch-up operations involve minimal paint quantities. Paint aerosols, although not expected to be present, are removed through filters before exhausting to the active ventilation system. Due to small quantities, painting methods, and the use of appropriate filtration, no significant emissions of paint materials are expected. Welding fumes from DSC seal welding operations are additionally exhausted through the HEPA filtered active ventilation system; therefore minimal welding contaminants are expected in PWMF emissions. The conventional air emissions are approved under the Pickering Nuclear MOECC air approval.

7.3.2 Active Liquid Waste Effluent Monitoring

Liquid effluent is generated from water collected in the floor drains (e.g., from floor wash water, precipitation such as snow or rainwater that enters the building during DSC transfer and from condensate from the air conditioners). No contamination is expected in the sump water from the DSC operations, however, the sump water is analyzed for tritium and gross beta-gamma activity prior to being transferred to the station's active liquid waste management system to ensure it meets the station's acceptance criteria.

7.3.3 Retube Components Storage Surface Area Drainage Monitoring

Water is sampled from the RCS surface drainage system for gross beta-gamma activity for confirmation purposes. This provides assurance that any radioactive contamination of the surface water originating from the storage area is detected, however, no contamination of the surface water is expected from the DSMs.

7.4 ALARA – Occupational Radiological Safety Management Assessment

An ALARA – Occupational Radiological Risks and Safety Management Assessment (OH94) was originally carried out during the detailed design phase of the PWMF. The purpose of this pre-operational assessment was to identify radiological and non-radiological hazards associated with the UFDS operations, and thereby provide a baseline for radiological safety performance.

This assessment was subsequently updated (OH98b) after the first year in service for the PWMF, to compare actual operational experience against the selected ALARA targets. This resulted in adjustment in the collective and individual ALARA dose targets downwards.

An updated shielding analysis (OPG00c) was performed to re-evaluate dose rates from DSCs and to evaluate gamma radiation dose rates inside the DSC Storage Building 2 and outside the PWMF as a result of increased storage capacity.

In 2003, operating experience at the PWMF was again compared with the ALARA targets (OPG03a). The assessment concluded that doses had remained below the collective and individual worker dose targets and recommended that the same targets be extended to cover operations at both the PWMF Phase I and Phase II sites.

Another post-operational ALARA assessment for the PWMF was conducted in 2007 (OPG07b). The assessment concluded that the design, operational and procedural measures in place at the PWMF for radiation/contamination prevention and control, have been effective in meeting the existing ALARA targets set during its first post-operational ALARA assessment. Annual collective and individual worker doses received as a result of the facility's operations have been well below the ALARA targets. Operational controls and task procedures employed at the facility have been effectively and promptly revised when necessary to further reduce worker dose expenditure in line with the objectives of the RP program.

The 2007 assessment recommended that, in light of the low doses received by workers, the ALARA targets for individual workers be further reduced to 3 mSv/yr.

For collective doses received by workers, the 2007 assessment recommended that the ALARA targets be changed to describe doses received from a single DSC. This would permit the ALARA dose target to reflect DSC throughput, which can change over time. The assessment recommended that the ALARA target for the collective worker dose due to the loading, processing, and storing of a single DSC be 0.3 Person-millisievert (P-mSv), and that the accompanying ALARA target for collective worker dose due to all other activities at the PWMF be 1.4 P-mSv/year. Brought together, the recommended ALARA target for collective worker dose is calculated as follows:

$$(\text{annual DSC throughput} \times 0.3 \text{ P-mSv}) + 1.4 \text{ P-mSv/year}$$

Actual collective doses at PWMF are lower than the bounding ALARA assessments outlined above and are reported in the quarterly operations reports. Aggressive ALARA targets are prepared annually with input from Operations and Radiation Protection and reviewed at a division level on a monthly basis.

7.4.1 Occupational Radiological Hazards Identification and Quantification

Potential occupational radiological hazards associated with PWMF operations can be categorized as follows:

- Chronic radiological hazards associated with normal operations of the PWMF; and
- Acute radiological hazards associated with some normal and abnormal operating conditions of the PWMF.

7.4.1.1 Chronic Radiological Hazards Associated with Normal Operations of the Pickering Waste Management Facility

The following radiological hazards to PWMF personnel may exist throughout the DSC processing, including surveying and monitoring activities. These hazards would consist of potential external and internal exposures arising from (i), irradiated fuel inside the DSC, and (ii), surface contamination on the DSC. The breakdown of these hazards with respect to type, process and location is given below.

7.4.1.1.1 External Gamma Radiation Hazards

Personnel working on or in the vicinity of a loaded DSC are exposed to external gamma radiation from its surface, originating from the used fuel inside. Radiation fields from a loaded DSC have been discussed in Section 4.3.2.1.

7.4.1.1.2 Surface Contamination Emitting Beta-Gamma Radiation

Surface contamination in PWMF operations presents an external radiation hazard exposure potential for fixed contamination, and an internal radiation exposure potential for loose contamination. The primary source of surface contamination for PWMF operations is the contamination picked up by the DSC from the IFB water during the process of used fuel loading. Other potential sources of surface contamination are discussed below.

(a) Process-Generated Surface Contamination on Material and Equipment

Surfaces of loaded DSCs are decontaminated, and checked that no loose contamination is present, before leaving the station IFB. One exception is the DSC bottom that is decontaminated but for reasons of practicality and the minimal contamination hazard involved, no further checks for loose contamination are made to the DSC bottom after it has been successfully decontaminated at the IFB.

The welding and painting process should not generate significant levels of loose surface contamination. The overall extent of surface contamination in the PWMF workshop should be very small.

(b) Surface Contamination on Personnel

Surface contamination on personnel would arise either directly from contact with DSC surface contamination, or indirectly from contact with material and equipment contaminated by the processes discussed above.

(c) Airborne Contamination

Airborne contamination hazards from PWMF operations may present a hazard if DSC loose surface contamination becomes airborne, or through leakage of the DSC internal volume gas (e.g., could contain krypton-85 as well as radioactive particulates). The processes that could potentially give rise to this airborne hazard (via leakage) are:

- DSC draining and drying;
- Transfer clamp and seal removal; and
- DSC back-filling with helium.

An airborne particulate monitor and gamma monitor is used to detect any abnormally high levels.

(d) Liquid Contamination

Liquid contamination in UFDS operations would be a low-level hazard with potential for internal and external exposure. The following sources could potentially generate liquid contamination in the form of the following:

- Gross particulate beta-gamma emitters:
    - DSC vacuum drying;
    - DSC decontamination;
    - DSC processing building drainage; and

    - DSC storage building floor drainage.

- Hot particles (localized, discrete sources of beta-gamma activity):

    - DSC vacuum drying.

7.4.1.2 Acute Radiological Hazards during Normal Operating Conditions

7.4.1.2.1 Removal of Dry Storage Container Drain Plug

It was recognized in the early design phase of the DSC that upon removal of the DSC steel drain plug to connect and disconnect the drainage and vacuum line connection, a rise in the gamma radiation field would occur near the drain port. A design modification consisting of adding a 90° bend to the DSC drain pipe was implemented, and a subsequent shielding calculation indicated a significant reduction in the gamma dose rate at the opening of the unshielded drain port. This has been confirmed with operational experience.

7.4.1.3 Acute Radiological Hazards during Abnormal Operating Conditions

Abnormal or accident conditions have the potential for the release of radioactive material and loss of shielding. Section 4.4 provides a detailed assessment of the potential impact of postulated events both inside and outside the PWMF.

7.4.2 ALARA Targets

The following targets are used as operations performance measures for the PWMF RP Program towards achieving the objective of keeping radiation exposures and contamination hazard risks ALARA.

7.4.2.1 Occupational Individual and Collective Dose Targets

Individuals attached to PWMF operations must comply with OPG requirements for Dose Limits and Exposure Control.

The recommended ALARA target for individual worker dose is 3 mSv/yr. This is reduced from the previous target of 5 mSv/year to set a more effective target based on historically low doses received by workers.

7.4.2.2 Dose Rate Limits at Pickering Waste Management Facility Boundaries

A dose rate target of 0.5 μSv/h on a quarterly average basis has been set at the security fence adjacent to the PWMF, based on the current regulatory annual dose limit for non-NEWs (i.e., members of the general public), with a maximum occupancy factor of 2,000 hours per year. Operational experience has shown that this requirement is met. As per references (OPG02, OPG03d), since 2003, TLDs placed on the PWMF Phase I east perimeter wall have a target perimeter dose rate of 1.75 μSv/h which is equivalent to the dose rate target 0.5 μSv/h described above but calculated for their location at the east perimeter wall.

An administrative dose target of 100 μSv/year has been set at the station site boundary. This target is ten percent of the CNSC annual public dose limit (Table 1-1).

7.4.2.3 Derived Release Limits

The Derived Release Limit (DRL) for a given radionuclide is the release rate to air or surface water during normal operation of a nuclear facility that would cause an individual of the most highly exposed group around the facility to receive and be committed to a dose equal to the annual regulatory dose limit over the period of the calendar year. The dose limit for calculating DRLs for members of the general public is 1000 μSv/year as set out in the Radiation Protection Regulations.

The PWMF uses the DRLs established for PNGS and the associated methodology for determining their calculation are documented in P-REP-03482-00006.

7.4.2.4 Action Levels

7.4.2.4.1 Radiation Protection Action Levels

RP Action Levels have been developed and implemented according to license requirements for the PWMF. Action Levels for PWMF are captured in N-REP-03420-10011, Occupational Radiation Protection Action Levels for Nuclear Waste Management Facilities (OPG17c).

7.4.2.4.2 Environmental Action Levels

Environmental Action Levels for the PWMF, as well as the methodology for their calculation, are documented in P-REP-03482-00006.

7.4.2.5 Contamination Control Targets

No detectable loose surface contamination is permitted in accessible areas. If any contamination is found it is remediated promptly and reviewed for potential reportability by the Facility Health Physicist. Fixed contamination is permitted under limited, controlled circumstances.

PWMF operations must comply with OPG requirements for the monitoring, identification and control of fixed and loose contamination hazards.

7.5 Occupational Non-Radiological (Conventional) Safety and Environmental Management System

7.5.1 Occupational Non-Radiological Health and Safety Management System

NWM and its contractors are committed to the prevention of workplace injuries/illnesses and to continuous improvement in the management of health and safety. The OPG Health and Safety Management System (HSMS) establishes the process requirements that must be implemented and maintained to ensure that health and safety risks to workers are being mitigated. It also outlines the responsibilities of the various levels of the organization to ensure these activities are carried out.

The Health and Safety Management System addresses:

- Occupational conditions and factors that could affect the health and safety of workers, in all workplaces or from work-related activities under the control of OPG.
- Non-occupational health-related conditions and factors that could affect the health of OPG workers where it impacts achievement of OPG's business objectives.
- Contractor safety.

OPG's HSMS program and associated operational control procedures define the requirements for:

Health and Safety Planning

- Hazard identification
- Risk assessment
- Determining controls
- Identification of legal and other health and safety requirements
- Setting objectives, targets and plans

Implementing the Requirements of the Health and Safety Management System

- Identifying roles, responsibilities, accountability and authority
- Setting standards for competence, training and awareness
- Setting standards for communication, participation and consultation
- Setting standards for control of documents
- Defining controls for the hazards
- Defining an emergency preparedness plan

Checking and Corrective Action

- The HSMS utilizes a systematic approach for monitoring and measuring occupational health and safety performance.
- The monitoring and measurement evaluates the extent to which OPG objectives have been met; the effectiveness of controls; reactive measures of performance that monitor ill health incidents (including accidents, near misses etc.) and other historical evidence of deficient occupational health and safety (OH&S) performance.
- An auditing process and procedure has been developed to measure compliance to legal requirements and compliance to the HSMS.
- A procedure has been established for the control of records.
- A procedure has been established for incident investigation/management, corrective action and for capturing operating experience from internal and external sources.

Management Review of the HSMS and OH&S Performance

Corporate Health and Safety annually assesses the need for a management review of aspects of the HSMS to ensure that the system is appropriate and operating effectively. Opportunities are recommended and appropriate action plans are developed.

Inputs to review may include:

- OPG safety performance.
- OPG strategic and operational risk registers.
- Assurance audit findings.
- Corporate and business' health and safety assessment results.
- A review of the extent to which safety objectives and targets have been met.
- Status of safety improvement initiatives.
- Review of stakeholder issues.

Recommendations may include:

- Revisions to the Employee Health and Safety Policy.
- Improvements to the HSMS.
- Revisions to the safety objectives and targets.
- Identification of aspects of the HSMS requiring further assessment.

7.5.2 Environmental Management System

The Environmental Policy (OPG-POL-0021) sets out OPG's environmental performance and environmental management commitments. OPG strives to improve its environmental performance by committing to the following:

- OPG shall establish an environmental management system and maintain registration for this system to the ISO 14001 Environmental Management System standard.
- OPG shall work to prevent or mitigate adverse effects on the environment with a long-term objective of continual improvement in its environmental management system and its environmental performance.
- OPG will work with its community partners to support regional ecosystems and biodiversity through science-based habitat stewardship.
- OPG shall set environmental performance objectives as part of its annual business planning process. Performance against these environmental objectives will be monitored and associated documented information will be maintained.
- OPG shall communicate its environmental performance to employees, governments, local communities, and other stakeholders.

The policy is supported by OPG-PROG-005, Environmental Management System and N-PROG-OP-0006, Environmental Management. The Environmental Management System (EMS) defines the requirements for:

Planning the work

- Environmental aspect identification associated with our activities, products and services
- Risk assessment
- Identification of compliance obligations
- Setting objectives and targets

Implementing the Requirements of the Planning Process

- Identifying roles, responsibilities, accountability, and authority
- Setting standards for competence, training and awareness
- Setting standards for control of documents
- Defining requirements for communications to internal and external stakeholders

- Defining procedures and operating criteria associated with OPG significant environmental aspects consistent with the policy, objectives and targets

- Defining an emergency preparedness plan

<u>Monitoring, Measurement, Analysis, and Evaluation of the Planning Process</u>

- A systematic approach for monitoring, measuring, analyzing, and evaluating environmental performance is an integral part of the EMS.

- The extent to which OPG objectives have been met is monitored and measured. The effectiveness of controls; environmental incident monitoring and response (including spills, near misses etc.) and other historical evidence of deficient environmental performance, feed into the analysis and evaluation of process effectiveness.

- An auditing process and procedure has been developed to measure compliance to legal requirements and compliance to the EMS.

- A procedure has been established for the control of records.

- A procedure has been established for incident investigation/management, corrective action and for capturing operating experience from internal and external sources.

<u>Management Review of the EMS and Environmental Performance</u>

OPG management reviews the EMS on planned intervals to ensure its continuing suitability, adequacy and effectiveness. The review includes assessing the opportunities for improvement, the need for changes to the EMS, including the Environmental policy and EMS objectives and targets.

Inputs to the reviews include:

- Actions from previous management review meetings

- Changes in internal and external issues, including compliance obligations, risks and opportunities

- Significant Environmental Aspects

- Environmental Performance and Objectives

- Results of audits and evaluations of regulatory compliance

- Status of Corrective Actions

- Adequacy of resources, including Environmental Policy review

- Feedback from interested parties
- Opportunities for continual improvement

Outputs from management reviews shall be consistent with OPG commitment to continuous improvement and includes decisions and actions related to changes to:

- Environmental policy
- Objectives and targets
- EMS documentation
- Other elements of the EMS.

7.5.3 Hazardous Materials

OPG Nuclear programs are in place for the control and safe handling of hazardous materials. The handling of hazardous materials must meet provincial legislation, particularly the Occupational Health and Safety Act and the Environmental Protection Act. Material Safety Data Sheets for hazardous materials are readily available to PWMF staff as required by Workplace Hazardous Materials Information System (WHMIS) legislation.

The PWMF contains a variety of non-radiological hazardous materials typically found in industrial buildings, including the following:

(a) Compressed gases: The main gases used include an inert weld cover gas discussed in Section 3.4.4.3.1, and helium, discussed in Section 3.4.4.3.4.

(b) Paint: As described in Section 3.4.4.4, touch-up paint is applied to areas on the DSC that have been affected by the welding process and to scrapes or scuffs on the DSC that may have resulted from handling.

(c) Consumables for maintenance: These include items such as welding rods, adhesives, paints, abrasives, various solvents, and lubricants for equipment.

(d) Janitorial and cleaning supplies.

7.5.4 Fire Protection

OPG has a process in place to ensure adequate fire protection at the nuclear waste facilities. The purpose of this process is to minimize both the probability of occurrence and the consequences of fire at these facilities.

The objective of this process embodies the commitment to:

(a) Protect OPG personnel, contractors, and visitors from the hazards of a fire;

(b) Ensure that a fire does not significantly increase the risk of radiological releases or other hazardous substances to the environment and public; and

(c) Minimize economic loss and delays in OPG operations resulting from a fire.

As part of design, each NWMD facility maintains fire protection documentation regarding the physical details of fire prevention, detection and suppression systems and features of the facility. OPG will design and install all fire protection systems in accordance with the NFCC and NBCC and all applicable CSA and NFPA standards.

For fire protection during operations, Fire Safety Plans, as required by the NFCC, are prepared for each licensed facility to describe the fire protection measures at the facility and reviewed periodically to ensure they remain current. The Facility Fire Safety Plan identifies the specific type of fire detection and suppression systems located in various buildings or areas. Procedures and maintenance call-ups are in place to ensure the operations, maintenance, testing, and inspection of fire related activities and equipment meet the requirements of the NFCC, the NBCC and applicable CSA and NFPA standards. All staff involved in the responsibilities of the fire protection procedure are qualified to the requirements identified in the relevant OPG governance.

Pre-Fire Plans and Evacuation plans are prepared for all OPG's nuclear waste facilities, identifying the building floor layout, and location of fire related information (e.g., location of extinguishers, fire hoses, fire exits, etc.) These are reviewed periodically and used by the emergency responders in the event of a fire.

Emergency response services at the PWMF are provided by the Pickering NGS as identified in the Partnering Agreements between Pickering Nuclear and PWMF, as stipulated in the Memorandum of Understanding with the City of Pickering. PWMF Phase II has services provided by Pickering Fire Services, as stipulated in the Memorandum of Understanding with the City of Pickering.

Fire drills are regularly scheduled to test occupant responses and test the capabilities of the ERT as per the NFCC.

8.0 NUCLEAR MANAGEMENT SYSTEM

OPG has an on-going program in place to assure that the required quality of products and services is properly defined and effectively achieved in activities associated with OPG's nuclear facilities. This program provides a disciplined approach in determining, communicating, and attaining the required level of safety, reliability, maintainability, environmental protection, and performance for OPG's nuclear facilities. The program defines requirements for the work to be done and provides for the integration and co-ordination of pertinent activities.

The Nuclear Management System (N-CHAR-AS-0002), in conjunction with Nuclear Waste Management program document (W-PROG-WM-0001), the Nuclear Safety Policy (N-POL-0001), and the Health and Safety Management Program (OPG-PROG-0010) establishes the overall managed system requirements for the management and operation of OPG's nuclear waste facilities.

The Nuclear Waste Management program (W-PROG-WM-0001) encompasses the operation of the nuclear waste facilities. This program also establishes the overall system for nuclear waste facility operations and incorporates, directly or by reference, the controls necessary to meet the requirements of CSA N286-12 and ISO 14001.

The Independent Assessment program (N-PROG-RA-0010) provides independent assessment (internal and external) processes to perform a comprehensive and critical evaluation of all activities affecting OPG nuclear facilities, including OPG's nuclear facilities. These independent audits and assessments cover the overall Nuclear Management System requirements. The independent audits and assessments monitor compliance with governing codes, standards and technical requirements, and confirm that program requirements are being effectively implemented. Independent audit and assessment results are documented, reported to and evaluated by a level of management having sufficient breadth of responsibility to assure actions are taken to address the findings thereof, which are performed by the Nuclear Oversight organization.

Additional oversight of OPG's nuclear waste activities is provided through self-assessment and the corrective action program. In particular, the corrective action program assures that adverse conditions are identified, documented, reported, evaluated, and corrected in a timely manner.

8.1 Employee Training

OPG has established and implemented a training program, following a systematic approach to training, to provide personnel with the knowledge and skills necessary to meet the performance requirements of their jobs.

The objective of the training program is to provide sufficient qualified personnel to operate and maintain the nuclear waste facilities in a safe and efficient manner and to ensure compliance with applicable regulations, operating licences and established operational limits.

The skill and knowledge of personnel is developed through initial training and job performance is maintained and improved through requalification and refresher training.

The overall training program and its individual elements are periodically assessed. The training program is updated to reflect the results of program evaluations, internal and external operating experiences and changes to equipment, procedures, and regulations.

9.0 COMMUNITY RELATIONS AND PUBLIC INFORMATION PROGRAM OBJECTIVES

OPG believes in open and transparent communication in a timely manner to maintain positive and supportive relationships and confidence of our key stakeholders

Corporate Affairs staff adhere to the principles and process for external communications as governed by the nuclear standard “External Communications”. This document guides our external community stakeholder activities, public response requirements of issues or significant events and OPG’s standards to respond to concerns expressed by the public. For operational status changes or unscheduled operations that may cause public concern or media interest, OPG follows a Public Interest Notification protocol to notify key community stakeholders in a timely manner. To support this protocol, Corporate Affairs maintains a duty on-call position 24/7 to manage this requirement.

OPG regularly and proactively provides information to the public on its on-going facility activities; public and environmental impact; and nuclear waste transportation program and consults with key stakeholders and the public on future planned activities such as the Deep Geologic Repository (DGR) for low and intermediate level waste.

OPG’s community relations and public information program has been recognized as a strength by national and international utility peers. OPG benchmarks current practices amongst other industries to ensure continuous performance improvement.

9.1 Program Highlights and Accomplishments

Each year a community engagement and consultation plan is developed to support OPG’s business strategy to build community awareness and support of OPG and site operations. The objective of the strategy is sustained company reputation, positive community relations and support for operations. The community relations program proactively provides information to our stakeholders on our nuclear station and waste operations and effects on the community or environment that may result from our operations.

Our community programming reflects our value and commitment to the key tenets of safety, performance, and environment and as a strong corporate citizen within our host community.

OPG’s Corporate Affairs staff manages communications and relationships between the waste facilities and our host communities by fostering healthy, open relationships and sustainable partnerships with community stakeholders, including government, media, business leaders, educational institutions, interest groups, and community organizations. OPG ensures transparent disclosure of our operations and potential impacts, both positive and negative that may occur as a result of our operations.

Additionally, we target key segments of the public with a variety of activities to encourage them to interact with OPG staff and to visit our Information Centres. We view public education as a key component of our program: Two-way dialogue with our community stakeholders and residents is facilitated through personal contact, community newsletters, speaking engagements, educational outreach, the Internet and many other products and programs.

Over the current licensing period, OPG has conducted an extensive and responsive community relations program for OPG/NWM as a whole and specifically for the PWMF.

OPG regularly provides milestones and regular waste management updates to key stakeholders. Presentations are regularly made at the Pickering Community Advisory Committee and Durham Nuclear Health Committee. In addition, presentations and informal meetings are held with local elected officials and community leaders a number of times each year to provide updates on performance and other activities taking place both at the stations and waste facilities and the DGR.

To increase the understanding of nuclear waste management, community presentations and facility tours have been provided at the PWMF, to key stakeholder groups, media and community and community-of-interest leaders, with strict oversight by OPG nuclear security. Any and all station tours also include information and discussion on the management of nuclear waste.

OPG's Corporate Citizenship Program, provides financial support for community-based programs to foster sustainable partnerships and to benefit the social fabric of a community

Nuclear waste management is regularly discussed at community updates with the public as part of the nuclear generation station projects.

OPG also receives and manages inquires raised by stakeholders and the public and has a managed process in place to respond and track actions and resolution of issues. This record is used as a gauge to monitor public attitudes and potential issues.

Public opinion polling in Durham Region for Pickering station and PWMF waste operations following the Fukushima event showed an increase in the level of support for OPG and nuclear power. This speaks to the strength of the relationships and trust OPG has built up with the community and community leaders. Quantitative and qualitative research in Durham two months after the events in Japan point to higher-than-ever perceptions of OPG's station and used fuel operations and nuclear power in Ontario.

A number of media communication products are produced. *Pickering Neighbours* is distributed quarterly to approximately 110,000 residents and businesses in the Municipality of Pickering. Periodically, nuclear waste management articles are highlighted in these editions.

Quarterly Performance Reports are produced for NWMD's waste operations which include performance of PWMF and are shared with key stakeholders and available on the OPG website.

The OPG corporate website www.opg.com provides online access to information on the Nuclear Waste Management program and projects. All waste information brochures and videos are available on the website and provides an opportunity for users to email questions, comments and concerns to nwmd@opg.com. The nuclear waste management community-focused website undergoes continuous improvements and updates to ensure current information on our facilities and to identify opportunities for the community to meet us face to face.

Numerous presentations and discussions are held each year with interested municipalities and communities in which our waste transportation program travels. Presentations are made to first responders to explain the safety of our nuclear waste transportation program and emergency response procedures in place.

Over the past two years considerable interest by the community has been expressed around OPG's proposed DGR and the long-term storage of low and intermediate level waste. Additionally, the activity of the NWMO Adaptive Phased Management (APM) process has elevated interest waste operations and the transportation of waste. This has provided for numerous engagement opportunities to stakeholders and communities to share the story of our nuclear waste operations.

In mid-2012 OPG reviewed the issuance of RD-99-3 *Requirements for Public Information and Disclosure.* To ensure full compliance OPG conducted a review of existing communication protocols and polices in place and revised procedures to ensure compliance. To further guide our actions, OPG has developed a Nuclear Public Information Disclosure and Transparency Protocol. This protocol sets forth our commitment to high standards of information disclosure and reporting. It also ensures information is provided in a timely manner to the communities in which we operate, the public and First Nations and Métis peoples with an interest in our nuclear station and waste operations. This protocol applies to all operations at Pickering, Darlington and Western Waste Management Facilities.

9.2 First Nations and Métis Relations

OPG has a board-level First Nation and Métis policy and active community relations program that focuses on:

- Community relations and outreach;
- Capacity building support with communities;
- Employment/business contracting opportunities; and
- Settlement of past grievances or legacy issues.

Building positive, community-minded relationships with First Nation and Métis is important to OPG nuclear with respect to current operations and the planning of new projects. We recognize close consultation with community members and leaders is an essential part of the process. OPG Nuclear continues to engage in active dialogue with First Nations and Métis people on a number of issues and operational decisions related to our nuclear operations. Discussions and information sharing are undertaken to build long-term mutually beneficial working relationships with Indigenous communities near our nuclear host communities and along our radioactive transportation routes.

Nuclear Waste operations is an important issue for First Nations and Métis people. OPG has on-going discussions and information sharing with Williams Treaty First Nations (Alderville First Nation, Chippewas of Georgina Island First Nation, Curve Lake First Nation, Hiawatha First Nation, and Mississaugas of Scugog Island First Nation) and more detailed discussions with Alderville First Nations, Métis Nation of Ontario and Oshawa and Durham Region Métis Councils. During the recent Pickering licence hearing OPG engaged the Métis Nation of Ontario – Toronto/Durham Councils with an information sharing session, community exchange and tours to discuss station and waste operations. This engagement will continue.

10.0 PRELIMINARY DECOMMISSIONING PLANNING

10.1 Introduction and Background

The PWMF is composed of two sites. The PWMF Phase I site is located within the Pickering NGS protected area, southeast of Pickering NGS Unit 8, adjacent to the east side of the station security fence. The PWMF Phase II site is located approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”, but still within the existing Pickering NGS site boundary. After shutdown of the Pickering NGS in 2024, all used fuel from the station will be loaded into DSCs and transferred to the PWMF until a long term used fuel management strategy is implemented. The contents of the DSMs will also be removed prior to decommissioning of the PWMF facility. The PWMF will remain in operation after the cessation of power production at Pickering NGS, and decommissioned in a subsequent time frame.

This chapter presents an overview of the preliminary planning for decommissioning of the PWMF. A Preliminary Decommissioning Plan (PDP) (OPG16) is in place for the PWMF to fulfill the associated licensing requirements. The PDP is prepared in accordance with guidance provided in CSA N294-09, Decommissioning of facilities containing nuclear substances (CSA14c), and CNSC Regulatory Guide G-219, Decommissioning Planning for Licensed Activities (CNSC00).

An overview of the characteristics predicted to exist at the time of decommissioning is provided in Section 10.5. This section also includes a discussion of the general types of hazards (including hazards potentially impacting the environment) associated with conditions that may be encountered during decommissioning. The decommissioning work required, sorted into work packages or planning envelopes as defined by CSA N294-09 is discussed in Section 10.6. Waste management, safety, human factors considerations, security, safeguards, and quality assurance are set out within the remainder of the chapter.

10.2 Long Term Management of Used Fuel

The Government of Canada passed the Nuclear Fuel Waste Act in 2002. The legislation required nuclear energy corporations to establish the NWMO to study the options available to recommend a long-term management approach for used fuel. On June 14, 2007, the Government of Canada – based on the NWMO’s recommendations – selected Adaptive Phased Management (APM) as the best plan for Canada for safeguarding both the public and the environment over the very long term in which used nuclear fuel must be managed.

APM involves the containment and isolation of used nuclear fuel in a deep geologic repository in a suitable rock formation. Used nuclear fuel will be safely and securely contained and isolated from people and the environment in the repository using a multiple-barrier system.

A detailed discussion of long-term management options for used fuel is beyond the scope of the PWMF preliminary decommissioning plan.

10.3 Long-Term Management of Low and Intermediate Level Waste

OPG is in the process of seeking a licence for the construction of a Deep Geologic Repository (DGR) for L&ILW at the Bruce site. The L&ILW DGR is planned to store operational and refurbishment waste however for planning purposes, it is assumed that any radioactive waste generated from decontamination and dismantling activities will be emplaced in an expanded portion of the L&ILW DGR.

It is assumed that the refurbishment waste stored in the DSMs in the RCS area will be removed prior to the start of decommissioning activities.

10.4 Pickering Waste Management Facility Preliminary Decommissioning Schedule

Decommissioning of the PWMF will occur after removal and transfer of all DSCs from the facility to a long term used fuel management facility. The contents of the DSMs will also be removed and transferred to the L&ILW DGR prior to permanent shutdown and dismantling of the PWMF. A detailed decommissioning plan will be submitted to the CNSC, in accordance with regulatory requirements, with decommissioning being completed within several years subsequent to obtaining the necessary regulatory approvals.

10.5 Predicted Characteristics of the Pickering Waste Management Facility at Decommissioning

This section summarizes the expected radiological conditions present at the PWMF, as well as hazards that may require consideration during decommissioning activities.

10.5.1 Dry Storage Container Processing Building

No significant radiological or chemical hazards are expected to be present in the DSC processing building at the time of decommissioning. General hazards that could be encountered during decommissioning include conventional hazards that would be associated with any structural demolition project.

The processing building has an active ventilation system due to the potential for low levels of contamination being present during DSC processing activities. If contamination is present, it will be detectable by practical survey methodologies and disposed of appropriately.

Due to the nature of operations in the processing building, a variety of consumables are utilized, some of which are classified as hazardous; these are assumed to be consumed or removed prior to the start of decommissioning activities. There is no laboratory or chemical store in the building.

10.5.2 Dry Storage Container Storage Buildings and Retube Component Storage Area

The DSC storage buildings are not expected to become contaminated through PWMF operations. No radiological or chemical hazards are expected to be present in the storage buildings at the time of decommissioning. Nevertheless, OPG will perform a site characterization prior to decommissioning to validate this assumption.

For planning purposes, it is assumed that the contents of the DSMs will be repackaged in certified transport packages for off-site transportation. Special facilities to control contamination and handle repackaging will be built as necessary. The certified transport packages will then be transported off-site with decommissioning ILW from the Pickering NGS dismantling. The empty DSMs will be decontaminated and surveyed. They will then be sectioned or reduced to their constituent materials for disposal as non-radioactive waste, or recycled as scrap.

10.6 Decommissioning Work Required

10.6.1 Decommissioning Planning Envelope

PWMF detailed decommissioning plans may be divided into the following four decommissioning planning envelopes:

- The DSC processing building;
- The DSC storage buildings;
- The RCS area; and,
- The remaining PWMF site (i.e. Phase I and Phase II outdoor surfaces).

Decommissioning approaches are discussed in the following sections.

10.6.1.1 Decommissioning of the Dry Storage Container Processing Building

All equipment, furniture, and supplies in radiologically zoned areas will be surveyed and decontaminated as necessary prior to clearance for non-radiological use, recycling, or disposal. The active ventilation system in the processing building will be dismantled, and depending on the feasibility of decontamination, disposed of as low-level radioactive waste.

Remaining inventories of non-radioactive hazardous materials, such as small quantities of unused paint supplies and remaining chemical waste in the processing building (if any) will be disposed in accordance with the applicable Ministry of Environment (MOE) regulatory requirements in force at the time of decommissioning.

Interior structural surfaces will be surveyed to identify areas requiring decontamination. The characterization survey will determine the nature and extent of contamination, for the purpose of hazard assessment as well as to facilitate and control decontamination work (if required). On completion of decontamination work, a final clearance survey will be carried out.

Non-radiologically zoned office space in the processing building will be appropriately surveyed prior to clearance. If contamination is found in office areas, decontamination, followed by a clearance survey will additionally be carried out.

While there is no reason to expect particulate contamination to be released from the HEPA-filtered active ventilation system, a confirmatory survey of roof surfaces as appropriate will be carried out prior to radiological clearance of the DSC processing building. Drains will be surveyed and if found to be contaminated, they will be removed separately prior to demolishing the building.

10.6.1.2 Decommissioning of the Dry Storage Container Storage Buildings

The DSC storage buildings are expected to be free from radiological contamination. Nevertheless, a radiological survey will be carried out prior to radiological clearance of each structure. Before decontamination or dismantling of structural components takes place, a survey will be carried out to identify the presence of detectable contamination in each building. Since the buildings are unlikely to be contaminated, this initial contamination survey should be designed to meet the full requirements of a final clearance survey.

Spot decontamination, followed by a final survey of potentially affected areas will be carried out if necessary. When the structure has been verified to meet clearance levels, and after CNSC approval, it will be dismantled and demolished for recycle and disposal of structural materials.

10.6.1.3 Decommissioning of the Retube Component Storage Area

As stated previously, it is assumed that the contents for the DSMs will be repackaged in certified transport packages for off-site transportation prior to the decommissioning of the RCS area. The empty DSMs will be decontaminated and surveyed. They will then be sectioned or reduced to their constituent materials for disposal as non-radioactive waste, or recycled as scrap.

Spot decontamination, followed by a final survey of potentially affected areas will be carried out if necessary. When the RCS area has been verified to meet clearance levels, and after CNSC approval, it will be dismantled and the materials disposed as non-radioactive waste.

10.6.1.4 Decommissioning of the Remaining Pickering Waste Management Site (Outdoor Surfaces)

It is unlikely that contamination will be present on outdoor surfaces (i.e. on asphalt or concrete yard surfaces); nevertheless, a final clearance surveys will be carried out across the Phase I and Phase II sites for confirmation on non-radiological status.

10.7 Waste Management

Prior to decommissioning of the PWMF, used fuel will be transferred to a long term used fuel management facility and the DSM contents will be transferred to the L&ILW DGR. For planning purposes it is assumed that all radioactive waste from decontamination and dismantling activities will be emplaced in an expanded portion of the L&ILW DGR.

It is not anticipated that the decommissioning envelopes identified in this section (i.e., the DSC processing building, the DSC storage buildings, the RCS area, and the PWMF outdoor areas) will contain significant quantities of radioactive contamination. Small quantities of low-level radioactive waste could be generated through decontamination and dismantling of active ventilation in the processing building, decontamination of the DSMs and storage structures.

10.8 Occupational, Public, and Environmental Safety

10.8.1 Safety Under Normal and Abnormal Operating Conditions

The design of the PWMF and the policies and principles for its operation serve to protect workers, the public, and the natural environment. Administrative and engineering controls will be established as necessary for the decommissioning of the PWMF, such that dose consequences are within established radiological criteria.

10.8.2 Occupational Safety during Decommissioning

10.8.2.1 Radiological and Conventional Safety Policies

Operational history will provide a very good preliminary indication of whether the potential for contamination exists within buildings, or on yard surfaces. The results of initial facility surveys will identify radiological and conventional safety hazards that may be present when decontamination, dismantling, and demolition work is carried out. Detailed safety assessments for individual work packages will be carried out prior to commencement of decommissioning work, based on PWMF operating history and these surveys.

The principles of radiation protection and occupational health and safety, set out within the operating policies and principles for the PWMF will be applied to site decommissioning work.

Building services such as ventilation, electrical power, fire prevention and control, and drainage will be maintained as required before demolition commences.

10.8.2.2 Occupational Exposure Assessment

Occupational dose limits are not expected to be exceeded throughout decommissioning activities.

10.8.3 Public and Environmental Protection during Decommissioning

OPG radiation protection procedures will be maintained to prevent the unlikely occurrence of loose contamination on persons, packages, or containers leaving the site.

Environmental monitoring of the site and surrounding area will be maintained throughout the decommissioning activities.

The potential public and environmental impacts from transportation of used fuel, DSM contents, and any contaminated decommissioning waste from the PWMF site to the final disposal facilities will be minimized by the use of approved packages or containers, trained staff, and approved procedures. Federal and provincial regulations relating to radioactive waste transportation and disposal will be observed.

In accordance with decommissioning work requirements, final clearance surveys will be carried out to confirm that clearance objectives have been attained for areas and materials released for unrestricted public use.

10.8.4 Non-Radiological Environmental Aspects

Non-radiological aspects of decommissioning nuclear facilities could include dust, noise, and increased traffic resulting from decommissioning and demolition activities. OPG will implement measures to mitigate these and other potentially adverse effects. OPG will assess and address socio-economic effects in the context of conditions pertaining to the affected communities at the time of decommissioning.

10.9 Human Factors

Human factors, such as the change of mission from operation to decommissioning, and possible loss of competence as facility staff leave for other duties, resulting in a small size of operating staff, are not expected to have an impact on the course of PWMF decommissioning. OPG will, however, ensure that human factors are considered throughout the planning and execution of the PWMF decommissioning. Attention will be given to staffing and training in order to minimize potential problems resulting from the loss of experienced personnel over time.

10.10 Security

During decommissioning, OPG will continue to abide by the terms of the security regulations on the physical security of nuclear facilities in force at the time of decommissioning.

10.11 Safeguards

In accordance with an agreement between the Government of Canada and the IAEA, nuclear safeguards are implemented at OPG’s nuclear generating stations. These international safeguards apply to the PWMF until the used fuel is removed from the site. The decommissioning of the PWMF will not be impacted by safeguards requirements since all used fuel would have been removed from site prior to the implementation of the decommissioning program.

10.12 Quality Assurance

OPG will incorporate quality assurance programs to ensure that all appropriate requirements (such as CSA N286) including occupational, public, and environmental protection, are met during the decommissioning of the PWMF.

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11.0 SAFETY AND ENVIRONMENTAL EFFECTS SUMMARY

11.1 Radiological Safety – Normal Operations

11.1.1 Gamma Radiation Dose Rates

11.1.1.1 Dose Rates Inside the Pickering Waste Management Facility

Detailed discussions of dose rates in the PWMF are set out in Chapters 4 and 6 of this report. Under the PWMF RP program, dose rates in work areas at the PWMF are maintained to levels that are ALARA.

Dose rates in routinely occupied areas inside the PWMF are at or near ambient background radiation levels. Although higher dose rates are normally found when walking between stored DSCs and DSMs, they are well below calculated estimates. Workers employ survey monitoring, and appropriate dosimetry is required to be carried out in areas having potentially elevated dose rates.

11.1.1.2 Dose Rates Outside the Pickering Waste Management Facility

As discussed in Section 4.3.2.3, the doses from DSC Storage Buildings 1-3, when filled to nominal design capacity, are estimated at about 2.08 μSv/year at the Pickering NGS site boundary based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the dose is estimated to be 0.72 μSv/year based on 1,000 hours occupancy. This is below the facility administrative dose target at the Pickering NGS site boundary and is expected to be indistinguishable from the variations in natural background.

Compared with the average annual background dose rate of about 1,400 μSv/year in the vicinity of the Pickering NGS site (OPG17b) and the CNSC regulatory dose limit of 1,000 μSv/year, these dose rates are very small.

The total annual background doses to members of the public will vary considerably, from individual to individual, according to lifestyle, housing type and location, occupation, and medical requirements. The typical resident in the vicinity of the Pickering NGS site receives a yearly average effective dose commitment of 565 μSv from radon and thoron daughters, 152 μSv from terrestrial radiation, 306 μSv from internal radionuclides, and 293 μSv from cosmic radiation. In addition to naturally occurring radiation, the public also receives about 70 μSv effective dose from man-made sources such as nuclear weapon test fallout and exposures from technological processes and consumer products and services. About 1,100 μSv on a per capita basis is received from medical exposures (OPG03b).

11.1.2 Radiological Contamination

11.1.2.1 Surface Contamination

The external surfaces of DSCs and DSMs in storage are routinely demonstrated to be free from loose contamination. Similarly, there is no loose contamination in the DSC storage buildings or the RCS area.

There have been no contamination incidents in the PWMF office area or coffee shop.

Minor incidences of loose contamination have occurred related to hoses connected to the DSC vacuum drying system. Dedicated hoses have been assigned to DSC processing systems, to prevent cross-contamination between systems that are directly connected to the internal DSC cavity and the helium leak detection system.

Contamination monitoring will be performed during UFDS operations.

11.1.2.2 Airborne Releases and Liquid Effluents

Active ventilation and active drainage systems are provided in the DSC processing building. Active ventilation exhaust is filtered using HEPA filtration to remove radiological particulates. Monitoring of the stack sampler has routinely demonstrated no significant levels of particulates in the active ventilation exhaust.

The contents of the holding tanks in the active drainage system are monitored and periodically pumped to the existing active liquid waste treatment system at Pickering NGS. Monitoring results show no significant levels of activity in active drainage effluent transferred to the station system.

Under normal operating conditions, no airborne emissions are expected from the seal-welded DSCs during on-site transfer to or storage in the DSC storage buildings. As each DSC is fully vacuum-dried, helium back-filled, seal welded, and leak-tested at the DSC processing building, there will be no liquid emissions expected from DSCs.

Monitoring of the RCS area catch basins has demonstrated that there are generally no detectable levels of activity in surface water runoff from the RCS area.

11.2 Radiological Safety – Abnormal Operating Conditions and Credible Accidents

Under postulated accident scenarios presented in this report, dose rates and emissions from the PWMF are within allowable limits and risk to the public, the workers and the environment is negligible. The combined effects of credible accident scenarios are not expected to exceed the dose rates resulting from bounding scenarios for single effects.

11.3 Cumulative Environmental Effects

Radiation dose resulting from the combined operations of the PWMF is very low in comparison to background levels and is expected to remain well below CNSC limits. The PWMF has been performing satisfactorily for storage of radioactive materials.

Estimated releases during normal operations, based on conservative assumptions, are negligible.

EAs have been conducted for the DSC processing building and DSC Storage Building 1; DSC Storage Building 2; and DSC Storage Buildings 3 and 4. The assessment documented the results of the analysis of potential effects associated with the staged construction and operation of the PWMF as follows:

- Analyses of non-radiological effects on the natural environment;
- Assessment of the potential radiological environmental effects on humans and the non-human environment;
- Public health effects due to both radiological and non-radiological aspects of the facility;
- Socio-economic impacts;
- Environmental effects of alternative options;
- Cumulative effects due to facility construction and operation in combination with the effects of other existing and planned activities; and
- The significance and uncertainty of the assessed environmental effects.

The EAs concluded that the staged construction and operation of the PWMF are not expected to result in any significant incremental effects on the local or regional environment. The assessment indicated no adverse cumulative effects due to conventional pollutants and radiation dose on the valued ecosystem components (VECs). It concluded that cumulative radiation doses to humans from all sources at the Pickering NGS site were expected to remain very low in comparison to background levels, and that the cumulative radiation effects on the most sensitive non-human species would be several orders of magnitude less than the threshold values documented in the literature.

The PWMF is, therefore, considered to have no significant adverse cumulative environmental effects. Also, adverse effects to the environment are not expected to occur during PWMF operations.

11.4 Human Performance and Human Factors

The human performance program establishes a systematic framework for human performance management. The program is specifically designed to achieve higher levels of Nuclear, industrial and environmental safety, higher unit reliability, and reduced operating costs through event-free operation. The goal of the program is to continually reduce human performance events and errors in pursuit of recognition as an event-free operator via consistent application of event prevention tools.

This program addresses human performance management and improvement by improving human performance through individual behaviours (all employees and contractors), organizational processes, and management and leadership behaviours.

Human Factors provides inputs and recommendations to the building and systems designs using a tailored approach that is based on industry accepted Human Factors Engineering processes, standards and guidelines. The aim is to ensure that the storage buildings, their equipment, tasks and work environment are compatible with the sensory, perceptual, cognitive and physical attributes of the people who are operating and maintaining the facility. Human Factors in design is applied to all facility modifications. Radiological and non-radiological issues are considered for improvement of the general operability, maintainability and safety of the facilities.

The training program and practices to support the provision of qualified staff is discussed in Section 8.1.

12.0 CONCLUSIONS

The information presented in this Safety Report addresses the health and safety of workers and the public, and the protection of the environment. It shows that the PWMF is a safe undertaking. Specifically:

- Over the last ten years the hypothetical public dose from all PWMF operations has been less than the administrative dose target. These doses are negligible compared to the total background dose of 1,400 μSv/year in the vicinity of the Pickering NGS site. PWMF's contribution to public dose is small.

- Accident analyses show that hypothetical public doses resulting from PWMF credible accident scenarios are below the 1 mSv public dose limit.

- No significant adverse effects on the natural or the social environment have been identified due to operation of the PWMF.

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14.0 GLOSSARY

APM	Adaptive Phased Management
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
BNPD	Bruce Nuclear Power Development
CANDU	Canada Deuterium Uranium (trademark of AECL)
CCW	Condenser Cooling Water
CEX	Controlled Experiment
CN	Canadian National (rail line)
CNEP	Consolidated Nuclear Emergency Plan
CNSC	Canadian Nuclear Safety Commission
CRNL	Chalk River Nuclear Laboratories
C/S	Containment/Surveillance
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DGR	Deep Geologic Repository
DNGS	Darlington Nuclear Generating Station
DRL	Derived Release Limit
DSC	Dry Storage Container
DSM	Dry Storage Module
DWMF	Darlington Waste Management Facility
EA	Environmental Assessment
ECA	Environmental Complication Approval
EMP	Environmental Monitoring Program
EMS	Environmental Management System
EPD	Electronic Personal Dosimeter
ERB	Easily Retrievable Basket
ERT	Emergency Response Team
GMAW	Gas Metal Arc Welding
GSS	Guaranteed Shutdown State
GTA	Greater Toronto Area
GTAW	Gas Tungsten Arc Welding
HAZ	Heat Affected Zone
HEPA	High Efficiency Particulate Air
HSMS	Health and Safety Management System
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
IFB	Irradiated Fuel Bay
IGLD	International Great Lakes Datum
ILW	Intermediate-Level Waste
ISO	International Organization for Standardization
LAN	Local Area Network
L&ILW	Low and Intermediate Level Waste
LSFCRP	Large Scale Fuel Channel Replacement Program
M	Magnitude
MDA	Minimum Detectable Activity
MOECC	Ministry of the Environment and Climate Change

MOE	Ministry of Environment
NBCC	National Building Code of Canada
NCSA	Nuclear Safety and Control Act
NEW	Nuclear Energy Worker
NFCC	National Fire Code of Canada
NFPA	National Fire Protection Agency
NGS	Nuclear Generating Station
NPT	The Treaty on the Non-proliferation of Nuclear Weapons
NWMD	Nuclear Waste Management Division
NWMO	Nuclear Waste Management Organization
OH&S	Occupational Health and Safety
OPG	Ontario Power Generation
PA	Public Address
PAUT	Phased Array Ultrasonic Testing
PDP	Preliminary Decommissioning Plan
PHTS	Primary Heat Transport System
PM	Particulate Matter
P-mSv	Person-millisievert
POR	Points of Reception
PWMF	Pickering Waste Management Facility
RCS	Retube Components Storage
RP	Radiation Protection
SCC	Stress Corrosion Cracking
SSC	Structures, Systems, and Components
TIBL	Thermal Inversion Boundary Layer
TLD	Thermoluminescent Dosimeter
TMB	Training and Mock-up Building
TSP	Total Suspended Particulates
UFDS	Used Fuel Dry Storage
UPS	Uninterruptible Power Supply
VEC	Valued Ecosystem Component
VOC	Volatile Organic Compound
WHMIS	Workplace Hazardous Materials Information System
WPCP	Water Pollution Control Plant
WWMF	Waste Management Facility

15.0 DRAWINGS

- NK30-D0A-10200-0001 R036, Pickering NGS Building Development – Site Plan.
- 00104-DRAW-79171-10001 R007, Dry Storage Container (DSC) – General Arrangement.
- 00104-DRAW-79171-10024 R001, Long-Module Dry Storage Container – General Arrangement.
- 00104-DRAW-79171-10051 R002, Long Module Dry Storage Container – MKII General Arrangement.
- 92896-D0A-29660-1001 R008, Used Fuel Dry Storage Building: Floor Plans.
- 92896-D0A-29660-1003 R005, Used Fuel Dry Storage Building: Elevations.
- 92896-DRAW-10200-10003 R000, LSFCRP-Site Layout, Shielded Storage Module Loading & Storage Area, General Arrangement.
- 92896-DRAW-79400-10005 R000, LSFCRP Dry Storage Module, Storage Area Support Pads & Surface Drainage, Concrete & Reinforcing.
- 92896-DRAW-29651-10075 R000, Pickering Waste Management Facility Dry Fuel Storage Building #3: Architectural Site Plan.
- 92896-DRAW-29651-10065 R000, Pickering Waste Management Facility Dry Fuel Storage Building #3 & 4: Overall Ground Floor Plan.

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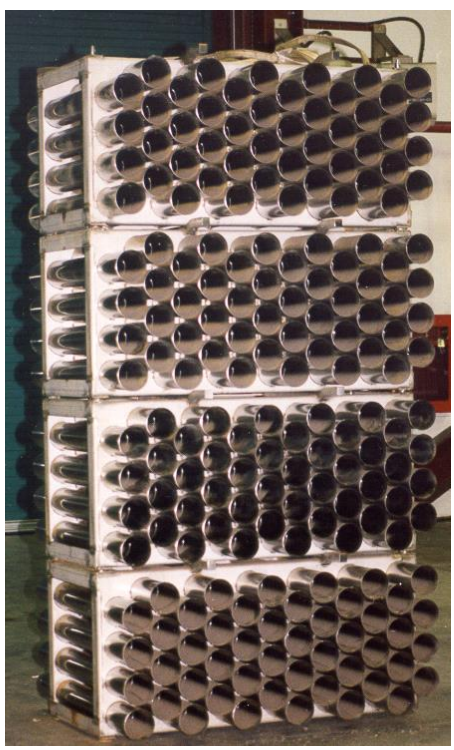

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B.1.0 PURPOSE

A safety assessment of the Pickering Waste Management Facility (PWMF) and the Used Fuel Dry Storage (UFDS) process is required to support the operating licence for the PWMF. This safety assessment is to assess the dose consequences to the public and workers under normal operation and postulated credible accident scenarios and to confirm that the dose rates are below the regulatory dose limits for workers and members of the public (B-CNSC17).

This safety assessment encompasses:

- The on-site transfer of loaded Dry Storage Containers (DSCs) from the Irradiated Fuel Bays (IFBs) at the Pickering Nuclear Generating Station (NGS) A and B to the DSC processing building at the PWMF;
- The processing of DSCs inside the DSC processing building;
- The transfer of seal welded DSCs from the DSC processing building to a DSC storage building at either the PWMF Phase I site or Phase II site, the transfer of seal welded DSCs from DSC storage buildings at the PWMF Phase I site to PWMF Phase II site; and
- The storage of DSCs within the DSC storage buildings at either PWMF Phase I or Phase II sites.

The loading, decontamination, and vacuum drying of DSCs at the station IFBs prior to transfer are part of the Pickering NGS safety reports and licensing basis and are not covered by this document.

B.2.0 ASSESSMENT BASIS

B.2.1 Used Fuel from the Pickering Nuclear Generating Station Reactors

The Pickering NGS reactors use 28-element Canadian Deuterium Uranium (CANDU) fuel bundles. Approximately 3,000 bundles are discharged each year from each of the reactors at Pickering NGS. After a minimum of 10 years of cooling[13], fuel bundles may be transferred to DSCs for interim dry storage.

B.2.1.1 Bundle Design

The Pickering fuel bundles are comprised of 28 cylindrical fuel elements, arranged in concentric rings of 16, eight, and four elements and held in place between two zirconium-alloy end plates (see Figure B-1). Each fuel element is made up of natural uranium in the form of sintered pellets of high-density uranium dioxide (UO_2), sealed in zirconium-alloy tubes. The sheath of the tubes is designed to minimize sheath strain during operation. The zirconium alloy (called “Zircaloy-4”,

[13] A minimum of 10 years of cooling can include residence time in fuel channels during GSS followed by IFB storage

referred to herein as "zircaloy") contains the elements tin, iron, chromium and sometimes nickel. In addition, a braze alloy containing beryllium is used to attach the appendages to the fuel sheath (i.e., bearing pads, inter-element spacers) and CANLUB, a commercial graphite coating, is applied to the inner surface of some fuel elements. Table B-1 shows the nominal dimensions for a Pickering fuel bundle.

B.2.1.2 Bundle Burnup[14]

The core is divided into different fuelling regions which may employ different fuelling schemes: four-bundle or eight-bundle shifting. Figures B-2 and B-3 show the statistical data available for the exit burnup of Pickering NGS-A and -B fuel bundles, produced by the computer code SORO (B-OPG13b). The historical maximum bundle discharge burnup is 595 MWh/kgU.

Although the maximum exit burnup of some bundles could be as high as 595 MWh/kgU, statistically those bundles represent a very small percentage of the total number of discharged bundles. The data used for Figures B-2 and Figure B-3 show less than one percent of the bundles discharged from Pickering NGS-B to date have burnups between ~250 MWh/kgU and 595 MWh/kgU.

The reference burnup for the reference fuel bundle used for the safety assessment was chosen as [redacted].

B.2.1.3 Reference Used Fuel Bundle

The primary factors that determine the characteristics of used fuel are physical attributes, power and burnup histories, and decay time. These factors are, in turn, influenced by fuelling strategies and reactor conditions. Therefore, for the purpose of performing the safety assessment of UFDS at the PWMF, a reference fuel bundle has been defined.

B.2.1.3.1 Reference Used Fuel Bundle Dimensions

The fuel bundle used at the Pickering NGS reactors is 495 mm in length, has an outer diameter of 100 mm, and has a nominal total bundle mass of 24.6 kg. The complete dimensions of the reference fuel bundle are given in Table B-1.

B.2.1.3.2 Reference Used Fuel Bundle Age

This safety assessment assumes that used fuel bundles are kept in the Pickering IFBs for 10 years before they are transferred to the PWMF. In practice, the age of fuel loaded in DSCs generally exceeds 10 years as operational procedures are employed to select older fuel first. Therefore, the average age of used fuel bundles in dry storage at the PWMF is greater than 10 years.

[14] Burnup is the fission energy generated per unit mass of heavy element initially in the fuel (unit: MWh/kgU).

B.2.1.3.3 Reference Used Fuel Bundle Burnup

Given statistical data of the fuel discharged from the Pickering NGS reactors (see Figures B-2 and B-3), [redacted] has been retained as the burnup for the Pickering reference used fuel bundle. Due to fuelling strategies and reactor operating conditions, the distribution of the fuel bundles within the IFBs is such that the average burnup of the fuel loaded into each DSC is expected to be below the chosen reference used fuel bundle burnup.

B.2.1.3.4 Reference Used Fuel Bundle Power

The total core fission power for a Pickering NGS reactor is 1,744 MW(f). The average bundle power is [redacted] and the average fuel bundle residence time[15] is [redacted] Full Power Days[16].

B.2.2 Nuclide Inventory

The radionuclide inventory for the defined Pickering reference fuel bundle was calculated using the computer code ORIGEN-S (B-ORNL11). Table B-2, Table B-3 and Table B-4 show the radionuclide inventories for the actinides[17] ,fission products[18] and light elements found in a reference fuel bundle.

B.2.3 Source Term

Table B-5 gives the gamma spectrum for a 10-year-cooled Pickering reference fuel bundle. The computer code ORIGEN-S was used to obtain this gamma spectrum.

B.2.4 Decay Heat

The energy produced by radioactive decay is released from the fuel bundle in the form of heat and radiation. The decay heat released from a Pickering reference fuel bundle for different cooling periods, as calculated using the ORIGEN-S computer code, is shown in Figure B-4. The decay heat for the 10-year-cooled reference fuel bundle is approximately 5.8 W.

B.2.5 Chemical and Physical Characteristics of Radionuclides in Used Fuel

The location of radionuclide species in a fuel element depends on their chemical and physical behaviours and where they were produced. The majority of new radionuclides, such as the actinides (Table B-2) and fission products (Table B-3) in 10-year-cooled used fuel are embedded within the lattice of uranium and oxygen atoms, very close to where they were produced. They substitute for uranium in the uranium dioxide lattice. Activation products that are produced in the zircaloy

---

15 Fuel Bundle Residence Time is the time for which a fuel bundle resides in the reactor.
16 Full Power Day is a 24 hours of reactor operation at nominal 100 percent reactor power.
17 The term Actinides is used by the computer code ORIGEN-S to group fuel nuclides and their decay products, plus ^{4}He (since it results from alpha decay).
18 Fission Products are the isotopes produced by fission and their decay products.

sheath are primarily trapped by the zirconium alloy and cannot diffuse any significant distance from the site of their formation.

Cracking and grain growth occurs in the ceramic fuel pellets at the high temperatures and temperature gradients in the reactor (400°C to 2,000°C). A diagram of the cross section of a ceramic fuel pellet is shown in Figure B-5. About 2 percent of the gaseous radionuclides, or those that are volatile at fuel irradiation temperatures, are released to the cracks in the pellets and to the gap between the pellets and the fuel sheath. A further 6 percent segregates to the grain (crystal) boundaries within the uranium dioxide pellets. Laboratory studies have been used to predict the quantities of radionuclides in the fuel-sheath gap and at the grain boundaries.

The release of radionuclides from failed fuel depends on the volatility of the chemical forms found in the fuel at the maximum fuel temperature and their ability to migrate through the fuel grains. Radionuclides born in the fuel may remain in elemental form or combine with other nuclides, the uranium dioxide fuel, the zircaloy sheath, or excess oxygen. The chemical forms of fission products have been studied in the past to determine their respective contributions to releases from failed fuel (B-OH86).

The maximum sheath temperature of a used fuel bundle with a decay heat of 6.4 W/bundle is not expected to exceed 150°C in dry storage in a helium atmosphere. For Pickering, it is expected to be less due to smaller decay heat load (5.8 W/bundle compared to 6.4 W/bundle assumed in the analysis). At these low temperatures, only the volatile fission products would be released should the fuel sheath become damaged. These volatile fission products include krypton-85, tritiated water vapour (HTO), and tritiated hydrogen gas (HT) should there be any tritium in the fuel-sheath gap that has evaded hydriding of the zirconium sheath. Release of these radionuclide vapours upon sheath failure is used for this safety assessment.

As discussed above, actinide series radionuclides along with most of the remaining fission and activation products will be embedded within the uranium dioxide grains or in the zircaloy. A small number of radionuclides are potentially volatile; therefore assessment of their characteristics relative to release is appropriate. The following is a brief review of the most likely chemical forms of the potentially volatile, radiologically significant fission and activation products found in used fuel at 10 years after discharge from the reactor.

Krypton-85 ($T_{1/2}$ = 10.7 y)

Krypton-85 is a member of the noble gas family, and as such is characterized by high ionization potential and small Van der Waals forces. Consequently, the element is chemically inactive, shows low solubility in water and requires very low temperatures and/or high pressures for liquefaction. Krypton has a melting point of -156.5°C, and a boiling point of -153°C.

Krypton-85 exists as elemental gas in the fuel. Due to its high volatility, krypton-85 residing in the fuel-sheath gap and other open voids in the fuel will be available for release almost instantaneously upon sheath failure.

Tritium ($T_{1/2}$ = 12.3 y)

The tritium generated during irradiation of the fuel is located in one of four regions: the fuel matrix (fuel grains), fuel grain boundaries, the fuel-sheath gap, and the zircaloy sheath. During irradiation, the tritium in the fuel grains migrates to the grain boundaries as tritium or HT, or combines with oxygen as oxygen-T (TO) or HTO (tritiated water vapour). The measured diffusion coefficient for tritium in UO_2 is orders of magnitude lower than the corresponding diffusion coefficient for molecular hydrogen (B-SCAR78). A slight increase in oxygen availability in the fuel due to the fission process may shift the equilibrium towards the formation of TO or HTO, both of which are volatile at used fuel temperatures in a DSC. When gas bubbles at the grain boundary reach saturation, tritium, together with other species stored in the bubbles, will be released to the fuel-sheath gap region. The final major chemical forms of tritium in the gap are HT and HTO, both of which are volatile.

These species will diffuse and/or react with the zircaloy sheath resulting in the increase of the sheath inner surface oxide layer thickness and a large tritium pickup by the zircaloy. On reaction with the sheath, the less volatile compound zirconium tritide (ZrT_2) will form, thereby holding up the free inventory of tritium in the cladding. The fractional tritium inventory resident in the fuel-sheath gap is the balance between two competing processes: the fuel-to-gap release process that increases the gap fraction, and the sheath pickup that decreases the gap fraction.

Fuel element puncture tests (B-GOODE70) have determined that the gap fraction is only 10^{-5} of the total tritium in fuel elements. Therefore, the majority of tritium atoms released to the gap will be held up in the cladding (B-GOODE70). Very little tritium will be available for release to the environment from the gap in case of fuel sheath failure.

The final tritium grain-boundary fraction and gap fraction are postulated to be similar to the values for the non-reacting species krypton-85, even though krypton-85 has a cumulative fission yield 100 times higher and a gap fraction about 1,000 times larger.

Carbon-14 ($T_{1/2}$ = 5,730 y)

The activation product carbon-14 is non-volatile in elemental form, and volatile when combined with oxygen as carbon dioxide ($^{14}CO_2$). The majority of carbon-14 atoms produced in the fuel remain within the uranium dioxide grains, but some segregate to grain boundaries within the UO_2 pellets and to the gap between the pellets and the fuel sheath. Much less than 0.1 percent of the carbon-14 is released to the gap and grain boundary regions (B-AECL94). Carbon-14 is also generated in the fuel sheath as a result of activation of the nitrogen impurity in the sheath.

Iodine-129 ($T_{1/2}$ = 1.6 × 10^7 y)

Iodine is one of the most widely studied fission products. Release experiments have demonstrated that only a very small fraction of iodine (i.e., ranging from a few thousandths of one per cent to less than 0.5 percent of the total amount available for release from the fuel) can be in volatile form at the temperature conditions of used fuel when it is discharged from the reactor core.

These experimental results, along with numerous other investigations have concluded that almost all of the iodine in the fuel-sheath gap exists as cesium iodide (CsI) in the condensed and vapour phases. Since the free energy of formation of CsI is strongly negative, the reaction will proceed to completion and essentially all of the iodine will participate in the reaction. Cesium iodide is semi-volatile with a boiling point of 1,280°C. At lower temperatures, condensable fission products such as CsI are shown to undergo complete deposition onto metal surfaces in the immediate proximity of the fuel. If iodine is in contact with the zircaloy sheath, the volatiles ZrI_3 and ZrI_4 may form in very small concentrations. Diatomic iodine (I_2), a stable and volatile form of the element, will not be found in fuel as the CsI reaction dominates.

Cesium-134 ($T_{1/2}$ = 2.1 y)
Cesium-135 ($T_{1/2}$ = 2.3 × 10^6 y)
Cesium-137 ($T_{1/2}$ = 30 y)

Several chemical forms of cesium in fuel are possible as there is a high concentration of cesium isotopes relative to other fission products. The most volatile species of cesium are the monatomic and the diatomic forms, Cs and Cs_2, together with CsI. However, these species behave as non-volatile at 10-year-cooled fuel temperatures.

Strontium-89 ($T_{1/2}$ = 51 d)
Strontium-90 ($T_{1/2}$ = 29 y)

Strontium has a very stable oxide and is predicted to be entirely in this form in the fuel. Strontium has a high affinity for oxygen, and will readily oxidize to strontium oxide (SrO), a non-volatile compound with a boiling point of about 3,000°C. The oxide is insoluble in the fuel matrix but will form compounds with zircaloy. Because of the extremely low volatility of SrO, no releases of strontium are expected in the event of fuel element damage.

B.2.5.1 Deposits on the Exterior of Fuel Elements

Corrosion products in the Primary Heat Transport System (PHTS) are present in low concentrations in the reactor coolant, and subsequently become radioactive as a result of neutron activation in the reactor core. These are removed from the PHTS coolant by purification systems; however, some of these corrosion products will deposit on the surfaces of the fuel bundles.

Fission products, which originate from defective fuel or the fission of trace levels of uranium present in the reactor coolant, also deposit on the surfaces of fuel bundles.

An experimental study has characterized surface deposits on fuel elements, for the purpose of assessing the radioactive handling hazards associated with the dry storage of fuel bundles (B-CHEN86). The study had two parts: first, surface deposits were removed from nineteen Bruce and Pickering fuel elements using aggressive chemical treatment, and the chemical solutions were analyzed for corrosion products. In the second part of the study, four Pickering used fuel bundles were stored in individual pressure vessels in moist air at 150°C (moisture was provided by a small quantity of water in the bottom of each pressure vessel). In two of the bundles, all of the outer elements except a control element in each were intentionally defected (i.e., the bundles were damaged under relatively cool fuel temperatures, some time after discharge from the reactor core), to determine the effect of storage conditions on defected versus undamaged bundles. After a 30-month storage period, the water from the pressure vessels was analyzed.

Conclusions from analyses are summarized as follows:

(a) Visual examination of all the elements showed no detectable evidence of surface deposits due to corrosion products.

(b) Corrosion deposits removed by chemical treatment of fuel elements were found to consist mainly of iron (concentration range of 11 to 300 mg/m^2). Nickel, copper, and chromium were found in lower concentrations, on fuel element outer surfaces. Cobalt-60 was detected in concentrations of about a factor of 1000 lower than iron. Cesium-137 and strontium-90 were the major fission products detected as corrosion products on fuel element outer surfaces.

(c) Analysis of the residues from the pressure vessel solutions after storage indicated the presence of iron as its major constituent. The major fission product on the outer sheath surface, due to corrosion product deposits, was cesium-137. The levels of surface deposit activities measured were consistent with those obtained by chemical treatment of fuel elements prior to storage. Higher levels of fission products were leached from defected fuel than intact bundles (due to leaching of the fuel matrix inside the element).

(d) The activities of cesium-137, cesium-134, strontium-90, and cobalt-60 corrosion products on the outer surfaces of fuel elements after storage are smaller by at least four orders of magnitude compared with the activities of krypton-85 or tritium available for release (from the fuel matrix inside a damaged element). The measured activities of cesium-137 surface deposits on a used fuel bundle were smaller by about six orders of magnitude compared with the total cesium-137 inventory in a 10-year-cooled reference fuel bundle.

(e) No significant activities of iodine-129 were measured in pressure vessel solutions after 30 months of storage.

The study showed that particulate contamination is present on used fuel bundles in the form of corrosion and fission product surface deposits. However, the corrosion products detected in the study are not volatile at the temperatures of 10-year cooled used fuel during handling in the station IFB or during dry storage in a helium atmosphere.

The study used aggressive chemical techniques or 30 months of leaching to remove surface deposits; hence, the conclusions of the study are conservative indications of the relative levels of particulate contamination that could be released into IFB water or be made airborne during dry storage operations.

Activated corrosion and fission products that have been deposited on used fuel external surfaces while in the reactor core, and that have remained adhered under the flow of PHTS coolant and during subsequent storage in the IFB, will be fixed to the outer surfaces of fuel elements and will require either physical abrasion or leaching to become released.

B.2.6 Defective and Damaged Fuel

When a sheath fails in the reactor core at high temperatures, the free inventory of volatile radionuclides residing in the fuel-sheath gap and other open voids in the fuel is released almost instantaneously. Leaching of water-soluble radionuclides from the fuel matrix occurs slowly over a longer-term, while the fuel element remains submerged in the fuel bay.

In the event that reference used fuel (i.e., 10-year cooled) should become damaged during dry storage operations, the only significant radionuclide species that are volatile are krypton-85, tritium and carbon-14. For a fuel element damaged under abnormal operating conditions, it is postulated that 100 percent of the krypton-85 and tritium present in the gap will be released, along with 10 percent of the krypton-85 and tritium present in the grain boundary. 0.1 percent of the carbon-14 activated in the sheath is assumed to be released as $^{14}CO_2$.

Used fuel with visible or known defects is not transferred to DSCs.

B.2.7 Free Inventories Available for Release

For a fuel element damaged under abnormal operating conditions, it is postulated that the free inventory of tritium and krypton-85 is the radionuclide inventory in the gap between the fuel matrix and the zircaloy sheath, plus 10 percent of the inventory in the grain boundary. The gap fraction is assumed to be 0.0095 for tritium and for krypton-85. The grain-boundary fraction is assumed to be 0.123 for tritium and for krypton-85. These numbers are based on the discussion in Section B.2.5. This is an appropriate assessment of the grain boundary release, at the lower grain-boundary gas pressures that would be associated with the lower temperatures of 10-year-cooled used fuel. For carbon-14, 0.1 percent of the inventory is assumed to be available for release.

Conservatively, for the assessment of airborne emissions, the calculation of tritium , krypton-85 and carbon-14 inventories takes into account the activation in the reactor core of small quantities of impurities present in the used fuel. The adjusted tritium inventory for a Pickering reference used fuel bundle is [redacted] Bq per bundle. The krypton-85 inventory remains essentially unchanged by impurities at [redacted] Bq per bundle. The carbon-14 inventory is [redacted] Bq per bundle.

For each radionuclide, the release from failed fuel can be written as:

$$\mathbf{R_e = (f_{gap} + 0.1f_{gb})I_e}$$

Where,

R_e = radionuclide released per failed used fuel element (Bq/element),
f_{gap} = fraction of radionuclide inventory in the gap,
f_{gb} = fraction of radionuclide inventory in the grain boundary, and
I_e = radionuclide inventory per used fuel element (Bq/element).

For carbon-14, the amount released to the gap and grain boundary is set to 0.1 percent.

B.2.8 Breathing Rates

Breathing rates for the public dose calculations were based on the 95th percentile of the breathing rate for the representative individual. This is consistent with the guidance given in the CSA N288.2-14 standard (B-CSA14c). The breathing rate value from the CSA N288.2-14 standard is 0.31 m^3/h (8.61 × 10^{-5} m^3/s) for an infant and 0.96 m^3/h (2.67 × 10^{-4} m^3/s) for an adult.

The breathing rate for a NEW is based on the ICRP 89 male adult breathing rate performing light exercise (1.5 m^3/h or 4.17 × 10^{-4} m^3/s) (B-ICRP02).

B.2.9 Dose Conversion Factors

As discussed in Section B.2.5, only krypton-85,tritium and carbon-14 associated with the used fuel in the DSC can contribute to airborne emissions. Therefore, the dose assessment methodology and the derivation of an emission source term are focused on releases of krypton-85, tritium in the form of HTO and carbon-14.

The inhalation dose conversion factors for tritium (HTO) used in this assessment are based on the recommended values from the ICRP 119 (B-ICRP12).

- 1.8 × 10^{-11} Sv/Bq for adult; and
- 6.4 × 10^{-11} Sv/Bq for infant.

The immersion skin absorption factor for tritium is taken as 1.5 (B-ICRP95).

The cloudshine dose conversion factors for krypton-85 used in this assessment are based on the recommended values from the Health Canada publication (B-HC99):

- 2.55×10^{-16} Sv.s^{-1}.Bq^{-1}.m^{3} for adult; and
- 3.83×10^{-16} Sv.s^{-1}.Bq^{-1}.m^{3} for infant.

The inhalation dose conversion factors for carbon-14 used in this assessment are based on the recommended values from the ICRP 119 (B-ICRP12).

- 6.2×10^{-12} Sv/Bq for adult; and
- 1.9×10^{-11} Sv/Bq for infant.

The cloudshine dose conversion factors for carbon-14 used in this assessment are based on the recommended values from the Health Canada publication (B-HC99).

- 2.6×10^{-18} Sv.s^{-1}.Bq^{-1}.m^{3} for adult; and
- 3.9×10^{-18} Sv.s^{-1}.Bq^{-1}.m^{3} for infant.

B.2.10 Dry Storage Container

B.2.10.1 Dimensions

The DSC dimensions and design details used for this assessment are given in the following drawing:

00104-DRAW-79171-10024, *Used Fuel Dry Storage Ontario Power Generation Long Module Dry Storage Container General Arrangement*.

An illustration of the modified DSC design can be found in 00104-DRAW-79171-10051. The safety assessment results are also applicable to the original long module design, as the MKII constitutes the bounding DSC design.

The direct radiation fields from a loaded DSC are discussed in Section B.6.1.1.

B.2.10.2 Concrete Density

The concrete used as radiation shielding in the DSC design is reinforced heavy concrete. For extra support, a series reinforcing steel bars are inserted into the heavy concrete area. The concrete has a nominal density of 3.5 Mg/m³; the presence of the reinforcing bars increases the homogenized concrete density to a nominal density of 3.57 Mg/m³.

B.2.10.3 Fuel Module

Used fuel bundles are placed into storage modules. The modules are made of [redacted] with dimensions [redacted]. Each empty fuel module weighs approximately [redacted] and can hold 96 bundles, two bundles in each of 48 [redacted] storage tubes. The fuel storage module design is shown in Figure B-7. Each DSC has the capacity to store four modules.

Freedom of Information and Protection of Privacy Act (FIPPA) S. 18 and Access to Information and Privacy (ATIP) S.13.

B.2.11 Pickering Used Fuel Dry Storage

UFDS at the PWMF currently consists of industrial-type buildings that are designed and constructed to provide facilities for the safe processing and storage of DSCs. The DSC processing building and storage buildings are not safety-related structures at the PWMF since they are not credited in the safety assessment as a barrier to the release of radiation.

The PWMF Phase I site consists of two stages. The PMWF Phase I Stage I site contains a DSC processing building, including a DSC workshop, offices and utilities, and a DSC storage building to accommodate 185 DSCs; PWMF Phase I Stage I buildings are a single-story structure with a two-story office area in the DSC processing building. The PWMF Phase I Stage II is a single-story building that shares the north wall of the PWMF Phase I Stage I building to form a single structure. The PWMF Phase I Stage II can accommodate 469 loaded DSCs and includes an area for receiving new empty DSCs. The PWMF Phase I is located within the Pickering NGS protected area.

The PWMF Phase II site consists of one new DSC storage building and a site for future DSC storage. Storage Building 3 is based on the DSC storage building designs of the WWMF and the Darlington Waste Management Facility (DWMF). The site of Storage Building 3 is in the eastern half of the Pickering NGS property and is situated on level ground adjacent to the shoreline of Lake Ontario directly east of Pickering NGS B, in the general area of the East Complex and well inside the Pickering NGS property boundary fence.

The PWMF has been designed to:

- Handle empty DSCs, which includes:
    - Receiving and offloading new empty DSCs and their lids from the DSC delivery vehicle;
    - Performing receipt inspections/preparation of DSCs and preparation/ cleaning the DSC and the lid seal weld area as required;
    - Housing and support of DSC handling equipment; and
    - Preparing empty DSCs for transfer to the Pickering IFBs.
- Handle DSCs loaded with fuel, including housing equipment and providing support for:
    - Receiving a loaded DSC from the DSC transporter vehicle;
    - DSC lid seal weld pre-heating;
    - Seal welding the DSC lid to the DSC body;

- DSC cool down;
- Inspection of the lid seal weld;
- Repairing welds as necessary;
- DSC vacuum drying and helium backfilling;
- Welding of the drain plug;
- Inspection of drain plug weld;
- Helium leak detection;
- Painting of scuffs and welds; and
- Application of International Atomic Energy Agency (IAEA) safeguards seals.

- And to provide services and space for personnel and equipment, such as:
    - OPG administrative offices;
    - IAEA personnel office;
    - Guard station;
    - Personnel amenities;
    - Gas container storage areas;
    - Janitor's rooms;
    - Tool storage area;
    - Electrical and mechanical rooms;
    - Equipment maintenance area; and
    - Welding rooms.

The PWMF Phase I building walls are made of 8 inch-thick (20 cm) ordinary concrete. This is sufficient to meet the dose rate limits at the station security fence, which is located 5 m from the building, and the dose rate target at the station site boundary, which is located 850 m east of the building inland and 420 m east on the lakeside.

Inside the PWMF Phase I DSC storage buildings, DSCs are placed with a minimum spacing of 63.5 cm north to south and 23 cm in the east to west directions. The narrower side of the containers face the east and west directions. The peripheral rows of containers are 1 m away from the building wall. In the Stage 2 building, a corridor of approximately 10 m width divides the floor layout into east and west blocks. Each block of containers is divided into north and south sections that are separated by a much narrower corridor of approximately 2.1 m wide. For the shielding analysis, the PWMF Phase I DSC storage buildings were assumed to be filled to nominal design capacity with stored DSCs carrying an average of 10-year-cooled used fuel (occupational dose rate calculations were based on DSCs carrying 10-year-cooled used fuel, since a worker can directly handle a DSC or work near DSCs containing used fuel of this minimum age). Operating experience has shown that the calculated dose rates for the assumed fuel ages were conservative and that actual dose rates are lower. The narrower sides of the DSCs form the peripheral row facing the eastern wall.

The PWMF Phase II DSC storage building walls are designed with 12 inch-thick (30 cm) ordinary concrete. This provides sufficient shielding to meet the dose rate target at the perimeter fence and to meet the administrative dose target at the site boundary, which is located about 330 m east of the building inland. This also ensures that the dose target at the lakeside exclusion zone boundary is met, which is located about 340 m south-east.

Inside PWMF Phase II DSC Storage Building 3, a corridor of approximately 13 m width divides the floor layout into east and west blocks. For the shielding analysis, the PWMF Phase II DSC storage building was assumed to be filled with 480 DSCs consisting of 84 DSCs filled with 28-year cooled used fuel and 396 DSCs filled with 26-year cooled used fuel. The younger DSCs were placed on the east side of the building to provide a conservative dose rate calculation at the site boundary. Operating experience has shown that the calculated dose rates are conservative.

B.3.0 NORMAL OPERATION ASSESSMENT METHODOLOGY

The methodology used to calculate the dose rates inside the PWMF and outside the facility are described below.

The computer code ORIGEN-S (B-ORNL11) together with a burnup-dependent library for natural uranium oxide fuelled CANDU reactor 28-element bundle design (B-OPG18) was used to determine the radioactivity content of the irradiated fuel. After calculating the buildup of actinides and fission products during a specified irradiation period, the code then calculates their inventory as a function of the time after discharge from the reactor (i.e., cooling time). The photon and neutron spectrum as a function of the cooling time is also calculated by the code. Verification and validation of the ORIGEN-S code has been performed (B-OPG01).

The gamma dose rate calculations for a single long module DSC (MKII) and for dose points inside and outside of the PWMF were carried out using the Monte

Carlo code (MCNP 6.1[19]) (B-LANL11). This code is capable of rigorously simulating the stochastic nature of gamma, neutron and electron transport by explicitly modelling the physical nature of their travel through space and their interactions with matter. The MCNP code captures gamma dose rate contributions from irradiated fuel in large arrays of storage containers in different storage buildings to provide an accurate, integrated shielding analysis, taking into account all gamma radiation dose pathways. This type of model has been used in the shielding analysis for the PWMF, DWMF and WWMF.

The MCNP code applies the 'Monte Carlo' method of analysis, simulating photon histories explicitly in the modelled geometry. A characteristic of this method is that all of the results ('detector tally' results of dose rate values at the specific locations) are always statistical quantities in the form of an estimated mean value and an estimated standard deviation. MCNP utilizes a sufficient number of 'photon histories' that the statistical uncertainty is very small. In cases where the estimated statistical uncertainties in the MCNP results are greater than five percent, special care has been taken in the consideration of the results. In such cases, the margin between the target dose rate and the computed value was considered related to the estimated uncertainty, and a judgement was made as to whether the values can be accepted as not exceeding the targets. This methodology is documented in Reference B-OPG17b.

The methodology used to develop the MCNP model of UFDS buildings and DSCs at waste facilities has been validated/benchmarked (B-OPG17a) through simulation of TLDs surrounding the UFDS buildings. The results show that the predictions using the MCNP model, which is based on the reference methodology, is conservative by 35-60%. This conservative outcome justifies the application of this methodology and the use of MCNP to model large arrays of DSCs loaded with irradiated fuel in normal operations safety assessments.

In the normal operation assessment, a bounding fuel source term is used for DSCs such that the decay of the fuel is minimized and the burnup is maximized. The overall outcome of the normal operations safety assessment is that the predicted dose rates from the UFDS building loaded to full nominal design capacity at the PWMF site and Pickering NGS site boundary are considered to be conservative and below acceptance criteria.

B.4.0 MALFUNCTIONS AND ACCIDENTS ASSESSMENT METHODOLOGY

Based on the distinct stages of the Pickering UFDS process, this assessment was divided into the following operational stages:

(a) On-site transfer operations;

[19] MCNP® and Monte Carlo N-Particle® are registered trademarks owned by Los Alamos National Security, LLC, manager and operator of Los Alamos National Laboratory

(b) Operations inside the DSC processing building; and

(c) Storage.

B.4.1 Identification of Initiating Events

For each stage of the UFDS operations, release of radiation can occur due to the failure of the systems and components being used.

There are two general categories of initiating events that can result in abnormal conditions or accidents: internal events and external events.

- **Internal events** are abnormal conditions generated within the area of operation as a result of equipment failure or human error.
- **External events** are natural and man-made phenomena originating outside the area of operation that may, as a result, cause an internal event or perhaps even multiple, wide-spread events.

The list of initiating events identified for the on-site transfer stage of UFDS operations are given in Table B-6; at the DSC processing building in Table B-7; and at the DSC storage building in Table B-8.

B.4.2 Event Frequency

If the frequency of occurrence estimated for any postulated accident scenario is less than 10^{-6} events per year (B-CSA14b), it is considered incredible.

B.4.3 Screening of Events

Each event was screened following the OPG screening criteria for internal hazards (B-OPG16b) and external hazards (B-OPG16a) to establish if it could result in any radiological impact to the public, the workers and/or the environment. Design provisions and procedural measures that could prevent the event or mitigate its consequences were also considered.

B.4.4 Dose Consequences

In the absence of Nuclear Waste specific standard regulatory documents, the guidance documented in CSA N288.1-14 (B-CSA14a) and CSA N288.2-14 (B-CSA14c) was used in performing the safety assessment.

B.4.4.1 Public

Doses to individual members of the public from the postulated chronic release from DSCs during processing were calculated using the IMPACT code (B-COG15), which follows the guidance given in the CSA N288.1-14 standard.

The potential doses to individual members of the public were calculated using the ADDAM code (B-COG11). The ADDAM code considers the inhalation, cloudshine, and groundshine pathways. The 95th percentile individual dose for an exposure period of 30 days was calculated consistent with the guideline given in the CSA N288.2-14 standard.

ADDAM is a safety analysis computer program developed by the Atomic Energy of Canada Limited (AECL) for use by the CANDU Owners Group (COG) community. ADDAM calculates doses to the public due to a postulated accidental release of radioactive material to the atmosphere from a nuclear facility. Radionuclides being released can be in the form of gases, vapours or small particles. The radionuclides will disperse as a result of the effects of atmospheric turbulence. The dispersion of the release is affected by the characteristic of the release, the prevailing meteorological conditions, the surrounding terrain and the nearby buildings. The concentrations in the cloud and on the ground take into account factors such as the nature of the releases, decay, build-up and deposition. Doses are calculated for various age groups and receptor locations, and categorized by release pathways (stack, inlet, leakage, or hole) and exposure pathways (inhalation, cloudshine, groundshine). The calculations of atmospheric dispersion and doses are based on CSA N288.2-M91.

The potential dose to an adult and an infant from airborne tritium, krypton-85 and carbon-14 emissions for all exposure pathways are determined using ADDAM. The bounding doses associated with the malfunction or accident scenario for used fuel dry storage involving used fuel bundles in the DSC are 6.5×10^{-3} mSv (adult) and 7.91×10^{-3} mSv (infant). These dose consequences are associated with the malfunction/accident scenario of a DSC drop during on-site transfer where 100 percent of the fuel elements are assumed to be damaged.

B.4.4.2 Occupational

The worker is assumed to be present in the vicinity of the accident location wearing no protective clothing or respiratory protection at the time of the accident. The worker's response time to remove himself or herself from the accident location (i.e., under emergency back-out conditions) is assumed to be two minutes.

The resulting dose rate is assessed using the semi-infinite cloud model. The cloud volume is assumed to be 500 m^3 and the exposure time 120 seconds.

The inhalation dose from exposure of tritium is calculated using the following equation:

$$D_{inhalation} = \sum_{n=1} \left(R_n \times BR \times sk_{a,n} \times DCF_{inhalation,n}\right) \times T/V$$

where;

$D_{inhalation}$ = worker dose from inhalation (Sv);

R_n = release amount (Bq) of nuclide n during the exposure time;

BR = worker breathing rate = 4.17×10^{-4} m^3/s;

$sk_{a,n}$ = skin absorption factor for nuclide *n*. $sk_{a,n}$ = 1.5 for tritium and = 1 for other nuclides;

$DCF_{inhalation,n}$ = inhalation dose coefficient (Sv/Bq) of nuclide n. 1.8×10^{-11} Sv/Bq for HTO and 6.2×10^{-12} Sv/Bq for carbon-14;

T = exposure time (120 s); and

V = contaminated cloud volume (500 m^3)

The cloudshine dose from krypton-85 exposure is calculated using the following equation:

$$D_{cloudshine} = \sum_{n=1} \left(R_n \times DCF_{cloudshine,n}\right) \times T/V$$

where;

$D_{cloudshine}$ = worker dose from cloudshine (Sv);

$DCF_{cloudshine,n}$ = cloudshine dose coefficient ($Sv.m^3.Bq^{-1}.s^{-1}$) of nuclide n.
2.55×10^{-16} $Sv.m^3.Bq^{-1}.s^{-1}$ for Kr-85 and 2.60×10^{-18} $Sv.m^3.Bq^{-1}.s^{-1}$ for carbon-14;

The bounding occupational dose associated with the malfunction or accident scenarios involving used fuel bundles in the DSC is 5 mSv, which came from the following contributors:

- 4.48 mSv from inhalation (including skin absorption) of tritium;
- 0.56 mSv from cloudshine of Kr-85

Carbon-14 was included for completeness, however, the contribution from carbon-14 was not significant. The calculated worker dose is 10.0% of the worker dose limit.

B.5.0 ACCEPTANCE CRITERIA

The radiation safety requirements under normal operation for the PWMF are the following:

- The OPG administrative dose target to the public at the Pickering site boundary shall be less than 100 μSv/y, which is ten percent of the regulatory limit of 1 mSv/y for members of the public.
- The dose rate to non-NEWs, derived from the regulatory limit of 1 mSv/y for members for the public and on 2,000 hours of work per year, is 0.5 μSv/h. This

dose rate is applied at the PWMF Phase I site security fence (located 5 m from the building wall), at the PWMF Phase II site perimeter fence, and anywhere between this fence and the Pickering NGS fence (where non-NEWs could be working).

- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over five years.

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose target to the public at or beyond the Pickering NGS site boundary due to an abnormal event/credible accident shall be 1 mSv.
- The dose target to a worker due to an abnormal event/credible accident shall be 50 mSv.

B.6.0 ASSESSMENT OF DRY STORAGE CONTAINER ON-SITE TRANSFER

New DSCs are received from the manufacturer and are inspected and checked for component fit at the PWMF before being sent to the stations for loading.

At the stations, each DSC is wet-loaded with four used fuel storage modules in the fuel bay, decontaminated, drained, and vacuum dried. The transfer clamp and seal are installed to secure and seal the lid during on-site transfer. The loaded DSC is then transported to the DSC processing building using a transporter.

When a loaded DSC is scheduled to be transferred from an IFB to the DSC processing building, the transporter picks up the loaded DSC with a transfer clamp installed. The transfer clamp is designed to maintain the lid secured to the DSC base during all normal operations and abnormal events/credible accidents. The process uses OPG radiation protection and security procedures. Security is provided in accordance with the approved security plan. IAEA monitoring and surveillance is performed in accordance with the safeguards requirements.

The DSC transporter is a multi-wheeled vehicle for on-site transfer of DSCs. The transporter is self-loading and self-powered by a diesel engine, and does not require the assistance of a crane when picking up or depositing a DSC. The tires on the transporter are designed not to deflate if punctured.

The DSC is lifted and transferred via lifting trunnions mounted on the upper vehicle frame. A mechanical lock prevents the DSC from being inadvertently lowered to the ground upon hydraulic failure.

While traveling at full speed of up to 12 km/h, the transporter is capable of stopping within 4.8 m when emergency stop buttons are depressed, front or rear bumpers are displaced by impact, sensors are activated, or when the seat occupancy switch is activated. Emergency stop when traveling at minimal speeds (for example, when moving DSCs within the PWMF) is achieved almost instantaneously.

A transfer route assessment has been carried out to assess the route by which loaded and seal welded DSCs are transferred from the PWMF Phase I site to the PWMF Phase II DSC Storage Buildings 3. The assessment included an evaluation of the road condition for the expected load and frequency of transfer, assessment of the layout of the roads in relation to buildings, obstacles and intersections which the transfer will have to negotiate, and identification of culverts and ditches along the route to confirm that an accidental roll over of the transporter would not exceed a 2 m drop.

The assessments also looked at possible hazards and obstructions that the transporter may encounter along the route, including hazards such as oil tanks. The assessment has shown that there are no hazards identified that jeopardize the safe transfer of DSCs. The assessment has confirmed that there are no grade changes or obstructions where the bottom of the DSC would scrape the ground due to its low (6 in) clearance, and the comparatively long wheelbase of the transporter.

B.6.1 Normal Operating Conditions during Dry Storage Container On-Site Transfer

B.6.1.1 Direct Radiation Fields

Figure B-8 shows the gamma dose rates calculated as a function of distance from the top, front, and side surfaces of the reference DSC design (that is, the modified design – see Section B.2.10.1). The calculations assume the DSC was filled with 384 Pickering reference fuel bundles as described in Table B-1. The label ‘front’ corresponds to the wider face of the DSC, the face bearing the lift plates.

B.6.1.2 Radioactive Emissions

Under normal operating conditions, no airborne emissions are expected from DSCs during transfer from the Pickering IFBs to the PWMF. This is because the uranium dioxide matrix, the used fuel sheath and the transfer elastomeric seal provide multiple barriers towards preventing the release of radioactive materials:

- The UO_2 matrix effectively contains the radionuclides present in 10-year cooled used fuel (either under wet or dry storage conditions), except for the free fractional inventory of tritium (in vapour form) and krypton-85 (which is a gas);
- The fuel sheath additionally contains the free fractional inventory of tritium, krypton-85 and carbon-14 that would otherwise be available for release; and
- The transfer clamp and elastomeric seal on the lid provide an additional barrier against the release of tritium, krypton-85 and carbon-14 in the event of fuel sheath failure during transfer.

B.6.1.3 Thermal Assessment

The thermal power for a reference fuel bundle was calculated using the computer code ORIGEN-S. Figure B-4 shows the thermal power per reference fuel bundle as a function of the fuel age.

The thermal analysis carried out for the DSC, for 10-year-cooled used fuel with a heat load of 6.4 W/bundle, demonstrated that the fuel would be adequately cooled. 10-year-cooled Pickering reference used fuel has a lower heat load of 5.8 W/bundle (see Figure B-4); therefore the thermal analysis is considered to envelope the conditions for storage of Pickering used fuel in DSCs. The experimental measurements obtained during the thermal performance verification program at the PWMF in summer 1998 are consistent with the results of the thermal analysis (B-OPG04a)

A structural integrity assessment for DSCs (B-OPG14b) considered fuel bundles with a significantly higher decay heat of 7.4 W, which is conservative with respect to the PWMF conditions. The resulting thermal gradient in the concrete base of the DSC was estimated to be 54°C (B-OPG14a). The predicted stresses generated in the concrete by the thermal gradient of 54°C indicate that through wall cracking will not occur and thermal expansion does not compromise the structural integrity of the DSC.

B.6.2 Malfunctions and Accidents Assessment for Operations during On-Site Transfer

B.6.2.1 Dry Storage Container Drop during On-Site Transfer

Transporter design features and administrative control requirements are expected to ensure that the transporter will not collide with another vehicle during DSC transfer. However, a bounding assessment has been carried out to envelope the radiological consequences resulting from the drop of a DSC due to an unforeseeable accident during on-site transfer.

B.6.2.1.1 Drop of a Dry Storage Container during Transfer from an Irradiated Fuel Bay to the Dry Storage Container Processing Building

Consider the case where the transporter collides with another vehicle during a DSC transfer from an IFB to the DSC processing building. The transfer clamp has been designed to withstand the impact resulting from collision with another vehicle, and will ensure that the lid will stay on the DSC. Therefore, only the airborne release of tritium, krypton-85 and $^{14}CO_2$ from the DSC cavity is considered for this assessment.

Failure of 100 percent of a DSC's used fuel content is assumed i.e., 100 percent of the fuel elements in all the 384 fuel bundles, for a total of 10,752 failure fuel elements. The free inventory of tritium, krypton-85 and $^{14}CO_2$ in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barriers provided by the transfer clamp, the elastomeric seal and the sub-atmospheric pressure inside the DSC cavity, it is assumed that these radionuclides are released immediately into the environment.

Based on the methodology presented in Section B.4.4.1, the total dose to the public due to this event was assessed to be 5.10×10^{-3} mSv for an adult and 6.14×10^{-3} mSv for an infant at the Pickering site boundary. Based on the methodology presented in Section B.4.4.2, the dose to the worker due to this event was assessed to be 5 mSv.

B.6.2.1.2 Drop of a Dry Storage Container during Transfer from the Pickering Waste Management Facility Phase I Site to the Pickering Waste Management Facility Phase II Site

Another possible scenario involving collision of the DSC transporter with another vehicle could take place during DSC transfer between the PWMF Phase I and Phase II sites. Along this transfer route, a Transporter would carry a seal welded DSC. Although the seal weld is extremely robust, the collision is postulated to compromise the seal weld.

Failure of 100 percent of a DSC's used fuel content is assumed i.e., 100 percent of the fuel elements in all the 384 fuel bundles, for a total of 10,752 failed fuel elements. The free inventory of tritium, krypton-85 and $^{14}CO_2$ in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barriers provided by the seal weld and the sub-atmospheric pressure inside the DSC cavity, it is assumed that the radionuclides are released at once into the environment.

Based on the methodology presented in Section B.4.4.1, the total dose to the public due to this event was assessed to be 6.50×10^{-3} mSv for an adult and 7.91×10^{-3} mSv for an infant at the Pickering site boundary. Based on the methodology presented in Section B.4.4.2, the dose to the worker due to this event was assessed to be 5 mSv.

B.6.2.2 Transporter Failure

In the event of transporter failure, the containment barrier provided by the transfer clamp and elastomeric seal is assumed to fail as a result of the longer than expected time taken to transfer the DSC from the IFBs to the DSC processing building.

Conservatively, it is assumed that the free inventory of tritium, krypton-85 and carbon-14 in four damaged fuel elements is released into the DSC cavity (if one percent of all bundles contain one damaged element, there would be approximately four damaged elements in each DSC). The barrier provided by the transfer clamp and elastomeric seal are ignored and these radionuclides are considered to be released at once into the environment.

Based on the PWMF and WWMF operating experience, the frequency for this event is three events per year. The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.6.2.3 Fire

The potential for an accident involving DSC contact with a source of combustible material during on-site transfer has been considered. Fire sources directly along the transfer route of the DSC following the north side of the station units could include hydrogen cylinders, hydrogen trailers and stationary tank set, compressed gas bottles, oil storage tanks and the fuel tanks of other vehicles. The combustible materials that could be contributed by the transporter itself are the diesel fuel in the tank, engine lubricating oil and hydraulic oil. It is expected that such a fire would be of short duration. The duration of the fire would be further limited as a result of the fire detection and suppression systems in the transporter design and the expected response of the Pickering NGS Emergency Response Team.

The effect of a fire could potentially be to increase the temperature of the DSC and the used fuel bundles inside the DSC. Given the large thermal inertia of the DSC and the limited duration of the event, it is concluded that a fire along the transfer route could not cause overheating or damage to the used fuel.

No releases of radioactivity are expected from this scenario and, as such, there would be no public or occupational dose consequences.

In the event of vehicle failure, the dose consequence to the public and workers has been described in Section B.6.2.2.

B.6.2.4 Criticality

Criticality considerations for used fuel stored in DSCs can be based on the criticality assessments previously carried out for the PWMF (B-OH98) and the WWMF (B-OPG04b). The internationally accepted criterion for assuring subcriticality in such storage facilities is that k_{eff} should be less than 0.95. Consistent with expectations for irradiated natural uranium fuel, the earlier analyses and assessments have yielded adequate subcriticality margin and have demonstrated that there will be no criticality of used CANDU fuel, even in DSCs filled with water.

The specific cases previously analyzed using the WIMS-AECL code for a DSC containing Pickering used fuel are given below:

Environment Inside DSC	k∞	keff
Dry inert atmosphere (187 MWh/kgU)	0.4734	0.2114
Flooded with H20 (187 MWh/kgU)	0.7815	0.7327
Flooded with H20 (27 MWh/kgU)	0.8119	0.7610

[redacted]

[redacted]

The reference fuel burnup for PWMF shielding analysis purposes is [redacted] MWh/kgU. The level of conservatism in dose rate calculations increases as the reference burnup is increased. For criticality considerations, it is more conservative to assume a lower burnup, since CANDU fuel reactivity reaches a maximum around the plutonium peak (~ 27 MWh/kgU) and decreases with further fuel irradiation. The criticality analysis used a nominal discharge burnup of 187 MWh/kgU and also considered, as a limiting case, the plutonium peak burnup of 27 MWh/kgU. These assumptions in the criticality assessments provide a conservative upper bound for the potential for criticality in Pickering reference used fuel stored in DSCs. It is therefore concluded that 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under normal conditions or under any postulated accident scenario.

B.6.2.5 Common Mode Incidents

B.6.2.5.1 Adverse Road Conditions

Procedural controls are in place to prohibit DSC transfer under poor road conditions or until potentially slippery conditions can be corrected by, for example, sanding or salting of the transfer route. Even if the transfer vehicle were to lose traction on a slippery surface resulting in the vehicle leaving the road, a release of radioactivity from a clamped or seal-welded DSC is not expected given the robust design of a DSC, which is intended to withstand transportation accident loads.

In the event the on-site transfer of a DSC takes longer than expected a result of adverse road conditions, the radiological consequences would be within the envelope of those in Section B.6.2.1.

B.6.2.5.2 Earthquake

The Design Basis Seismic Ground Motion for the Pickering site is defined as an earthquake with a Peak Ground Acceleration (PGA) of 0.05g. The frequency of a Design Basis Earthquake (DBE) with an acceleration of 0.05g is approximately 1×10^{-3} events per year; this categorizes this event as credible.

Since the transporter with a DSC is not on the road 100 percent of the time, the combined occurrence of having a DBE and the transporter on the road simultaneously was calculated. The following assumptions were taken into account:

(a) An average of 70 DSCs are transferred each year between the station IFBs and the PWMF DSC processing building.

(b) The transporter is assumed to travel at a speed of up to 12 km/h.

(c) The greatest distance the transporter needs to travel between the IFBs and the DSC processing building is approximately 1 km.

(d) The transporter is conservatively assumed to take a much longer time during transfer and be on the road for 1 hour.

With these assumptions, the probability of finding a loaded DSC in transit during a 1-year period would be:

$$70 \times 1 \times (1/24) \times (1/365) = 8 \times 10^{-3}$$

The frequency of a DBE occurring at a time when a DSC is being transferred is:

$$(1 \times 10^{-3}) \times (8 \times 10^{-3}) = 8 \times 10^{-6} \text{ events/year}$$

If the on-site transfer of the DSC from the IFBs to the DSC processing building occurs during an earthquake, the DSC will not topple over due to the forces from the DBE for horizontal and vertical PGA of 0.12g, which is higher than the 0.05g corresponding to a postulated DBE at Pickering B.

In the event the on-site transfer of the DSC from the IFBs to the DSC processing building may take longer than expected as a result of an earthquake, the consequences would be within the envelope of those in Section B.6.2.1.

DSCs transferred from the PWMF Phase I site to the PWMF Phase II site will be seal-welded; this scenario is, therefore, not expected to result in any radioactivity dose consequence to workers or to the public.

B.6.2.5.3 Tornadoes

Tornadoes normally occur in unstable atmospheric conditions when warm moist air comes into contact with cold air. A tornado is a rotating thunderstorm with a vortex of air extending downward from a thundercloud. The strong updraft in a thunderstorm interacts with strongly sheared winds causing rotation of the updraft that intensifies to become a tornado. The Design Basis Tornado (DBT) defined for the Darlington nuclear site (B-OPG12) is defined as follows:

- Rotational wind speed of 322 km/h,
- Translational wind speed of 96 km/h,
- Pressure drop of 9.6 kPa,
- Rate of pressure drop of 5.6 kPa/s and
- Radius of maximum rotational wind speed of 46 m.

These parameters are considered to be large enough to envelope any credible tornadoes in Southern Ontario. Based on the Pickering NGS site wind speed frequencies, the DBT-definition rotational wind speeds correspond to a mean frequency of 3.13 × 10^{-6} events per year.

During tornado winds, objects can be picked up by the wind forces and accelerated to high velocities. Reference (B-OPG12) has established a spectrum of tornado-generated missiles considered in the Darlington NGS design as part of the DBT:

(a) Woodplank,102 mm × 305 mm × 3.7 m, mass 91 kg, velocity 335 km/h (80 percent of total tornado velocity, rotational plus translational).

(b) Steel pipe, 76 mm diameter, schedule 40, 3 m long, mass 35.4 kg, velocity 168 km/h (40 percent of total tornado velocity).

(c) Steel rod, 25 mm diameter × 914 mm long, mass 3.6 kg, velocity 251 km/h (60 percent of total tornado velocity).

(d) Steel pipe, 152 mm diameter, schedule 40, 4.6 m long, mass 129 kg, velocity 168 km/h (40 percent of total tornado velocity).

(e) Steel pipe, 305 mm diameter, schedule 40, 4.6 m long, mass 337 kg, velocity 168 km/h (40 percent of total tornado velocity).

(f) Utility pole, 343 mm diameter, 10.7 m long, mass 676 kg, velocity 168 km/h (40 percent of total tornado velocity).

(g) Automobile, frontal area 1.9 m^2, mass 1,800 kg, velocity 84 km/h (20 percent total tornado velocity).

The effect of tornado-generated missiles listed by the Southern Ontario DBT impacting on the DSC has been evaluated (B-AECL03). The analysis showed that the transfer clamp will keep the lid in place, the containment will not be breached, and the DSC will not overturn under the impact of postulated missiles during on-site transfer. The safety factor against overturning was found to be greater than 5 for the DSC loaded with used fuel.

The probability of finding a loaded DSC in transit from the IFBs to the processing building is 8 × 10^{-3} in a year. Therefore, the frequency of a tornado occurring at a time when a DSC is being transferred is $8 \times 10^{-3} \times 3.13 \times 10^{-6} = 2.5 \times 10^{-8}$ events per year, which is below the cut-off frequency of 10^{-6} per year. This event is therefore considered incredible.

B.6.2.5.4 Thunderstorms

Thunderstorms can potentially involve lightning striking a loaded DSC on the transporter during on-site transfer. Should an unexpected storm occur resulting in a lightning strike during transfer, the outer metal frame of either the transporter or the DSC outer metal shell are expected to carry most of the electricity to ground. In such an event, it is expected that the lid will stay in place, since the thunderstorm conditions affecting the DSC during on-site transfer would be much less than the DBT and DBE.

Given the results from the analyses carried out to study the effects of the DBT and DBE on the DSC during on-site transfer, it is expected that the integrities of both the DSC and DSC transfer clamp would not be affected if subjected to thunderstorms.

In case the on-site transfer of a DSC takes longer than expected as a result of thunderstorms, the consequences would be within the envelope of those in Section B.6.2.1.

For the transfer of DSCs from the PWMF Phase I site to the PWMF Phase II site, DSCs will be seal-welded and, consequently, no radiological dose consequence is expected to result from this accident scenario.

B.6.2.5.5 Floods

The only possibility for flooding at the Pickering site would be as a result of extreme local meteorological events. A Review Level Condition (RLC) for Probable Maximum Precipitation (PMP) has been developed to be used at OPG sites, which represents a rainfall of 420 mm in a 12-hour period, of which 51% (214 mm) falls within a one hour period.

Transfer procedures require that loaded DSCs not be transferred during anticipated extremely adverse weather conditions. In addition, sufficient warning time should be available for site staff to prevent this scenario from occurring.

If transport of a DSC during an extreme rainfall were to occur, extensive flooding would likely affect the operation of the transporter, however there would be no detrimental effect on the DSC. The DSCs are designed to tolerate water immersion at 2 MPa (B-OPG11), so the temporary water levels would not be of a concern to the radiological safety. In this event any potential radiological consequences would be within the envelope of those in Section B.6.2.1.

B.6.2.5.6 Explosions Along the Transfer Route during Dry Storage Container Transfer

There are several sources of explosion along the on-site transfer route of the DSCs from the IFBs to the Phase I processing building, such as acetylene cylinders and compressed gas bottle storage facility. Explosions originating from handling accidents of acetylene cylinders, compressed gas bottle explosion and pressure vessel burst leading to missiles due to normal wear and tear of oxygen, nitrogen or air cylinder have been assessed and the combined hazard frequency has been calculated to be 9.74×10^{-8} events per year (B-OPG18), which is lower than the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

Explosion hazards along the onsite transfer route of the DSCs from the Phase I processing building to the Phase II storage building have been assessed. The following hazard scenarios have been taken into consideration to be capable of toppling a passing DSC:

- Acetylene cylinder detonation

- Propane storage tank Boiling Liquid Expanding Vapour Explosion (BLEVE)
- Vapour Cloud Explosion (VCE) due to a propane storage tank rupture.

The combined explosion hazard frequency has been determined to be 5.2×10^{-8} events per year (B-OPG18), assuming 20 DSC shipments a week. Even based on this extremely conservative approach, the explosion hazard frequency is lower than the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

B.6.2.5.7 Turbine Missile Strike

The frequency of turbine missiles impacting structures, systems and components (SSC) is about 6×10^{-6} events per year.

The probability of transferring a loaded DSC between the IFBs and the processing building is 8×10^{-3} over a year.

The frequency of turbine missiles impacting a loaded DSC while the DSC is being transferred from either the IFB to the processing building or from the processing building to Storage Building 3 is: $6 \times 10^{-6} \times 8 \times 10^{-3} = 4.8 \times 10^{-8}$ events per year, which is below the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

B.6.2.5.8 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of PWMF Phase I and Phase II.

PWMF Phase I DSCs/DSMs occupy a total area of approximately 533’ (162.5 m) x 312’ (95 m) and the building height is 45’ (13.7 m). Phase I was assumed to be a rectangular shape that includes the Processing Building, Storage Buildings 1 and 2 and the RCS area.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase I was determined to be 5.95×10^{-7} events per year (B-OPG19).

PWMF Phase II DSCs occupy a total area of approximately 238’ (72.5 m) x 278’-9” (84.9 m), and the building height is 31.5’ (9.6 m). Phase II was assumed to be a rectangular shape that only includes Storage Building 3.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase II was determined to be 2.92×10^{-7} events per year (B-OPG19).

The LiftKing transporter, which is slightly larger than the Gen4 transporter, has been used to assess this hazard with respect to DSC in transit. The transporter has an overall length of 8.5 m, an overall width of 3.3 m and an overall height of 4.7 m.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency impacting the transporter has been determined, considering the limited time that a loaded DSC transporter will be in transit and taking into account that the transporter is a small moving target.

The frequency of an aircraft crash impacting the DSC while being transferred is 3.68×10^{-10} events per year (B-OPG18).

The total frequency for an aircraft crashing into a DSC/DSM at either Phase I or Phase II is $5.95 \times 10^{-7} + 2.92 \times 10^{-7} + 3.68 \times 10^{-10} = 8.87 \times 10^{-7}$ events per year, which is below the event cut-off frequency of 10^{-6} events per year. Therefore, the aircraft crashing into a DSC/DSM at the PWMF is considered incredible.

B.6.2.5.9 Toxic Gas Release – Chlorine Originating from Ajax Water Treatment Plant

The Ajax Water Treatment Plant uses chlorine cylinders for water treatment. The facility is located at approximately 4.1 km from the Phase II Storage Building 3 and the route of the DSC transfer from the processing building. The Screening Distance Value (SDV) for chlorine is 4.4 km, hence this hazard cannot be screened out based on distance.

Chlorine leakage from the Ajax Water Treatment may have an impact on the transporter operator's ability to keep the transporter safely on the road. Even if the operator illness were to result in the transporter leaving the road, a release of radioactivity from a seal-welded DSC is not expected. The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.6.2.5.10 Soil Failures/Slope Instability

In the event the on-site transfer of a DSC takes longer than expected as a result of adverse road conditions due to soil failure or slope instability, the radiological consequences would be within the envelope of those in Section B.6.2.1.

B.7.0 ASSESSMENT OF THE DRY STORAGE CONTAINER DURING PROCESSING OPERATIONS

The transporter takes DSCs from the IFBs to the DSC processing building. DSCs are then taken through the following steps before they are ready for storage:

(a) The DSC lid is welded and the inspection of the lid seal weld is completed.

(b) The DSC is subjected to vacuum drying and helium backfill.

(c) The drain port plug is installed and seal welded in place.

(d) The DSC is helium leak-tested by using a vacuum chamber (bell jar).

(e) A radiological survey is performed on the DSC and spot cleaning is carried out, as required.

(f) Paint is applied to the welded areas and to any scrapes or scuffs on the DSC exterior.

(g) IAEA safeguards seals are applied.

(h) The transporter moves the DSC to the DSC storage building for final storage.

B.7.1 Normal Operating Conditions during Dry Storage Container Processing

B.7.1.1 Dose Rates Inside the Dry Storage Container Processing Building

The DSC processing building is a single-story structure with a two-story amenities area. The processing building's workshop space is approximately 830 m², divided into the workspaces that support the various DSC processing operations.

Conservatively, one can assume that there are approximately 20 loaded DSCs occupying the designated staging and processing stations in the workspaces. With approximately 20 DSCs in the building, the occupational dose rate of individual workers is dominated by their proximity to a single loaded DSC in any of the designated stations. Workers may routinely handle DSCs containing 10-year-cooled used fuel, the minimum age of used fuel permitted to be loaded into a DSC for routine storage.

Figure B-8 shows the gamma dose rates calculated as a function of distance from the top, bottom, front and side surfaces of a DSC loaded with reference used fuel (10-year-cooled). In the Figure, the label "front" corresponds to the wider face of the container (i.e., the face bearing the lift plates), and the label "side" indicates its narrower face.

Dose rates inside the PWMF have been demonstrated to be acceptably low (most working areas are normally at or near ambient background radiation levels). The actual dose rates from working with DSCs have consistently been found to be much lower than predicted. Results of historical radiation monitoring at the PWMF indicate that contact dose rates on the front and side surfaces of DSCs loaded with used fuel that has cooled for 10 years or more have varied between 9 μSv/h and 13 μSv/h.

As shown in the drawing 92896-D0A-29660-1001, there are some offices in the building on the second floor. Workers in offices are not likely to be in line of sight with any one of the approximately 20 DSCs that can be present in the workshop area.

Analysis has shown 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.

B.7.1.2 Radioactive Emissions during Dry Storage Container Processing

There is a small potential for airborne emissions resulting from DSC processing operations such as welding and vacuum drying. Surface contamination on DSC exterior surfaces is effectively controlled through prevention measures and decontamination at the Pickering NGS IFBs. Nevertheless, small quantities of fixed surface contamination may become airborne during welding operations. Such airborne particulate contamination, if present, would be effectively removed by the High Efficiency Particulate Air (HEPA) filters in the active ventilation system.

Radioactive contamination may be present on the outside of the fuel cladding. Although this contamination is expected to adhere to the fuel during storage, there is some potential for it to become airborne during vacuum drying of the DSC cavity. A dedicated hose is used for DSC vacuum drying operations to prevent the spread of such contamination to other workshop systems. Vacuum skid discharge is directed to the active ventilation system, where particulate contamination is removed by HEPA filters. Fuel temperatures during vacuum drying have been assessed and are expected to remain well within safe temperatures for maintaining the long-term integrity of the used fuel cladding.

Additionally, PWMF experience demonstrates that particulate emissions in exhaust from DSC processing operations have been typically below the Minimum Detectable Activity (MDA) level.

However, because DSCs are unclamped and unwelded at some stages during processing, an assessment of the chronic radioactive emissions during processing is presented below.

A very small quantity of fuel elements may have minor defects in their fuel cladding. OPG fuel performance experience has demonstrated that cladding defects are present in less than 0.01 percent of fuel bundles (representing < 0.001 percent of fuel elements). Residual releases from these defective elements have been assessed and described below.

For the purpose of evaluating the potential emissions under normal operating conditions, the following conservative assumptions are used to obtain an estimate for chronic airborne emissions:

(a) One fuel element in 1 percent of fuel bundles is damaged during handling (four elements per DSC) and for each failed fuel element the free inventory of tritium, krypton-85 and carbon-14 is released into the DSC cavity.

(b) The barrier provided by the DSC lid seal-weld is ignored and these radionuclides are released into the environment.

These assumptions are deemed conservative for the following reasons:

- The postulated defect rate is more than 20 times higher on a per element basis than OPG fuel performance experience.

- Fuel element defects occur primarily in the bundle manufacturing process or as a result of debris fretting in the reactor core. At high fuel temperatures during irradiation, the free inventory of tritium, krypton-85 and carbon-14 in elements with cladding defects would have been released within the reactor core.

- Used fuel is stored for at least 10 years in the Pickering IFBs prior to transfer to a DSC. Leaching of grain-boundary inventory and release of gap inventory would have additionally occurred over this period for bundles with minor cladding defects.

- Should free inventory remain in the fuel-sheath gap or grain boundaries subsequent to in-bay storage, its release would have occurred during initial vacuum drying inside the Pickering IFBs.

- Since the facility processes about 70 DSCs per year, it is postulated that a total of 280 fuel elements fail during 1 year under normal operating conditions (a very conservative scenario).

- The chronic off-site dose consequences from this scenario for the most exposed age group and location at the Pickering NGS site boundary and beyond is estimated to be less than 1.0×10^{-3} µSv/year

Since the above assumptions are very conservative, the assessed consequence is considered an upper bound for any possible chronic emissions during normal operating conditions.

B.7.2 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Processing

B.7.2.1 Drop of a Dry Storage Container during Handling

Failure of the lifting beam, lift plates, the crane, or the DSC trunnions could potentially result in dropping a loaded DSC while it is being lifted during operations at the DSC processing building. Fuel sheath failure is not expected from a DSC drop from the low lift height of DSC processing building operations.

The failure probability of crane lifting very heavy loads, based on US nuclear plant operating experience (B-NRC03), is estimated to be 5.6×10^{-5} per demand. Based on the combined operating experience of the PWMF, DWMF and WWMF, the number of lifts to be carried out at the DSC processing building using the crane is approximately 600 per year. Therefore, the total frequency of crane failure would be 3.36×10^{-2} events per year.

A handling accident involving the dropping or tip over of multiple DSCs is not considered to be a credible event. Should a crane accident result in the drop of a clamped DSC or seal welded DSC, the low lift height inside the DSC processing building would prevent the container from tipping over and striking a second DSC. In two instances, for a very short duration, the seal-welded DSC is lifted above low-lift height: when it is loaded into the bell jar (1.5 m) and when it is placed on the drain port welding stands. In these instances, it is assumed that the DSC could drop and tip over as a result of a crane failure. A release from this event is not expected.

However, even in the worst-case scenario, dropping a DSC during handling is not expected to result in failure of more than 30 percent of a DSC's used fuel elements, a total of 3,226 failed fuel elements. Realistically, fuel sheath failure is not expected to result from an accidental DSC drop from the low lift height of the transporter or from the crane in the DSC processing building.

The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements 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 dose consequences for this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.2 Equipment Drop onto a Dry Storage Container

The crane auxiliary hoist is used to handle other processing equipment, including the DSC transfer clamp, lid welding equipment, and vacuum bell jar lid. A structural failure of any lifting/rigging equipment such as slings, shackles, or other specialty equipment lifting points or lifting beams while suspended by the auxiliary crane could result in a drop of equipment onto the lid of a loaded DSC.

These accident scenarios are unlikely given that the rated load capacity of the auxiliary hoist and the lifting/rigging equipment are not exceeded and routine inspections and pre-operational checks are performed.

However, to calculate the event frequency of equipment dropping onto a DSC, the following assumptions were made:

(1) An average of 70 DSCs could be processed each year;

(2) A total of four pieces of equipment have the potential to drop onto a DSC lid; and

(3) With the exception of the transfer clamp, each structure is lifted twice over the DSC (installation and removal), totalling seven lifts per DSC.

Assuming that the probability of the equipment failure is 5.6×10^{-5}, the total frequency of a drop of equipment onto a loaded DSC lid would be $5.6 \times 10^{-5} \times 70 \times 7$, which equals 2.74×10^{-2} events per year.

The dose consequences for this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.3 Dry Storage Container Collision during Craning

A DSC craning accident due to operator error could result in a loaded DSC colliding with another DSC (loaded or empty) on the DSC processing building floor or other process building equipment or structure.

Assuming that 70 DSCs are processed per year, the total number of times a loaded and unwelded DSC is lifted would be approximately 80 (one lift per DSC plus 10 percent of them are assumed to have weld failure and required weld repairs). The postulated frequency of a loaded and unwelded DSC craning collision accident is $(1.0 \times 10^{-3}) \times 80 = 8 \times 10^{-2}$ events per year.

Given that the overhead crane bridge and trolley maximum speeds are limited by design, the dose consequences from this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.4 Transporter Collision with a Loaded Dry Storage Container or Another Transporter

Operator error during transporter vehicle operations could result in a collision with a loaded DSC on the DSC processing building floor or with another transporter in the DSC processing building. The transporter collision could occur while it is carrying a loaded or empty DSC.

It is assumed that an average of 70 DSCs are loaded each year and that the transporter is used four times for each DSC during processing (pick up of an empty DSC for transfer to the IFBs, transfer of a loaded DSC to the DSC processing building, transfer of a seal-welded DSC to the paint station at the DSC Storage Building 1, and transfer to storage). Therefore, the probability of a transporter collision with a loaded DSC or with another transporter in the processing building due to operator error would be $(1.0 \times 10^{-3}) \times 70 \times 4$, which equals 2.8×10^{-1} events per year.

Given that the transporter vehicle speed is limited by design and that it is equipped with front and rear bumper emergency stops or sensors, the dose consequences from this scenario would be bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.5 Equipment Collision with a Loaded Dry Storage Container during Craning

A craning accident due to operator error could result in process equipment colliding with a loaded DSC while suspended from the auxiliary hoist. To calculate the frequency of this event, the following assumptions were made:

(1) An average of 70 DSCs could be processed each year;

(2) A total of 4 pieces of equipment have the potential to collide with a loaded DSC while suspended from the auxiliary hoist (transfer clamp, lid welding equipment, and vacuum bell jar lid); and

(3) With the exception of the transfer clamp, each structure is lifted twice over the DSC (installation and removal), totalling seven lifts per DSC.

The total postulated frequency of this event due to operator error is $(1.0 \times 10^{-3}) \times 70 \times 7$, which equals 4.9×10^{-1} events per year.

Given that the overhead crane bridge and trolley maximum speeds are limited by design, the dose consequences from this scenario are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.6 Criticality

See Section B.6.2.4.

B.7.2.7 Dry Storage Container Processing Building Fire

The DSC processing building has been designed in accordance with the National Building Code of Canada (NBCC) and the National Fire Code of Canada (NFCC).

The Fire Hazards Assessment (FHA) for PWMF (B-OPG17c) demonstrated that the bounding fire scenario for the processing building was a fire involving one DSC transporter. The DSC processing building contains a limited quantity of combustible materials: for example, the transporter contains a small quantity of diesel fuel. The welding cover gases used at the workshop are inert and will not burn or explode. As a result, a DSC transporter fire would likely ignite the adjacent combustibles located in the workshop. The fire would be of short duration due to the limited quantity of combustible material, fire detection, protection systems in the building, and the expected prompt arrival of emergency response personnel.

The effect of a fire would be to increase the temperature in the proximity of the DSC. Given the large thermal inertia of the DSC and the limited duration of the event, it is expected that a fire inside the processing building is not expected to cause fuel overheating or fuel damage. Therefore, no releases are expected from this scenario and therefore there would be no public or occupational dose consequences as a result of this event.

B.7.2.8 Common Mode Incidents

B.7.2.8.1 Earthquake

The DSC processing building has been designed to NBCC-1990 seismic requirements; it would not be expected to collapse in the event of an earthquake with a ground motion equal to or small than 0.05g.

An analysis was performed to determine the impact of a collapsing DSC processing building on an unclamped and un-welded DSC lid (B-OPG13a). It was concluded that there is some potential for damage to the DSC lid and base, although the DSC outer liner remains intact. There is no likelihood of the contained fuel being exposed due to the complete removal of the lid or tipping of the DSC container.

Calculations were performed (B-OH92) to assess the DSC seismic stability for the lower probability Pickering ‘A’ DBE ground motion parameters of 0.12g horizontal PGA and 0.08g vertical PGA. These parameters bound both the Pickering ‘B’ DBE and the NBCC ground motion parameters for the Pickering site. The calculations demonstrated that the safety factor against overturning of the DSC is 3 and the safety factor against sliding is 1.54.

The hazard for the DSC overturning or tipping under the loads described for an earthquake scenario is bounded by the case when the processing building collapses.

Nonetheless, in the event that an earthquake causes a DSC to be dropped during handling, the dose consequences are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.8.2 Tornadoes

As discussed in Section B.6.2.5.3, a DSC can resist overturning in tornado winds of up to 425 km/h. This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the interceding building structures.

The effect of tornado-generated missiles on a clamped DSC has also been considered in Section B.6.2.5.3; the transfer clamp will keep the lid in place, the containment will not be breached, and the DSC will not overturn under the impact of postulated missiles.

No radiological dose consequence is expected to result from either of the above accident scenarios.

It has been postulated that the processing building is subject to a tornado at the time that an unclamped DSC stands in preparation of seal-welding, and that an unclamped DSC is struck by a tornado-generated missile.

The DSC has already been shown to be able to withstand the direct impact of a tornado missile without any release; therefore, a missile striking the DSC is expected to have negligible consequences, even if the DSC is un-welded. In addition, the likelihood of both the tornado (3.13×10^{-6} events per year) and preparation of the DSC for seal welding scenarios occurring simultaneously is below the cut-off frequency of 10^{-6} events per year. This event is therefore considered incredible.

B.7.2.8.3 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC processing building. As per design requirements, the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during processing. Thunderstorms would result in no releases from DSCs in the processing building and there would be no public or occupational dose consequences.

B.7.2.8.4 Floods

Water entry into the facility originating from a PMP event is possible but the consequences are assumed to be negligible. The simulated water depth is too shallow to be anywhere near the level of the DSC lid. As well, no loose contamination is permitted on the exterior DSC surfaces or on accessible surfaces within the processing building, therefore there is no potential for waterborne contamination.

B.7.2.8.5 Turbine Missile Strike

The PWMF Phase I is located southeast of Unit 8, with Storage Building 2 situated the closest at a distance of 30 m. The DSC Storage Building 2 is attached to the north wall of the processing building and the Storage Building 1.

The frequency of turbine missiles impacting SSCs has been determined to be 6×10^{-6} events per year. Based on the location of the DSC processing building with reference to the Unit 8 turbine, a turbine missile striking the processing building and then the DSC is considered to be an incredible event.

B.7.2.8.6 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of PWMF Phase I and Phase II.

PWMF Phase I DSCs/DSMs occupy a total area of approximately 533’ (162.5 m) x 312’ (95 m) and the building height is 45’ (13.7 m). Phase I was assumed to be a rectangular shape that includes the Processing Building, Storage Buildings 1 and 2 and the RCS area.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase I was determined to be 5.95×10^{-7} events per year (B-OPG19).

PWMF Phase II DSCs occupy a total area of approximately 238’ (72.5 m) x 278’-9” (84.9 m), and the building height is 31.5’ (9.6 m). Phase II was assumed to be a rectangular shape that only includes Storage Building 3.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase II was determined to be 2.92×10^{-7} events per year (B-OPG19).

The LiftKing transporter, which is slightly larger than the Gen4 transporter, has been used to assess this hazard with respect to DSC in transit. The transporter has an overall length of 8.5 m, an overall width of 3.3 m and an overall height of 4.7 m.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency impacting the transporter has been determined, considering the limited time that a loaded DSC transporter will be in transit and taking into account that the transporter is a small moving target.

The frequency of an aircraft crash impacting the DSC while being transferred is 3.68×10^{-10} events per year (B-OPG18).

The total frequency for an aircraft crashing into a DSC/DSM at either Phase I or Phase II is $5.95 \times 10^{-7} + 2.92 \times 10^{-7} + 3.68 \times 10^{-10} = 8.87 \times 10^{-7}$ events per year, which is below the event cut-off frequency of 10^{-6} events per year. Therefore, aircraft crashing into a DSC/DSM at the PWMF is considered incredible.

B.7.2.8.7 Release of Oxidizing, Toxic, Corrosive Gases and Liquids Stored in the Processing Building

For a chemical release to have an impact on nuclear safety, the chemical must fall into one of the following categories under Part IV of the Canadian Controlled Products Regulations (B-CAN87):

- Acute Toxicity;
- Corrosive;
- Oxidizing/Reactive; and
- Asphyxiant.

All chemicals held within the PWMF Phase I were categorized and screened based on quantities held on site. Direct exposure to the categorized chemicals may cause an operator to become incapacitated, leading to container mishandling errors. The dose consequence from this event is bounded by the drop of a dry storage container.

B.8.0 DRY STORAGE CONTAINER STORAGE BUILDINGS ASSESSMENT

Once the DSC processing is completed, the transporter moves the DSC to one of the DSC storage buildings for storage.

B.8.1 Normal Operating Conditions

B.8.1.1 Direct Radiation Fields

B.8.1.1.1 Dose Rates Outside the Pickering Waste Management Facility PWMF Phase I Site

When the PWMF DSC storage buildings 1 to 3 are filled to nominal design capacity, the dose rate at the site boundary is calculated to be 1.04×10^{-3} μSv/h. This is equivalent to an annual dose of 2.08 μSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 7.23×10^{-4} μSv/h. This is equivalent to an annual dose of 0.72 μSv based on 1,000 hours occupancy; this is a conservative occupancy assumption for boaters and fishermen.

Dose rates at the perimeter fence east of Phase 1 are calculated to be less than 0.24 μSv/h. The predicted dose rates are less than 50% of the dose rate target of 0.5 μSv/h.

These results indicate that the PWMF dose rate target of ≤0.5 μSv/h at the station security fence on a quarterly average basis and the administrative dose target ≤100 μSv/y at the station site boundary (ten percent of the CNSC regulatory dose limit for members of the public), as set out in Section B.5.0, are met during UFDS operations.

B.8.1.1.2 Dose Rates Outside the Pickering Waste Management Facility PWMF Phase II Site

The calculated dose rates at the perimeter fence, [redacted] from the DSC storage building 3, is 0.29 μSv/h. This dose rate is well within the criterion of ≤0.5 μSv/h established for limited occupancy (i.e., up to 2,000 hours per year) by non-NEW personnel at the perimeter fence.

As discussed in Section 3.4.3, 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 TMB are below the dose rate target of 0.5 μSv/h (see Section B.5.0). The dose rate at the TMB was estimated to be 6.58×10^{-2} μSv/h.

When the PWMF DSC storage buildings 1 to 3 are filled to nominal design capacity, the dose rate at the site boundary is calculated to be 1.04×10^{-3} μSv/h. This is equivalent to an annual dose of 2.08 μSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 7.23×10^{-4} μSv/h. This is equivalent to an annual dose of 0.72 μSv based on 1,000 hours occupancy.

B.8.1.1.3 Dose Rates Inside the Dry Storage Container Storage Buildings

The predicted dose rates from a row of DSCs facing the corridor in the middle of Phase I storage building 2 loaded with 10-year cooled fuel in storage, are presented in Figure B-9. The results show that the dose rates on the west side are approximately 30 μSv/h, this drops to 12.5 μSv/h in the middle of the corridor and increases to 28 μSv/h on the east side. Note the dose rates presented for Storage Building 2 will bound those of Storage Building 1.

The predicted dose rates from a row of DSCs facing the corridor in the middle of Phase II Storage Building 3 loaded with an average of 26 year old fuel in storage, are presented in Figure B-10. The DSC configuration is symmetric on the east and west side. The results show that the dose rates on the west or east side are approximately 18 μSv/h and this drops to 5 μSv/h in the middle of the corridor.

B.8.1.2 Radioactive Emissions during Storage

There are no mechanisms for airborne releases to occur under normal operating conditions during storage of seal welded DSCs.

B.8.2 Malfunctions and Accidents Assessments for Operations during Dry Storage Container Storage

B.8.2.1 Seal Weld Failure during Storage

Both the fuel sheath and the DSC lid 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.

The DSC lid and base are sealed with a full-penetration groove weld. Once the weld has cooled sufficiently for inspection, a 100 percent inspection of the weld is performed to check for any defects. The DSC is subsequently filled with inert helium and leak test prior to storage.

As the seal welds are inspected and pressure tested, and there is no external force acting upon DSCs in storage, it is concluded that a random weld failure is not a credible event. No instances of random-weld failures have been identified.

B.8.2.2 Dry Storage Container Drop during Transfer to Storage

Failure of the transporter or the DSC lift plates while the DSC is lifted by the transporter during transfer to placement of a loaded DSC in a DSC storage building could result in a DSC drop. This scenario is unlikely given the independent mechanical locking mechanism on each side of the transporter to prevent DSC drop.

Assuming that 70 DSCs are processed per year, the probability of both mechanical locking mechanisms failing simultaneously while carrying a loaded DSC in the storage building would be $(1.0 \times 10^{-4}) \times (1.0 \times 10^{-4}) \times 70 = 7 \times 10^{-7}$ events per year.

Given that the transporter is equipped with front and rear bumper emergency stops or sensors, the low-lift height of the DSC while in the Transporter, and that the loaded DSC has already been seal-welded at this stage of the process, no releases would result from this scenario. There would be no public or occupational dose consequences as a result of this event.

B.8.2.3 Transporter Collision with a Dry Storage Container or Another Transporter

Operator error during transporter operations could result in a collision with a loaded DSC on the DSC storage building floor or with another transporter in the DSC storage building.

It is assumed that an average of 70 DSCs are loaded and transferred each year. The transporter is used once in the DSC storage building for each DSC. The probability of a DSC drop or collision event in the DSC storage building due to operator error would be $(1.0 \times 10^{-3}) \times 70 = 7 \times 10^{-2}$ events per year.

Given that the transporter is equipped with front and rear bumper emergency stops or sensors, the low lift height of the DSC while in the transporter, and that the loaded DSC has already been seal-welded at this stage of the process, no releases would result from this scenario. There would be no public or occupational dose consequences as a result of this event.

B.8.2.4 Criticality

See Section B.6.2.4.

B.8.2.5 Dry Storage Container Building Fire

The DSC storage buildings have been designed in accordance with the NBCC and the NFCC.

The Fire Hazards Assessment (FHA) for PWMF (B-OPG17c) demonstrated that the bounding fire scenario for both Phase I and Phase II storage buildings is a fire involving one DSC transporter, with two transporters located in the storage building. The fire scenario was assessed crediting the available fire detection, suppression and emergency response. Due to the fire detection and alarm systems in all three storage buildings, and the expect prompt arrive of the emergency response personnel, the fire would be of short duration and localized. The effect of the fire would be to increase the temperature in the proximity of the DSC. Given the large thermal inertia of the DSC and the limited duration of the event, the fire inside the DSC storage building will not cause fuel overheating or fuel damage. Therefore, no releases would result from DSCs in storage and there would be no public or occupational dose consequences as a result of this event.

B.8.2.6 Common Mode Incidents

B.8.2.6.1 Earthquake

The earthquake scenario was described previously in Section B.6.2.5.2.

The DSC has a safety factor of 7 against overturning and 4 against sliding under the loads described for the earthquake scenario (B-OH92). The structure of the container is adequately strong to ensure the integrity of the DSC in case of an earthquake.

Additionally, the impact of a building truss collapse (in different orientations) onto an array of DSCs has been assessed (B-OPG11). It was concluded that the seal-welded DSCs would maintain their integrity if an earthquake caused a DSC storage building to collapse directly on top of them.

An earthquake would result in no releases from DSCs in storage and there would be no public or occupational dose consequences.

B.8.2.6.2 Tornadoes

As discussed in Section B.6.2.5.3, a DSC can resist overturning in tornado winds of up to 425 km/h. This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the interceding building structures.

The effect of tornado-generated missiles on a seal-welded DSC is bounded by a similar discussion regarding clamped DSCs in Section B.6.2.5.3; the containment will not be breached and the DSC will not overturn under the impact of postulated missiles.

A tornado would result in no releases from DSCs in storage and there would be no public or occupational dose consequences.

B.8.2.6.3 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC storage buildings. As per design requirements, the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during storage. Thunderstorms would result in no releases from DSCs in storage buildings and there would be no public or occupational dose consequences.

B.8.2.6.4 Floods

As discussed in Section B.6.2.5.5, given the characteristics of the Pickering NGS site, extensive flooding affecting a DSC storage building is not a credible event.

B.8.2.6.5 Hazardous Material Building Explosion

A hazardous materials storage building is located west of the PWMF Phase II site, within about 55 m of Storage Building 3.

With regard to the safety of hazardous materials in storage, OPG has a comprehensive Hazardous Materials (HazMat) Management Program in place to control the procurement, assessment, receipt, segregation, warehousing, use, storage in the field, and disposal of hazardous materials. Consequently, only materials that are considered ‘low flash’ are stored at this location. The building design has also been evaluated against NFCC and NBCC requirements.

Given the relatively small quantities of combustibles and the safety provisions already incorporated into the building, an accident leading to an explosion at the hazardous materials storage building is extremely unlikely. The scenario of an explosion from the hazardous materials building has been considered in Section B.6.2.5.6. This event is not expected to result in any releases from DSCs in storage and there would be no public or occupational dose consequences.

B.8.2.6.6 Toxic Materials Stored in Storage Building 3

The only toxic chemical stored in Storage Building 3 in a quantity greater than 5 liters is the HYVOLT II transformer oil, which, based on its Material Safety Data Sheet (MSDS) can be fatal if entering the airways. Inhalation of acute toxicity chemicals can lead to container mishandling caused by human error.

Strict safety procedures and processes are in place for storage and handling of the hazardous chemicals within the Processing Building. The handling of hazardous materials must meet provincial legislation, particularly the Occupational Health and Safety Act and the Environmental Protection Act.

The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.8.2.6.7 Turbine Missile Strike

The frequency of turbine missiles impacting SSCs has been determined to be 6×10^{-6} events per year.

The probability of a high-trajectory missile strike due to a turbine over speed failure is about an order of magnitude less than the number calculated above because an ejected missile would have greater energy and would, therefore, “peak” at a greater height. This effectively reduces the target area by an order of magnitude.

Based on the low frequency of a turbine missile impacting a SSC and taking into account the location of the Phase I DSC Storage Buildings and Phase II storage building with reference to the Unit 8 turbine, this hazard is considered to be an incredible event.

B.8.2.6.8 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of PWMF Phase I and Phase II.

PWMF Phase I DSCs/DSMs occupy a total area of approximately 533’ (162.5 m) x 312’ (95 m) and the building height is 45’ (13.7 m). Phase I was assumed to be a rectangular shape that includes the Processing Building, Storage Buildings 1 and 2 and the RCS area.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase I was determined to be 5.95×10^{-7} events per year (B-OPG19).

PWMF Phase II DSCs occupy a total area of approximately 238’ (72.5 m) x 278’-9” (84.9 m), and the building height is 31.5’ (9.6 m). Phase II was assumed to be a rectangular shape that only includes Storage Building 3.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase II was determined to be 2.92×10^{-7} events per year (B-OPG19).

The LiftKing transporter, which is slightly larger than the Gen4 transporter, has been used to assess this hazard with respect to DSC in transit. The transporter has an overall length of 8.5 m, an overall width of 3.3 m and an overall height of 4.7 m.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency impacting the transporter has been determined, considering the limited time that a loaded DSC transporter will be in transit and taking into account that the transporter is a small moving target.

The frequency of an aircraft crash impacting the DSC while being transferred is 3.68×10^{-10} events per year (B-OPG18).

The total frequency for an aircraft crashing into a DSC/DSM at either Phase I or Phase II is $5.95 \times 10^{-7} + 2.92 \times 10^{-7} + 3.68 \times 10^{-10} = 8.87 \times 10^{-7}$ events per year, which is below the event cut-off frequency of 10^{-6} events per year. Therefore, aircraft crashing into a DSC/DSM at the PWMF is considered incredible.

B.9.0 CONCLUSIONS

The above sections describe the safety assessment of the PWMF. The radiological dose consequences to the public and the workers during normal operations and under abnormal and credible accident conditions were evaluated.

It is concluded that under normal operation the dose consequences to the public would be well below 100 µSv per year, an OPG administrative dose target at the Pickering site boundary and ten percent of the regulatory dose limit. The dose consequences would also be less than 0.5 µSv per hour, an OPG target based on a quarterly average over a maximum of 2,000 hours per year occupancy at the PWMF Phase I and Phase II site fences. Occupational doses were also found to be below the dose limit of 50 mSv per year in any single year and 100 mSv over 5 years.

The radiological dose consequences to the public as a result of abnormal and credible accident conditions were concluded to be well below the dose limit of 1 mSv at or beyond the OPG property boundary; the associated dose consequences to a worker were concluded to be well below the target of 50 mSv. Tables B-6, B-7 and B-8 show the dose consequences for the postulated abnormal and accident conditions during on-site transfer, DSC processing, and DSC storage, respectively.

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B.11.0 GLOSSARY

ADDAM	Atmospheric Dispersion and Dose Analysis Method
AECL	Atomic Energy of Canada Limited
BLEVE	Boiling Liquid Expanding Vapour Explosion
CANDU	Canada Deuterium Uranium (a registered trademark of AECL)
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners' Group
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DCF	Dose Conversion Factor
DSC	Dry Storage Container
DSM	Dry Storage Module
DWMF	Darlington Waste Management Facility
FHA	Fire Hazards Assessment
HazMat	Hazardous Materials
HEPA	High Efficiency Particulate Air
HT	Tritiated Hydrogen Gas
HTO	Tritium Oxide or Tritiated Water Vapour
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IFB	Irradiated Fuel Bay
MCNP	Monte-Carlo N-Particle
MDA	Minimum Detectable Activity
MSDS	Material Safety Data Sheet
NBCC	National Building Code of Canada
NEW	Nuclear Energy Worker
NFCC	National Fire Code of Canada
NGS	Nuclear Generating Station
OPG	Ontario Power Generation
ORIGEN-S	Oak Ridge Isotope Generation Code
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
PWMF	Pickering Waste Management Facility
RLC	Review Level Conditions
SDV	Screening Distance Value
SORO	Simulation of Reactor Operation
SSC	Structures, Systems and Components
TMB	Training and Mock-Up Building
UFDS	Used Fuel Dry Storage
VCE	Vapour Cloud Explosion
WIMS	Winfrith Improved Multigroup Scheme
WWMF	Western Waste Management Facility

B.12.0 TABLES AND FIGURES

Table B-1: Pickering Reference Used Fuel Bundle Properties

Number of fuel elements	28
Length	495 mm
Mass of UO_2	22.87 kg
Mass of Zircaloy	1.67 kg
Mass of U	20.16 kg
Mass of the bundle	24.54 kg
Average bundle power	[redacted]
Exit burnup	[redacted]
Time after discharge	10 years

Table B-2: Pickering Waste Management Facility Reference Fuel Bundle Actinides and Heavy Metals Inventory (10-year-cooled fuel)

[redacted]

[20] 1 TBq = 10^{12} Bq.

Table B-3: Pickering Waste Management Facility Reference Fuel Bundle Fission Products Inventory (10-year-cooled fuel)

[redacted]

Table B-4: Pickering Waste Management Facility Reference Fuel Bundle Light Elements Inventory (10-year-cooled fuel)

[redacted]

Table B-5: Gamma Spectrum for a 10-year-cooled Pickering Reference Used Fuel Bundle

Lower MeV	Upper MeV	Photons/Second/Bundle
0.01	0.02	3.83E+12
0.02	0.03	2.17E+12
0.03	0.06	3.78E+12
0.06	0.1	1.72E+12
0.1	0.2	1.57E+12
0.2	0.4	7.66E+11
0.4	0.6	5.31E+11
0.6	0.7	1.65E+13
0.7	0.8	6.08E+11
0.8	1	1.59E+11
1	1.5	2.23E+11
1.5	2	9.67E+09
2	3	7.26E+08
3	4	4.20E+06
4	5	5.14E+03
5	6	1.73E+03
6	7	5.82E+02
7	8	1.96E+02
8	10	8.55E+01
10	14	4.05E+00
	Total	**3.41E+13**

Table B-6: Postulated Malfunctions or Accidents during Dry Storage Container On-Site Transfer

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Transporter failure	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
Transporter Operator Health-Related Emergency	credible	$<5.10 \times 10^{-3}$	$<6.14 \times 10^{-3}$	<5
DSC drop during on-site transfer from IFB to DSC Processing Building	credible	5.10×10^{-3}	6.14×10^{-3}	5
DSC drop during on-site transfer between the PWMF Phase I and Phase II sites	credible	6.50×10^{-3}	7.91×10^{-3}	5
Fire	incredible[21]	—	—	—
Criticality	incredible	—	—	—
Adverse road conditions	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5
Earthquake	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5
Tornado	incredible	—	—	—
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Explosion along transfer route	incredible	—	—	—
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—
Toxic gas releases – chlorine from Ajax water treatment plant	credible	0	0	0
Soil failure/slope instability	credible	$<6.50 \times 10^{-3}$	$<7.91 \times 10^{-3}$	<5

[21] The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.

Table B-7: Postulated Malfunctions or Accidents during Dry Storage Processing

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Drop of a DSC during handling	credible	<$5.10x10^{-3}$	<$6.14x10^{-3}$	<5
Equipment drop onto a DSC	credible	<$5.10x10^{-3}$	<$6.14x10^{-3}$	<5
DSC collision during craning	credible	<$5.10x10^{-3}$	<$6.14x10^{-3}$	<5
Transporter collision with a loaded DSC or another Transporter	credible	<$5.10x10^{-3}$	<$6.14x10^{-3}$	<5
Equipment collision with a loaded DSC during craning	credible	<$5.10x10^{-3}$	<$6.14x10^{-3}$	<5
Criticality	Incredible	—	—	—
DSC Processing building fire	credible	0	0	0
Earthquake	credible	<5.10×10^{-3}	<6.14×10^{-3}	<5
Tornado	incredible	—	—	—
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—
Release of oxidizing, toxic, corrosive gases and liquids stored in the Processing Building	credible	0	0	0

Table B-8: Postulated Malfunctions or Accidents during Dry Storage Container Storage

Malfunction or Accident	Potential for Occurrence	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
		Adult	Infant	
Seal weld failure during storage	incredible	—	—	—
DSC drop during transfer to storage	credible	<$6.50x10^{-3}$	<$7.91x10^{-3}$	<5
Transporter collision with a DSC or another Transporter	credible	<$6.50x10^{-3}$	<$7.91x10^{-3}$	<5
Criticality	incredible	—	—	—
DSC storage building fire	credible	0	0	0
Earthquake	credible	<$6.50x10^{-3}$	<$7.91x10^{-3}$	<5
Tornado	credible	<$6.50x10^{-3}$	<$7.91x10^{-3}$	<5
Thunderstorm	credible	0	0	0
Flood	credible	0	0	0
Hazardous Material Building explosion	credible	0	0	0
Toxic Material stored in SB3	credible	0	0	0
Turbine missile strike	incredible	—	—	—
Aircraft crash	incredible	—	—	—

Table B-9: Calculated Dose Rates from a Dry Storage Container of the Modified Long Module Design, Fully-Loaded with Pickering 10-year-cooled Used Fuel Bundles

Distance from DSC	Position	Radiation Fields (μSv/h)
Contact	Side	33.8
	Front[22]	39.7
	Top	26.7
	Bottom	109.5
1 m	Side	16.3
	Front	20.9
	Top	9.5
	Bottom	40.9
2 m	Side	8.4
	Front	11.0
	Top	4.1
	Bottom	16.3

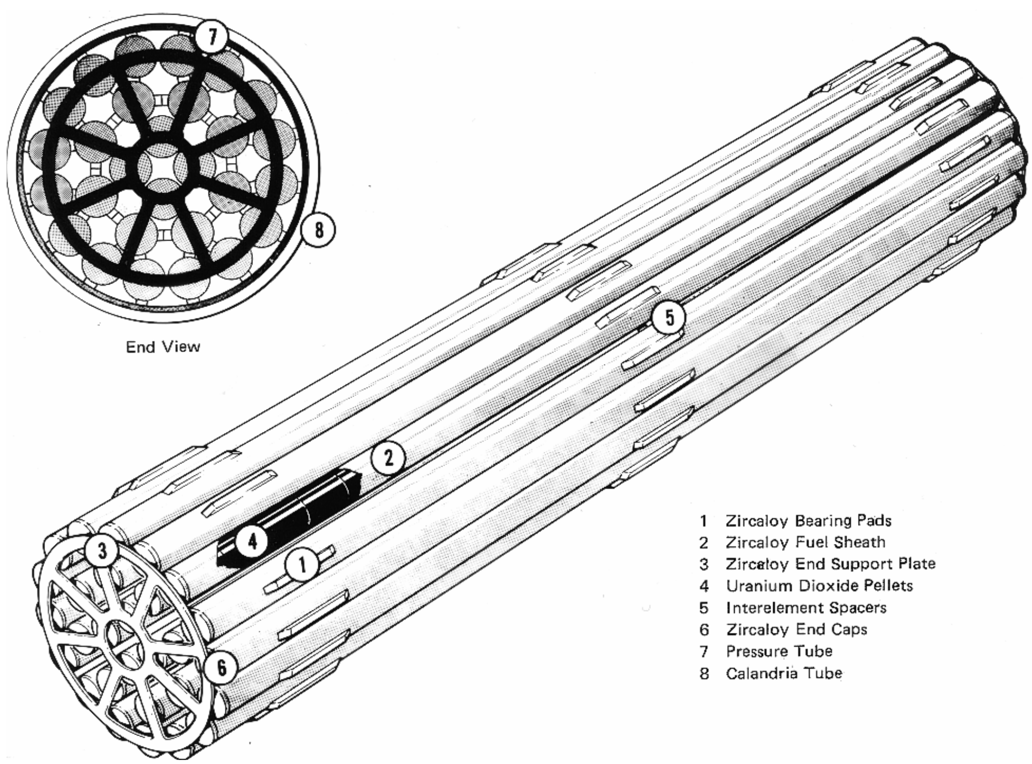

Figure B-1: 28-Element Fuel Bundle

22 The label 'front' corresponds to the wider face of the DSC and 'side' indicates the narrower face.

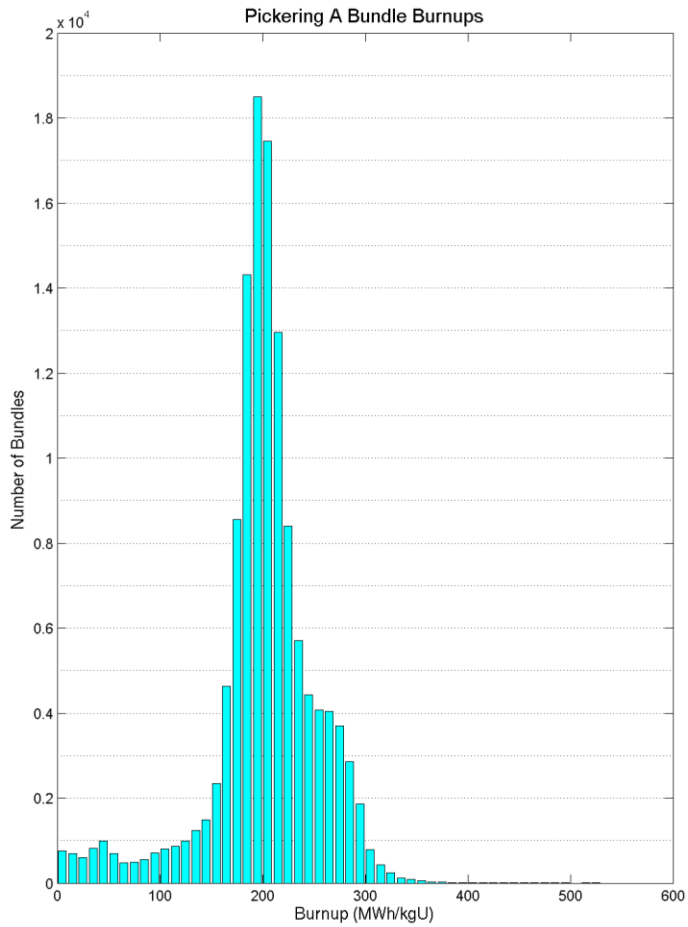

Figure B-2: Pickering Nuclear Generating Station A Reactors Exit Burnup Distribution during 1980 to 2012

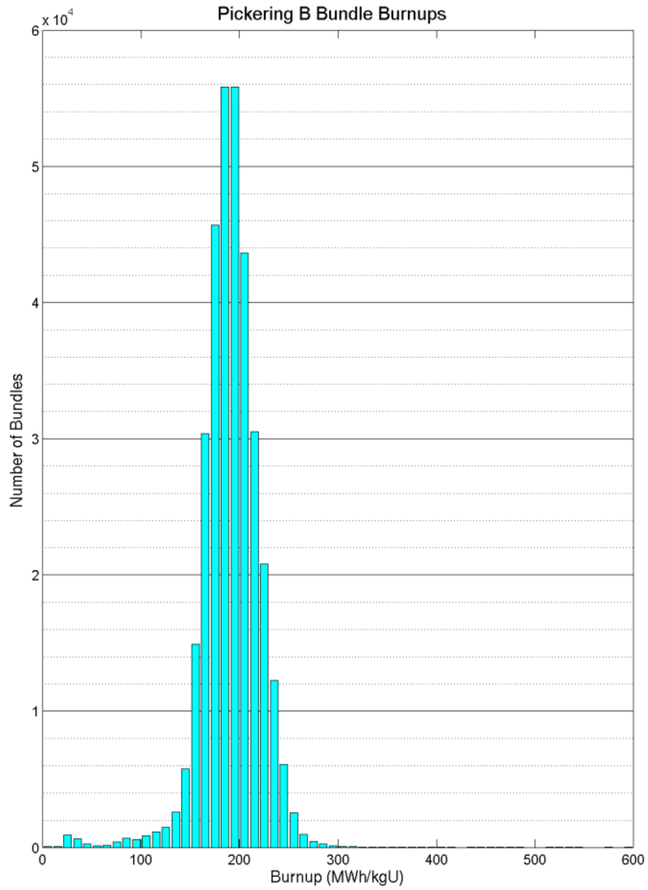

Figure B-3: Pickering Nuclear Generating Station B Reactors Exit Burnup Distribution during 1980 to 2012

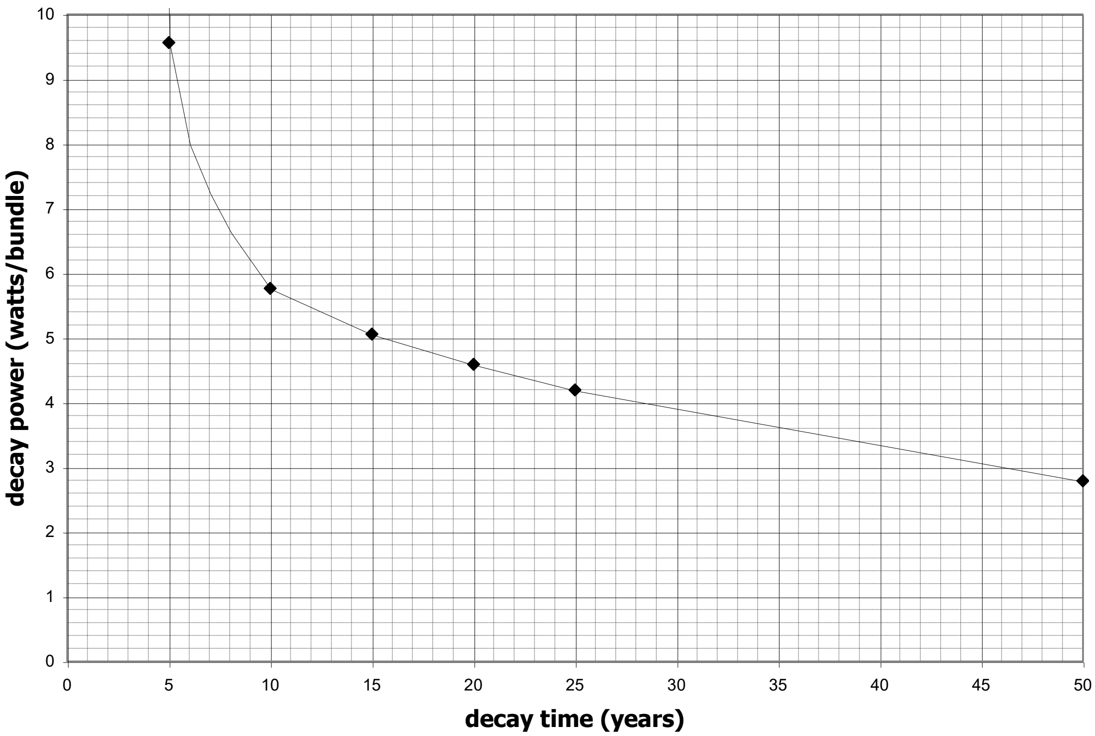

Figure B-4: Total Thermal Power per Pickering Waste Management Facility Reference Fuel Bundle as a Function of Decay Time After Discharge from the Reactor

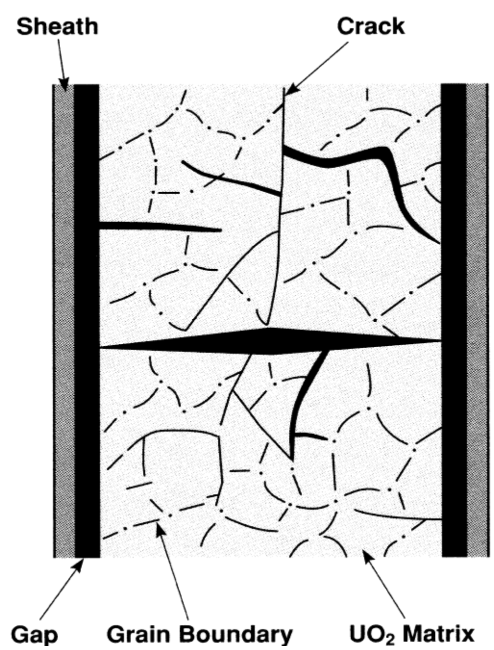

Figure B-5: Locations of Radionuclides in Used Fuel (B-AECL94)

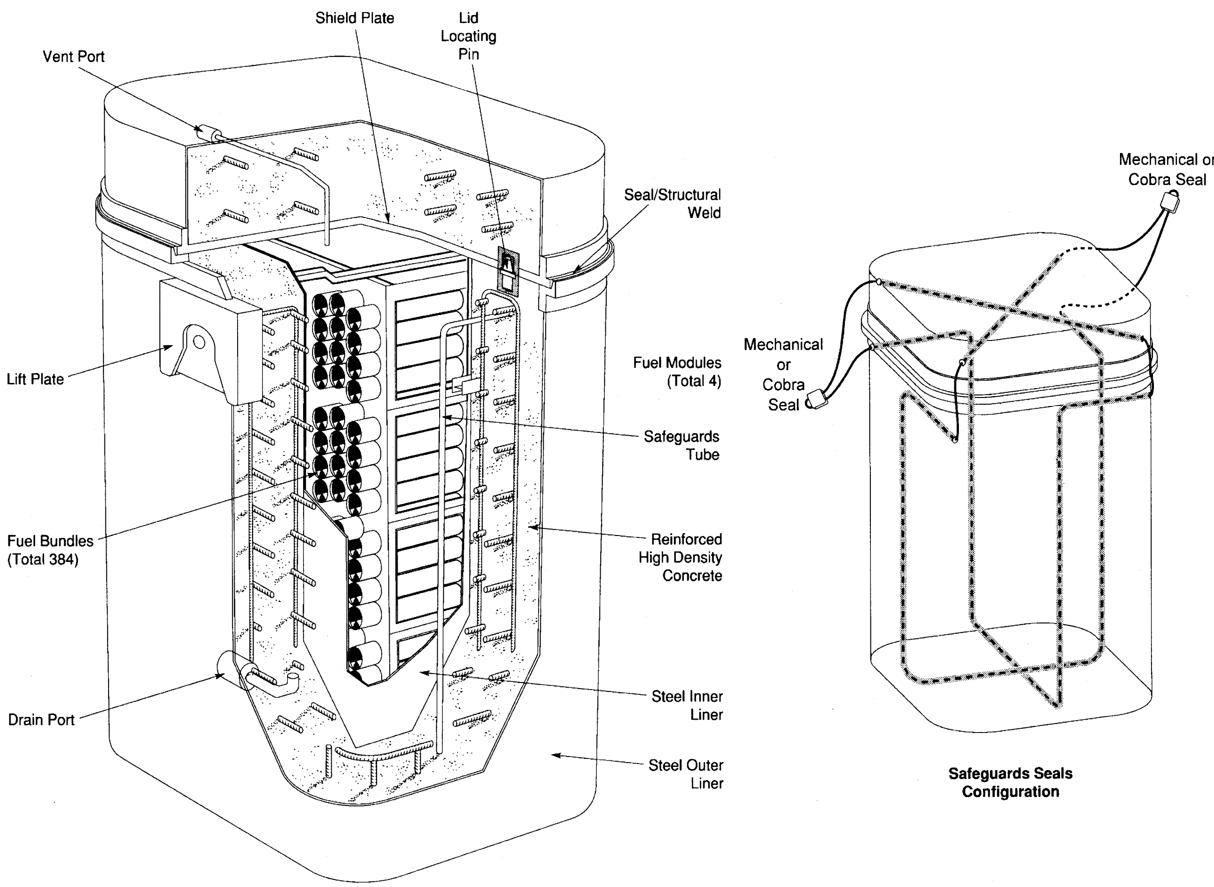

Figure B-6: Dry Storage Container

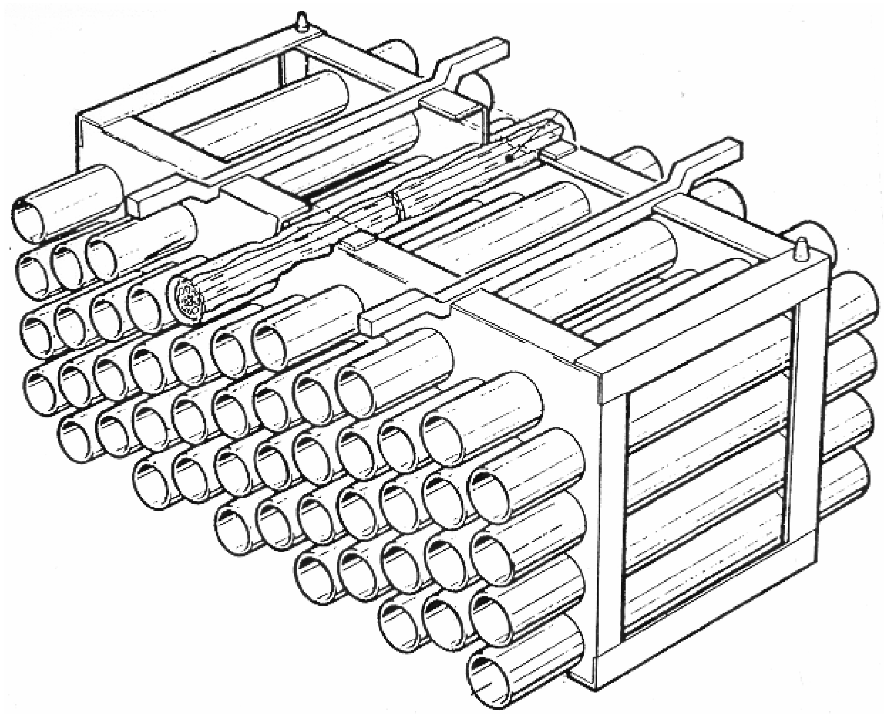

Figure B-7: Storage Module

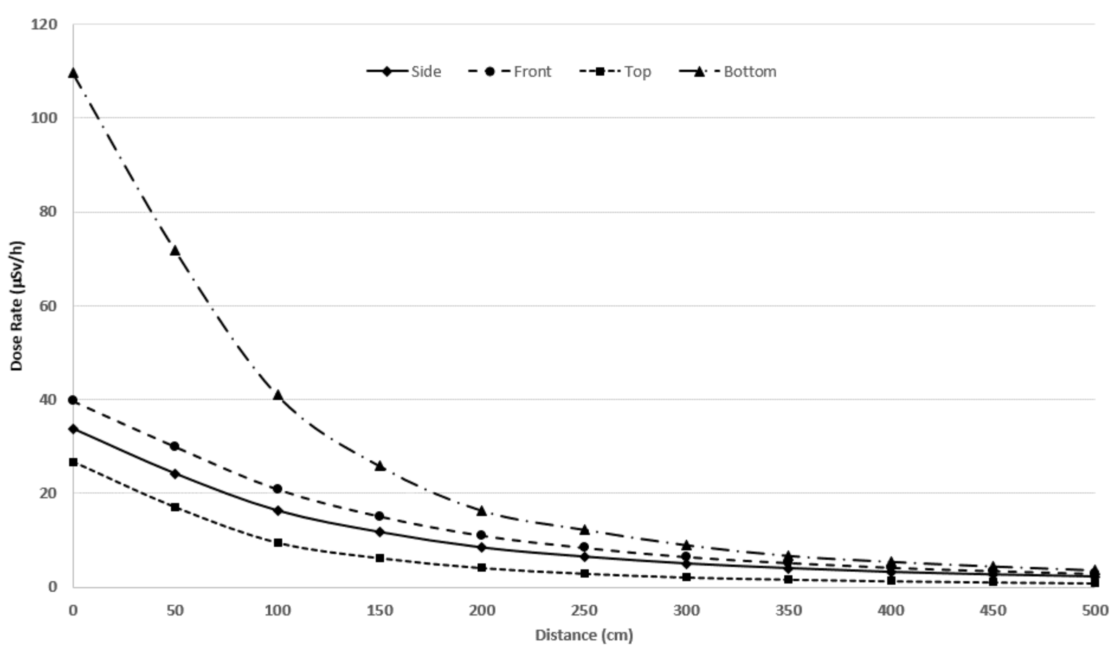

Figure B-8: Dose Rate versus Distance from the Surface of a Single Dry Storage Container with 384 10-year-cooled Pickering Reference Fuel Bundles. The Front Corresponds to the Wider Face of the Dry Storage Container.

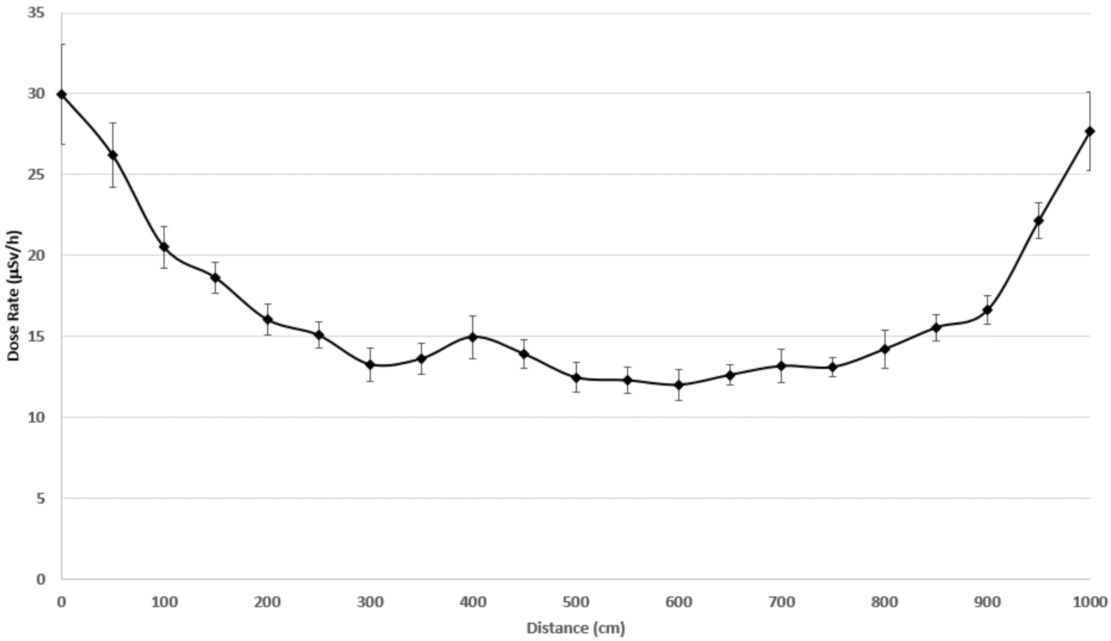

Figure B-9: Calculated Dose Rates Across the Width of the North-South Corridor of Phase I Dry Storage Container Storage Building 2

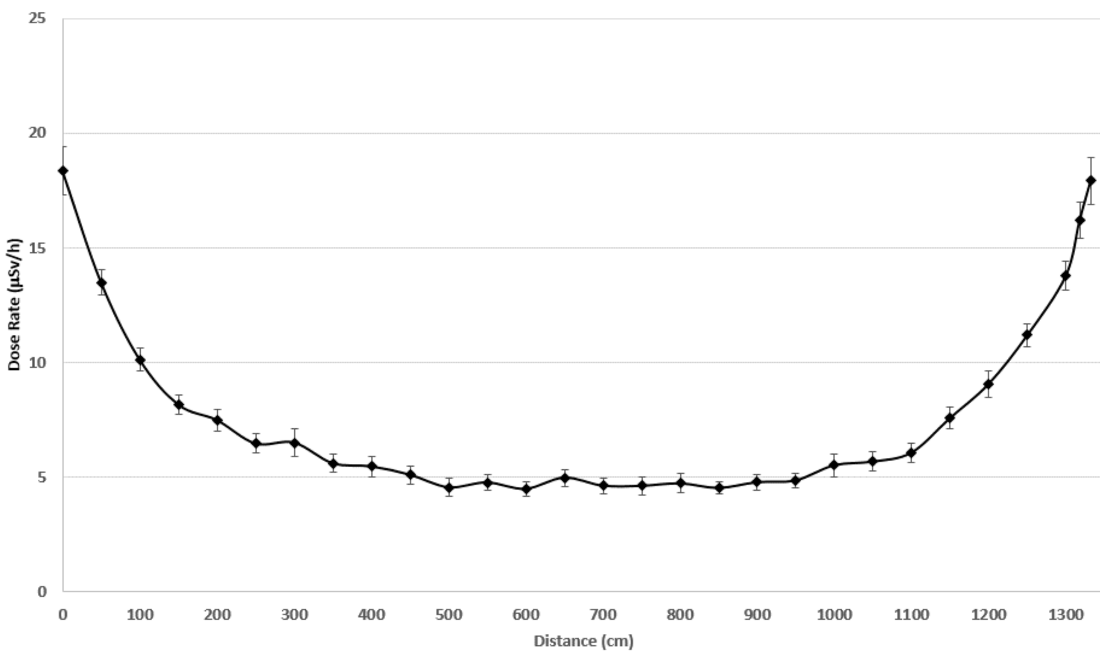

Figure B-10: Calculated Dose Rates Across the Width of the North-South Corridor of Phase II Dry Storage Container Storage Building 3

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Appendix C: Retube Component Storage Area Safety Assessment

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C.1.0 INTRODUCTION

This chapter provides a summary of the PWMF Retube Component Storage (RCS) area radiological safety assessment during storage. Conservative estimates of public dose rates due to releases resulting from hypothetic failures of Dry Storage Modules (DSMs) for normal and abnormal operating conditions, and credible accident conditions are also presented.

C.1.1 Safety Assessment Approach

Under normal operating conditions during storage, DSMs are expected to provide reasonable assurance that the waste can be stored without undue risk to workers, members of the general public, or the environment.

In the absence of Nuclear Waste specific standards and regulatory documents, the guidance document in CSA N288.1-14 (C-CSA14a) and CSA N288.2-14 (C-CSA14c) was used in performing the safety assessment.

RCS waste operations comply with OPG requirements to keep total radioactive emissions under normal operating conditions below regulatory limits and As Low As Reasonably Achievable (ALARA).

The safety assessment of normal and abnormal operating conditions and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions that are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions

C.1.1.1 Safety Assessment Hazard Screening

Potential hazards for the RCS area were identified and the events were screened in or out based on the OPG hazard screening process for internal hazards (C-OPG16c) and external hazards (C-OPG16a). Events that were screened out were deemed to have a negligible contribution to risk. This process followed the OPG screening criteria against which the events were assessed and summarily dismissed.

First a qualitative screening was conducted to identify hazards that were judged to have minimal or negligible impact on risk without the need to perform any detailed quantitative assessments. Some of the identified hazards may not have an impact on the RCS due to its location with respect to the rest of the PWMF Used Fuel Dry Storage (UFDS) buildings or certain hazards may not be applicable due to consideration of those hazards during design. Hazards that have been determined to not impact radiological hazards were screened out.

A quantitative screening criteria based on event frequency has also been applied if the event was not already screened out based on qualitative criteria. CSA N292.0-14 (C-CSA14b), which provides guidance for the management of radioactive waste and irradiated fuel, defines a credible abnormal event as a naturally occurring or human generated event or event sequence that has a frequency of occurrence equal to or greater than 10^{-6} per year. Using this definition of a credible event, an event screening frequency of 10^{-6} was applied to quantitative screening

C.1.1.2 Computer Codes used in Safety Assessment

The potential doses to individual members of the public were calculated using the ADDAM code (C-COG11). The ADDAM code considers the inhalation, cloudshine and groundshine exposure pathway. An exposure period of 30-days was applied, consistent with CSA N288.2-14.

ADDAM is a safety analysis computer program developed by the Atomic Energy of Canada Limited (AECL) for use by the CANDU Owners Group (COG) community. ADDAM calculates doses to the public due to a postulated accidental release of radioactive material to the atmosphere from a nuclear facility. Radionuclides being released can be in the form of gases, vapours or small particles. The radionuclides will disperse as a result of the effects of atmospheric turbulence. The dispersion of the release is affected by the characteristic of the release, the prevailing meteorological conditions, the surrounding terrain and the nearby buildings. The concentrations in the cloud and on the ground take into account factors such as the nature of the releases, decay, build-up and deposition. Doses are calculated for various age groups and receptor locations, and categorized by release pathways (stack, inlet, leakage, or hole) and exposure pathways (inhalation, cloudshine, groundshine). The calculations of atmospheric dispersion and doses are based on CSA N288.2-M91.

The IMPACT code (C-COG15), which is in compliance with CSA N288.1-14, was utilized to analyse the public dose resulting from routine airborne and waterborne releases from PWMF. Pathways include in the assessment follow the recommendations given in the CSA N288.1-14 standard. Doses were reported by age group.

C.1.2 Acceptance Criteria

The radiation safety requirements under normal operation for PWMF are as follows:

- ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.
- ≤ 100 μSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose target for the public at or beyond the PNGS site boundary due to an abnormal even/accident is 1 mSv.
- The dose target for a worker due to an abnormal event/accident is 50 mSv.

The target of 50 mSv for abnormal event/accident refers to NEWs. The equivalent target for non-NEWs/members of the public is 1 mSv.

C.2.0 RADIONUCLIDE INVENTORY

Radionuclide inventories were estimated at the time of loading, for increasing decay times from unit shutdown through 50 years. To date, radionuclide inventories in DSMs have decayed over periods ranging from 20 to 27 years (i.e., since the completion of the Pickering NGS A retubing of the respective Units), and will continue to undergo radioactive decay. Estimates of the radionuclide contents and activities for different decay periods are presented in this section.

Dose rate monitoring at the RCS area fence has confirmed that the original estimates remain conservative. The calculated radionuclide content due to the activation of the Pickering unit 1 (P1), Pickering unit 2 (P2), Pickering unit 3 (P3) and Pickering unit 4 (P4) retube components are given in Table C-1 and Table C-3. The estimated total activities due to activation of the P1/P2 and P3/P4 components are given in Tables C-2 and C-4, respectively and were calculated using the computer code ORIGEN (C-BELL73).

Table C-1: Radionuclide Content of each Pressure Tube, Shield Plug, and End Tube Fitting for P1/P2

[redacted]

Table C-2: Total Activity from P1/P2 Components (TBq)

[redacted]

Table C-3: Radionuclide Content of Each Pressure Tube, End Fitting and Shield Plug for P3/P4

[redacted]

Table C-4: Total Activity from P3/P4 Components (TBq)

[redacted]

The activities for P1/P2 and P3/P4 components differ because of differences in composition of the components. For example, the P3/P4 pressure tubes are made of Zr-2.5 percent Nb whereas the P1/P2 pressure tubes are made of Zircaloy-2 and do not contain niobium. In addition, the degree of conservatism used in the calculations for P1/P2 was greater than that for P3/P4 calculations, which were also based on conservative assumptions. These assumptions are given in Section C.3.3.1.

The values given in Tables C-1 to C-4 do not include loose contamination.

Prior to removal of components, the CANDECON process was used to remove adherent activated corrosion products (crud). On P1/P2, the reactor cooling system was then drained and flushed with demineralized water to remove heavy water. On P3/P4, vacuum drying was used without flushing the reactor cooling system. For Pickering Units 3 and 4, the annulus gas system was oxygenated to convert carbon-14 dust to $^{14}CO_2$, which is removed as a gas.

The following residual contaminants may be present in small amounts associated with the stored components:

(a) Carbon-14 particulate, in the case of Pickering Units 1 and 2, present on the outside of the pressure tubes, end fitting inboard stubs and garter springs.

(b) Activated corrosion products deposited on the inside of the pressure tubes and end fittings, and on the outside of the shield plugs (in the form of loose crud).

(c) Metallic swarf generated during cutting of the pressure tubes and end fittings. Due to the use of roll cutting, the swarf was in the form of slivers, with very little dust.

(d) Tritium in the form of zirconium tritide, which is a very stable compound requiring temperatures of several hundred degrees Celsius for its dissociation.

The expected amounts of these contaminants are conservatively estimated in Section C.2.1.1. Each year the remaining contaminants are less due to radioactive decay.

C.2.1.1 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/bags will keep the loose contamination from mixing in the free atmosphere of the DSM.

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.

A conservative estimate of the inventory of loose radioactivity in DSMs is given below. Assessments are for inventories at the time of initial loading. Short-lived radionuclides will have significantly decayed since then.

C.2.1.1.1 Carbon-14

During the course of retubing, it was discovered that some reactor systems had become contaminated with carbon-14, produced in the annulus gas as a result of the reaction ^{14}N (n,p) ^{14}C. There has been carbon-14 contamination of the P1/P2 components stored in DSMs. For P3/P4 retubing, oxygen was added to the annulus gas system to oxidize the carbon-14. Therefore, only a relatively small amount of carbon-14 is expected in the DSMs loaded with components from P3/P4 retubing. At this time, any remaining carbon-14 is expected to be present predominantly in particulate form.

Carbon-14 measurements on pressure tubes and end fittings for P1/P2 have indicated a maximum of 5.6 kBq/cm^2 (C-AECL91). Using a surface area of 2 m^2 for a pressure tube and for 90 pressure tubes per DSM, this gives a total inventory of 10^{10} Bq per DSM for loose carbon-14 activity.

C.2.1.1.2 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.

C.2.1.1.3 Loose Crud

Data on loose crud (defined as crud that can be removed ultrasonically) were obtained for Pickering Units 1, 2 and 3. These data indicate a maximum initial activity of 2.6×10^7 Bq cobalt-60 per m^2 for zirconium alloy surfaces, and 6.7×10^7 Bq cobalt-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.

C.2.1.1.4 Activated Swarf

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.

C.3.0 NORMAL OPERATION CONDITIONS

C.3.1 Public Dose

Dose to members of the public from normal operation of the PWMF have been determined based on the latest information on radionuclide emissions, representative group locations, and meteorological data (C-OPG18a). The calculated doses are listed in Table C-5 below. Gamma radiation dose rates under normal operating conditions are discussed further in Section C.3.3. The potential for dose due to hypothetical chronic radioactive emissions is discussed in Section C.3.3.4.

Table C-5: Annual Individual Dose from PWMF Normal Operation

Radiation Source	Max. annual individual dose (μSv/year)	Dose Receptor Location
External gamma radiation from DSCs[1] and DSMs	2.08	Landside boundary
	0.72	Lakeside boundary
Chronic particulate emission from PWMF measurements reported in quarterly reports	$1.37\text{x}10^{-4}$	Landside boundary
	$8.90\text{x}10^{-7}$	Lakeside boundary
Postulated volatile releases from DSC[1] processing	$2.28\text{x}10^{-4}$	Landside boundary
	$5.40\text{x}10^{-4}$	Lakeside boundary
Postulated release from DSMs	$5.70\text{x}10^{-4}$	Landside boundary
	$8.10\text{x}10^{-5}$	Lakeside boundary
Total annual individual dose	2.08	Landside boundary
	0.72	Lakeside boundary

[1] Dry Storage Containers

C.3.2 Worker Dose

The actual worker doses received during normal operation of the PWMF are reported in the PWMF quarterly reports. Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported during the 2007-2016 period are 11.2 person-mSv and 1.6 mSv.

The maximum effective dose to NEWs working at the PWMF is well below the regulatory dose limits for NEWs; 50 mSv in a one-year dosimetry period and 100 mSv in a five-year dosimetry period.

C.3.3 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 RCS normal operation are reported below.

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 in Section C.3.3.1. Results of dose rate monitoring at the RCS area and predicted dose rates at protected area perimeter fences are presented in Section C.3.3.2 and C.3.3.3, respectively. These dose rates measurements are representative of over more than a decade decay time. Dose rates continue to decrease with increasing facility life.

C.3.3.1 Estimated Dose Rates from Dry Storage Modules

The estimated gamma dose rates from a DSM (containing irradiated components), at various distances as a function of time after removal from the reactor are given in Figure C-1 for DSMs with P1/P2 components and Figure C-2 for DSMs with P3/P4 components. A worst case payload, i.e., the DSM contents giving the highest dose rate, was used.

These dose rates were calculated using the state-of-the-art radiation transport code, MCNP 6.1[23] (C-LANL11).

This code is capable of rigorously simulating the stochastic nature of gamma, neutron and electron transport by explicitly modelling the physical nature of their travel through space and their interactions with matter. The MCNP code captures gamma dose rate contributions from irradiated fuel channel components in large arrays of storage containers to provide an accurate, integrated shielding analysis, taking into account all gamma radiation dose pathways.

The MCNP code applies the ‘Monte Carlo’ method of analysis, simulating photon histories explicitly in the modelled geometry. A characteristic of this method is that

[23] MCNP® and Monte Carlo N-Particle® are registered trademarks owned by Los Alamos National Security, LLC, manager and operator of Los Alamos National Laboratory

all of the results ('detector tally' results of dose rate values at the specific locations) are always statistical quantities in the form of an estimated mean value and an estimated standard deviation. MCNP utilizes a sufficient number of 'photon histories' that the statistical uncertainty is very small. In cases where the estimated statistical uncertainties in the MCNP results are greater than five percent, special care has been taken in the consideration of the results. In such cases, the margin between the target dose rate and the computed value was considered related to the estimated uncertainty, and a judgement was made as to whether the values can be accepted as not exceeding the targets. This methodology is documented in Reference C-OPG17b.

The methodology used to develop the MCNP model of DSMs at PWMF has been validated/benchmarked (C-OPG17a) through simulation of TLDs surrounding the Used Fuel Dry Storage (UFDS) buildings. The results show that the predictions using the MCNP model, which is based on the reference methodology, is conservative by 35-60%. This conservative outcome justifies the application of this methodology and the use of MCNP to model large arrays of heavily shielded waste containers in normal operations safety assessments.

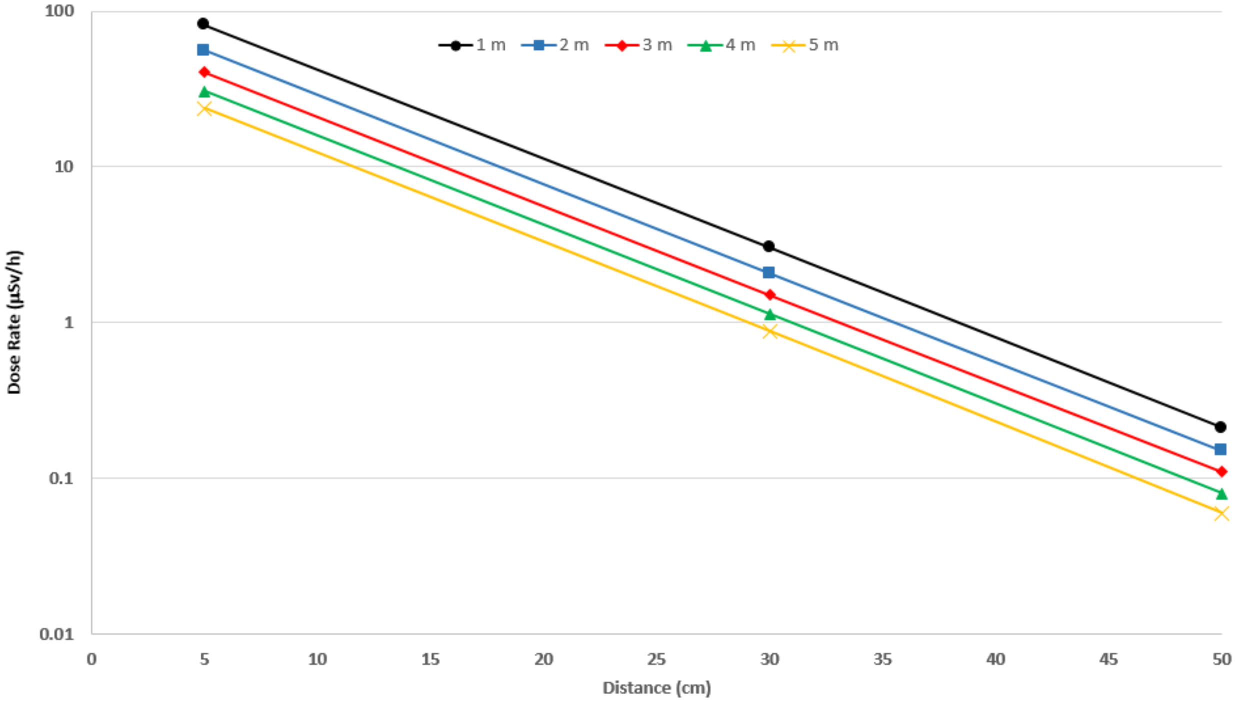

Figure C-1: Estimated Dose Rates from a Dry Storage Module Loaded with P1/P2 Components (C-OPG18b)

Note: For Figure C-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.
- Radiation fields from storage modules containing end fittings are very similar to those from modules containing pressure tubes.
- These dose rates were calculated using conservative early estimates of the activation product inventories.
- The earliest time after shutdown that a DSM was initially loaded was 18 months.

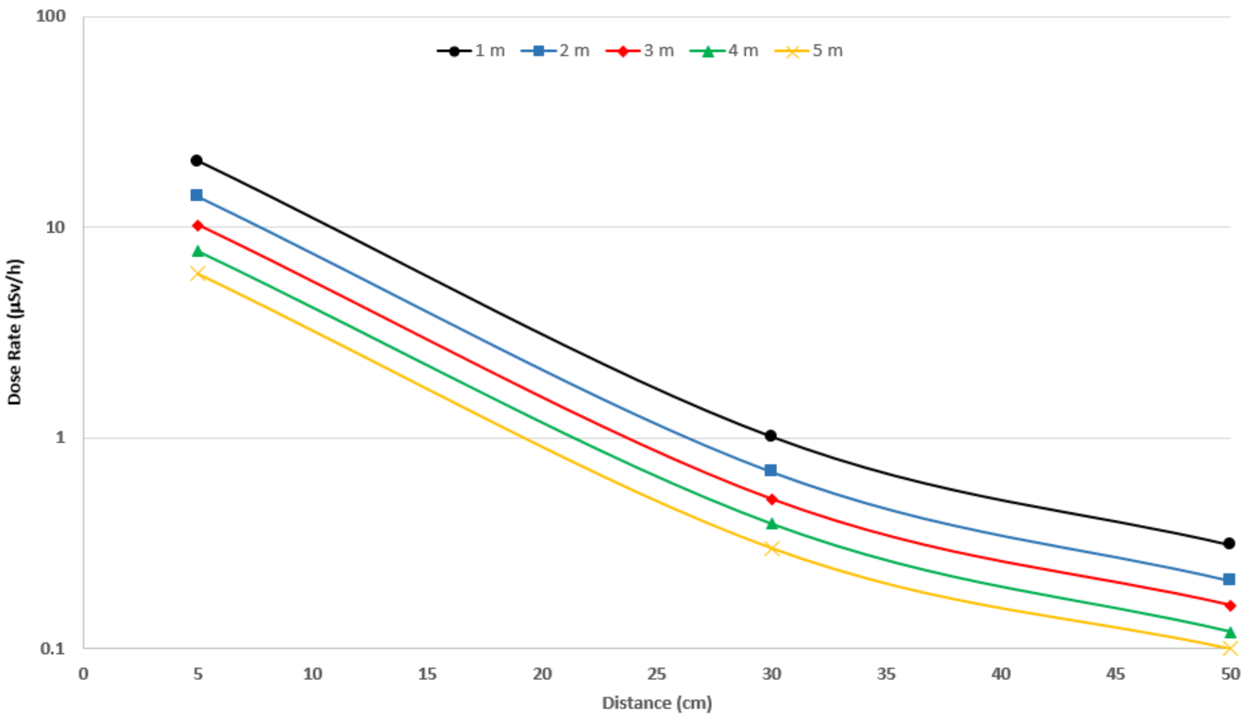

Figure C-2: Estimated Dose Rates from a Dry Storage Module Loaded with P3/P4 Components (C-OPG18b)

Note: For Figure C-2:

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

C.3.3.2 Results from Dry Storage Module Dose Rate Monitoring

Comparisons between available measurements and the corresponding calculated quantities have historically yielded good agreement. The activity content and radiation fields associated with pressure tubes and end fittings have been shown to be well predicted by the analytical methods used.

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.

In ten DSMs containing P3/P4 pressure tubes with shield plugs, at some specific locations, notably toward the underside, measurements of contact dose rates exceed calculated estimates by a factor of 3 to 4 over limited areas of the surface. These locations of higher than expected dose rates occur at about 1.8 m from the loading end, towards the saddle area, within a 0.5 m wide band.

Due to correlation with position, as well as evidence from other shielded flasks, these elevated dose rates are believed attributable to the shield plugs pushed in toward one end of each pressure tube. Higher than expected activity levels are suspected to be the result of greater cobalt impurity content in the shield plugs' stainless steel than was assumed in the calculations.

High contact dose rates (up to 6.5 mSv/h) have been measured in localized spots near the top of the shield doors of six DSMs. These areas of elevated dose rate are only a few centimetres in width at the DSM surface, and are sufficiently scattered such that there are no discernible beams near the DSM face. Maximum dose rates were assessed to be below about 700 μSv/h at 30-cm distance, and below about 100 μSv/h at 1 m. The highest contact dose rate from routine surveys of the designated survey locations on the DSMs is currently 79 μSv/h (7.9 mrem/hr).

The localized spots probably correspond to small gaps in the DSM shielding wall, perhaps at the gasket seal near the top of the door. The dose rates drop to ambient background at ground level, and are therefore relevant only for employees using a ladder or platform to perform close work such as painting the shield door. Where contact work near the shield doors is required, controls such as dose rate survey monitoring, and minimizing exposure time in elevated radiation fields, are employed to keep occupational dose ALARA.

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 because 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.

C.3.3.3 Dose Rates from the Retube Components Storage

(a) 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 TLDs 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 2007-2017 TLD survey monitoring results, the maximum dose rates are 0.16 μSv/h at the south fence, 0.11 μSv/h at the east fence, 0.12 μSv/h at the west fence and 0.34 μSv/h at the north fence.

(b) Calculated Dose Rates

The dose rate at the limiting location where non-NEW may be present was calculated (C-OPG16b) and found to be well within the 0.5 μSv/h dose rate target. The highest dose rate at locations east of the DSC storage buildings 1 and 2 is 0.24 μSv/h. The dose rate at locations east of the RCS is 0.22 μSv/h.

C.3.3.4 Chronic Radioactive Emissions

Annual public doses from releases during normal operation were calculated at the Pickering NGS site boundary and at the critical group locations, which include urban residents, dairy farm and farm residents, industrial/commercial worker, correctional institute resident and sport fisher.

The public dose estimate was performed using the methodology outlined in the CSA N288.1-14 standard. The methodology covers releases to the atmosphere and to surface water (both fresh and marine).

C.3.3.4.1 Potential Airborne Emissions

Conservative estimates were made previously for emissions of particulate contamination. However, routine contamination survey monitoring conducted since the RCS area at the PWMF 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 canisters;
- The gasketed seal of the DSMs; and
- 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 container).

Carbon-14 particulate 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 percent per year or about 0.2 percent per week (C-OH84). This conservative estimate assumes that moisture may be present in the DSM atmosphere.

Based on conservative assumptions (C-OH84), 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 P1/P2 components were considered. Since carbon-14 is expected to be in stable form after over 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 carbon-14 as $^{14}CO_2$ is:

$$1.0 \times 10^{10} \text{ (Bq/DSM)} \cdot 0.2\%/\text{wk} \cdot 16 \text{ DSMs} = 3.2 \times 10^{8} \text{ Bq/wk (8.6 mCi/wk)}$$

The chronic off-site dose consequence resulting from this postulated scenario, for a member of the public at the most exposed age group/location is estimated to be 5.70×10^{-4} μSv/year.

C.3.3.4.2 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 RCS area basins confirms that contamination levels are generally below the Minimum Detectable Activity (MDA) of 14 Bq/L (3.8×10^{-7} μCi/mL).

The storage modules 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 storage 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 storage 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. Routine monitoring of surface water in RCS area basins confirms that contamination levels are generally below the MDA.

C.3.3.4.3 Comparison with the Derived Release Limits

Compared with the Derived Release Limits (DRLs) of carbon-14 for Pickering NGS B, the emission of carbon-14 (as $^{14}CO_2$) from the RCS is only 8×10^{-4} percent of the Pickering NGS carbon-14 operational DRL for airborne releases.

C.4.0 RADIOLOGICAL SAFETY – ABNORMAL OPERATIONS

C.4.1.1 Abnormal Events

C.4.1.1.1 Earthquakes

The DSMs were designed in accordance with CSA N289 (Seismic Design for CANDU Nuclear Power Plants). The DSM supports have been designed to withstand a horizontal ground acceleration of 10 percent of gravity, and the DSM design is qualified to an acceleration of 3 percent of gravity and for a 1.2 m drop.

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 B Design Basis Earthquake (DBE) ground motion as the design basis event. It was found that the required safety function of the DSMs was met.

The reinforced concrete footings of the DSM supports are founded in a compacted gravel “working surface”, installed when the area was used for construction storage and laydown. The loading from the DSMs has not resulted in detectable settlement. The foundations are considered adequate to support the DSMs under normal static loads. However, the variable fill deposit underlying the DSM foundations could amplify earthquake ground motion. In order to address potential effects of dynamic loads on the foundations, a detailed seismic capacity assessment of the DSMs was undertaken, and is described below.

The DSM storage site foundation conditions are similar to those for the adjoining UFDS buildings and are described in Section 2.8.1 of this report. Amplified ground motion resulting from the soft foundation conditions was accounted for in the seismic analysis. A finite element model of the DSM, its support frame, concrete footings and the soil foundation was subjected to the Pickering NGS B DBE ground motion using the response spectral analysis method (C-LEE94). Conservative soil and structural component damping values were used in the analysis.

The structural members of the support frames were found to have adequate seismic capacity, and the DSM container itself to have low seismic stresses. The assessment indicated seismic stresses in anchor bolts at the base of the support pedestals and in field welds at the top of pedestal connections to the underside of the DSM support frame which were at, or marginally exceeded, code allowable seismic stress levels. If these connections were to fail, structural stability of the DSM supports could be jeopardized, allowing the DSM container to fall to grade.

The structural implications of the DSM container falling to grade as a consequence of an earthquake were examined. ABAQUS/Explicit finite element computer code (C-HIBBIT94) was used to simulate dynamic impact of the DSM container on the concrete footing and to analyze non-linear material/ geometry effects at the point of impact (C-LEE95). It was found that plastic deformation of the outer steel shell would occur with crushing of the adjacent concrete of the DSM container in a limited zone at the point of impact. Crushing of concrete occurred to a depth less than one-half the concrete wall thickness. The damage was very limited in extent and would be repairable. The inner liner and concrete surrounding it remained intact and in the elastic range. The loading port area also was not distorted. No path for radiological release to the environment would, therefore, be created by the impact.

Based on the above assessments it was concluded that there are no radiological implications for DSM storage in the PWMF for the seismic design basis event.

C.4.1.1.2 Flooding

The RCS area is located at an elevation of 77.4 m. Each DSM is set on 1 m high pedestals, atop a 1.8 m x 5.5 m x 0.6 m concrete foundation.

Review Level Conditions (RLC) were developed for natural hazards, and the RLC for Probable Maximum Precipitation (PMP) was determined to be 12-hour precipitation, equivalent to 420 mm of total rainfall with 51% occurring in the 6th hour. The RCS area shall have a maintenance free, well drained non-water-ponding surface and the storage saddles shall not trap water at the surface where they contact the DSM. Taking into account the location of the RCS and the elevated storage of the DSMs, the probability of flooding-induced failures from PMP is negligible.

The 500-year maximum still water level is 76.6 m and the maximum recorded wave uprushes is 2.20 m, so the maximum wave run-up heights would reach 78.8 m. This means that the DSMs are still 0.2 m above the maximum recorded lake levels, including wave uprushes. In addition, a 3.3 m high embankment protects the RCS area from wave action in case of a storm. The probability of flood water coming in contact with a DSM is considered 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.

C.4.1.1.3 Fires and Explosions

Two fuel oil tanks located in a dyked area approximately 50-60 m west of the PWMF are the only potential fire and explosion hazard near the site. A detailed assessment of this hazard has been carried out (C-NATHWANI84) 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 storage modules, it was assumed that the initiating event caused the oil tanks to rupture and all the fuel draining into the dyked area was involved in the fire. It was judged that a fire which 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 as a result of the analysis 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 under storage at the PWMF.

In addition, based on the latest PWMF Fire Hazard Assessment (FHA) (C-OPG17c), a fire involving the fuel oil storage tanks would result in a maximum heat flux of 6.1 kW/m^2. Therefore it was concluded that based on the magnitude of the heat flux, the fire originating from the fuel oil storage tanks would not impact the integrity of the DSMs.

The RCS area is protected from the lake by a 3.3 m high 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 under storage at the PWMF.

C.4.1.1.4 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of PWMF Phase I and Phase II.

PWMF Phase I DSCs/DSMs occupy a total area of approximately 533’ (162.5 m) x 312’ (95 m) and the building height is 45’ (13.7 m). Phase I was assumed to be a rectangular shape that includes the Processing Building, Storage Buildings 1 and 2 and the RCS area.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase I was determined to be 5.95×10^{-7} events per year (C-OPG19).

PWMF Phase II DSCs occupy a total area of approximately 238’ (72.5 m) x 278’-9” (84.9 m), and the building height is 31.5’ (9.6 m). Phase II was assumed to be a rectangular shape that only includes Storage Building 3.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency for Phase II was determined to be 2.92×10^{-7} events per year (C-OPG19).

The LiftKing transporter, which is slightly larger than the Gen4 transporter, has been used to assess this hazard with respect to DSC in transit. The transporter has an overall length of 8.5 m, an overall width of 3.3 m and an overall height of 4.7 m.

Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency impacting the transporter has been determined, considering the limited time that a loaded DSC transporter will be in transit and taking into account that the transporter is a small moving target.

The frequency of an aircraft crash impacting the DSC while being transferred is 3.68×10^{-10} events per year (C-OPG18a).

The total frequency for an aircraft crashing into a DSC/DSM at either Phase I or Phase II is $5.95 \times 10^{-7} + 2.92 \times 10^{-7} + 3.68 \times 10^{-10} = 8.87 \times 10^{-7}$ events per year, which is below the event cut-off frequency of 10^{-6} events per year. Therefore, aircraft crashing into a DSC/DSM at the PWMF is considered incredible.

C.4.1.1.5 Extreme Climatic Conditions

(a) Thunderstorms

Lightning would be the only consequence of a thunderstorm which could pose concern for the RCS, from the safety viewpoint. However, because of the round shape, the DSMs should not be susceptible to a lightning strike. The DSMs shall be designed to survive exposure to extreme environmental conditions, such as “tornado winds, high lake water levels and lightning strikes”, as per DSM design requirements. The lightning hazard is credible; however, since lightning strikes are within the design basis of the DSM, this hazard can be screened out.

(b) Tornadoes

The effect of DBT-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 skin could be penetrated, but that the concrete would stop the missile. The maximum penetration calculated for a variety of missiles was 145 mm, compared with the DSM wall thickness of 570 mm. The maximum scabbing thickness calculated (i.e., the thickness required to prevent spalling of concrete from the inner surface) was about 570 mm, however, the inner steel liner would prevent loss of concrete.

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

C.5.0 CONCLUSION

This appendix describes the safety assessment of the RCS area at the PWMF. The dose consequences to the public and the workers under normal operations and considering potential accidents and malfunctions were evaluated.

It is concluded that under normal operation the dose consequences to the public are well below the OPG administrative dose target at the site boundary (ten percent of the regulatory dose limit) and at perimeter fences. Occupational dose were also found to be below the acceptance criteria.

All potential malfunctions and accidents that could impact the DSMs in storage have been found to be non credible given the event frequency cut off of 10^{-6} events per year and therefore there are no radiological consequences to the public.

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C.7.0 GLOSSARY

ADDAM - Atmospheric Dispersion and Dose Analysis Method
AECL - Atomic Energy of Canada Limited
ALARA - As Low As Reasonably Achievable
CNSC - Canadian Nuclear Safety Commission
COG - CANDU Owners Group
CSA - Canadian Standards Association
DBE - Design Basis Earthquake
DBT - Design Basis Tornado
DRL - Derived Release Limit
DSC - Dry Storage Container
DSM - Dry Storage Module
FHA - Fire Hazard Assessment
MDA - Minimum Detectable Activity
PMP - Probable Maximum Precipitation
PNGS - Pickering Nuclear Generating Station
PWMF - Pickering Waste Management Facility
RCS - Retube Component Storage
RLC - Review Level Condition
TLD - Thermoluminescent Dosimeter
UFDS - Used Fuel Dry Storage

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Appendix D: Dry Storage Module Storage Area Utilization Summary

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ATTACHMENT 2

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Redacted CD# 92896-CORR-00531-01487, "*CNSC Staff's Prior Written Notification of Document Changes: 92896-SR-01320-10002, Nuclear Sustainability Services – Pickering Waste Management Facility Safety Report, R007*"

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Subject:	CNSC Staff"s Prior Written Notification of Document Changes: 92896-SR-01320-10002, Pickering Waste Management Facility Safety Report, R007
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Importance:	High

OPG Proprietary

CD# 92896-CORR-00531-01487

CNSC Staff's Prior Written Notification of Document Changes: 92896-SR-01320-10002, Nuclear Sustainability Services – Pickering Waste Management Facility Safety Report, R007

Dear Ms Petseva:

The purpose of this e-mail is to provide advance notification to CSNC staff of Revision 007 to 92896-SR-01320-10002, *Nuclear Sustainability Services - Pickering Waste Management Facility - Safety Report*

R007 will be issued on or about December 8, 2023. This written notification is in accordance with the PWMF licence condition which requires OPG to give written notification of changes made to the documents submitted to support the licence application.

OPG has revised this document to:

- Incorporate identified enhancements to safety analysis;
- Reflect organizational changes and/or changes to other governance;
- Incorporate information on DSC Storage Building 4

Specific changes are described in the Revision Summary section of the document.

A comparison of calculated doses against the previous Safety Report is provided in the table below;

Parameter Unit Previous Current Remark on change
Normal operation:
Public dose, landside mSv per year 0.002 0.003 The dose is dominated by the dose from gamma radiation from the DSCs in storage.

Occupancy factor = 2000 h per year in previous and current Safety Report.
Higher dose due to DSCs in Storage Building 4 (not present in previous Safety Report).
Normal operation:
Public dose, lakeside mSv per year 0.0007 0.0014 The dose is dominated by the dose from gamma radiation from the DSCs in storage.
Occupancy factor = 1000 h per year in previous and current Safety Report.
Higher dose due to DSCs in Storage Building 4 (not present in previous Safety Report).
Malfunction/accident:
Public dose mSv per year 7.91E-03 2.90E-03 Reduced dose due to change in meteorological data and lower tritium inventory due to revised cross-section library.

Malfunction/accident
Worker dose, individual mSv per year 5 4.7 Reduced dose due to lower tritium inventory due to revised cross-section library.

As this revision was performed in accordance with OPG document change management requirements, OPG maintains this revision:

- Has no adverse impact on the health and safety of persons, security, the environment, or Canada's international obligations.

- Has no adverse impact on the licensing basis.

Please update the PWMF LCH accordingly.

Regards,

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Report

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Title:
NUCLEAR SUSTAINABILITY SERVICES - PICKERING WASTE MANAGEMENT FACILITY - SAFETY REPORT

Unless otherwise stated, all information redacted in this document is considered prescribed information under the General Nuclear Safety and Control Regulations

NUCLEAR SUSTAINABILITY SERVICES - PICKERING WASTE MANAGEMENT FACILITY - SAFETY REPORT

2023-10-19

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Revision Summary

Revision Number	Date	Comments
R7	2023-10-19	**General Revisions** • Updated facility name to Nuclear Sustainability Services – Pickering Waste Management Facility (NSS-PWMF) • Revised cover page to include staff changes • Addition of information relevant to DSC Storage Building 4 (SB4) • N288.1-14 superseded by N288.1-20 • N288.2-14 superseded by N288.1-19 • N-REP-03420-10011 R001 replaced by N-STD-RA-0045 R000 • Various editorial changes for the sake of clarification
		Chapter 1 • Figure 1-1 updated with a more recent image • Section 1.2.1: Updated number of DSCs stored at PWMF Phase I and II sites by December 2022 • Section 1.3: Added footnote relevant to REGDOC 2.4.4 • Section 1.6.1: Added reference CAN21
		Chapter 2 • Section 2.1: Included information for SB4 • Section 2.4: Replaced OPG17b with OPG22f, Replaced OPG18 with OPG22j • Section 2.5.1: Temperature data updated for the 5 year period of 2017-2021 • Section 2.5.2: Precipitation data updated based on 2021 measurement data • Section 2.5.3: Wind data updated for the 5 year period of 2017-2021 • Section 2.5.4: Updated in accordance with CSA N288.2-19 and the 5 year period of 2017-2021 • Section 2.5.5: Updated to reflect wind data for the 5 year period of 2017-2021 • Section 2.5.7: Updated based on input from OPG Subject Matter Experts • Section 2.5.8: Updated based on input from OPG Subject Matter Experts • Section 2.8.1: Added reference OPG22e • Section 2.8.1.3: Updated information relevant to groundwater monitoring • Section 2.9.1.2: Included information for SB4 • Section 2.9.2: Updated with 2021 population data • Section 2.10: Updated as per input from OPG Indigenous Relations
		Chapter 3 • Section 3.2.4: Added reference OPG04 • Section 3.3.1: Small updates to DSC design dimensions to better align with design documentation (no actual change in design) • Section 3.3.2.7: Updated the DSC lift height to better align with onsite procedures • Section 3.4.1: Included information for SB4 • Section 3.4.2.2: Included information for SB4

		• Section 3.4.3.2: Included information for SB4 • Section 3.4.5.2: Included information for SB4 • Section 3.4.5.3: Added reference OPG22i. Included information for SB4 • Section 3.5.3.6: Revised as per comment #15 from 92896-CORR-00531-01333 • Section 3.6.1: Included reference to both DSC Transporter designs • Section 3.6.4: Replaced OPG17a with OPG19a • Section 3.7: Revised as per comment #16 from 92896-CORR-00531-01333 • Section 3.8: Updated DSC inventory as per OPG23c
		Chapter 4 • Section 4.1.1: Added reference OPG22c. Re-worded to address comment #13 from 92896-CORR-00531-01333 • Section 4.1.2: Added reference CSA22 • Section 4.2: For credible abnormal events and/or credible accidents clarified the “targets” previously mentioned to “dose limits”. Added reference CSA22 • Section 4.3.1: Updated as per the most recent safety assessment OPG22c • Section 4.3.2: Added references OPG22a, OPG23c. Sub-sections updated as per the most recent safety assessment OPG22c • Section 4.3.2.1: Updated dose rates in Table 4-1 and also Figure 4-1. • Section 4.3.2.3.2: Added context with respect to conservative dose calculations at perimeter fences • Section 4.4.1: Updated as per the most recent safety assessment OPG22c • Section 4.4.2: Updated as per the most recent safety assessment OPG22c • Section 4.4.3: Updated as per the most recent safety assessment OPG22c. Replaced OPG17a with OPG19a • Section 4.5.1: Added reference to N-PROG-RA-0013. Added accountability of Director of Radiation Safety • Section 4.5.2: Updated to include data up to 2021 • Section 4.7.1: Added statement that the DSC has been designed to maintain used fuel integrity under normal and abnormal operating conditions, along with credible accident conditions • Section 4.7.2.4: Added reference to I-PROG-AS-0001 to address comment #24 from 92896-CORR-00531-01333
		Chapter 5 • Section 5.1.3.1: Added references to Inspection and Maintenance Programs to address comment #24 from 92896-CORR-00531-01333 • Section 5.1.2.1: Replaced OPG17a with OPG19a
		Chapter 6 • Updated safety assessment and accident analysis related to RCS as per the most recent safety assessment OPG22c
		Chapter 7 • Section 7.2.4: Included information for SB4 • Section 7.2.5: Replaced 92896-INS-09071-00002 instruction with N-INS-09071-10025 • Section 7.4.2.4.1: Added reference OPG23b • Section 7.5.1: Added references to OPG-POL-0001 and OPG-PROG-0005 to address comment #24 from CNSC 92896-CORR-00531-01333. • Section 7.5.4: Added reference CSA22

		Chapter 8 • Section 8.0: Added references to N-STD-MP-0028, W-PROG-WM-0003, N-PROG-RA-0003 to address comment #24 from 92896-CORR-00531-01333 • Section 8.1: Added references OPG17, OPG22h, OPG23a
		Chapter 9 • Sections 9.0, 9.1: Updated as per input from OPG Stakeholder Relations • Section 9.2: Updated as per input from OPG Indigenous Relations
		Chapter 10 • Revised to reflect most recent PWMF PDP; OPG22d • Added references CNSC00a, OPG20a, OPG22d
		Chapter 11 • Section 11.1.1.2: Added reference OPG22f • Section 11.1.1.2: Updated as per the most recent safety assessment OPG22c
		Chapter 12 • Updated as per the most recent safety assessment OPG22c
		Chapter 13 • Added references CAN21, CNSC00a, CSA22, OPG17, OPG17a, OPG20a, OPG21c, OPG22c, OPG22e, OPG22g, OPG22h, OPG22i, OPG23a, OPG23b, OPG23c • Replaced OPG16 with OPG22d • Replaced OPG17a with OPG19a • Replaced OPG17b with OPG22f • Replaced OPG18 with OPG22j
		Chapter 14 • Updated to reflect changes in the report text
		Appendix A • Updated with more recent photos
		Appendix B • Section B.2.1.2: Added Table showing bundle statistics from B-NWMO21 • Section B.2.7: Updated as per the most recent safety assessment B-OPG22b • Section B.2.8: Added reference B-CSA19a • Section B.2.11: Updated as per the most recent safety assessment B-OPG22b. Included information for SB4 • Section B.3.0: Added reference B-OPG22a • Section B.4.2: Replaced B-CSA14b with B-CSA19b • Section B.4.3: Replaced B-OPG16a with B-OPG18a, Replaced B-OPG16b with B-OPG18b • Section B.4.4: Replaced C-CSA14a with C-CSA20. Replaced C-CSA14c with C-CSA19a • Section B.4.4.1: Updated as per the most recent safety assessment B-OPG22b. Added references B-EcoMetrix18, B-CSA20, B-CSA19a

		• Section B.4.4.2: Updated as per the most recent safety assessment B-OPG22b • Section B.6: Included information for SB4 • Section B.6.2.1.1: Updated as per the most recent safety assessment B-OPG22b • Section B.6.2.1.2: Updated as per the most recent safety assessment B-OPG22b • Section B.6.2.4: Added reference B-CSA22 • Section B.6.2.5.2: Updated as per the most recent safety assessment B-OPG22b • Section B.6.2.5.3: Updated as per the most recent safety assessment B-OPG22b. Replaced reference B-OPG12 with B-OPG22e • Section B.6.2.5.4: Updated as per the most recent safety assessment B-OPG22b • Section B.6.2.5.5: Replaced B-OPG11 with B-OPG22g • Section B.6.2.5.7: Updated as per the most recent safety assessment B-OPG22b • Section B.6.2.5.8: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG22b • Section B.6.2.5.9: Included information for SB4 • Section B.7.1.1: Added reference B-OPG22f • Section B.7.1.2: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.1: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.2: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.3: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.4: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.5: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.7: Added reference B-OPG22d • Section B.7.2.8.1: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.8.3: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG22c • Section B.7.2.8.4: Added reference B-OPG22b • Section B.7.2.8.5: Added reference B-OPG21 • Section B.7.2.8.6: Added reference B-OPG22b • Section B.7.2.7: Added reference B-OPG22d • Section B.7.2.8.5: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.8.6: Updated as per the most recent safety assessment B-OPG22b • Section B.7.2.8.7: Added reference B-CAN18 • Section B.8.1.1.1: Updated as per the most recent safety assessment B-OPG22b. Included information for SB4

		• Section B.8.1.1.2: Updated as per the most recent safety assessment B-OPG22b. Included information for SB4 • Section B.8.1.1.3: Updated as per the most recent safety assessment B-OPG22b. Included information for SB4 • Section B.8.2.2: Updated as per the most recent safety assessment B-OPG22b • Section B.8.2.3: Updated as per the most recent safety assessment B-OPG22b • Section B.8.2.5: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG22d • Section B.8.2.6.1: B-OPG11 replaced with B-OPG22g • Section B.8.2.6.3: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG22c • Section B.8.2.6.4: Updated as per the most recent safety assessment B-OPG22b. B-OPG11 replaced with B-OPG22g • Section B.8.2.6.5: Updated as per the most recent safety assessment B-OPG22b for storage of toxic materials in SB3 • Section B.8.2.6.6: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG21. • Section B.8.2.6.7: Updated as per the most recent safety assessment B-OPG22b. Added reference B-OPG22b • Added references B-CAN18, B-CAN21, B-CSA19a, B-CSA19b, B-CSA20, B-CSA22, B-EcoMetrix18, B-OPG00, B-OPG03, B-OPG18a, B-OPG18b, B-OPG18c, B-OPG21, B-OPG22a, B-OPG22b, B-OPG22c, B-OPG22d, B-OPG22e, B-OPG22f, B-OPG22g, , B-ORNL11, B-Sandia83, B-NWMO21, • Removed references B-CAN87, B-CNSC17, B-CSA14a, B-CSA14b, B-CSA14c, B-COG15, B-OPG11, B-OPG12, B-OPG16a, B-OPG16b, B-OPG17c, B-OPG18, B-OPG19 • Table B-1.1: Data from B-NWMO21 added in new Table • Tables B-2 – B-9: Updated as per the most recent safety assessment B-OPG22b • Figure B-9: New figure
		Appendix C • Section C.1.1: Replaced C-CSA14a with C-CSA20, Replaced C-CSA14c with C-CSA19a • Section C.1.1.1: Replaced C-OPG16c with C-CSA19b, Replaced C-OPG16a with C-CSA19a • Section C.1.2: Added reference C-EcoMetrix18 • Section C.3.1: Updated as per the most recent safety assessment C-OPG22a • Table C-5: Updated as per the most recent safety assessment C-OPG22a • Section C.3.2: Updated to include data up to 2021 • Section C.3.3: Updated to include data up to 2021 • Section C.3.3.4.1: Updated as per the most recent safety assessment OPG22a • Section C.3.3.4.3: Updated as per the most recent safety assessment C-OPG22a • Section C.4.1.3: Added reference C-OPG22c

		• Section C.4.1.4: Updated as per the most recent safety assessment C-OPG22a • Section C.4.1.5: Updated as per lightning strike assessment C-OPG22b. Added reference C-OPG02 • Removed references C-CSA14, C-CSA14a, C-CSA14b, C-OPG16a, C-OPG16b, C-OPG16c, C-OPG17c, C-OPG19 • Added references C-CSA19a, C-CSA19b, C-CSA20, C-EcoMetrix18, C-OPG18a, C-OPG18b, C-OPG22a, C-OPG22b, C-OPG22c • Replaced C-CSA14a with C-CSA20 • Replaced C-CSA14b with C-CSA19b • Replaced C-CSA14c with C-CSA19a

1.0 INTRODUCTION

1.1 Purpose

This report provides the information to support the Nuclear Sustainability Services - Pickering Waste Management Facility (NSS-PWMF) Operating License, as required by the Nuclear Safety and Control Act (NSCA) and associated Regulations. The NSS-PWMF consists of the Used Fuel Dry Storage (UFDS) area for the storage of used fuel in Dry Storage Containers (DSCs), the Retube Component Storage (RCS) area, and land that has been reserved for future expansion.

1.2 Overview of the Pickering Waste Management Facility

Ontario Power Generation (OPG) currently operates the NSS-PWMF. The NSS-PWMF is composed of 2 sites. The NSS-PWMF Phase I site is located within the Pickering Nuclear Generating Station (NGS) protected area, southeast of Pickering NGS Unit 8, adjacent to the east side of the station security fence. The NSS-PWMF Phase II site is located approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”.

The NSS-PWMF Phase I site consists of the following sub-facilities: UFDS for interim storage of Pickering used fuel in DSCs; and RCS for interim storage of Pickering NGS A irradiated reactor components in Dry Storage Modules (DSMs).

The NSS-PWMF Phase II site contains a security kiosk, DSC Storage Buildings 3 and 4.

A photograph depicting the layout of the NSS-PWMF is shown in Figure 1-1. Aerial views of the Phase I and Phase II sites are shown in Figure 1-2 and 1-3 respectively.

1.2.1 Used Fuel Dry Storage at the Pickering Waste Management Facility

When fully constructed, the NSS-PWMF is expected to provide sufficient capacity to store the used fuel from the Pickering reactors until the end of the service life of the stations. UFDS at the NSS-PWMF consists of Phase I and Phase II sites.

The NSS-PWMF Phase I site was constructed in two stages.

- Stage 1 became operational in January 1996 and contains a DSC processing building and DSC Storage Building 1. DSC Storage Building 1 has a design capacity of up to 185 DSCs.

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

The Phase II site is located 500 m Northeast of the Pickering Nuclear Generating Station (PNGS) site in the East Complex within a distinct protected area. The Environmental Assessment (EA) for the NSS-PWMF DSC Storage Buildings 3 and 4 was completed and Construction Approval was given on December 23, 2004, by the Canadian Nuclear Safety Commission (CNSC). The Phase II was also constructed in two stages.

- Stage 1 was operationally in service in 2009, and contains a DSC storage building 3. DSC storage building 3 has a design capacity of 480 DSCs.

- Stage 2 became operationally in service in 2021, and contains a DSC storage building 4. DSC storage building 4 has a design capacity of 624 DSCs.

As of December 2022, 411 loaded DSCs have been stored in dry storage at the NSS-PWMF Phase I site and 772 loaded DSCs have been placed in storage at the NSS-PWMF Phase II site. Thus a total of 1,183 loaded DSCs have been stored in dry storage at the NSS-PWMF.

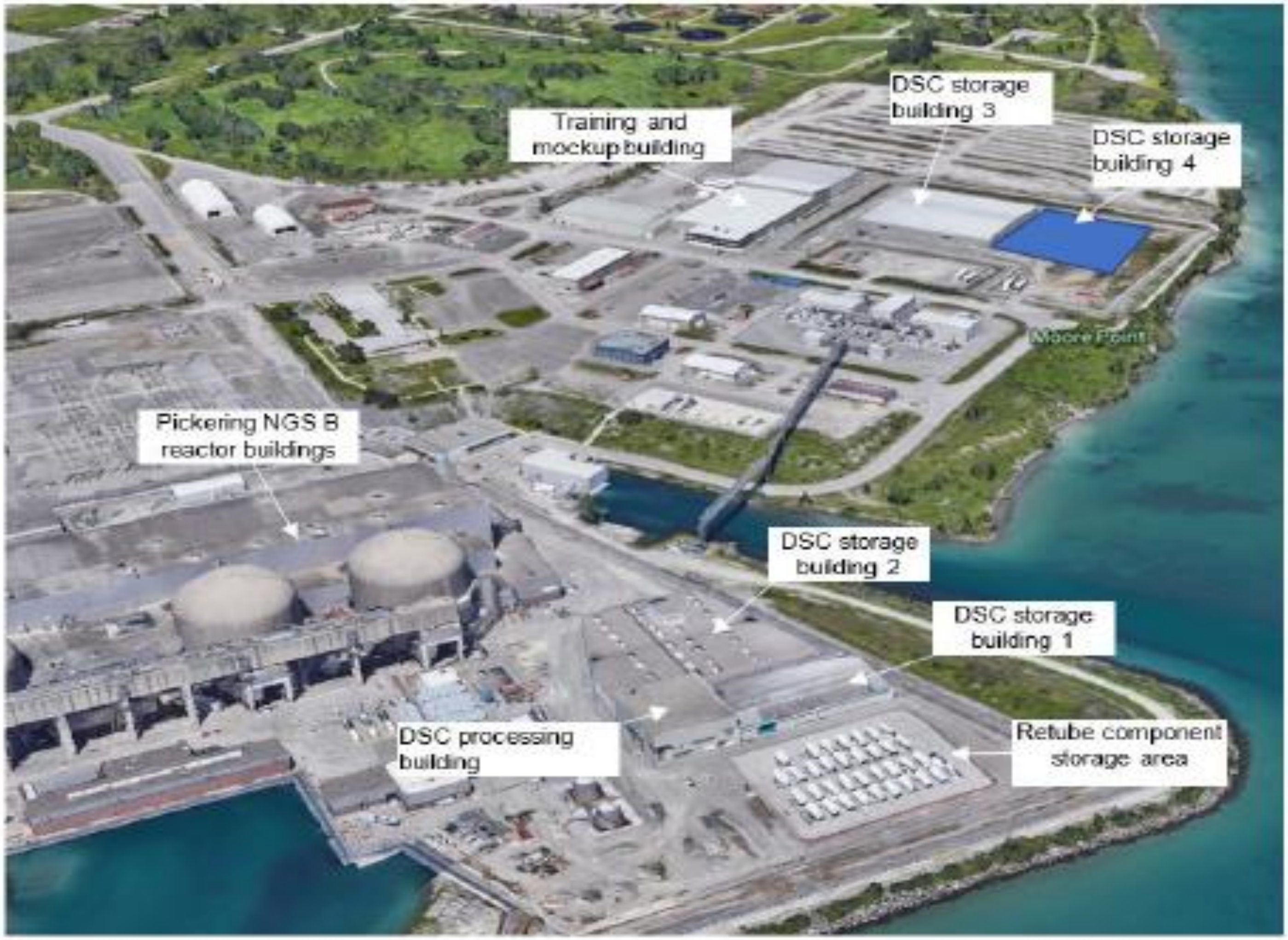

Figure 1-1: Pickering Waste Management Facility Layout and Surrounding Buildings

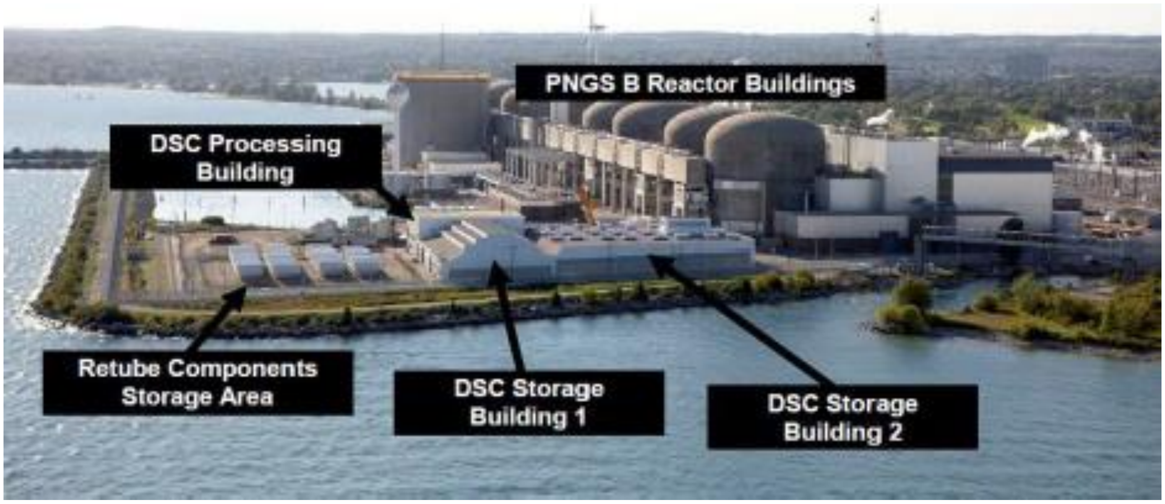

Figure 1-2: Aerial View of NSS-PWMF Phase I

Figure 1-3: Aerial View of NSS-PWMF Phase II

1.2.2 Dry Storage of Used Fuel

Since 1996, used fuel that has been cooled for at least 10 years in the Pickering Irradiated Fuel Bays (IFBs) has been routinely transferred into DSCs for dry storage. The DSC is a rectangular container made of a double carbon-steel shell filled with reinforced high density concrete. The DSC has been designed for above ground storage and transportability. Each DSC has a storage capacity of 384 used fuel bundles. The used fuel in the seal-welded DSCs is stored in an inert helium atmosphere. The design life of UFDS systems and components is at least 50 years.

Dry storage of used fuel, initially cooled in IFBs for several years after discharge from the reactor, is used worldwide as the preferred method for safe and economical storage of used fuel. In addition to UFDS at the NSS-PWMF, there are other dry storage facilities in operation in Canada. There are facilities located at Atomic Energy of Canada Limited's (AECL's) Douglas Point Generating Station on the Bruce Nuclear Power Development (BNPD) site, Hydro Quebec's Gentilly Nuclear Power Plant, and New Brunswick Power's Point Lepreau NGS. OPG has UFDS at its Western Waste Management Facility (NSS-WWMF) on the Bruce site receiving used fuel from Bruce Power since 2003. OPG's Darlington Waste Management Facility (NSS-DWMF) at the Darlington Nuclear Generating Station (DNGS) site became operational in early 2008. The purpose of the NSS-DWMF is to store DSCs with used fuel from the DNGS reactors.

1.2.3 Retube Components Storage at the Pickering Waste Management Facility

The RCS at the NSS-PWMF provides interim storage of irradiated reactor components in DSMs. These components were removed during the retube of the Pickering NGS A reactors in the period of 1984 to 1992. With the exception of periodic inspection, monitoring, and maintenance of DSMs and the RCS area, there have been no operational activities for RCS since 1993.

1.2.4 Storage of Retube Waste in Dry Storage Modules

DSMs contain irradiated fuel channel components from the Pickering NGS A retubing operation. The DSMs are stored outdoors in the RCS area, situated south of the NSS-PWMF Phase I UFDS buildings, as shown in Figure 1-1.

The irradiated components, consisting of pressure tubes, end fittings, shield plugs, and miscellaneous identified components, are stored in 34 DSMs. Two empty DSMs are stored in the RCS area for contingency purposes.

1.3 Scope of the Safety Report

The Safety Report describes the NSS-PWMF and its operations, including the on-site transfer of DSCs between the station IFBs and the NSS-PWMF, the processing of DSCs at the NSS-PWMF, DSC storage at the NSS-PWMF, DSC transfer between the NSS-PWMF Phase I and Phase II sites, and the storage of retube waste in DSMs. Within the scope of the safety assessment, the report discusses used fuel and retube waste characteristics, long-term performance of the DSCs and DSMs, and the potential public and occupational dose consequences and environmental effects of facility operations under normal, postulated abnormal and credible[1,2] accident conditions.

A site description is presented, encompassing the atmospheric, aquatic, terrestrial, geophysical and social environments, as well as the use of the land. The report gives an overview of safety and environmental monitoring programs and the nuclear management system applicable to the NSS-PWMF.

The Safety Report aspects of the loading, decontamination, and vacuum drying of DSCs at the station IFBs are covered in the Pickering NGS Safety Reports and the off-site transportation of DSCs will be covered in a Safety Report by the Nuclear Waste Management Organization (NWMO) in the future.

Public views regarding the storage of used fuel and components from retubing operations have been considered from the early stages of the NSS-PWMF. Public communication activities relevant to the NSS-PWMF are outlined in this report. The report includes an overview of the preliminary decommissioning planning for the NSS-PWMF.

1.4 Waste Management Strategy

1.4.1 Used Fuel Storage Strategy

OPG is committed to safe interim storage of its used nuclear fuel on its reactor sites until a long-term waste management approach becomes available. Alternative long-term waste management options for used fuel have been investigated by the NWMO in compliance with the federal Nuclear Fuel Waste Act. Further details on the long-term waste management plan is provided in Section 10.0.

The intended purpose of UFDS at the NSS-PWMF is to store used fuel from Pickering NGS reactors only. DSCs are capable of being transported off-site although this is not currently a planned or licensed activity.

1.4.2 Retube Waste Storage Strategy

The DSMs provide safe interim storage for the irradiated retube components.

There are no liquid emissions and no significant airborne emissions from the DSMs. OPG plans to safely store retube reactor components in the DSMs at the Pickering site until the Pickering NGS reactors are decommissioned; allowing the radioactivity levels to decay on site is considered a safe option for the interim management of this waste.

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[1] An accident scenario is considered “credible” if its probability of occurrence is deemed to be 10^{-6} per year or higher.

[2] This safety analysis did not reference REGDOC 2.4.4. A gap assessment and subsequent implementation plan will need to be completed for REGDOC 2.4.4 before it is referenced in safety analysis for NSS-PWMF.

1.5 Performance Objectives

Objectives of the NSS-PWMF are:

- To provide safe interim storage of the used fuel from Pickering NGS reactors in DSCs until all the used fuel is transported to an alternative long-term used fuel storage or disposal facility; and
- To provide safe storage of the retube reactor components for Pickering NGS A in DSMs until they are transported to a disposal facility.
- To ensure radioactive waste and used fuel stored at the site is retrievable.

OPG is committed to managing radioactive waste in an environmentally, socially, and financially responsible way, to ensure the protection of workers, the public, and the environment, and to ensure full compliance with regulatory and licensing requirements.

1.6 Performance Criteria

During normal NSS-PWMF operations and under abnormal and credible accident conditions, the dose consequences to the public and the workers will be within the criteria described in the following sections.

1.6.1 Normal Operations

General radiological protection requirements are established in OPG's governing documents on Radiation Protection (RP). Any radiation dose resulting from NSS-PWMF operations will be within the regulatory dose limits and kept As Low As Reasonably Achievable (ALARA). The CNSC regulatory dose limits for the public and Nuclear Energy Workers (NEWs) are shown in

Table 1-1.

OPG has based the NSS-PWMF radiation dose rate targets on the public dose limits in the SOR/2000-203 Regulations (CAN21), which are promulgated under the Nuclear Safety and Control Act that came into force on May 31st, 2000. As per SOR/2000-203, every licensee, like NSS-PWMF, must ensure that the effective dose received by and committed to a person described in column 1 of Table 1-1, during the period set out in column 2, does not exceed the effective dose set out in column 3.

Table 1-1: Canadian Nuclear Safety Commission Effective Dose Limits

Person	Period	Effective Dose (mSv)
NEW, including a pregnant NEW	1-year dosimetry period.	50
	5-year dosimetry period.	100

Person	Period	Effective Dose (mSv)
Pregnant NEW	Balance of the pregnancy (after the licensee is informed of the pregnancy).	4
A person who is not a NEW (i.e. the public)	One calendar year.	1

The dose/dose rate targets for NSS-PWMF operations, derived from Table 1-1 for a member of the general public, are as follows:

≤ 0.5 µSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public (CAN21), over a maximum of 2,000 hours per year occupancy for non-NEWs.

≤ 100 µSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year (CAN21) for a member of the public.

The administrative dose target is an internal target used to ensure that the facility is designed to protect the public during normal operations. The OPG target is applied at the site boundary, which is located closest to the NSS-PWMF on the east side as shown in Figure 2-1. This location is a walking trail and is not occupied by the representative most exposed member of the public. There is a fence that runs along the site boundary and overlaps with the PNGS exclusion zone on the east side of the property.

With the future expansion of the NSS-PWMF Phase II site to include Storage Building 5/6 (SB5/6), construction of used fuel dry storage buildings will be closer to the site boundary. The previous administrative target of 10 µSv/year was implemented early in the NSS-PWMF Phase I development and is now overly restrictive for expansion activities. An administrative dose target of 100 µSv/year has been adopted and will be applied during the design of future Storage Buildings.

Doses to the public from NSS-PWMF will continue to be well below the CNSC annual regulatory dose limit of 1000 µSv/year.

1.6.2 Potential Abnormal Operating Conditions and Credible Accident Conditions

The radiological doses from radionuclide releases and direct radiation, either to members of the public at the Pickering NGS site boundary or to workers, following abnormal operating conditions or a credible accident are targeted to remain below the annual dose limits given in Table 1-1.

2.0 SITE DESCRIPTION

2.1 Site Location

The NSS-PWMF Phase I site is located within the Pickering NGS protected area at the southeast corner of the Pickering site, as shown in Figure 2-1. Figure 2-1 also indicates the location of the NSS-PWMF Phase II site containing DSC Storage Buildings 3 and 4 (shown as NSS-PWMF II), approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”.

Pickering NGS is located in the City of Pickering within the Regional Municipality of Durham. The Pickering NGS property is on the north shore of Lake Ontario, 32 km east-northeast of downtown Toronto and 21 km southwest of the City of Oshawa. The two major watercourses closest to the site are Duffins Creek, 2.2 km to the east and the Rouge River, 4 km to the west.

The Pickering NGS property is approximately 240 ha in size with a continuous landscaped buffer paralleling all adjacent municipal roads. The property is fenced and access is restricted and controlled by OPG. There is a 914 m exclusion zone around Pickering NGS. This exclusion zone restricts the type of land uses that can occur within its confines. The exclusion zone is predominately owned by OPG. These lands are primarily used for industrial purposes related to power generation. Two public outdoor recreation parks, Alex Robertson Park and Kinsmen Park, are located approximately 600 m northwest of Pickering NGS A, on lands leased to the City of Pickering and outside the Pickering NGS site boundary fence.

2.2 Site Accessibility

The Pickering site is well serviced by road. Two major highways, Highway 401 and Highway 2, and the main Canadian National (CN) rail line run in the east-west direction, at a closest distance of 2.8 km to the site. The site is accessible from either highway via Brock Road.

Vehicular access to the NSS-PWMF Phase I is through the north side paved road in front of the Pickering NGS site, leading to the facility through an existing road. Vehicular access to the Phase II site is through the Brock Road south into the east side of Pickering NGS.

Pedestrian access to the NSS-PWMF Phase I site is via a paved road on the east side of Pickering NGS. Pedestrian access to the Phase II site is via the paved pedestrian lane along the Brock Road.

2.3 On-Site Facilities

Emergency, medical aid and fire prevention facilities for the NSS-PWMF Phase I site are provided by Pickering NGS. A 12 inch (30 cm) buried ring header loops around the reactor buildings and powerhouse, providing water for fire protection purposes.

Emergency, medical aid and fire prevention facilities for the NSS-PWMF Phase II site are provided by Pickering Fire Services and Durham Emergency Medical Services with backup support provided by the Pickering NGS Emergency Response Team (ERT).

OPG maintains its own full-time fire crews. It also has a standing agreement with the City of Pickering's fire department. Together, these ensure a quick response to all emergencies at the Pickering site.

OPG has developed and maintains a comprehensive on-site and off-site emergency response plan, referred to as the Consolidated Nuclear Emergency Plan (CNEP). Response teams are trained and equipped to respond to fires, as well as emergencies involving the non-routine release of radioactivity.

The CNEP is a consolidated plan that applies to both the Pickering and Darlington nuclear sites.

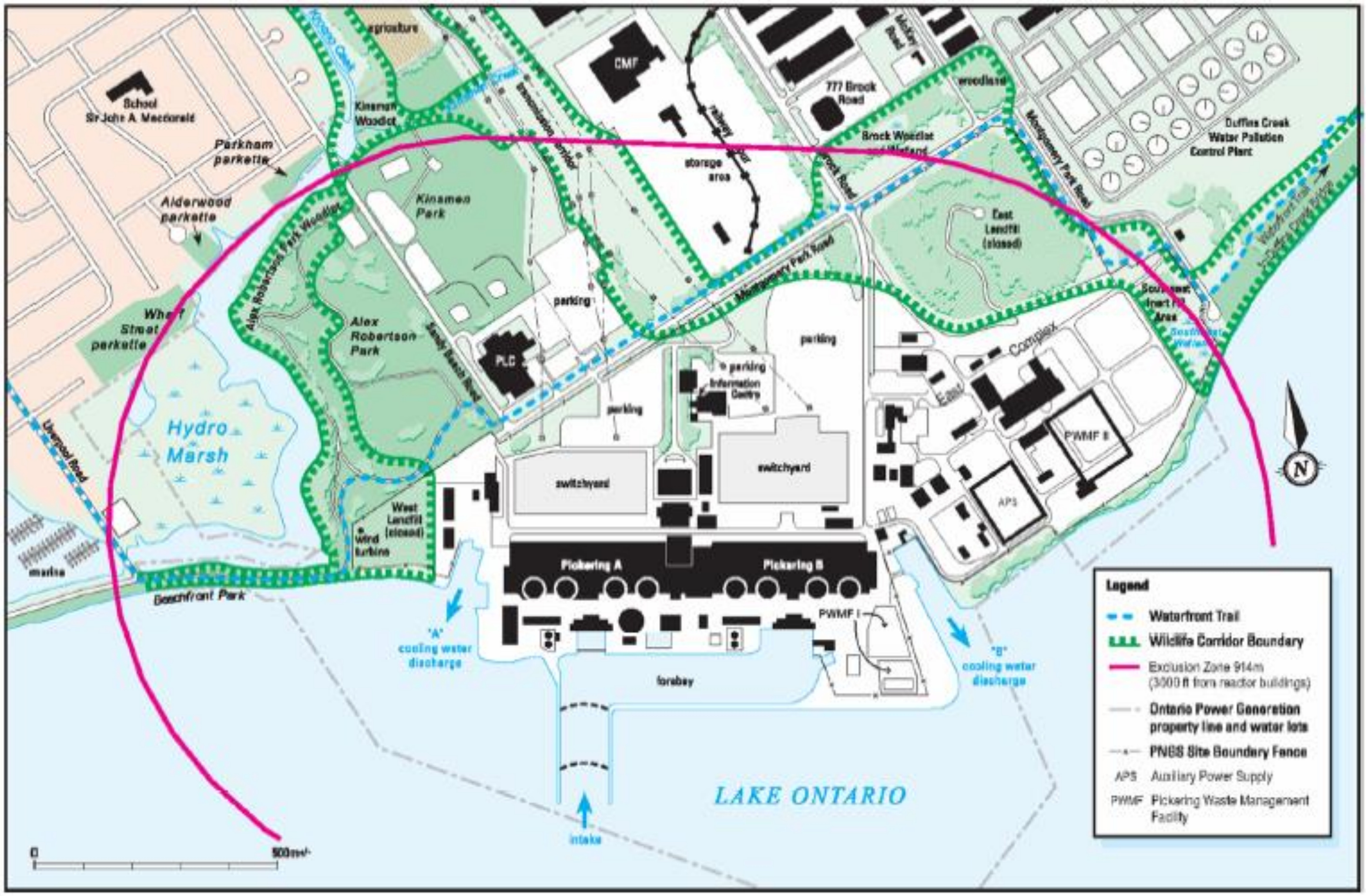

Figure 2-1: Pickering Nuclear Generating Station Site Layout Showing the Pickering Waste Management Facility Sites for Phase I and Phase II Relative to the Pickering Nuclear Generating Station and the Exclusion Zone Boundary

2.4 Environmental Studies

OPG has been conducting environmental studies at the Pickering NGS site since 1969. Studies have focused on environmental effects of PNGS and NSS-PWMF operations, routine radiological environmental monitoring, and EAs.

Environmental monitoring program reports that include Pickering NGS and the NSS-PWMF are prepared annually (OPG22f). Environmental radiation monitoring at the NSS-PWMF is also conducted and the results are reported to the CNSC on a quarterly basis.

The Environmental Risk Assessment Report for Pickering Nuclear (OPG22j) evaluated the risk to relevant human and ecological receptors resulting from exposure to contaminants and physical stressors related to the Pickering Nuclear site and its operations. The report also recommended potential monitoring or assessment as needed based on the results of the Environmental Risk Assessment. Overall, the report concluded that the Pickering Nuclear site is operating in a manner that is protective of human health and ecological receptors residing in the surrounding area.

The Pickering NGS B Refurbishment EA Report (OPG07a) summarizes much of the historical information, and results of an extensive study of the site environment at local and regional levels. The Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report (OPG03b) identified no significant residual adverse environmental effects of the NSS-PWMF Phase II project, which involved the site preparation, construction, operation and maintenance of two additional storage buildings (DSC Storage Buildings 3 and 4) at the NSS-PWMF site. Screening EA (OH98a) prepared for the Stage II expansion document environmental information for the site.

2.5 Atmospheric Environment

The meteorology in the vicinity of Pickering NGS is affected by meso-scale/synoptic factors consisting of the general circulation of air masses and the effects of the Great Lakes, and micro-scale factors that include off-shore/on-shore winds (due to temperature difference between land and lake surfaces), terrain and topography. OPG has been gathering on-site meteorological data at Pickering NGS since the 1970s.

2.5.1 Temperature

The mean temperature measured at the Pickering NGS site's meteorological tower is presented in Table 2-1. Mean daily temperatures fall below freezing in January and February. The coldest recorded daily temperature measurement was -16°C, and the warmest recorded daily temperature measurement was 31.9°C during this period.

Table 2-1: Mean Temperature Information for the Pickering Nuclear Generating Station Site[3]

Month	Based on measurements taken between the years 2017 and 2021		
	Maximum Daily (°C)	Minimum Daily (°C)	Mean Daily (°C)
January	4.8	-14.4	-2.2
February	8.2	-16.0	-4.4
March	14.8	-14.4	2.2
April	19.5	-4.6	7.0
May	26.3	1.2	12.3
June	30.5	8.8	18.5
July	31.9	11.7	20.7
August	30.2	12.9	22.7
September	26.5	8.0	17.3
October	25.0	5.2	15.5
November	20.2	0.0	10.0
December	20.0	-4.9	8.4
Annual Average	**21.5**	**-0.5**	**10.7**

2.5.2 Precipitation

The Pickering site weather tower does not record the precipitation amount and fog occurrence. The Frenchman's Bay meteorological station is the closest weather station to the Pickering NGS site and represents the best record of precipitation at the Pickering NGS between the years 1971 and 2000 (EC06). Prior to 2004, precipitation data was taken from the Frenchman’s Bay meteorological station. Post 2004, precipitation data was taken from Oshawa, which is 21 km from the Pickering site. The precipitation data is presented in Table 2-2.

Measurable precipitation occurred an average of approximately 121 days per year. More precipitation is recorded in the second half of the year. The annual precipitation is approximately 709 mm (OPG22c).

[3] Measurements were taken at 10 m elevation at the Pickering NGS site Meteorological Tower. Minimum and maximum temperatures were measured on the coldest and warmest days of that month.

Table 2-2: Precipitation Data from Oshawa Meteorological Station (2021)

Month	Precipitation (mm), monthly average	Precipitation (mm), daily maximum
January	19	7.7
February	29	10.8
March	32	20.4
April	62	23.8
May	20	8.4
June	62	14.2
July	129	41.5
August	41	15.5
September	145	48.1
October	70	20.9
November	43	9.9
December	58	19.8
Annual	**709**	

2.5.3 Wind

Table 2-3 contains the wind direction and wind speed joint frequencies measured at the 10 m elevation Pickering NGS Meteorological Tower. The data was collected between the years 2017 and 2021, and was measured at a height of 10 m. Wind speeds are grouped into sectors according to compass direction (the direction from which the wind is blowing). Figure 2-2 presents wind roses derived from this data. From this figure, the prevailing winds in the vicinity of the Pickering NGS are shown to occur most commonly (greater than 30 percent of the time) from the northwesterly quarter; winds from the southwest and northeast occur 23 percent to 27 percent of the time, while winds from the southeast are least frequent. The average measured wind speed at the meteorological tower was approximately 2.6 m/s (9.4 km/h). Periods with little or no air movement, or “calms” (<1 km/h), were reported 5 percent of the time.

The meteorology in the vicinity of the plant is affected by the proximity of the station to the lake. This is commonly referred to as the lake effect, which causes lake breezes. Lake breezes result from temperature differences between land and water. An important characteristic of lake breezes is the formation of the Thermal Inversion Boundary Layer (TIBL).

In the spring and summer, when the skies are clear and the geostrophic winds are light, a strong temperature gradient exists between the air over the land and the air over Lake Ontario. This gradient begins forming in the morning as the land is heated at a higher rate relative to the water due to solar radiation. As a result the warm air over land rises and is replaced by the cooler lake air, thus producing a lake breeze. At night this situation is reversed, resulting in a land breeze. Lake breezes are usually stronger than land breezes. In the fall and winter, the lake is generally warmer than the land, resulting in more frequent land breezes.

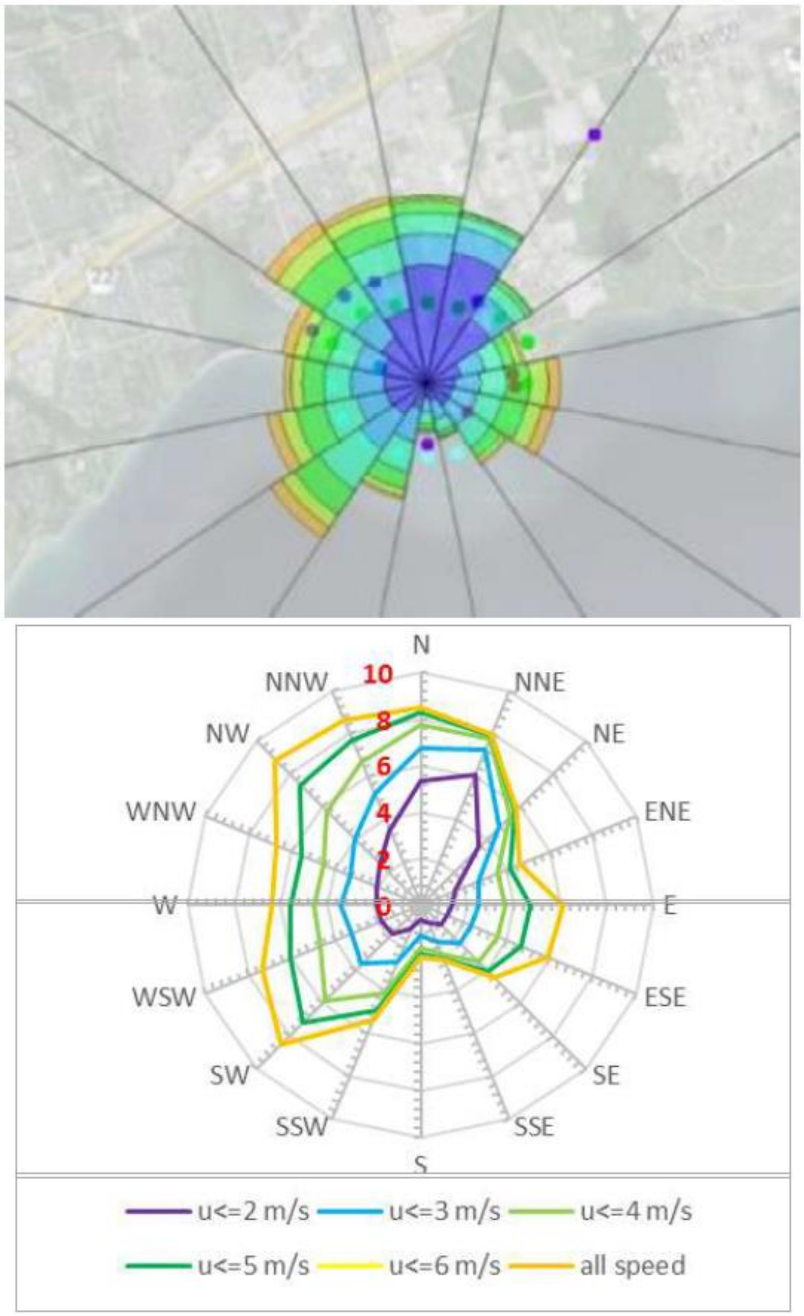

Figure 2-2: Wind Rose for the Pickering Site 10 m Tower

Table 2-3: Wind Direction and Wind Speed Joint Frequencies (%) between the years 2017 and 2021 at the 10 m elevation Pickering Meteorological Tower

Direction Wind Blowing		Wind Speed (m/sec)						
From	**To**	**<2**	**2-3**	**3-4**	**4-5**	**5-6**	**>6**	**Total**
N	S	5.37	1.40	0.97	0.53	0.19	0.05	8.52
NNE	SSW	6.08	1.16	0.50	0.18	0.05	0.00	7.97
NE	SW	3.56	1.24	0.67	0.27	0.07	0.01	5.81
ENE	WSW	1.61	1.11	0.88	0.52	0.30	0.17	4.59
E	W	1.32	1.14	1.16	1.06	0.87	0.54	6.09
ESE	WNW	1.20	1.15	1.18	1.10	0.74	0.55	5.93
SE	NW	1.21	1.15	1.15	0.58	0.25	0.13	4.47
SSE	NNW	0.81	0.89	0.57	0.20	0.08	0.03	2.58
S	N	0.64	0.67	0.56	0.23	0.10	0.05	2.25
SSW	NNE	1.16	1.52	1.42	0.81	0.32	0.15	5.37
SW	NE	1.73	1.86	2.24	1.35	0.68	0.60	8.47
WSW	ENE	1.86	1.32	1.53	1.33	0.74	0.58	7.35
W	E	1.89	1.52	1.19	1.00	0.50	0.33	6.43
WNW	ESE	2.03	1.28	1.20	1.01	0.61	0.55	6.69
NW	SE	2.49	1.53	1.67	1.61	0.89	0.65	8.84
NNW	SSE	3.45	1.75	1.50	0.99	0.64	0.31	8.63
Total		**36.41**	**20.69**	**18.40**	**12.79**	**7.01**	**4.70**	**100.00**

In warm seasons, due to solar heating, the air over the land is often 10°C (or more) warmer than the air over the water. When cold, stable lake air flows over warmer land, the resulting upward heat flux gives rise to a TIBL. This TIBL grows in depth with distance inland as the stable air is advected over land and adjusts to changes in surface roughness, heat and moisture input. The depth of the TIBL is typically hundreds of metres and extends on the order of 10 km inland before a new equilibrium is reached.

For emission sources near the ground, pollutants emitted into the unstable boundary layer would result in higher than expected ground level concentrations during on-shore flows with a TIBL because the stable layer aloft would limit vertical diffusion.

2.5.4 Atmospheric Stability

Atmospheric stability is a measure of atmospheric turbulence. The turbulent nature of the atmosphere strongly affects the concentration of contaminants downwind of the release point. As outlined in Canadian Standards Association (CSA) N288.2-19 (CSA19a), a highly turbulent atmosphere is referred to as Stability Class A and occurs under warm sunny conditions. A highly stable atmosphere is referred to as Stability Class F and occurs typically at nighttime and under low wind speed conditions. A neutral atmosphere, referred to as Stability Class D, is representative of average turbulence conditions and occurs typically under cloudy, windy conditions. All other things being equal for ground level releases, downwind contaminant concentrations are highest when the atmosphere is highly stable (F stability) and lowest when the atmosphere is highly unstable (A stability).

Using the modified sigma theta with daytime and nighttime correction method given in (CSA19a), meteorological data collected at the 10 m-elevation Pickering NGS Meteorological Tower between the years 2017 and 2021 produces the atmospheric stability distribution shown in Table 2-4.

Table 2-4: Frequency of Occurrence (%) of Atmospheric Stability Class (2017-2021)

Year	Stability Class					
	A	B	C	D	E	F
2021	4.3%	7.6%	21.2%	41.6%	13.4%	11.8%
2020	4.5%	8.0%	17.3%	45.4%	14.0%	10.7%
2019	4.8%	8.3%	18.0%	43.9%	14.5%	10.4%
2018	3.3%	8.3%	17.1%	47.8%	15.0%	8.6%
2017	3.4%	8.0%	19.1%	46.9%	14.4%	8.3%
2017-2021	4.1%	8.0%	18.2%	45.4%	14.6%	9.7%

2.5.5 Joint Frequency of Occurrence

The meteorological data used to predict stability class are measured at the 10 m elevation of the Pickering NGS meteorological tower. The joint frequency of occurrence distribution is given in Table 2-5.

Table 2-5: 10 m-elevation Pickering Nuclear Generating Station Meteorological Tower: Wind Speed and Wind Direction Frequencies by Stability Class (2017-2021)

Wind Direction (Blowing From)	Stability Class A						
	Wind Speed, u (m/s)						
	u ≤ 2	2 < u ≤ 3	3 < u ≤ 4	4 < u ≤ 5	5 < u ≤ 6	u > 6	Total
N	0.207	0.192	0.163	0.104	0.014	0.000	0.679
NNE	0.236	0.081	0.035	0.002	0.000	0.000	0.354
NE	0.156	0.046	0.016	0.000	0.000	0.000	0.218
ENE	0.125	0.030	0.009	0.000	0.000	0.000	0.165
E	0.106	0.023	0.000	0.002	0.000	0.000	0.132
ESE	0.076	0.011	0.007	0.000	0.000	0.000	0.095
SE	0.083	0.016	0.000	0.000	0.000	0.000	0.099
SSE	0.100	0.014	0.002	0.000	0.000	0.000	0.116
S	0.093	0.019	0.002	0.000	0.000	0.000	0.114
SSW	0.070	0.030	0.005	0.000	0.000	0.000	0.104
SW	0.132	0.028	0.028	0.002	0.000	0.000	0.190
WSW	0.119	0.058	0.039	0.009	0.000	0.000	0.225
W	0.125	0.072	0.030	0.009	0.000	0.000	0.237
WNW	0.123	0.058	0.058	0.014	0.000	0.000	0.253
NW	0.132	0.088	0.100	0.037	0.005	0.000	0.362
NNW	0.181	0.197	0.218	0.109	0.018	0.002	0.725
All directions	**2.064**	**0.963**	**0.713**	**0.289**	**0.037**	**0.002**	4.069

Wind Direction (Blowing From)	Stability Class B Wind Speed, u (m/s)						
	u ≤ 2	2 < u ≤ 3	3 < u ≤ 4	4 < u ≤ 5	5 < u ≤ 6	u > 6	Total
N	0.265	0.206	0.252	0.145	0.074	0.014	0.956
NNE	0.215	0.128	0.074	0.032	0.005	0.000	0.454
NE	0.150	0.085	0.046	0.005	0.000	0.000	0.286
ENE	0.086	0.042	0.026	0.000	0.000	0.000	0.153
E	0.072	0.054	0.007	0.007	0.000	0.000	0.140
ESE	0.107	0.069	0.007	0.000	0.002	0.000	0.185
SE	0.100	0.042	0.009	0.002	0.000	0.000	0.153
SSE	0.077	0.074	0.012	0.002	0.002	0.000	0.167
S	0.102	0.109	0.035	0.002	0.000	0.000	0.249
SSW	0.098	0.144	0.044	0.002	0.000	0.000	0.288
SW	0.132	0.137	0.042	0.014	0.005	0.002	0.332
WSW	0.104	0.067	0.097	0.033	0.005	0.005	0.311
W	0.140	0.142	0.156	0.140	0.070	0.028	0.675
WNW	0.116	0.137	0.212	0.282	0.086	0.035	0.867
NW	0.132	0.167	0.296	0.425	0.147	0.070	1.238
NNW	0.184	0.246	0.383	0.378	0.241	0.120	1.552
All directions	**2.080**	**1.849**	**1.697**	**1.470**	**0.636**	**0.274**	**8.006**
Wind Direction (Blowing From)	**Stability Class C Wind Speed, u (m/s)**						
	u ≤ 2	2 < u ≤ 3	3 < u ≤ 4	4 < u ≤ 5	5 < u ≤ 6	u > 6	Total
N	0.290	0.091	0.076	0.055	0.035	0.009	0.556
NNE	0.409	0.144	0.084	0.065	0.021	0.000	0.723
NE	0.250	0.199	0.132	0.042	0.021	0.000	0.644
ENE	0.183	0.203	0.168	0.136	0.092	0.066	0.847
E	0.168	0.278	0.375	0.324	0.256	0.188	1.589
ESE	0.177	0.319	0.274	0.308	0.173	0.108	1.359
SE	0.254	0.427	0.348	0.097	0.030	0.007	1.163
SSE	0.190	0.372	0.209	0.025	0.005	0.000	0.801
S	0.114	0.318	0.263	0.046	0.005	0.005	0.751
SSW	0.188	0.534	0.502	0.128	0.019	0.007	1.377
SW	0.237	0.482	0.679	0.297	0.049	0.035	1.778
WSW	0.160	0.238	0.407	0.323	0.170	0.056	1.355
W	0.139	0.204	0.257	0.260	0.187	0.146	1.193
WNW	0.114	0.172	0.240	0.288	0.272	0.223	1.310
NW	0.125	0.202	0.336	0.442	0.421	0.435	1.962
NNW	0.130	0.136	0.169	0.143	0.114	0.083	0.775
All directions	**3.127**	**4.320**	**4.517**	**2.979**	**1.868**	**1.369**	**18.180**
	Stability Class D Wind Speed, u (m/s)						
	u ≤ 2	2 < u ≤ 3	3 < u ≤ 4	4 < u ≤ 5	5 < u ≤ 6	u > 6	Total
N	1.247	0.278	0.464	0.231	0.069	0.023	2.312
NNE	2.334	0.476	0.304	0.079	0.023	0.005	3.220
NE	1.479	0.727	0.471	0.221	0.049	0.009	2.956
ENE	0.412	0.720	0.673	0.388	0.206	0.108	2.507
E	0.273	0.607	0.777	0.730	0.609	0.352	3.349
ESE	0.313	0.612	0.890	0.793	0.567	0.443	3.619
SE	0.382	0.568	0.792	0.478	0.216	0.127	2.564

SSE	0.143	0.391	0.338	0.174	0.070	0.026	1.141
S	0.073	0.165	0.253	0.177	0.093	0.047	0.807
SSW	0.358	0.686	0.859	0.679	0.298	0.140	3.019
SW	0.533	1.011	1.487	1.037	0.629	0.566	5.263
WSW	0.440	0.668	0.980	0.960	0.565	0.519	4.131
W	0.158	0.662	0.746	0.595	0.243	0.152	2.555
WNW	0.344	0.595	0.688	0.430	0.254	0.288	2.598
NW	0.431	0.621	0.939	0.709	0.313	0.146	3.160
NNW	0.356	0.432	0.732	0.361	0.263	0.102	2.246
All directions	**9.276**	**9.219**	**11.393**	**8.041**	**4.466**	**3.052**	**45.447**
Wind Direction (Blowing From)	**Stability Class E**						
	Wind Speed, u (m/s)						
	u ≤ 2	**2 < u ≤ 3**	**3 < u ≤ 4**	**4 < u ≤ 5**	**5 < u ≤ 6**	**u > 6**	**Total**
N	1.716	0.377	0.018	0.000	0.000	0.000	2.111
NNE	1.824	0.268	0.005	0.000	0.000	0.000	2.097
NE	0.904	0.157	0.000	0.000	0.000	0.000	1.061
ENE	0.387	0.088	0.002	0.000	0.000	0.000	0.478
E	0.384	0.141	0.005	0.000	0.000	0.000	0.530
ESE	0.309	0.106	0.005	0.000	0.000	0.000	0.420
SE	0.165	0.086	0.000	0.000	0.000	0.000	0.250
SSE	0.113	0.037	0.011	0.002	0.000	0.000	0.164
S	0.116	0.047	0.005	0.005	0.002	0.000	0.174
SSW	0.203	0.110	0.009	0.002	0.000	0.000	0.325
SW	0.405	0.196	0.007	0.002	0.000	0.000	0.609
WSW	0.592	0.244	0.002	0.000	0.000	0.000	0.839
W	0.824	0.390	0.002	0.000	0.000	0.000	1.216
WNW	0.848	0.287	0.002	0.000	0.000	0.000	1.138
NW	0.975	0.377	0.002	0.000	0.000	0.000	1.354
NNW	1.336	0.513	0.002	0.000	0.000	0.000	1.851
All directions	**11.102**	**3.425**	**0.079**	**0.012**	**0.002**	**0.000**	**14.619**
Wind Direction (Blowing From)	**Stability Class F**						
	Wind Speed, u (m/s)						
	u ≤ 2	**2 < u ≤ 3**	**3 < u ≤ 4**	**4 < u ≤ 5**	**5 < u ≤ 6**	**u > 6**	**Total**
N	1.650	0.256	0.000	0.000	0.000	0.000	1.906
NNE	1.063	0.062	0.000	0.000	0.000	0.000	1.125
NE	0.625	0.023	0.000	0.000	0.000	0.000	0.648
ENE	0.420	0.023	0.000	0.000	0.000	0.000	0.443
E	0.317	0.035	0.000	0.000	0.000	0.000	0.352
ESE	0.218	0.030	0.000	0.000	0.000	0.000	0.248
SE	0.229	0.012	0.000	0.000	0.000	0.000	0.241
SSE	0.187	0.002	0.000	0.000	0.000	0.000	0.190
S	0.144	0.012	0.000	0.000	0.000	0.000	0.156
SSW	0.242	0.016	0.000	0.000	0.000	0.000	0.258
SW	0.293	0.009	0.000	0.000	0.000	0.000	0.302
WSW	0.441	0.044	0.000	0.000	0.000	0.000	0.485
W	0.501	0.054	0.000	0.000	0.000	0.000	0.555
WNW	0.485	0.035	0.000	0.000	0.000	0.000	0.520
NW	0.691	0.077	0.000	0.000	0.000	0.000	0.768
NNW	1.261	0.223	0.000	0.000	0.000	0.000	1.484
All directions	**8.766**	**0.913**	**0.000**	**0.000**	**0.000**	**0.000**	**9.679**

2.5.6 Severe Weather

Severe weather events in the region generally include thunderstorms and lightning, ice storms, wind storms, heavy precipitation, and fog. Thunderstorms require low-level, warm, moist air which, when lifted, will release sufficient latent heat to provide the buoyancy necessary to maintain its upward movement in an extremely unstable atmosphere. Thunderstorms produce lightning and, on occasion, tornadoes. In Southern Ontario locations, thunderstorms normally occur 28 to 30 days a year.

Ice storms, including freezing rain and ice pellets, are associated with atmospheric conditions that are characterized by an elevated inversion having a maximum temperature above 0°C, overriding lower subfreezing. Freezing rain occurs in Southern Ontario, on average, 12 to 17 days per year. Ice storms are usually accompanied or followed by precipitation such as snow, wet snow, rain or fog.

Although infrequent, the most common high windstorm is a tornado. Tornadoes are caused by excessive instability and steep lapse rates in the atmosphere. Tornadoes most often occur along squall lines of a tropical cyclone (low pressure centre), in conjunction with cumulonimbus clouds and severe thunderstorms.

In Southern Ontario, a few tornadoes or near-tornadoes are reported every year, but these storms are not as intense, nor do they cause as much damage, as those in the United States south and west of the Great Lakes. The average tornado in Southern Ontario has a diameter of between 150 and 600 m, and typically travels at a speed of 50 to 70 km/h in a southwest to northeast direction. Tornadoes normally touch ground for less than 20 minutes. In the Pickering area, a maximum of one tornado per 10,000 km^2 can be expected annually (OPG11a).

2.5.7 Non-Radiological Air Quality

The air quality in the vicinity of the Pickering NGS site (which includes the NSS-PWMF) is typical of the general air quality in Southern Ontario within the Quebec-Windsor corridor and the Greater Toronto Area (GTA). Around the Pickering area, the Ministry of the Environment Conservation and Parks (MECP) operates a few ambient air quality monitoring stations to measure Ozone (O_3), sulphur dioxide (SO_2), nitrogen dioxide (NO_2) and particulate matter (PM) – PM for all "respirable" particulates (those measuring less than 2.5 microns in diameter, referred to as $PM_{2.5}$).

Examining the MECP air quality data from the years 2007 to 2023, air quality in the vicinity of the Pickering NGS site was found to be most influenced by NO_2, TSP, and $PM_{2.5}$.

An update to the Emission Summary and Dispersion Modelling (ESDM) report evaluated three emission scenarios for the Pickering Nuclear Generating station (OPG21b). Significant contaminants with MECP point of impingement standards that are discharged from the facility were found to have point of impingement concentrations less than the MECP Limits.

2.5.8 Noise

Pickering Nuclear Generating Station is operated with an Environmental Compliance Approval (ECA) condition to have an up-to-date noise assessment on site. There are three Point of Receptors (POR) considered in the noise assessment report (OPG22g). The first POR #1 is to the northwest of the site at 1443, Parkham Crescent. The second POR #2, is a town house on the west of the facility across from Alex Robertson Park and Alderwood Park. The third POR #3 is to the east, at the OPG property line bordering the Duffin Creek Water Pollution Control Plant. The area surrounding the site is classified as a Class 1 for POR #1 and POR #2 and Class 2 for POR #3 as per the Environmental Noise Guideline definition (NPC-300). The noise emissions were predicted at the PORs for the worst-case scenario “emergency equipment under testing conditions”. These predicted noise levels are all below limits, and the PNGS site is in compliance with the Environmental Noise Guideline definition (NPC-300).

2.6 Aquatic Environment

2.6.1 Drainage

Drainage in the Pickering NGS site is a mix of ephemeral swales, ditches, culverts, and storm sewers – no major watercourses traverse the site. The discharge points are approximately 6 m to 10 m above the Lake Ontario water level. No water body other than a small (0.5 ha) isolated wetland, known as the Southeast Wetland, is located within the Pickering site. This small isolated wetland, which was once farmland and created during the construction of Pickering NGS as a result of landfilling activities, lies in the southeast corner of the Pickering NGS property at the foot of Montgomery Park Road. This wetland receives drainage from the area around the former construction landfill within the Pickering site.

Surface drainage from the Pickering NGS site is enabled by 19 separate storm water drainage basins, or catchments. The NSS-PWMF Phase I site is part of a smaller drainage basin with an area of 3.7 ha. The NSS-PWMF Phase I site shares this area with the Pickering NGS B standby generators. Greater than 95 percent of this area is considered to be impervious. Runoff, including the NSS-PWMF Phase I roof drainage, is directed through the Pickering NGS drainage network into the Pickering NGS B discharge channel. Drainage from the RCS area is directed via catch basins to the Pickering NGS drainage system for discharge to the outfall.

Even though liquid effluents generated inside the NSS-PWMF Phase I site are infrequent and small in volume, they are sampled and pumped into the Pickering NGS Active Liquid Waste Management System.

Surface drainage from the NSS-PWMF Phase II site in the East Complex area drains to Lake Ontario, with drainage areas of between 0.4 ha and 19.8 ha.

2.6.2 Fish

More than 90 fish species are known to inhabit Lake Ontario, of which approximately 60 species have been found in impingement, gillnetting and electrofishing studies at the Pickering NGS site and are considered to represent the fish community in its vicinity. As there are no surface water features suitable as fish habitat within the site, the only aquatic habitat and biota are located within Lake Ontario. However, almost all of these species make use of the nearshore waters of the lake for one or more of spawning, rearing, feeding, migration and over-wintering. These nearshore waters include the Rouge River, Frenchman's Bay, and Duffins Creek; the Pickering B discharge channel is used by smallmouth bass as a spawning area.

Sport fish taken near the Pickering NGS include brown trout, walleye, chinook and coho salmon, rainbow trout, smallmouth bass, white bass and carp. Commercial fishing is not a significant industry in the area.

2.6.3 Lake Water Levels

Using the Marine Environment Data Source, a review of historic water levels at Toronto and Cobourg was undertaken as part of the Pickering A Return to Service EA (OPG00a). Based on the monthly average water levels of Lake Ontario from 1918 to 1998, the annual maximum daily average water levels at Toronto for the period 1908 to 1998 ranged from a low of 74.26 m International Great Lakes Datum (IGLD) (1935) to a high of 75.81 m IGLD (1952). At Cobourg for the period 1956 to 1998, the maximum value ranged from a low of 74.64 m IGLD (1958) to a high of 75.76 m IGLD (1973). Lake Ontario water levels have been regulated since the completion of the St. Lawrence Power Project in 1958.

For the post lake regulation period, portions of the Toronto and Cobourg data (1960 to 1998) were analyzed using a Gumbel Extreme Value analysis to provide a range of maximum daily average level estimates for different return periods. The results are shown in Table 2-6.

Table 2-6: Estimated Daily Maximum Water Levels along the Toronto-Cobourg Corridor

Return Period (years)	Maximum Daily Average Water Level Toronto (m, IGLD)	Maximum Daily Average Water Level Cobourg (m, IGLD)
10	75.5	75.4
50	75.8	75.8
100	76.0	76.0
200	76.1	76.0

The lowest bank height along the shoreline of Pickering NGS is at the southwest corner of the East Complex, which is at 76.7 m IGLD. The NSS-PWMF Phase I site is slightly higher at an estimated 77 m IGLD (with Phase II even higher). When compared to the estimated 1 in 200 year water level of 76.1 m IGLD and the observed historic fluctuations, there is a low probability of flooding at the site due to high lake levels.

A review of the deep water wave climate, in conjunction with the existing topography at the site, was undertaken as part of the Pickering A Return to Service EA (OPG00a) to assess the potential for flooding due to wave runup and overtopping. The review indicated that, with nearshore water depths of 2 to 2.5 m, the maximum wave runups will be approximately 2 m on a riprap shore. As the crest of the protection works at Pickering NGS exceeds 77.5 m IGLD for much of their length, the probability of flooding due to wave runup and overtopping is considered low. Periodic wetting of extreme events due to wave spray and splash will occur. However, it is considered unlikely that waves breaking directly over top of the foreshore works will occur. It is also reasonable to assume that the surface drainage system will have adequate capacity to deal with any spray that may occur.

2.6.4 Lake Ice Conditions

Observations of Lake Ontario ice conditions near the Pickering NGS site from 1969 to 1980 included airborne reconnaissance to map the extent of ice cover and ground observations to determine ice conditions along the shore. It was found that during Pickering NGS operations the extent and duration of ice accumulation depends on meteorological conditions and the extent of the thermal plume (OPG07a).

Ice conditions in Lake Ontario were found to vary substantially every year throughout the winters: a band of brash and slush ice was generally observed along the shore throughout the winter season, but the build-up was found to vary considerably from year to year. The minimum build-up of shore ice occurred during the winter of 1974/1975 as a result of mild weather. The maximum accumulation was observed during the 1976/1977 winter when an ice foot approximately 3 m high extended a distance of about 50 m offshore west of the Pickering NGS site.

Lake ice accumulation in the intake channel can block the supply of water to the Condenser Cooling Water (CCW) system, as well as to other service water systems. Large ice jams are rare and only two significant events have been reported since 1971 (OPG00a). In the past, ice jams have primarily affected Pickering NGS B because of the configuration of the forebay. Ice can accumulate at the Pickering NGS B intake because it is at the end of the forebay where the ice can flow no further. Ice booms installed at the intake channel are expected to prevent any ice jams from occurring as shore ice is pulled into the intake channel.

Pickering NGS has adopted procedures for clearing ice blockages, which include lowering flow rates in the CCW (if necessary, the reactors are powered down to reduce the need for cooling water), directing warm water from the outfall into the intake to melt the ice, and installation of ice booms.

2.7 Terrestrial Environment

The Pickering property is visited by birds during the spring and fall migration; a number of species have been identified as breeding on-site, particularly in association with the Hydro Marsh and adjacent Frenchman’s Bay Marsh. Recent Pickering Nuclear Site Biodiversity Monitoring Program Reports have been completed from 2018 to 2021 that provide updated site observations (BEACON18, BEACON22).

Regarding Species at Risk, Barn Swallow is an annual breeder at the Pickering NGS property including within the Protected Area, one pair of Peregrine Falcons has nested annually from 2014 to 2018 on Pickering NGS property with four fledglings observed at the nest site in 2018. Two territorial male Eastern Wood-Peewees were recorded in 2021 in the wooded area of Alex Robertson Park and one mature butternut tree was located by the entrance of the trail at Alex Robertson Park in 2018. Pairs of Least Bittern were calling during Hydro Marsh monitoring in 2019 and 2020 indicating likely breeding on site. Chimney Swifts were observed flying by Pickering site in 2019 and 2020.

No significant breeding bird populations have been identified in association with the NSS-PWMF.

2.8 Geophysical Environment

2.8.1 Geology

2.8.1.1 Regional Geology

On a regional basis, fairly thick 15 m to 30 m Pleistocene ground moraines overlie the Ordovician shale and limestone bedrock. The Ordovician sedimentary rocks are essentially flat lying and have a measured thickness of 212 m (OH82). These rocks rest unconformably on the Precambrian bedrock surface.

2.8.1.2 Site-Specific Geology

During the course of subsurface investigations and excavations for Pickering NGS, OPG has added substantially to the geological information of the site area. Further geological information for the NSS-PWMF Phase I site was obtained in a study conducted exclusively for this site (OH90). The borehole locations and a sample geotechnical data sheet are given in the subsoil investigation and evaluation report for the site (OH90). The ground surface at the NSS-PWMF Phase I site is at an elevation of approximately 3 m above the lake level.

The NSS-PWMF Phase I site is covered with a variable fill deposit, composed of coarse to fine sand and gravel, ranging in thickness up to 4.6 m. The state of compaction of this fill is erratic. Caissons were used to support the structure at the NSS-PWMF Phase I site. In general, the fill overlies a natural deposit of clayey silt till having stiff to hard consistency. The clayey silt till deposit extends to a depth of 13 m to 15.5 m to rest on a basal till deposit of very dense silt or clayey silt containing pieces of a weathered shale. Zones of hard to very stiff silty clay and dense sand are found sandwiched between the upper clayey silt till and the basal shale-till complex.

Shale bedrock at the NSS-PWMF Phase I site exists at depths varying from 16.9 m to about 20 m below present grade (OH90). The surface of this bedrock in the area ranges between elevation 58 and 64 m above sea level, and rises 2 to 3 m above this, inland from the Pickering NGS location. In the offshore area, the bedrock surface slopes off to an elevation of 49.4 m above sea level for a distance of 305 m.

The NSS-PWMF Phase II site is underlain by 1 m to 2 m of grading fill and 15 m to 20 m of stiff to hard upper and lower till complex soil deposits overlying shale bedrock. The grading fill for road bases and parking lots was derived from either reworked glacial till from on-site or imported granular soils. This shallow horizon is underlain by a hard, low permeability silty clay to clayey silt till horizon approximately 5 m thick which extends out beneath Lake Ontario. This upper till layer is a barrier that restricts downward seepage of groundwater. The upper till is underlain by a more permeable silty to sandy horizon over a lower till layer above the bedrock surface.

The Lake Ontario shoreline south of the NSS-PWMF Phase II site is a relatively undisturbed natural shoreline and bluff, rising an average of 3 to 4 m to the level plateau above. The bluff is composed of silty clay till, and has been prone to erosion in recent years due to elevated lake levels. The beach and foreshore are relatively flat, composed of sand, gravel and boulders associated with the parent tills. The bluff is routinely monitored by OPG for erosion to ensure safety and security of the Pickering NGS facilities.

During several borings and excavations carried out at and around the Pickering NGS site, no evidence was found of any structural weakness in the bedrock foundation. Neither mining activity nor withdrawal of fluids under the site has occurred that may affect the NSS-PWMF.

2.8.1.3 Groundwater

Subsurface investigations indicate a lower water-bearing zone of sand, silt and gravel within the lower till complex and local water-bearing lenses in the upper till complex. A relatively minor water-bearing zone occurs in the top part of the bedrock.

Generally on a regional basis, the groundwater level is about 1 m below the ground surface, which slopes gently towards Lake Ontario. The groundwater gradient is relatively flat and towards the lake. Groundwater movement, along with monitoring around Pickering NGS, have been summarized (OPG22e).

Pickering Nuclear Generating Stations has a mature and robust groundwater monitoring program in place to address the following three primary objectives.

Objective 1: Monitor changes to on-site groundwater quality to ensure timely detection of inadvertent releases to groundwater.

The groundwater data collected from the suspended sediment concentrations indicate stable tritium concentration consistent with results for previous years. An emerging issue was identified at the western Unit 5 to 8 irradiated fuel bays, which is likely a result of groundwater moving from Unit 6, beneath the IFB-B area, ultimately towards the Turbine Auxiliary Bay (TAB) Inactive Drainage (IAD) sumps.

In 2021, thickness of free-phase hydrocarbon product in select wells were measured and hydrocarbon concentrations in groundwater were measured in one well (MW-287-15). The 2021 results for PHC product thickness and hydrocarbons in groundwater were measured in one well (MW-287-15). The 2021 results for PHC product thickness and hydrocarbons in groundwater in these areas were consistent with monitoring results in recent years

Objective 2: Ensure there are no adverse off-site impacts from PNGS groundwater.

Tritium concentrations within the site boundary wells and shoreline wells are stable and are within historical ranges. Off-site impacts to groundwater are not observed.

Objective 3: Confirm predominant on-site groundwater flow characteristics at the PNGS site.

The predominant groundwater flow patterns remain unchanged in 2021 from the recent years' interpretation.

2.8.2 Seismicity

2.8.2.1 Regional Seismicity

The western Lake Ontario region lies within the tectonically stable interior of the North American continent, which is characterized by low rates of seismicity (OPG00b). The seismic zoning maps in the National Building Code of Canada (NBCC, see Reference CCBFC20a), for example, place the site in Zone 0 to 1, with Zone 6 corresponding to the most seismically active regions of the country. The region surrounding Pickering NGS experiences low to moderate levels of seismicity. Most events have magnitude (M, called Richter magnitude) less than 5, with rare occurrences of larger events. In general, earthquakes in stable interior regions, such as the Lake Ontario region, occur at depths of 5 to 20 km, on faults formed hundreds of millions of years ago during previous active tectonic episodes. These faults are widespread throughout the crust, and typically have little to no surface expression.

As a guide to levels of seismic activity in the region, the western Lake Ontario region experiences an earthquake of M = 5 or larger about once every 100 years. This estimate of the level of seismicity is well-established. A magnitude 5 is considered a moderate earthquake, which may cause significant damage to poorly built (unreinforced) structures in the epicentre area, but does not generally damage modern well-engineered structures or heavy industrial structures. More damaging earthquakes of M = 6 or greater are considered possible but the rate of occurrence is ten times lower. This means that the likelihood of a large potentially damaging event (of M = 6 or greater) occurring in the local area is less than 1 in 1,000 per year.

Seismic hazards and design issues for Pickering NGS have been summarized (OPG00b). The seismic load assessment of the DSC is described in Section 3.3.2.6.

2.8.2.2 Site-Specific Seismicity

Research which has drawn on geological investigations and geophysical surveys in Lake Ontario, as well as the development of probabilistically based methodologies have provided assessments of the seismic hazard for the area. Improved understanding of eastern North American magnitude attenuation relationships and the characteristics of seismic wave propagation in stable continental regions has aided quantification of the seismic hazard. Uniform hazard spectra, representing current evaluations of regional seismic hazard, are similar to the Pickering NGS B design basis seismic ground response spectra in the dominant response frequency range of the DSMs. The seismic capacity assessment of the DSMs described in Appendix C was based on the Pickering B Design Basis Earthquake (DBE) as the input ground motion.

2.9 Land Use and Social Environment

2.9.1 Land Use

2.9.1.1 Pickering Nuclear Generating Station Site

The Pickering NGS is part of the Brock Industrial Neighbourhood in the City of Pickering, immediately east of the Bay Ridges Neighbourhood, south of Highway 401, west of the Town of Ajax and north of Lake Ontario. The property is located on a 240 ha site bordering Lake Ontario. There is a 914 m non-residence radius, also known as the exclusion zone, around the development. Beyond this limit, structures include recreational, institutional, and park facilities.

The land use surrounding the Pickering NGS site boundary fence is largely urban, including industrial, residential and parkland uses. The notable exceptions are Frenchman's Bay marsh, Hydro Marsh and Duffins Creek Marsh, which are provincially significant wetlands found in the vicinity of the Pickering NGS property.

The Pickering NGS site has been developed since its inception in the early 1960s. The Pickering property is designated as a utility in the Region of Durham Official Plan. The property is zoned Industrial Zone M2. A small site located between the closed Brock Road right-of-way and the Pickering NGS B thermal discharge bay is zoned Public Service Zone M3.

The City of Pickering official Plan designates the property as a controlled access area. The entire property is fenced and access to the site is restricted and controlled by OPG. Within the Pickering NGS boundaries, existing land uses consist of buildings, structures, switchyards, and transportation access required to operate and support the station's functions.

The buildings include the reactor buildings, powerhouses, a variety of low rise office buildings, warehouse, maintenance, and storage facilities as well as structures designed specifically for the technical functions required for generating and transmitting electricity from nuclear power. Other land uses within the Pickering NGS boundaries include landfills and waste management facilities.

The Duffin Creek Water Pollution Control Plant (WPCP) is located on lands immediately east of the Pickering property. The remaining portions of the Brock Industrial neighbourhood contain a variety of employment uses and vacant land.

The Waterfront Trail, an active recreational path paralleling Lake Ontario across the City of Pickering, with connections to municipalities to the east and west, runs adjacent to the Pickering NGS site boundary fence along Montgomery Park Road, on lands leased to the municipality by OPG.

The Bay Ridges, West Shore and Rosebank neighbourhoods west of Pickering NGS contain predominately residential and compatible ancillary uses (e.g., schools), with some employment and commercial uses generally along Bayly Street and Liverpool Road.

The southern boundary of the Pickering NGS property extends as a water lot into Lake Ontario. The lake is used locally for sport fishing, as well as recreational swimming and boating. The lake also provides water supply to the adjacent municipalities, the nearest water supply plant being in Ajax, 4 km to the east.

2.9.1.2 Pickering Waste Management Facility Site

The NSS-PWMF is distributed across the Phase I and Phase II sites. The RCS area is at the southern end of the NSS-PWMF Phase I site, covering about 0.43 ha. At the northern end of the NSS-PWMF Phase I site are the DSC processing building, offices, and DSC Storage Buildings 1 and 2. A berm to the east of the NSS-PWMF Phase I site prevents direct discharge of runoff to the Pickering NGS B outfall channel. The NSS-PWMF Phase II site provides Storage Buildings 3 and 4 for DSCs. Space on the Phase II site is reserved for future DSC storage requirements.

2.9.2 Population

The Regional Municipality of Durham is one of the largest municipalities in Canada and is also among the fastest growing. Durham Region includes the City of Pickering, the City of Oshawa, the Towns of Ajax and Whitby, the Townships of Brock, Scugog and Uxbridge, and the Municipality of Clarington.

Population growth in Durham Region has been closely linked to development and economic growth in Toronto and surrounding regions (i.e., Peel and York Regions). Buoyed by the strong economy in the GTA, strong population and economic growth is projected for Durham Region and its area municipalities (OPG07a). Detailed population data for Durham Region, including historical, current, and projected, are shown in Table 2-7.

Table 2-7: Historic, Existing, and Projected Population Growth – Durham Region and Area Municipalities (2001-2060)

Region	Area (km²)	2001		2016	2021	Population density 2021 (persons/km²)	2025	2060
Town of Ajax	67.1	76,675		119,677	126,666	1888	131,279	154,665
City of Oshawa	145.7	144,560		159,458	175,383	1204	214,325	331,086
Town of Whitby	146.5	90,875		128,377	138,501	946	193,937	349,694
City of Pickering	231.6	90,595		91,771	99,186	428	171,134	324,274
Municipality of Clarington	611.1	72,605		92,013	101,427	166	150,854	282,294
Uxbridge Township	420.7	18,065		21,176	21,556	51	23,649	29,056
Scugog Township	474.6	21,025		21,617	21,706	46	25,803	30,982
Brock Township	423.7	12,590		11,642	12,567	30	16,655	23,821
Total Regional Municipality of Durham	—	526,990		645,731	696,992	—	927,636	1,525,872

Forecasts indicate that the populations and housing stock of Durham Region and the City of Pickering will continue to grow by an average of approximately 2.5 percent per year over the next 20 years.

Agriculture has historically been an important component of the local and regional economies, environment and social fabric, and represents the most significant land use throughout most of the northern portions of the City of Pickering and Durham Region. Notwithstanding this agricultural heritage, both the City and the Region have developed an industrial platform with major industries in energy, automotive manufacturing, plastics/packaging, pharmaceuticals, aerospace/defense, chemicals/rubber and environmental technologies. Pickering NGS is a major employer within the Region. Approximately 3,000 persons are employed at the Pickering NGS site. Approximately 70 percent of those employed at the station live within Durham Region, while 8 percent live within the City of Pickering.

Pickering NGS is one of the City of Pickering's most distinctive features. It is the most visible component of the Brock Industrial area, where most of the manufacturing industries in the city are located (i.e., south of Highway 401). The areas nearest to Pickering NGS are also characterized by a number of established residential neighbourhoods. At present, new developments are being located in the small remaining undeveloped areas within these neighbourhoods, some providing mixed residential and commercial uses.

As a mature city, residents of Pickering have access to a wide variety of educational, community and recreational facilities and services. The area nearest to Pickering NGS contains several municipal parks and the Waterfront Trail, which are heavily used by the local community for recreation. The lands and ravines associated with Frenchman's Bay, immediately west of the Pickering site boundary fence, provide the greatest concentration of recreational amenities in the City of Pickering. The river mouths, Frenchman's Bay and the shore area in the vicinity of Pickering NGS are also used for recreational fishing and boating. Pickering NGS contributes to these recreational activities, by maintaining approximately 120 hectares of land north of the station and outside the Pickering NGS site boundary fence as natural areas and parklands through lease agreements with the City.

2.9.3 Social Environment

The social environment relevant to the NSS-PWMF has been described in detail in the Pickering A Return to Service EA (OPG00a) and is also discussed in the Pickering B Refurbishment EA (OPG07a).

While there is a significant impact of Pickering NGS on the community, NSS-PWMF is a small but integral part of the Pickering NGS. The NSS-PWMF has only a minimal impact on the social and community environment, mainly due to its limited size and number of employees. However, the NSS-PWMF supports the continued operation of Pickering NGS, and thus contributes indirectly to the social economic benefits of Pickering NGS to the neighbouring communities.

2.10 Indigenous Interests

PNGS is located on the traditional and Treaty territory of the Michi Saagiig and Chippewa Nations, collectively known as the Williams Treaties First Nations (WTFN). The seven Nations comprising the WTFN are Alderville First Nation, Curve Lake First Nation, Hiawatha First Nation, Mississaugas of Scugog Island First Nation, Chippewas of Rama First Nation, Georgina Island First Nation, and Beausoleil First Nation. The WTFN have a relationship with the lands along the north shore of Lake Ontario (from the Bay of Quinte) and north to Lake Simcoe and Rice Lake as a result of their occupation and traditional use of these lands prior to European settlement and the subsequent signing of treaties.

The WTFNs are all within a 170 km radius of the Pickering NGS facility and are detailed as follows:

Nation	Distance from PNGS (km)	Direction from Pickering
Alderville First Nation	100	East
Curve Lake First Nation	130	Northeast
Hiawatha First Nation	110	Northeast
Mississaugas of Scugog Island First Nation	50	Northeast
Chippewas of Beausoleil First Nation	170	Northwest
Georgina Island First Nation	75	North
Rama First Nation	115	North

While there are no historic Metis settlements proximate to the Pickering NGS, there are Metis citizens within the Durham region and OPG engages with the Metis Nation of Ontario as they have an interest in OPG nuclear operations.

2.11 Security

As a Class IB Nuclear Facility that is used to handle and store Category II nuclear material, all buildings belonging to the NSS-PWMF are located within “protected areas” and are provided with appropriate security and alarm systems, in accordance with the CNSC Nuclear Security Regulations. OPG Nuclear has established a comprehensive and effective security program for the two different “protected areas” belonging to the NSS-PWMF. A description of the program, in the form of a Security Report, has been submitted to the CNSC in response to applicable regulations. The Security Report falls under the category of Prescribed Information, as defined by the General Nuclear Safety and Control Regulations, and is protected from public disclosure. The Report is not released to the general public, and is distributed by OPG Nuclear on a strict need-to-know basis.

Access to protected areas is restricted to authorized persons and all entry is controlled by code-access locking devices and monitored by security.

Security systems, staff, and equipment are available to monitor the transfer of DSCs from the station IFBs to the NSS-PWMF Phase I site, which is within the Pickering NGS protected area, and from the NSS-PWMF Phase I site to the NSS-PWMF Phase II site, which is within a separate nearby protected area. The on-site transfer of loaded DSCs is performed in accordance with the Nuclear Security Regulations.

Cyber security requirements are governed by the OPG cyber security policy (OPG-POL-0035) and program (OPG-PROG-0042).

Access to the RCS area is through a locked gate and access requires controlling authority approval.

3.0 USED FUEL DRY STORAGE

3.1 Used Fuel Dry Storage Process Overview

The UFDS process encompasses the facilities, structures, systems, and components (SSCs), and the operations necessary to transfer used fuel from the Pickering NGS IFBs to dry storage at the NSS-PWMF. This includes UFDS SSCs and operations inside the Pickering NGS, the NSS-PWMF, and en-route between the associated buildings. The in-station UFDS process, DSC handling, and safety assessment at the Pickering IFBs form part of the Pickering NGS licensing basis documentation. The summary below provides a complete overview of the dry storage process.

New, empty DSCs are received from the manufacturer at the NSS-PWMF Phase I processing building, where they are prepared and then transferred to the station for subsequent loading. The dry storage process, beginning with the preparation of new DSCs at the processing building and ending with the storage of loaded hermetically sealed DSCs in a DSC storage building, is summarized in Figure 3-1. Details of the modified DSC MKII design are given in Section 3.3.1.

At Pickering NGS, each of the IFBs are filled with demineralized water and contain the used fuel. Irradiated fuel bundles are placed into 96-bundle storage modules. Modules with 10-year or older fuel may be loaded into a DSC. The 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 and drained while being raised, 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 NSS-PWMF. The loaded DSC is transferred on Pickering NGS site roads to the NSS-PWMF Phase I site for further processing.

In the DSC processing building, the DSC lid is seal welded and the integrity of the lid seal is inspected by Phased Array Ultrasonic Testing (PAUT). Final vacuum drying and helium backfilling of the DSC cavity is then performed, followed by installation of the drain port plug; the drain port plug is seal welded and inspected. The DSC is then placed in a vacuum chamber and helium leak tested. On successful completion of this leak test, International Atomic Energy Agency (IAEA) seals are applied and touch-up painting is completed. The application of the IAEA seals is completed in either the DSC processing building or in a designated IAEA surveillance area inside DSC storage building #1.

Details of dry storage operations are provided in Section 3.5. Note that many operations have logical pre-cursors, while others could be done in a different sequence to achieve the same results. The important objective is to complete all steps; minor variations to sequence may occur due to staffing and scheduling constraints, or may result from continuous improvement initiatives.

Used fuel is discussed in detail in Section 3.2, and details of the DSC are provided in Section 3.3. The UFDS area and equipment design descriptions are provided in Section 3.4 and the

UFDS operations are described in Section 3.5. The systems specific to DSC handling are described in Section 3.6. An overview of safeguards provisions for the UFDS area at the NSS-PWMF is provided in Section 3.7. Illustrative photographs are shown in Appendix A.

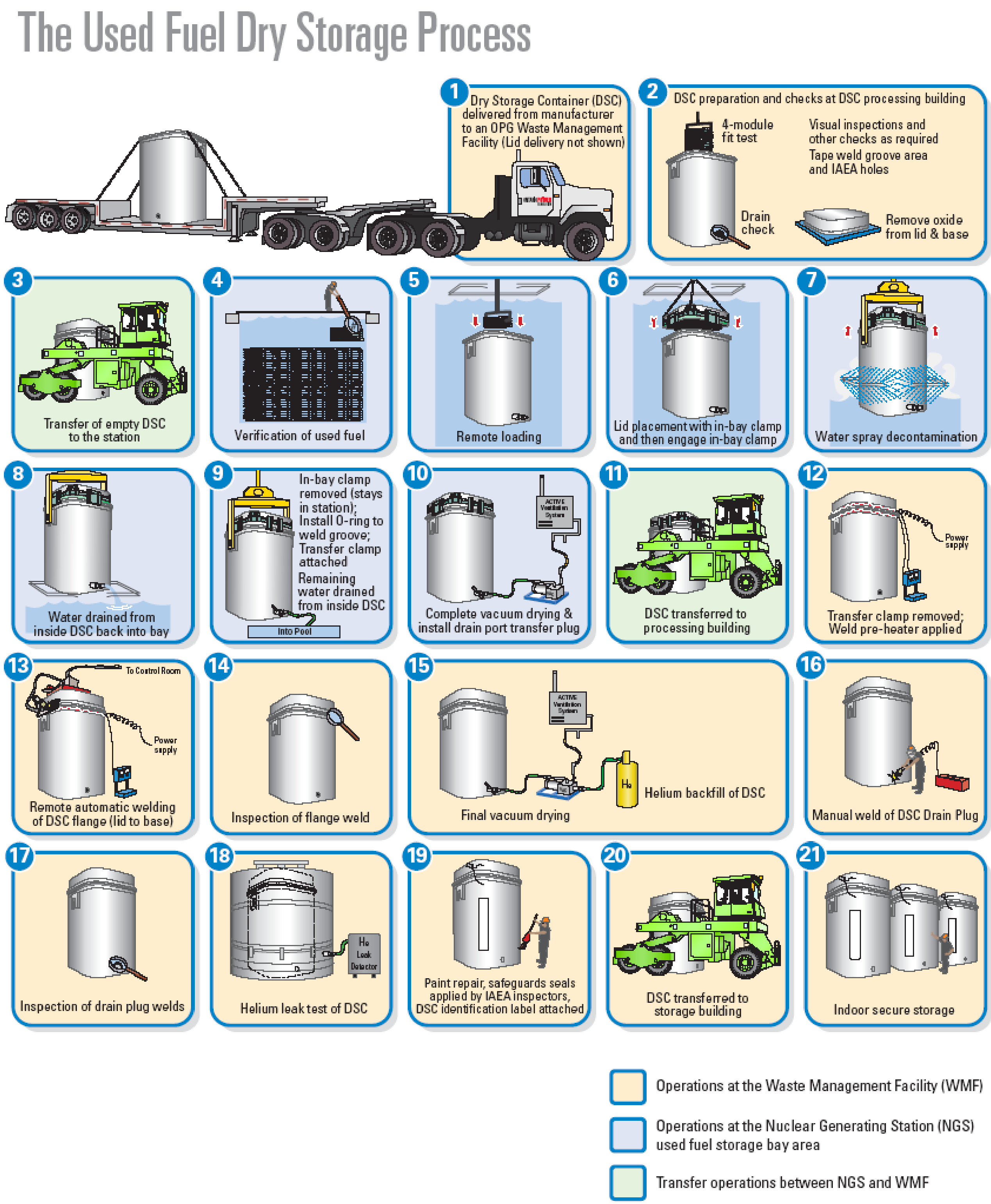

Figure 3-1: Used Fuel Dry Storage Process for Dry Storage Container MKII Design

3.2 Pickering Used Fuel Description

The fuel bundles used at the Pickering NGS reactors are 28-element Canadian Deuterium Uranium (CANDU) type fuel. Approximately 3,000 bundles are discharged annually from each of the reactors at Pickering NGS. After a minimum of 10 years of cooling[4], fuel bundles may be transferred to DSCs for interim dry storage.

3.2.1 28-Element Fuel Bundle

The fuel bundles are assemblies of 28 cylindrical fuel elements, arranged in concentric rings of 16, 8, and 4 elements (see Figure 3-2). Each fuel element contains high-density natural UO_2 pellets in a zirconium-alloy (Zircaloy-4) tube sheath[5]. CANLUB, a commercial graphite coating, is applied to the inner surface of fuel elements to minimize sheath strain during operation. The Zircaloy-4 sheath (hereafter referred to as Zircaloy) contains the alloy elements tin, iron, chromium, and sometimes nickel. In addition, beryllium is used to form a braze alloy to attach the appendages to the fuel sheath (i.e., bearing pads, inter-element spacers).

3.2.2 Reference Used Fuel Bundle

The primary factors that determine the characteristics of the used fuel are physical attributes, power and burnup histories, and decay time. These factors are in turn influenced by fuelling strategies and reactor conditions. Therefore, for the purpose of performing the safety assessment of the NSS-PWMF structures, systems, and processes, a reference used fuel bundle has been defined. Table 3-1 presents the characteristics of the Pickering reference used fuel bundle that has been used for the safety assessment (Appendix B) presented in this safety report.

Table 3-1: Pickering Waste Management Facility Reference Used Fuel Bundle Properties

Number of fuel elements	28
Length	495 mm
Mass of UO_2	22.87 kg
Mass of Zircaloy	1.67 kg
Mass of U	20.16 kg
Mass of the bundle	24.54 kg
Reference bundle power	[redacted]
Exit reference burnup	
Time after discharge	10 years

[4] A minimum of 10 years of cooling can include residence time in fuel channels during GSS followed by subsequent IFB storage
[5] The fuel sheath is also referred to as clad or cladding. These terms are interchangeable in the nuclear industry.

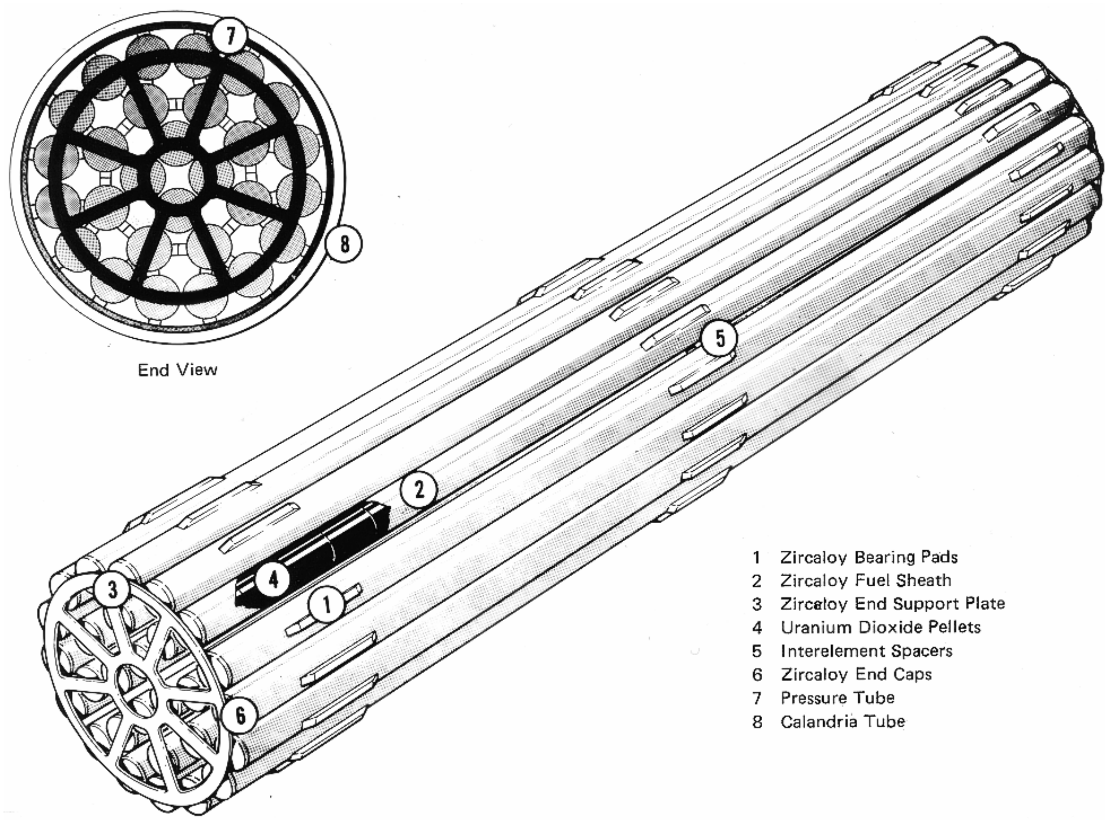

Figure 3-2: Pickering Fuel Bundle

3.2.2.1 Reference Used Fuel Bundle Dimensions

The fuel bundle used at the Pickering NGS reactors is 495 mm in length, has an outer diameter of 100 mm, and has a nominal total bundle mass of 24.6 kg. The complete dimensions of the reference fuel bundle are given in Table 3-1.

3.2.2.2 Reference Used Fuel Bundle Age

The reference fuel age for dry storage of Pickering used fuel is 10 years. In practice, however, the age of fuel loaded in DSCs will generally exceed 10 years, as operational procedures require the loading of the oldest available fuel, when feasible. Therefore, the average age of used fuel bundles in dry storage at the NSS-PWMF is greater than 10 years.

3.2.2.3 Reference Used Fuel Bundle Burnup[6]

Given statistical data of the fuel discharged from the Pickering NGS reactors (see Appendix B), [redacted] has been retained as the burnup for the Pickering reference used fuel bundle. Due to fuelling strategies and reactor operating conditions, the distribution of the fuel

6 Burnup is the fission energy generated per unit mass of heavy element initially in the fuel (unit: MWh/kgU).

bundles within the IFBs is such that the average burnup of the fuel loaded into each DSC is below the chosen reference used fuel bundle burnup.

3.2.2.4 Reference Used Fuel Bundle Power

The total core fission power for a Pickering NGS reactor is 1,744 MW (f). The average bundle power is [redacted] and the average fuel bundle residence time[7] is [redacted] Full Power Days[8].

3.2.3 Reference Used Fuel Bundle Radionuclide Inventories

After irradiated fuel is discharged from the reactor, nuclear fission essentially ceases. The fuel bundle radionuclide inventories and their associated radiation fields and decay heat decrease rapidly after discharge, due to decay of short-lived fission products and actinides with short half-lives.

At the time of discharge from the reactor core, the radionuclide inventory in a bundle is determined by the bundle power history. After 10 years of decay time following discharge from the reactor, the inventory of radionuclides in the fuel depends on the final burnup achieved and not on the power level or irradiation history of the fuel (see Appendix B for more information).

3.2.4 Reference Used Fuel Bundle Decay Heat

The energy produced by radioactive decay is released from the fuel bundle in the form of heat and radiation. The heat produced by the decay of actinides and fission products for the 10-year-old reference fuel bundle is 5.8 W. This value was retained from R006 of the Safety Report.

The maximum expected sheath temperature for the reference used fuel bundle is less than 150°C. The thermal analysis carried out for the sheath (OPG04) assumed that the DSC was loaded with 10-year-old used 37-element fuel bundles with a decay heat of 6.4 W/bundle. Therefore the thermal analysis is considered to be conservative with respect to the NSS-PWMF conditions.

3.2.5 Chemical and Physical Characteristics of Radionuclides in Reference Used Fuel

The location of radionuclide species in a fuel element depends on their chemical and physical behaviour and where they were produced. The majority of new radionuclides, such as fission products, actinides and heavy elements in 10-year-cooled used fuel, are embedded within the lattice of uranium and oxygen atoms, very close to where they were produced. Activation products that are produced in the zircaloy sheath are primarily trapped by the zirconium alloy and cannot diffuse any significant distance from the site of their formation.

As discussed in Appendix B, at maximum expected sheath temperatures for 10-year-cooled Pickering used fuel (less than 150°C in dry storage in a helium atmosphere), krypton-85,

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[7] Fuel Bundle Residence Time is the time for which a fuel bundle resides in the reactor.
[8] Full Power Day is defined as 24 hours of reactor operation at 100 percent of full power.

tritium and carbon-14 would be the nuclides released within the DSC cavity should the fuel sheath become damaged. Carbon-14 is included for completeness, however, the dose contribution is significantly lower than tritium or krypton-85.

3.2.6 Deposits on the Exterior Surfaces of Fuel Elements

Corrosion products in the Primary Heat Transport System (PHTS) are present in low concentrations in the reactor coolant and subsequently become radioactive as a result of neutron activation in the reactor core. These are removed from the PHTS coolant by purification systems. However, some of these corrosion products will deposit on the surfaces of fuel bundles.

Experimental studies (see Appendix B) have shown that activated corrosion and fission products that have been deposited on fuel surfaces while in the reactor core, and that have remained adhered under the flow of PHTS coolant and during subsequent storage in the IFB, will be fixed to the outer surfaces of fuel elements and will require either physical abrasion or chemical dissolution to be released.

3.2.7 Defective and Damaged Fuel

When a sheath fails in the reactor core at high temperatures, the free inventory of volatile radionuclides residing in the fuel-sheath gap and other open voids in the fuel is released almost instantaneously upon sheath failure. Leaching of water-soluble radionuclides from the fuel matrix occurs slowly over a longer term, while the fuel element remains submerged in the fuel bay.

Used fuel with visible or known defects affecting the integrity of the fuel bundle is not transferred to DSCs.

3.3 Dry Storage Container

3.3.1 Dry Storage Container Description

The DSC is a free-standing reinforced concrete container, with an inner carbon-steel liner and an outer carbon-steel shell, for the storage, on-site transfer and off-site transportation (with an outer packaging) of used 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 is a safety-related structure because failure of the DSC to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (OPG19a).

The DSC has the capacity to store 384 used CANDU fuel bundles in four storage modules; each module has the capacity to hold 96 fuel bundles. The DSC provides the necessary radiation shielding, heat removal path, and containment of radioactive materials.

There are currently three configurations of the DSC in service:

1. The original DSC (also called Standard DSC) design for standard fuel modules. Refer to OPG Drawings 92896-D0H-29642-0001 and 00104-DRAW-79171-10001;

2. The Long Module DSC design with an enlarged interior cavity for long fuel modules and standard fuel modules. Refer to OPG Drawing 00104-DRAW-79171-10024;

3. The current configuration, the Long Module DSC Mark II design, which was evolved from the Long Module DSC design, with the vent port removed, and a smaller drain port. Refer to OPG Drawing 00104-DRAW-79171-10051.

The standard module DSC is a double-shell rectangular container with exterior nominal dimensions of 2.121 m × 2.419 m by 3.547 m in height (excluding the flange), and an inside cavity nominal dimensions of 1.023 m × 1.321 m × 2.518 m. The nominal thickness of each carbon-steel shell is 12.7 mm. The DSC walls consist of 523.975 mm high-density 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 range of 3.5 to 3.7 Mg/m^3 and a compressive strength of at least 40 Mpa. The total mass (including the lid of 11 Mg) is approximately 60 Mg when empty and approximately 70 Mg when loaded with four modules (384 used fuel bundles).

The outer dimensions of the DSCs remain unchanged for all the three DSC configurations. The long module DSC design was modified slightly from its original standard design: the enlarged inner liner results in a thinner high-density concrete shielding thickness by 11 mm on each side of the longer sides (i.e., those containing the lifting plates). In all other respects, the original standard DSC and the long module DSC designs are identical. The design change was implemented to facilitate storage of fuel bundles that are slightly longer and are used only at the Bruce NGS and at the Darlington NGS and are stored at the NSS-WWMF and the NSS-DWMF, respectively. NSS-PWMF stores only standard length fuel bundles and has adopted the long module DSC design to permit consistency in DSC designs across all three OPG waste management facilities. In 2009, another modified DSC design (long module DSC MKII) was introduced at the NSS-PWMF. An illustration of the long module DSC MKII is provided in drawing 00104-DRAW-79171-10051. From a safety perspective there is no difference between the three DSC designs.

The DSC MKII, shown in Figure 3-3 constitutes the reference container design for the NSS-PWMF.

In the DSC MKII design the vent port has been eliminated and the size of the drain port has been reduced. The DSC MKII has been modified to take advantage of operating experience and to further reduce the radiological dose to personnel during DSC processing.

Not to be confused with the negative pressure containment system employed within the station, the "containment system" for all three DSC configurations is defined as the inner liner, the top plate of the base, the bottom plate of the lid, the lid locating pin housings, the lid-to-base seal-weld, and the drain port. The lid-to-base seal-weld is a full penetration groove weld between bottom plate of the lid and the top plate of the base. The weld is designed to secure the lid in place to provide the DSC with the required structural strength, and to complete the containment barrier. The drain port has a stainless steel shielding plug that is seal welded after the DSC is loaded with used fuel.

The DSC containment system is backfilled with helium gas. Helium is used as the inert cover gas in the DSC cavity to protect the DSC inner liner and the fuel bundles from potential oxidation reactions and to facilitate leak testing of the DSC containment boundary.

The outer shell is coated with a high performance protective coating system to facilitate decontamination of the DSC following wet-loading operations in the IFB and to provide corrosion protection of the carbon steel.

Lifting plates are designed to safely lift the DSC, with the dedicated lifting beam or the transporter vehicle.

The DSC is designed with provision for installing safeguards seals. Two separate U-shaped, 25.4 mm outer diameter stainless steel tubes are embedded in the DSC base 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 3-3. These tubes are used for attaching two different types of IAEA seals.

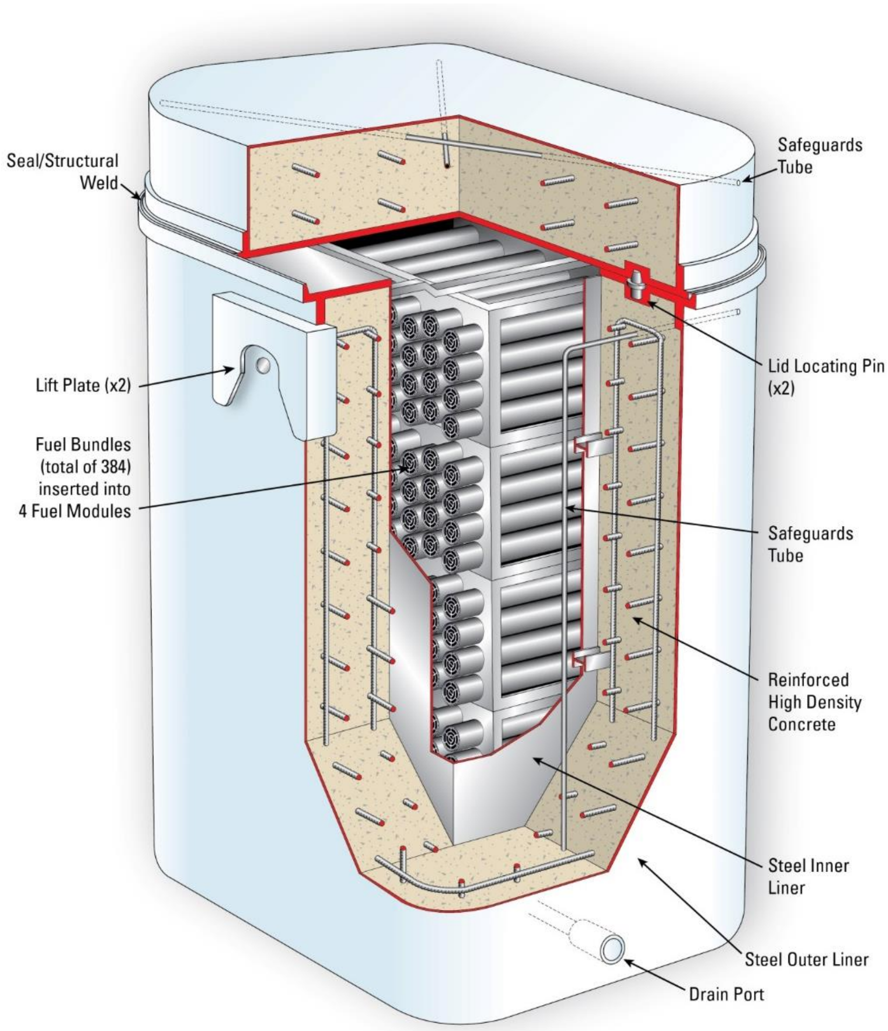

Figure 3-3: Ontario Power Generation Dry Storage Container

3.3.1.1 Used Fuel Storage Module

Used fuel bundles are placed into storage modules. [redacted] Each empty fuel module [redacted] and can hold 96 bundles, two bundles in each of the 48 [redacted] tubes. The storage module design is shown in Figure 3-4. A stack of four modules loaded with used fuel is placed inside the DSC inner cavity.

Freedom of Information and Protection of Privacy Act (FIPPA) S. 18 and Access to Information and Privacy (ATIP) S.13.

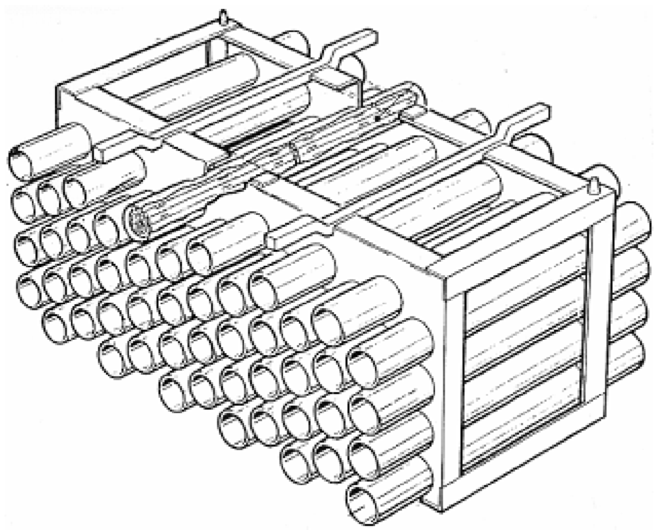

Figure 3-4: Storage Module

3.3.2 Key Dry Storage Container Parameters

Results of investigations performed on the DSC design regarding the integrity and stability of the DSC for different load cases are given below.

3.3.2.1 Decay Heat Removal

As discussed in Section 3.2.4, the thermal analysis carried out for the DSC (OPG04) assumed that the DSC was loaded with 10-year-old used fuel bundles with a decay heat of 6.4 W/bundle. The analysis demonstrated that the fuel would be adequately cooled. Ten-year cooled Pickering reference used fuel has a lower decay heat of 5.8 W/bundle, therefore the thermal analysis is considered to be conservative with respect to the NSS-PWMF conditions (see Appendix B).

3.3.2.2 Dry Storage Container Integrity under Thermal Load

The structural integrity assessment (OPG14b) for DSCs considered fuel bundles with a significantly higher decay heat of 7.4 W, which is conservative with respect to the NSS-PWMF conditions. The resulting thermal gradient in the concrete base of the DSC was estimated to be 54°C (OPG14a).

The predicted stresses generated in the concrete by the thermal gradient of 54°C indicate that through wall cracking will not occur and thermal expansion does not compromise the structural integrity of the DSC. For the Pickering reference used fuel bundle with a lower heat load of 5.8 W/bundle, the temperatures are expected to be lower. Therefore, no significant loss of either structural strength or shielding is expected to occur over the DSC design life.

3.3.2.3 Stability under Wind Load

The Design Basis Tornado (DBT) defined for the Darlington nuclear site (OPG22b) has a rotational wind speed of 322 km/h, a translational wind speed of 96 km/h, a pressure drop of 9.6 kPa, a rate of pressure drop of 5.6 kPa/s and a radius of maximum rotational wind speed of 46 m. These parameters are considered to be large enough to envelope any credible tornadoes in Southern Ontario.

Environment Canada database between 1980 – 2009 (EC13a) confirms no occurrence of F5 category tornado in Southern Ontario. The F5 category tornado has a wind speed of 420 – 510 km/h (EC13b).

Safety of the DSC against overturning was investigated for a severe wind load simulating a tornado wind speed of 425 km/h (OH92). The wind pressure on the DSC was calculated in accordance with the NBCC (CCBFC20a).

A safety factor against overturning of greater than four was found for an empty DSC and greater than five for the DSC loaded with used fuel (OH92).

3.3.2.4 Tornado-Generated Missile Impact

The design of the DSC, with respect to overturning, along with transfer clamp and seal-weld integrity, has been assessed for the impact of the following tornado generated missiles (AECL03):

Wood plank, 102 mm × 305 mm × 3.7 m, mass 91 kg, velocity 335 km/h (80 percent of total tornado velocity, i.e., rotational plus translational);

Steel pipe, 76 mm diameter, schedule 40, 3 m long, mass 35.4 kg, velocity 168 km/h (40 percent of total tornado velocity);

Steel rod, 25 mm diameter × 914 mm long, mass 3.6 kg, velocity 251 km/h (60 percent of total tornado velocity);

Steel pipe, 152 mm diameter, schedule 40, 4.6 m long, mass 129 kg, velocity 168 km/h (40 percent of total tornado velocity);

Steel pipe, 305 mm diameter, schedule 40, 4.6 m long, mass 337 kg, velocity 168 km/h (40 percent of total tornado velocity);

Utility pole, 343-mm diameter, 10.7 m long, mass 676 kg, velocity 168 km/h (40 percent of total tornado velocity); and

Automobile, frontal area 1.9 m^2, weight 1,800 kg, velocity 84 km/h (20 percent of total tornado velocity.

Analysis shows that the transfer clamp will keep the lid in place and, if welded, the DSC lid seal-weld integrity will not be impaired, DSC containment would not be breached and the DSC will not overturn, under the impact of tornado winds or the above postulated missiles (AECL03). Based on these results, it can be concluded that neither tornadoes nor missiles generated by them can cause significant damage to the containers that might affect used fuel during storage or transfer.

3.3.2.5 Effect of Creep and Shrinkage of Concrete

The maximum creep in DSC height at the end of a 50-year period was evaluated to be about 3.22 mm. The maximum creep in wall thickness was calculated to be about 0.67 mm and the maximum expected shrinkage of concrete to be 1.57 mm at the end of 50 years (OH92). These deformations are small enough to ensure that the structural integrity of the DSC will not be compromised.

3.3.2.6 Seismic Load

The DBE is defined as the earthquake which has an estimated probability of occurrence of not more than 0.001 events per year for the particular location. Using the seismic ground response spectra of the Pickering NGS B site, the applicable accelerations corresponding to a postulated DBE are evaluated as 6.25 percent of gravity (*g*) in the horizontal direction and 5 percent *g* in the vertical direction. The horizontal ground acceleration has been corrected for the container vibration frequency. The DSC has a safety factor of five against overturning and approximately two against sliding under these loads (OPG03c). Overturning of the DSC due to an earthquake is therefore considered incredible. Sliding of an unwelded DSC lid of overturning of the transporter would also not occur under the above loads (OPG03c). Hence the integrity of the DSC is not affected by the seismic loads.

Ground acceleration cannot be directly related to a Richter scale magnitude value since factors such as distance from the earthquake epicentre and site geology must be taken into consideration on a case-by-case basis. However, magnitudes up to six on the Richter scale, postulated to occur on the 50 to 80 km from Pickering NGS, have been considered (OPG00b).

3.3.2.7 Impact Load

As designed, the DSC can withstand an impact load of 45,608 kN, which is equivalent to about 65 *g* deceleration. This has been confirmed by the quarter-scale model drop tests in which the model survived intact under a 250-300 *g* deceleration (SMITH91; BOAG93), equivalent to about 62-75 *g* in full scale. In the half scale tests the model survived 230-310 *g* deceleration, equivalent to 115-155 *g* deceleration in full scale (BOAG93).

Within the NSS-PWMF, the normal lift height for a loaded unwelded DSC with transfer clamp installed is approximately 0.15 m. A seal welded DSC is raised about 1.3 m when lifted into the leak detection bell jar, or onto a raised platform for drain plug welding. Provisions have been made to limit the maximum height to which the DSC can be lifted during DSC transfer, processing and storage operations. It has been estimated that if dropped from 2.4 m on to a concrete floor, the deceleration would be equivalent to 40 *g*, well below the design basis of 65 *g* (SMITH91).

3.3.2.8 Immersion Load

The seal-welded DSC is designed to withstand a hydrostatic load of 1,965 kPa (that which would be imparted by 200 m depth of water) for more than one hour without rupture of the DSC containment system.

3.3.2.9 Internal Pressure

The DSC can withstand ±100 kPa(g) internal pressure.

3.4 Pickering Waste Management Facility Description

3.4.1 General

The NSS-PWMF Phase I site consists of two stages. The NSS-PWMF Phase I Stage I contains a DSC processing building, which also includes workshop, offices, utilities, and DSC Storage Building 1 to accommodate up to 185 DSCs (71,040 bundles); the DSC processing building and office area are attached to DSC Storage Building 1. The NSS-PWMF Phase I Stage II contains DSC Storage Building 2 to accommodate up to 469 loaded DSCs (180,096 bundles). DSC Storage Building 2 includes an area for the receiving of new, empty DSCs.

The NSS-PWMF Phase II site consists of two DSC storage buildings, referred to as DSC Storage Buildings 3 and 4. Storage Building 3 became operational in 2009 and can accommodate up to 480 DSCs. Storage Building 4 became operational in 2021 and can accommodate 624 DSCs. The Phase II site has allocated space for future DSC storage. Construction of additional DSC storage space at the NSS-PWMF will be staged, as additional storage space is required and as authorized by the current licence.

3.4.2 Layout of the Pickering Waste Management Facility Buildings

3.4.2.1 Pickering Waste Management Facility Phase I Site

The NSS-PWMF Phase I site consists of industrial type buildings that are designed and constructed to provide for the safe processing and storage of DSCs. Storage Building 2 shares the north wall of the Storage Building 1 to form a single structure.

The DSC processing building has a floor area of about 830 m² and DSC storage buildings 1 and 2 have floor areas of about 2,070 m² and 4,775 m², respectively. The total floor area occupied by the NSS-PWMF Phase I site buildings, including the ground floor office area, is approximately 8,000 m².

The NSS-PWMF Phase I floor plans are shown in Section 0.

3.4.2.2 Pickering Waste Management Facility Phase II Site

The NSS-PWMF Phase II site consists of industrial type buildings that are designed and constructed to provide for the safe processing and storage of DSCs. Storage Building 3 is a single storey, commercial-type, pre-engineered or precast concrete structure with a concrete slab-on-grade floor. Storage Building 3 can accommodate up to 480 DSCs. The layout of the DSC Storage Building 3 showing the placement of the DSCs is provided in Section 15.0.

DSC Storage Building 4 is a single storey structure, located immediately south of the Storage Building 3 with a shared wall and built as an extension of SB3. Unlike the Storage Building 3, Storage Building 4 does not have precast concrete panels as shielding walls. Storage Building 4 can accommodate up to 624 DSCs. The layout of a DSC Storage Building 4 showing the placement of the DSCs is provided in Section 0.

A small kiosk facilitates radiological and security monitoring of personnel at the Phase II site. A small services structure is used to house the Uninterruptible Power Supply (UPS), switchgear, and control cabinets. The backup diesel generator is located outside, immediately west of the services structure.

3.4.3 Structural Description of the Pickering Waste Management Facility Buildings

3.4.3.1 Pickering Waste Management Facility Phase I Site

The floors are designed and constructed for long service with minimal maintenance. Floors in the NSS-PWMF are sloped to provide drainage to floor drains.

Building walls consist of 0.2 m thick precast concrete panels from ground level to a 3.7 m height. The walls above the concrete panels consist of metal panels. Wall louvres are installed at upper wall elevations above the DSC height. The precast concrete wall system provides effective radiation shielding, however the storage and processing buildings are not safety related structures, as failure of the buildings to perform to their design intent will not result in any doses to workers/public exceeding regulatory limits (OPG19a).

The NSS-PWMF has a built-up roof design with provisions for drainage of rainwater and melted snow. Access to the roof is provided by use of an outside, all-weather permanent stairway and a fixed ladder system. The Storage Building 1 roof has three large steps on both sides of the roof trusses to accommodate roof ventilation louvres. Storage Building 2 has roof vents. The buildings provide weather protection for the DSCs in storage.

3.4.3.2 Pickering Waste Management Facility Phase II Site

The floors of the storage building are designed for long service with minimal maintenance. Floors have been designed to slope to provide drainage to floor drains. Building walls consist of precast concrete panels from ground level to a 4.2 m height, with the walls above the concrete panels consisting of metal panels.

Storage Buildings 3 and 4 are equipped with a passive ventilation system of wall louvres and roof vents. It also has provisions for drainage of rainwater and melted snow, as well as permanent access to the roof through an outside, fixed ladder system.

The north side of Storage Building 3 does not include wall louvres. Instead, to provide the adjacent Training and Mock-up Building (TMB) with increased shielding against direct gamma radiation from the DSCs, the concrete shielding panels have been extended in height to approximately 6.5 m high. The TMB is located north of the NSS-PWMF Phase II site and can be seen in Figure 1-1. Storage Building 4, located immediately south of Storage Building 3 sharing a wall, is not built with concrete shielding panels.

The storage buildings are not safety related structures as failure of the buildings to perform to their design intent will not result in any doses to workers/public exceeding regulatory limits (OPG19a).

3.4.4 Pickering Waste Management Facility Used Fuel Dry Storage Systems and Areas

3.4.4.1 Office and Utility Area

The NSS-PWMF Phase I, Stage I office and utility area has two stories. The ground floor accommodates an electrical room, heating, ventilation, and air conditioning (HVAC) equipment room, tool room, washrooms, a coffee shop, an inventory office, and office space for IAEA staff. Office space, washrooms, and a viewing gallery permitting a view of both the DSC Storage Building 1 and DSC processing building are provided on the second floor. The lobby has an elevator to provide barrier-free access to the second floor.

3.4.4.2 Receiving and Preparation Areas for New (Empty) Dry Storage Containers

New DSCs are received and inspected in the DSC processing building. Additionally, a receiving area in the DSC Storage Building 2 provides added flexibility to receive new, empty DSCs from the manufacturer. Preparation of new DSCs includes visual inspection for physical defects and component fit.

The receiving and preparation area has a 77.1 Mg (85-ton) capacity overhead crane to handle an empty DSC, lid, transfer clamp, and the DSC lifting beam.

3.4.4.3 Workshop

The DSC processing building, also referred to as a workshop, houses the following dedicated systems for DSC processing:

- Lid welding and welding-related systems;
- Welding inspection system;
- Vacuum drying system;
- Helium backfilling system;
- Helium leak detection system

The preparation of new (empty) DSCs is performed in the processing area. Additionally, as discussed in Section 3.4.4.2, an area for the receipt of new DSCs has also been provided in

DSC Storage Building 2. The DSC processing building is not a safety-related structure at the NSS-PWMF since it is not credited in the safety assessment as a barrier to the release of radiation.

3.4.4.3.1 Closure Welding and Welding-Related Systems

The following systems are used in closure welding of the DSC lid to the base:

- Weld preheat system;
- Seal welding system; and
- Weld cover-gas system.

Upper and lower pre-heaters are provided to preheat the flange and weld area of the DSC prior to and during the seal welding operation.

The seal welding system is designed to seal weld the DSC lid to the base. This weld is a full penetration, multi-pass groove weld that provides a permanent closure seal between the lid and the base.

The welding system is fitted with two Gas Metal Arc Weld (GMAW) weld heads using an inert shield gas. Each weld head has its own remote camera system and separate monitoring and control console. The weld head assembly is positioned by the overhead crane to rest on the DSC lid during the weld cycle.

The system is remotely operated from the control room. Leading and trailing closed-circuit television cameras provide views of the weld puddle to the operator during the welding process. Data monitors and warning systems are provided to monitor the essential welding parameters and report deviations outside the welding procedure tolerances.

The weld cover gas supply system supplies compressed shielding gas to the seal welding equipment. The system also provides a curtain screen of inert gas that protects the camera assemblies from damage by welding slag. Weld cover gas bottles are kept in the gas bottle storage room. This room is located in the northwest corner of the workshop and is only accessible from the outside.

Seal-welding of the DSC drain port, and weld repairs as necessary for lid weld, are accomplished using a gas tungsten arc welding (GTAW) process.

3.4.4.3.2 Weld Inspection System

A PAUT system is used for the inspection of the DSC lid-to-base seal-weld. The PAUT inspection system includes a scanner, two phased array probes, an equipment cabinet with the ultrasonic electronic motor drive control unit, and acquisition computer, and a storage cabinet with the couplant supply pump, calibration blocks stand, work bench and storage for tools. The weight of the PAUT scanner is approximately 14 kg.

All analysis variables (for example, position, detection, size) can be input into a file that is then utilized by the analysis routine. After completion of the inspection, the scanner system will be disengaged from the DSC and mounted into the storage cabinet.

A minimum clearance of 0.6 m is available to adjacent structures/equipment over the full circumference for application of the inspection scanner. All inspections are completed indoors in an environmentally controlled space at room temperature.

3.4.4.3.3 Vacuum Drying System

The DSC vacuum drying system evacuates and dries the DSC internal cavity through the drain port, after lid-to-base seal welding and weld inspection are complete.

The vacuum drying system filters particulate contamination that might be drawn from the DSC, prior to entering the vacuum pump. A dedicated hose is used in connection with the vacuum system to prevent the spread of contamination to other systems. The vacuum pump is connected to the active ventilation system.

3.4.4.3.4 Helium Backfilling System

The DSC cavity is backfilled with helium gas, after final vacuum drying and before seal welding the drain port. The helium gas facilitates leak detection for the seal welded DSC and creates an inert atmosphere for the stored used fuel.

Helium is piped from a bulk bottled supply, stored in the gas bottle storage room located in the northwest corner of the workshop. Helium is delivered to the DSC through the DSC vacuum system, using the hose dedicated for vacuum and helium backfilling operations.

3.4.4.3.5 Helium Leak Detection System

The helium leak detection system is designed to leak test the final welds on all DSC seal-welds, including the lid and drain port welds.

The leak detection system consists of the helium leak detection equipment cart and a vacuum chamber large enough to hold a DSC. The seal welded DSC is placed in the vacuum chamber after raising the container about 1.5 m and lifting it over the side of the vacuum chamber base. The vacuum chamber lid is then lifted over the DSC using the overhead crane.

The vacuum chamber is evacuated by the DSC vacuum drying system and the leak detector vacuum pump, using the hose dedicated for helium leak testing operations.

3.4.4.4 Paint Bay

A paint bay is provided in DSC Storage Building 1 for painting the DSC weld area after the container has been welded and inspected. Weld affected areas are cleaned and painted, and touch-up paint applied to scrapes or scuffs on the DSC that may have resulted from handling.

The paint bay consists of an area with walls, platforms and two electric roll-up doors. During the painting operations, the workshop ventilation and exhaust system maintains air quality in the paint bay. Painting can also be performed in the IAEA surveillance area in SB1.

3.4.4.5 Dry Storage Container Storage Buildings

The DSC storage buildings at the NSS-PWMF are provided to facilitate all-weather operation. The DSC storage buildings are not safety-related structures at the NSS-PWMF since they are not credited in the safety assessment as a barrier to the release of radiation.

DSCs are stored in a pattern that allows retrieval of any DSC, if needed. The layout of the storage areas permits placement of DSCs using a transporter to achieve the desired storage capacity.

A designated IAEA surveillance area is provided inside the DSC Storage Building 1 to temporarily store DSCs prior to be IAEA-sealed. DSC placement is discussed in Section 3.5.3.5.

3.4.5 Pickering Waste Management Facility Building Services

The building services provided for UFDS at the NSS-PWMF include the following:

(a) Ventilation;

(b) Drainage;

(c) Fire protection and detection;

(d) Electrical services;

(e) Instrument and service air;

(f) Heating and air conditioning;

(g) Overhead cranes;

(h) Security;

(i) Safeguards; and

(j) Radiological monitoring.

Items (a) to (f) are discussed in this section. The overhead cranes used for UFDS are discussed in Section 3.6.2. Site and facility security provisions have been described in Section 2.11. Safeguards requirements are discussed in Section 3.7, and radiological monitoring is further detailed in Chapter 7.

The building services also include domestic water, sewage, lighting, public address (PA) and telephone systems.

3.4.5.1 Ventilation

The DSC processing building is provided with active ventilation consisting of exhaust fans, radioactive filter assemblies and a discharge stack. This system provides active ventilation hook-up to DSCs for processing operations including vacuum drying. Airborne particulate contamination, if present, would be effectively removed by High Efficiency Particulate Air (HEPA) filters in the active ventilation system.

Localized ventilation exhaust, tied into the active ventilation system, is also provided for the welding station in the workshop. The ventilation exhaust from the DSC processing building also serves the paint bay of DSC Storage Building 1. Pre-filters installed in the paint exhaust hood collect paint aerosols, if present. Make-up air for the paint bay is drawn from DSC Storage Building 1 through a passive air passage in the paint bay wall.

The DSC Storage Buildings use passive ventilation through wall and roof louvres to dissipate decay heat from used fuel in storage to the atmosphere. The wall and roof louvres have been covered to prevent the ingress of snow and rain. The louvres can be uncovered if necessary to reduce ambient temperatures inside the facility.

Further, the louvres for Storage Building 3 are designed to prevent the ingress of rain, snow, and sand. Screens reduce the likelihood of small animals or birds entering buildings through the ventilation system. The roof vents are designed to minimize the retention and build-up of water, snow, or ice.

3.4.5.2 Drainage

Floor drains in DSC Storage Buildings 1 and 2 are connected to the active drainage system. However, these storage buildings are not expected to be contaminated and liquid effluents are not expected to be generated under normal conditions during DSC storage.

Drainage is directed to two underground active liquid sumps and transferred via sump pumps to two holding tanks, each with a 4 m^3 working volume, located in the workshop area.

After monitoring, the contents of the holding tanks are periodically transferred for routine treatment by pumping their contents, via underground piping, to the existing active liquid waste treatment system at Pickering NGS.

There are no provisions for active drainage within DSC Storage Buildings 3 and 4. Inactive drainage is provided in the floor for Storage Buildings 3 and 4. The inactive drain system within DSC Storage Building 3 and 4 is routed to a combined collection sump, where it can be monitored, if desired, and then released for discharge into the existing Pickering NGS site sewer system.

3.4.5.3 Fire Protection and Detection

The fire protection and detection systems at the NSS-PWMF are designed in accordance with CSA N393 (CSA22), the NBCC (CCBFC20a) and the National Fire Code of Canada (NFCC, see Reference CCBFC20b), and all applicable codes and standards referenced therein.

The fire protection and detection system provides fire detection at the NSS-PWMF, fire water supply to the fire hose cabinets in the NSS-PWMF Phase I facility, and hydrants servicing the NSS-PWMF Phase I and Phase II sites.

Fire protection provisions including material usage meet the OPG Fire Protection Requirements (OPG22i). The Storage Buildings are appropriately grounded to protect against lightning.

3.4.5.3.1 Pickering Waste Management Facility Phase I

Fire water for the office, processing, and storage areas is supplied by a ring header located in Pickering NGS south yard. Fire hose cabinets are located at each access door in the workshop and storage buildings. The office area has two fire hose cabinets: one on each level.

The fire protection system in the unheated storage buildings is a dry leg system. When initiated manually, fire system piping is charged with water from the fire protection system in the workshop and office area. The dry leg system is vented to the atmosphere unless filled with water.

Fire detectors are connected to the NSS-PWMF fire alarm panel. A manual alarm hand pull station is provided at each exit and connected to the fire alarm panel. Fire alarms associated with the NSS-PWMF Phase I site are displayed in the Pickering NGS main control room for dispatch of the Pickering NGS ERT. Upon arrival of the ERT, local panel alarms indicate the required response area.

3.4.5.3.2 Pickering Waste Management Facility Phase II

Fire protection for the NSS-PWMF Phase II site is provided by a ring of fire hydrants located outside Storage Buildings 3 and 4. Fire detectors for Storage Buildings 3 and 4 are connected to an addressable local fire alarm panel, which in turn provides trouble and alarm signals to an annunciating alarm panel located in the kiosk. A manual alarm hand pull station is provided at each exit and connected to the fire alarm panel.

Fire alarms associated with the NSS-PWMF Phase II site are relayed to the Auxiliary Security Building (ASB) for notifying the City of Pickering Fire Services. The initial responsibility for extinguishing fires in the NSS-PWMF Phase II rests with the Pickering Fire Services.

3.4.5.4 Electrical Services

The NSS-PWMF Phase I site at the NSS-PWMF is provided with three power supplies:

(k) Class IV Power

Class IV power is for general building loads, and electrical equipment within the DSC processing and storage buildings.

The Class IV power system supplies normal loads associated with the DSC processing building, such as office and shop area heating and ventilation, domestic and demineralized hot water systems, sewage pumps, active drainage pumps, overhead cranes, welding equipment and the air compressor. These loads can tolerate long-term power interruptions without impairment or have no safety implication following a power failure. The system receives its power from Pickering NGS Unit 8 via a 4.16 kV/600 V transformer.

(l) Class II Power

Class II power is for emergency lighting, IAEA Safeguards cameras and camera lighting, fire protection panel and alarms, and telephone and PA systems. The NSS-PWMF is provided with Class II power from the Unit 8 powerhouse. A transformer is used to provide the main source of Class II power to the emergency lighting. Also, a stepped down feed is used as a provision for IAEA Safeguards equipment.

(m) Class I Power

Class I power is provided from Unit 8 switchgear, and is used only to power the critical switchgear loads such as the protective and control circuits.

More essential loads are supplied by more reliable sources of power, to ensure continued electrical service during abnormal conditions.

The NSS-PWMF Phase II site is provided with 600V, 120V Class II, III, and IV power distribution systems.

(a) Class IV Power

Class IV power supply is taken from a 4.16kV/600V transformer to service the storage buildings, including the public address system.

(b) Class III Power

Class III power is supplied from a stand alone standby generator to meet the power requirements of the security systems, the exterior building lighting, the kiosk, the fire detection systems and the overhead door.

(c) Class II Power

Class II power is provided for emergency lighting and security monitoring equipment circuits. A suitably sized local battery powered UPS is included within the facility for Class II backup power supply requirements.

3.4.5.5 Instrument and Service Air

The instrument air system provides compressed air at sufficient pressure and flow capacity for the operation of pneumatically operated instrumentation and controls in the DSC processing building. The instrument air is supplied from Pickering NGS B Unit 8. Should normal supply of instrument air from Unit 8 fail, air is supplied to the instrument air receiver from the NSS-PWMF service air system. A service air system provides dry air at adequate pressure and flow for pneumatic tooling and general maintenance activities.

3.4.5.6 Heating and Air Conditioning

Office areas and the workshop are heated. The office area is additionally provided with air conditioning. Heating and air conditioning are not provided in DSC storage buildings.

The entrance kiosk to the NSS-PWMF Phase II site is provided with heating and air conditioning. The electrical and LAN rooms on the NSS-PWMF Phase II site are heated and the LAN room is additionally provided with air conditioning.

3.5 Used Fuel Dry Storage Operations

New DSCs are received from the manufacturer and are inspected and checked for component fit by the NSS-PWMF before being sent to the stations for loading.

At the stations, each DSC is wet-loaded with four used fuel storage modules in the fuel bay, decontaminated, drained and vacuum dried, and the transfer clamp and seal are installed to secure and seal the lid during on-site transfer. The loaded DSC is then transported to the DSC processing building using a special-purpose vehicle as described in Section 3.6.1.

At the DSC processing building workshop, the DSC is received, the transfer clamp and seal are removed, and the lid is seal-welded to the DSC body. The lid weld is subsequently inspected for defects. The DSC undergoes final vacuum drying and helium backfilling. The drain port is then welded and the weld is inspected, followed by helium leak testing. Finally, touch-up paint is applied to welded areas and scuffs or scrapes on the DSC exterior and the DSC is placed in a DSC storage building. IAEA Containment/Surveillance (C/S) of the DSC is maintained during the entire operation. Each out-of-station operation is described in the following text.

3.5.1 Preparation of New (Empty) Dry Storage Containers

Preparation of each new DSC consists of the following operations:

(a) Receiving new DSCs delivered by the manufacturer;

(b) Inspecting the DSCs for physical defects;

(c) Checking for component fit (e.g., lid fit, and lifting beam fit);

(d) The DSC drain tool and drain path are checked to ensure they are free of foreign material;

(e) Temporarily holding the DSC (with the lid) in a storage building until ready to be delivered to the station IFB for loading; and

(f) Weld groove inspection and grinding to remove excess rust and to prepare the DSC for processing. This is normally done within a day or two of transfer.

3.5.2 On-Site Transfer

After the DSC is loaded with used fuel at the station IFB and prepared for on-site transfer, a DSC transporter is used to pick up the DSC for on-site transfer. Labels are attached to the DSC identifying its contents, date of loading, and gamma dose rates.

Before the transporter exits from the IFB, both the vehicle and the DSC are monitored for contamination and if needed, they are decontaminated prior to being released into the unzoned area. The vehicle carrying the DSC travels from the IFB to the DSC processing building along the designated transfer route in accordance with security and safeguards requirements for on-site transportation.

3.5.3 Dry Storage Container Handling in the Processing Building

After the loaded DSC is received at the DSC processing building, it is prepared for storage as described below. After completion of lid weld inspection, partially processed DSCs may be transferred inside the DSC processing building and temporarily stored for up to one year from time of loading

3.5.3.1 Receiving a Loaded Dry Storage Container

The DSC transporter places the DSC on the floor in the DSC processing building. The DSC is lifted from the floor using the overhead crane and lifting beam and moved into one of the welding stations. A loaded DSC is not left unattended unless a transfer clamp is installed and engaged. The clamp must be in place for all DSC movements with fuel on board, unless the lid seal-weld has been applied. This includes craning a recently transferred DSC to a weld bay in preparation for processing.

3.5.3.2 Dry Storage Container Lid Seal-Welding

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 installed. The pre-heater is used to heat the DSC weld flange to a prescribed temperature.

The welding equipment is installed on the container and the DSC lid welded to the base by a full-penetration groove weld. This weld is deposited by a mechanized welding system using a

GMAW process. At the conclusion of lid welding, the weld machine is removed and the DSC allowed to cool.

3.5.3.3 Lid Weld Inspection

The PAUT system is used for the inspection of the DSC lid-to-base seal-weld.

PAUT can be performed in any location in the facility that has suitable AC power and a sufficient platform to allow access to the 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 percent of the weld as well as the heat affected zone (HAZ). The system drives two probes to scan the welds with a single pass, with one probe vertically placed on the vertical side and the other horizontally at the bottom of the DSC Base top flange. The bottom probe's position can be adjusted in the non-motorized axis of the scanner so that it is properly positioned to cover the weld and HAZ area. Inspection results are automatically saved electronically by the data acquisition system in the circumferential direction. All analysis variables are able to be input into a file that is then utilized by the analysis routine. After completion of the inspection, the scanner system is disengaged from the DSC and mounted into the storage cabinet.

3.5.3.2 Final Vacuum Drying, Helium Backfill, and Drain Port Seal-Welding

After successful completion of the weld inspection, the DSC is lifted into position for final vacuum drying and helium backfilling. The lifting beam is removed and the vacuum drying/helium backfilling system connected.

The interior of the container undergoes a final vacuum drying through the DSC drain port using the vacuum pump. Although the container is vacuum dried at the station before transfer to the DSC processing building, this final step removes residual moisture including moisture in the air that may have entered the DSC during processing. Pump discharge is directed to active ventilation. After final vacuum drying, helium is backfilled into the container to a pressure slightly below atmospheric pressure.

The drain port shield plug and tapered pin are inserted to permit the interspace behind the plug to fill with helium. The shield plug and tapered pin are welded and the welds are checked using dye penetrant inspection.

3.5.3.3 Helium Leak Testing

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 vacuum chamber. The vacuum chamber lid is craned over the DSC and sealed onto the base of the vacuum chamber. Using the helium leak detection system, air is first removed from the bell jar and then the helium leak detection system is activated.

This test simultaneously verifies the leak tightness of all DSC welds, including the lid closure weld and the drain seal-weld.

3.5.3.4 Decontamination, Paint Touch-Up and Safeguards Seals

Exterior DSC surfaces are checked for loose surface contamination at the time of receipt and decontaminated if needed.

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.

Documentation and identification labelling are completed and permanent safeguards seals are installed by IAEA in a designated IAEA surveillance area.

3.5.3.5 Dry Storage Container Placement and Storage

The DSC is moved, using a DSC transporter, to a designated storage location at a DSC storage building for storage. In the DSC storage building, the transporter is positioned to unload the DSC and place it on a designated storage location.

If the designated storage location is at the NSS-PWMF Phase II site, the DSC transporter will transfer the DSC between the NSS-PWMF Phase I and Phase II sites following the designated transfer route.

The container arrangement allows a minimum of 0.6 m spacing between the wider faces of the DSCs and a minimum of 0.2 m spacing between the narrower faces. This spacing between DSC rows is necessary to provide sufficient space for air circulation and cooling and to permit safe access to each DSC for periodic inspection by IAEA safeguards inspectors, radiation monitoring personnel, and maintenance personnel.

3.5.4 Dry Storage Container Records

The arrival and preparation of each new DSC, along with relevant operations carried out at the IFB and the NSS-PWMF, are recorded by NSS-PWMF Operations.

At the station IFB, a record is made of the identification number of the fuel modules loaded, the age of fuel loaded into the labelled DSC, and the time of loading. A record is also made of the time the vehicle leaves the bay.

3.6 Dry Storage Container Handling Systems Description

3.6.1 Dry Storage Container Transporters

The DSC transporters are specially designed multi-wheeled vehicles for the transfer of DSCs between the station IFBs and the DSC processing building, between the NSS-PWMF Phase I and Phase II sites, and for placement and retrieval of the seal welded DSCs inside the DSC storage buildings. There are two DSC transporter designs in service called the LiftKing and

MacLean (or Gen4). The transporters are also used to place DSCs into the paint bay. Appendix A provides a photograph of a transporter carrying a DSCs.

The transporters are powered by a diesel engine, and are self-loading; they do not require the assistance of a crane when picking up or depositing a DSC. The DSC is lifted and transported via lifting trunnions mounted on the upper frame of the machines. The DSC is carried at a low lift height during transfer (about 0.2 m). Locking arrangements prevent the DSC from being inadvertently lowered to the ground upon hydraulic failure. The tires on the transporters are designed not to deflate if punctured.

When travelling with a DSC, the transporters operate at low speed and have a short stopping distance (within 3 m). When travelling at minimal speeds (e.g., when moving DSCs within the DSC processing and storage buildings), stopping is essentially instantaneous.

The transporters are capable of forward and reverse motion and have a tight turning radius. Vehicle lighting is provided for operation on site roads, if necessary.

3.6.2 Overhead Cranes

The NSS-PWMF DSC processing building is equipped with a 77.1 Mg (85-ton) overhead crane that is fitted with a 9.1 Mg (10-ton) auxiliary crane.

3.6.3 Dry Storage Container Lifting System

The DSC lifting system consists of lifting plates on the DSC and a lifting beam with trunnions. The lifting beam has been designed for DSC handling and is compatible with the swivel hooks on the processing building overhead cranes and on the crane in the IFB. The lifting beam is designed to engage into the lifting plates attached on the DSC body and not to disengage from the DSC while the beam is under load.

A lifting beam is provided for use in the DSC processing building and in the IFBs. When not in use, the lifting beam is stored on a custom built frame.

3.6.4 Transfer Clamp

The transfer clamp is designed to prevent the lid from separating under credible accident scenarios during transfer of loaded DSCs between the stations and the DSC processing building, and during DSC handling inside the processing building prior to seal welding the DSC lid.

At the station IFB, once the in-bay clamp has been removed, the elastomeric seal is installed in the weld groove and confirmed seated before the transfer clamp is installed. The transfer clamp is used to securely attach the lid to the DSC base during on-site transfer of a loaded DSC between the stations and the NSS-PWMF Phase I site. The transfer clamp is used in conjunction with a seal between the lid and base, to permit the cavity of a loaded DSC to be vacuum dried.

The transfer clamp is a safety related structure because the failure of the transfer clamp to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (OPG19a).

3.7 Safeguards

In accordance with IAEA requirements, the Integrated Safeguards approach and the Additional Protocol to the Treaty on the Non-proliferation of Nuclear Weapons (NPT) have been implemented. These include the following elements:

- Provision of information, including:
    - Advanced Information of upcoming planned activities:
        - Operational Program – annual, quarterly and interim updates;
        - Weekly notifications of scheduled DSC loadings and planned transfers in the following week;
        - Schedule changes to DSC transfers; and
        - Force majeure notifications (e.g. malfunctions of a facility computer systems).
    - Declarations of activities performed, including:
        - Monthly integrated safeguards declarations;
        - Annual physical inventory taking;
        - Annual additional protocol updates;
        - Monthly nuclear material accounting and operational records (e.g. general ledgers);
        - Inventory summary of fissionable and fertile materials which have foreign obligations;
        - Inventory Change Documents (ICDs), and;
        - Operational records upon request.
- Inspections including:
    - Unannounced inspections of random DSC transfer-related activities at the discretion of IAEA;

    - Annual Physical Inventory Verification (PIV);
    - Design Information Verifications (DIVs);
    - Annual self-assessment as per N-PROC-RA-0136 section 1.9 (OPG21c);

- CNSC's Physical Inventory-Taking Evaluations (PIT-Es);
- Radiological profiling for each DSC transferred;
- Application of IAEA containment and surveillance measures, including video surveillance in the processing building and storage building, as well as applying dual containment seals (Cobra Seals and Laser Mapping Containment Verification – LMCV) on loaded, welded DSCs within the designated IAEA surveillance areas[9];
- Remote Monitoring by IAEA to access their real-time safeguards data of the facility; and
- Complementary and managed access to the facility as part of the additional protocol to the NPT.

The NSS-PWMF management stays current with the IAEA's safeguards requirements and is committed to meeting OPG's safeguards obligations (OPG21c) in an efficient and timely manner.

3.8 Pickering Used Fuel Dry Storage Operating Experience

Operating experience has demonstrated that the NSS-PWMF can be operated safely and without undue risk to workers, members of the general public, or the environment.

The NSS-PWMF has been operating since January 1996. The safety performance of the facility has been excellent during its operation over the entire period.

Doses to both the public and NSS-PWMF staff have remained below the regulatory limits. Collective occupational radiation exposures have been less than the predicted exposures by 30 percent or more. Emissions have remained below the regulatory limits. The NSS-PWMF routinely operates contamination free.

There have been no public safety events at the facility. In the past five years there have been no MECP-reportable spills. As of December 2022, 1,183 DSCs have been successfully and safely stored in the NSS-PWMF (OPG23c).

[9] In cases of LMCV temporarily not being suitable due to a reduced weld profile, metal seal should be an option.

4.0 USED FUEL DRY STORAGE SAFETY ASSESSMENT

4.1 Introduction

This chapter provides a summary of the NSS-PWMF UFDS radiological safety assessment during DSC on-site transfer, processing, and storage. Details of the UFDS safety assessment methodology, assumptions and results are given in Appendix B.

Estimated gamma radiation dose rates are presented in this section for distances from the NSS-PWMF to the Pickering NGS site boundary. Dose rate calculations consider UFDS at the NSS-PWMF to contain 1,692 DSCs. Although the site has a higher design capacity, a lower number of DSCs was assumed to account for wider spacing requirements of the new DSC transporter design as well as for physical limitations due to electrical panels and roof bracing.

Conservative estimates of public dose rates due to releases resulting from hypothetical failures of an assumed fraction of fuel elements for normal and abnormal operating conditions and credible accident conditions are also presented.

4.1.1 Safety Assessment Approach

The NSS-PWMF safety approach employs multiple barriers to ensure radionuclide emissions are kept within levels that are below regulatory limits and are ALARA. Each barrier independently provides a measure of safety toward preventing the release of radioactive materials as follows:

- The uranium dioxide (UO_2) matrix effectively contains the radionuclides present in 10-year-cooled used fuel (either under wet or dry storage conditions), except for the free fractional inventory of tritium (in vapour form), krypton-85 (in gaseous form) and carbon-14 (in gaseous form);

- The fuel sheath additionally acts as a barrier for the free fractional inventory of tritium, krypton-85 and carbon-14 that would otherwise be available for release;

- The seal-welded DSC is considered a safety related structure credited in the containment of radioactivity and provides an additional barrier against the release of tritium and krypton-85 in the event of fuel sheath failure (see Section B.2.5); and

- The reinforced concrete construction of the DSC base and lid provides a barrier against radionuclide release and also provides effective shielding for gamma radiation from used fuel.

Conditions relevant to UFDS operations are classified as normal and abnormal operating conditions, or credible accident conditions, as defined below:

(a) Normal operating conditions are routine. UFDS SSCs are expected to remain functional and to experience no unacceptable degradation.

(b) Abnormal operating conditions do not occur routinely, but can potentially occur over the lifetime of the facility. UFDS SSCs are expected to experience abnormal events and conditions without permanent degradation of functional capability (although operations are expected to be suspended or curtailed during abnormal conditions, unless the appropriate compensatory measures are taken).

Potential abnormal operating conditions may include failure of used fuel sheath or a DSC seal-weld, operator error, and equipment failure.

(c) Accident conditions are unlikely to occur over the lifetime of the facility; however, the safety implications resulting from accidents may exceed the potential consequences of abnormal operating conditions.

Accidents can result from events within a facility (e.g., equipment failure resulting in the drop of a loaded DSC), or from events that are external to the facility. External events include hazards relating to human activities (e.g., external fires and explosions or a small aircraft crash), and natural hazards (e.g., earthquakes, tornadoes, thunderstorms including lightning, and floods).

For each failed fuel element (as discussed in Appendix B), it is postulated that 100 percent of the tritium and krypton-85 present in the gap (i.e., the space between the fuel pellet and the fuel element sheath) will be released, along with 10 percent of the tritium and krypton-85 present in the uranium dioxide grain boundary. In addition, 100 percent of the carbon-14 inventory in each failed fuel element is released. Carbon-14 is included for completeness, however, the dose contribution is significantly lower than tritium or krypton-85.

4.1.2 Design and Operating Acceptance Criteria

Under normal operating conditions during storage, NSS-PWMF UFDS SSCs are expected to provide reasonable assurance that the used fuel can be stored in DSCs and the DSCs can be retrieved without undue risk to workers, members of the general public, or the environment.

NSS-PWMF operations comply with the OPG requirement to keep total radioactive emissions under normal operating conditions within regulatory limits and ALARA.

The NSS-PWMF SSCs have been designed to fulfill the following criteria under normal and abnormal operating conditions and credible accident conditions including external hazards:

(a) Comply with CNSC regulatory dose limits (occupational and public) and the ALARA principle;

(b) Maintain subcriticality;

(c) Maintain the integrity of used fuel in dry storage; and

(d) Recover safely from abnormal operating conditions and credible accident conditions, including protection criteria as outlined in CSA N393:22 (CSA22).

The safety assessment of normal and abnormal operating conditions, and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions and are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions.

4.2 Acceptance Criteria

The radiation safety requirements under normal operation for the NSS-PWMF are the following:

- ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.
- ≤ 100 μSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years.

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose limit for the public at or beyond the OPG property boundary due to an abnormal event/credible accident shall be 1 mSv.
- The dose limit for a worker due to an abnormal event/credible accident shall be 50 mSv.

4.3 Radiological Safety Assessment – Normal Operating Conditions

4.3.1 Radioactive Emissions and Contamination

Under normal operating conditions, no airborne emissions are expected from loaded DSCs during transfer from the stations 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. A dedicated active ventilation system is used to deal with any airborne emissions.

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

NSS-PWMF experience demonstrates that particulate emissions in exhaust from DSC processing operations are typically below the Minimum Detectable Activity (MDA). The dose resulting from the normal operation emissions has been assessed using the emission survey data from the 2007-2021 NSS-PWMF quarterly reports and the methodology based on the

CSA N288.1-14 standard (CSA14a). The highest annual individual dose to a member of the public during the evaluated period is calculated to be less than 10^{-3} µSv per year.

In addition to the dose from the recorded emission data, potential emissions under normal operating conditions have been evaluated (see Appendix B). Since each DSC has the capacity to hold 384 fuel bundles and assuming the facility processes about 150 containers per year, it is postulated that a total of 600 fuel elements (four elements per DSC, i.e., one fuel element in 1 percent of the fuel bundles is assumed to be damaged) fail during 1-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 boundary, are estimated to be 9.3×10^{-4} µSv/year. When combined with the dose from the recorded normal operation emissions and the potential airborne emissions from the DSMs, the upper bound estimate for dose consequence for airborne emissions is 1.88×10^{-3} µSv/year (OPG22c).

As the DSC is fully drained and vacuum dried after loading at the station IFBs, 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 NSS-PWMF.

The exterior surfaces of DSCs are decontaminated prior to their transfer from the IFBs to the NSS-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 DSC 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 NSS-PWMF operations.

Radioactive airborne emissions and liquid effluents from the NSS-PWMF are well within the restrictive administrative targets set for the facility and contribute a negligible fraction of the release limits for the Pickering site.

The dose to workers during normal operating conditions is discussed in Section 4.5.2. The processing building has an active ventilation system and welding is performed remotely, therefore worker dose from welding operations is unlikely. Furthermore, dose to workers is managed under the RP Program (N-PROG-RA-0013), which includes whole body counting to detect potential internal uptakes.

4.3.2 Radiation Fields

4.3.2.1 Dry Storage Container Dose Rates

Using the radionuclide inventory data and taking into consideration the support (and slight shielding) provided by the steel rebar used to reinforce the high-density concrete in the container, the radiation fields for a fully-loaded DSC have been calculated for 10-year-cooled Pickering used fuel. Table 4-1 shows the calculated gamma radiation dose rates for different distances from the top, side, and front surfaces of a DSC of the modified long module design, fully-loaded with 10-year-cooled Pickering reference fuel bundles as defined in Chapter 3 of this report. Figure 4-1 shows the calculated gamma radiation dose rates as a function of the distance from the top, side, and front surfaces of a DSC, fully-loaded with 10-year-cooled Pickering reference fuel.

Table 4-1: Calculated Dose Rates from a Dry Storage Container of the Modified Long Module Design, Fully-Loaded with Pickering 10-year-cooled Used Fuel Bundles

Distance from DSC	Position	Dose Rate (μSv/h)
Contact	Side	33.0
	Front[10]	37.93
	Top	27.22
1 m	Side	15.89
	Front	20.01
	Top	13.09
2 m	Side	8.10
	Front	10.57
	Top	4.06

Since the DSC is composed of high-density concrete which acts as neutron shield, the contribution of the neutron radiation to dose rates outside of a DSC is small compared to that of the gamma radiation (approximately 4% of the decay gamma dose rates on the side of a DSC and approximately 2% at 1 m from a DSC, as reported in OPG22a).

Calculated dose rate estimates have been demonstrated to be conservative compared with actual DSC dose rates measured during UFDS 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 compares with estimates of 33 to 38 µSv/h contact dose rates for 10-year-cooled fuel (at the DSC side or front) as set out in Table 4-1. At a 1 m distance, measured dose rates are about 5 to 7 µSv/h, compared with calculated dose rate estimates of 15 to 20 µSv/h.

[10] The label ‘front’ corresponds to the wider face of the DSC and ‘side’ indicates the narrower face.

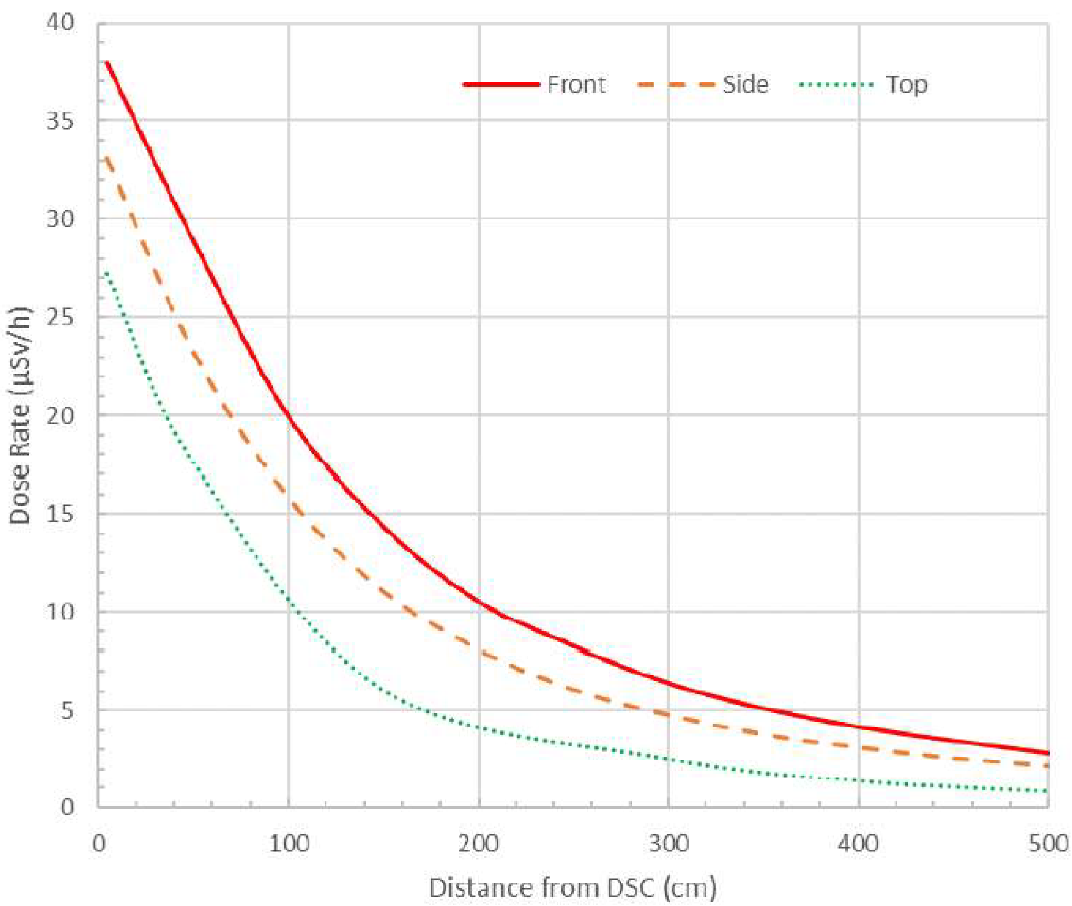

Figure 4-1: Calculated Dose Rate versus Distance from the Surface of a Single Dry Storage Container of the Modified Long Module Design, Fully-Loaded with 10-year-cooled Pickering Reference Fuel Bundles.

4.3.2.2 Dose Rates Inside the Dry Storage Container Storage Buildings

The predicted dose rates from a row of DSCs facing the corridor in the middle of Storage Building 2 loaded with 10-year cooled fuel in storage, are presented in Figure 4-2. The results show that the dose rates on the west side are approximately 30 µSv/h, this drops to 12.5 µSv/h in the middle of the corridor and increases to 28 µSv/h on the east side. Note that due to the increased number of DSCs, the dose rates presented for Storage Building 2 will bound those of Storage Building 1.

For the previous Revision 6 of this Safety Report (OPG20b), dose rates were predicted from a row of DSCs facing the corridor in the middle of Storage Building 3 loaded with an average of 26 year old fuel in storage. These results, which are now conservative since the fuel has aged since the dose calculations, are presented in Figure 4-3. The DSC configuration is symmetric on the east and west side. The results show that the dose rates on the west and east side are approximately 18 µSv/h and this drops to 5 µSv/h in the middle of the corridor.

The predicted dose rates from a row of DSCs facing the corridor in the middle of Storage Building 4 loaded with an average of 18 year old fuel in storage, are presented in Figure 4-4. The dose rates on the west and east side are approximately 34 µSv/h and these drop to 9 µSv/h in the middle of the corridor.

The reported doses shown in Figures 4-2, 4-3 and 4-4 are more impacted by the age of the DSCs directly facing the corridors, as opposed to the average age of DSCs within the separate buildings.

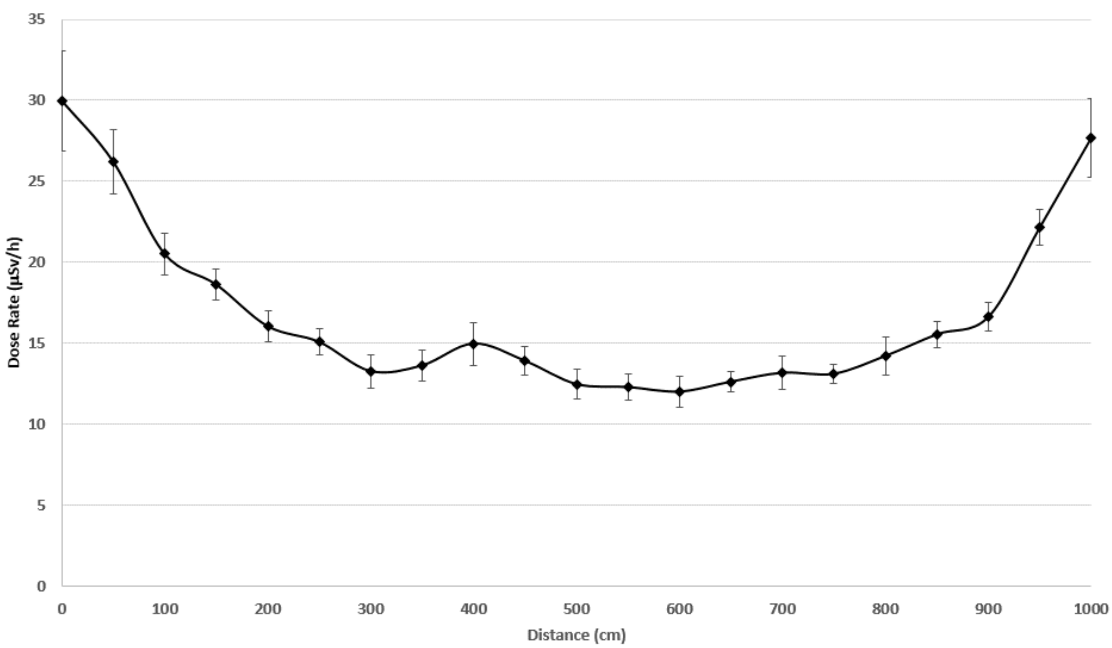

Figure 4-2: Calculated Dose Rates across the width of the North-South Corridor of Dry Storage Container Storage Building 2

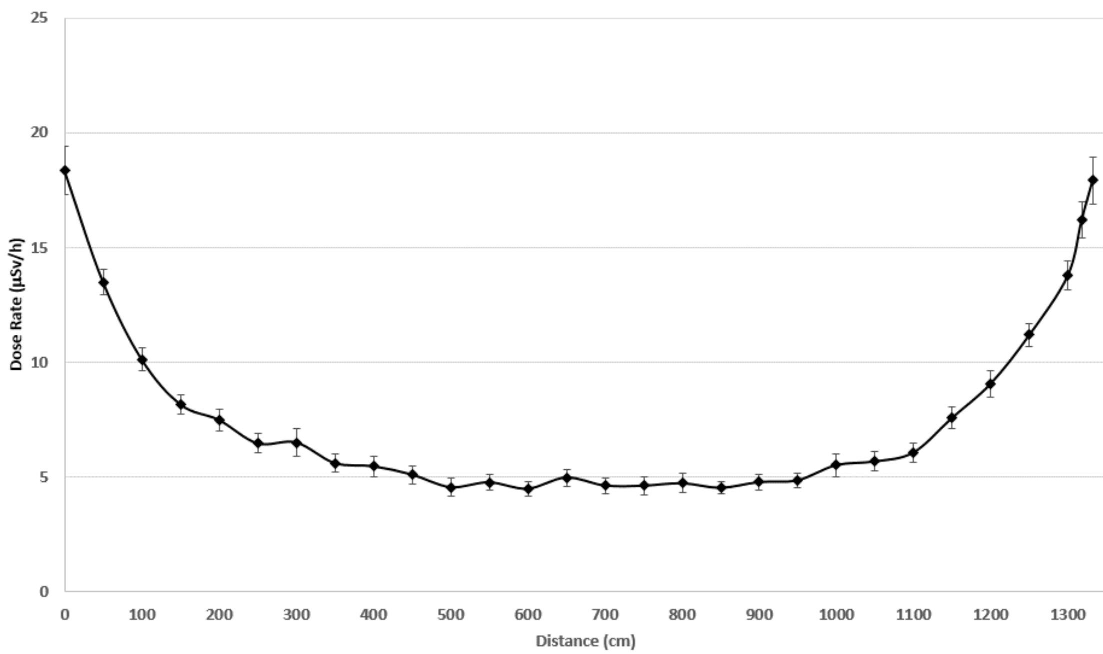

Figure 4-3: Calculated Dose Rates across the width of the North-South Corridor of Dry Storage Container Storage Building 3

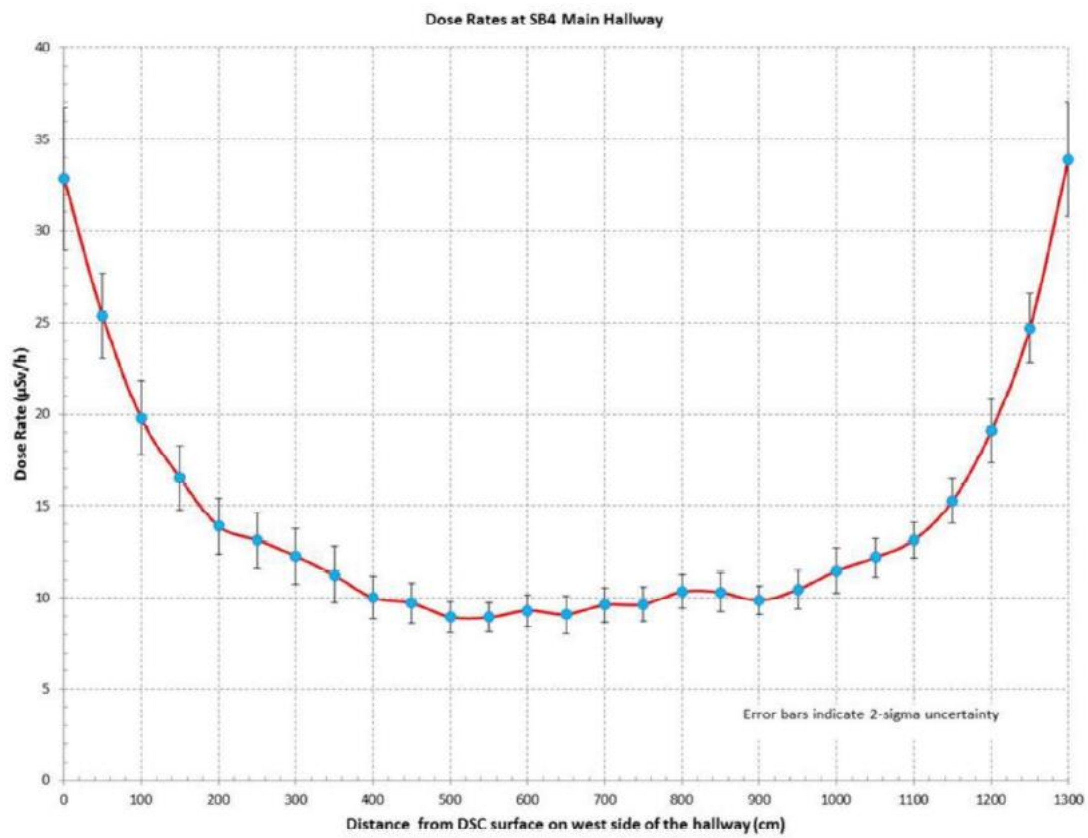

Figure 4-4: Calculated Dose Rates across the width of the North-South Corridor of Dry Storage Container Storage Building 4

4.3.2.3 Dose Rates Outside the Dry Storage Container Storage Buildings

When the NSS-PWMF DSC storage buildings 1 to 4 are filled with 1692 DSCs, the maximum dose rate at the site boundary is calculated to be 1.46×10^{-3} µSv/h.

This is equivalent to an annual dose of 2.92 µSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 1.41×10^{-3} µSv/h. This is equivalent to an annual dose of 1.41 µSv based on 1,000 hours occupancy. This is a conservative occupancy assumption for boaters and fishermen.

These results indicate that the NSS-PWMF dose target of ≤100 µSv/y at the station site boundary as set out in Section 1.6.1, is met during UFDS operations.

4.3.2.3.1 Pickering Waste Management Facility Phase I Site

Dose rates at the perimeter fence east of Phase 1 are calculated to be less than 0.24 µSv/h. The predicted dose rates are less than 50% of the dose rate target of 0.5 µSv/h.

These results indicate that the NSS-PWMF dose rate targets of ≤ 0.5 µSv/h at the station security fence on a quarterly average basis as set out in Section 1.6.1, are met during UFDS operations.

4.3.2.3.2 Pickering Waste Management Facility Phase II Site

The calculated dose rates at the north and east perimeter fences of the storage buildings 3 and 4 are within the criterion of ≤0.5 µSv/h established for limited occupancy (i.e., up to 2,000 hours per year) by non-NEW personnel at the perimeter fence.

The calculated maximum dose rates at the west and south perimeter fences of the Storage Buildings 3 and 4 are 0.69 and 0.64 µSv/h, respectively, which is above the 0.5 µSv/h criterion. These calculations are known to be conservative, and actual site measurements, which are recorded in quarterly reports and issued to the regulator, all remain below the 0.5 µSv/h criterion. Other, more appropriately placed, site fences can also be used as boundaries at which the target of 0.5 µSv/h can be applied if necessary. OPG will take every precaution to ensure that administrative dose rate targets are met. The dose consequences for personnel walking or driving on the Pickering site roads (i.e., at further distances and for short time durations) will be well within acceptance criteria.

As discussed in Section 3.4.3, 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 TMB are below the dose rate target of 0.5 µSv/h (see Section 1.6.1). The dose rate at the TMB was estimated to be 7.57×10^{-2} µSv/h.

4.4 Radiological Safety Assessment – Abnormal Incidents and Accidents

The operation of the NSS-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 within and external to the NSS-PWMF. A full description of this assessment is provided in Appendix B.

Given the very distinctive stages of the Pickering UFDS process, the assessment of malfunctions and accidents is divided into the following main stages of the out-of-station UFDS operations:

(a) On-site transfer operations;

(b) Operations inside the DSC processing building; and

(c) Storage.

For each stage of UFDS operations, release of radiation due to fuel sheath failure is assumed to occur during UFDS operations. The safety assessment for the in-station operations is part of the Pickering NGS licensing basis documentation.

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 UFDS operations at NSS-PWMF, safety analyses presume that abnormal operating conditions and 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.

4.4.1 Malfunctions and Accidents Assessment for Operations during Dry Storage Container On-Site Transfer

As described in Chapter 3, the DSC transporter is used to transfer loaded DSCs from the Pickering NGS A and B IFBs to the DSC processing building. It is also used to transfer seal-welded DSCs from the NSS-PWMF Phase I site to the NSS-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 described in Appendix B has taken into account postulated malfunctions and accidents that could potentially affect the on-site transfer of a loaded DSC.

Table 4-2 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., with a frequency of occurrence that is $\geq 10^{-6}$ events per year (CSA14b))[11] during DSC on-site transfer.

[11] The frequency of occurrence of 10^{-6} is consistent with the value given in CSA N292.0-14 for a credible abnormal event, “General Principles for the Management of Radioactive Waste and Irradiated Fuel”.

Table 4-2: Postulated Malfunctions or Accidents during Dry Storage Container On-Site Transfer

Malfunction or Accident	Potential for Occurrence (events per year)	Credible Event (Y/N) See Note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Transporter failure	3	Y	<1.60 x10^{-3}	<1.96 x10^{-3}	<4.7
Transporter Operator Health-Related Emergency	See Note 2	Y	<1.60 x10^{-3}	< 1.96 x10^{-3}	<4.7
DSC drop during on-site transfer from IFB to DSC Processing Building	See Note 2	Y	1.60x10^{-3}	1.96 x10^{-3}	4.7
DSC drop during on-site transfer between the NSS-PWMF Phase I and Phase II sites	See Note 2	Y	2.43 x10^{-3}	<2.90 x10^{-3}	4.7
Fire[12]	1.04 x 10^{-7}	N	—	—	—
Criticality	See Note 3	N	—	—	—
Adverse road conditions	See Note 2	Y	<2.43x10^{-3}	<2.90 x10^{-3}	<4.7
Earthquake	1.71 x 10^{-5}	Y	0	0	0
Tornado[12]	5.36 x 10^{-8}	N	—	—	—
Thunderstorm	See Note 2	Y	0	0	0
Flood	See Note 2	Y	0	0	0
Explosion along transfer route[12]	5.2 x 10^{-8}	N	—	—	—
Turbine missile strike[12]	1.03 x 10^{-7}	N	—	—	—
Aircraft crash[12]	6.17 x 10^{-10}	N	—	—	—
Toxic gas releases – chlorine from Ajax water treatment plant	See Note 2	Y	0	0	0
Soil failure/slope instability	See Note 2	Y	<2.43x10^{-3}	<2.90 x10^{-3}	<4.7

Notes:

1. The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.
2. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature or it is bounded by an assessed credible hazard.
3. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.

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[12] Low Potential for Occurrence due to limited period of DSC in 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. Although fuel sheath failure is not expected to result from a DSC drop from the low lift height of the transporter, analysis of the drop of a DSC during on-site transfer conservatively considered the bounding scenario involving failure of all the fuel elements inside a DSC.

Consequently, the free inventory from 10,752 failed fuel elements is assumed to be released into the environment. The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements is assumed to be released into the DSC cavity. As Phase II is closer to the site boundary the assessment conservatively assumes that the drop of an unwelded DSC occurs during transfer from the NSS-PWMF Phase I site to the Phase II site. In reality, all DSCs transferred from NSS-PWMF Phase I to Phase II are already seal welded.

Assuming that this event occurs at or near the NSS-PWMF Phase II site, the total dose to the public was calculated to be 2.43 µSv for an adult and 2.90 µSv for an infant at the Pickering site boundary. The dose to a NEW would be 4.7 mSv. All of these doses are below the acceptance criteria defined in Section 4.2.

4.4.2 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Processing

The processes and systems taken into account for this assessment (see Appendix B) encompass those at the DSC processing building once the transporter arrives at the NSS-PWMF with a loaded DSC and before the DSC is taken to storage, as described in Chapter 3.

Table 4-3 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible during DSC processing. Full details of this assessment are described in Appendix B.

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 percent of the fuel elements inside the DSC are damaged and the free inventory from 3,226 failed fuel elements is released from the NSS-PWMF into the environment.

The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements 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 and worker due to this event is bounded by the DSC drop during transfer discussed in Section 4.4.1.

Table 4-3: Postulated Malfunctions or Accidents during Dry Storage Container Processing

Malfunction or Accident	Potential for Occurrence (events per year)	Credible Event (Y/N) See Note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Drop of a DSC during handling	3.36×10^{-2}	Y	$< 1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
Equipment drop onto a DSC	1.68×10^{-2}	Y	$<1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
DSC collision during craning	1.50×10^{-1}	Y	$<1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
Transporter collision with a loaded DSC or another Transporter	6×10^{-1}	Y	$<1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
Equipment collision with a loaded DSC during craning	3×10^{-1}	Y	$<1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
Criticality	See Note 2	N	—	—	—
DSC Processing building fire	See Note 3	Y	0	0	0
Earthquake	See Note 3	Y	$<1.60\text{x}10^{-3}$	$<1.96\text{x}10^{-3}$	<4.7
Tornado	See Note 4	N	—	—	—
Thunderstorm	See Note 3	Y	0	0	0
Flood	See Note 3	Y	0	0	0
Turbine missile strike	See Note 5	N	—	—	—
Aircraft crash	6.93×10^{-7} See Note 6	N	—	—	—
Release of oxidizing, toxic, corrosive gases and liquids stored in the Processing Building	See Note 3	Y	0	0	0
Asphyxiants	See Note 3	Y	0	0	0

Notes:

1. The term incredible is used for those events with frequency of occurrence less than 10^{-6} events per year.
2. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.
3. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature or it is bounded by an assessed credible hazard.
4. Simultaneous occurrence of a tornado (3.13×10^{-6} events per year) and preparation of the DSC for seal-welding scenarios is sufficiently low to be considered an incredible scenario (frequency of occurrence less than 10^{-6} events per year).
5. Considering the location of the DSC processing building with reference to the Unit 8 turbine, a turbine missile striking the processing building and the DSC is considered an incredible event.
6. The calculated frequency of occurrence considers the Phase I, Phase II sites and DSM storage area together.

4.4.3 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Storage

Once the DSC processing is completed, the transporter moves the DSC from the DSC processing building to one of the DSC storage buildings for storage.

Table 4-4 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (that is, with a frequency of occurrence that is $\geq 10^{-6}$ events per year) during the storage process. Details of the assessment for each event are given in Appendix B.

Note that the storage buildings are not safety related structures credited in the containment of radioactive releases (OPG19a).

Table 4-4: Postulated Malfunctions or Accidents during Dry Storage Container Storage

Malfunction or Accident	Potential for Occurrence (Events per year)	Credible Event (Y/N) See note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Seal-weld failure during storage	See Note 2	N	—	—	—
DSC drop during transfer to storage	1.5×10^{-6}	Y	$<2.43 \times 10^{-3}$	$<2.90 \times 10^{-3}$	<4.7
Transporter collision with a DSC or another Transporter	1.5×10^{-1}	Y	$<2.43 \times 10^{-3}$	$<2.90 \times 10^{-3}$	<4.7
Criticality	See Note 2	N	—	—	—
DSC storage building fire	See Note 3	Y	0	0	0
Earthquake	See Note 3	Y	0	0	0
Tornado	See Note 3	Y	0	0	0
Thunderstorm	See Note 3	Y	0	0	0
Flood	See Note 3	Y	0	0	0
Toxic Material	See Note 3	Y	0	0	0
Turbine missile strike	3×10^{-7}	N	—	—	—
Aircraft crash	6.93×10^{-7} See Note 4	N	—	—	—

Notes:

1. The term incredible is used for those events with frequency of occurrence less than 10^{-6} events per year.
2. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.
3. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature or it is bounded by an assessed credible hazard.
4. The calculated frequency of occurrence considers the Phase I, Phase II sites and DSM storage area together.

4.5 Occupational Safety Assessment

4.5.1 Radiation Protection Program

The OPG RP Program (N-PROG-RA-0013) applies to all OPG nuclear facilities including the NSS-PWMF and is discussed in Section 7.2. ALARA dose targets set under the RP Program are reviewed annually for continuous improvement. The RP Program provides the framework and standards of performance for all the staff and operational activities.

The Director of Radiation Safety implements the standardized RP Program. All radioactive work is planned and executed in accordance with ALARA practices. Relevant procedures are followed to control contamination within the controlled work areas. The external and internal occupational radiation exposures are kept within the administrative limits and planned exposures for each job. The administrative and control levels established by the OPG RP program are found in N-PROC-RA-0019, Dose Limits and Exposure Control.

4.5.2 Worker Dose Assessment

In worker dose assessments for operations at NSS-PWMF through conservative bounding assumptions, it has been demonstrated that the facility can be operated well within regulatory dose limits.

Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported for NEWs during the 2007-2022 period are 12.6 person-mSv and 1.6 mSv. These values are both below all relevant safety limits. Worker doses are submitted to the CNSC as a part of the Pickering Waste Quarterly Operations Reports.

4.6 Summary of Pickering Waste Management Facility Safety Assessment

This radiological safety assessment for the UFDS process at the NSS-PWMF has addressed worker and public doses under both normal and abnormal operating conditions, and credible accident conditions.

Based on the NSS-PWMF operating experience, the environmental releases from the NSS-PWMF under normal operating conditions 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 boundary, are also well below the CNSC public dose limits.

Occupational doses arising from UFDS operations are managed under the RP Program and are expected to be well below regulatory dose limits.

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

4.7 Long-Term Integrity of Used Fuel and Dry Storage Container

4.7.1 Used Fuel Integrity

Dry storage is a passive mode of storage for used fuel and is the preferred interim option for the long-term management of used fuel at OPG. A requirement of dry storage is that the fuel will not be adversely impacted by the storage conditions to ensure that the fuel remains structurally sound, retrievable and can be safely handled during subsequent steps in its management.

The CANDU fuel bundle is a circular array of fuel elements made of seal-welded Zircaloy tubes known as “cladding” and containing inside the tubes a stack of natural UO_2 fuel pellets. The cladding wall thickness is about 0.42 mm thick allowing the cladding to collapse onto the fuel pellets under the operating pressure of the coolant of about 10 Mpa, thus, allowing an efficient thermal contact with the fuel allowing a high heat transfer from the fuel to the coolant. The fuel elements are held together by two endplates resistance welded to the tubes end caps, thus, providing stability and structural strength to the bundle. The fuel elements are 0.5 m in length and 15.2 mm in diameter for the 28-element Pickering bundle and 13.2 mm for the 37-element type bundle. Both bundles are about 10 cm in diameter.

Appendages in the form of bearing pads and inter-element spacers are brazed to the Zircaloy cladding at designated locations to maintain the bundle configuration. They also allow ease of handling during in-reactor service and subsequent management operations.

Key to the performance of the bundle is its structural integrity as an array of elements and the integrity of the Zircaloy cladding. The cladding acts as the primary barrier for the containment of the fission products generated during the bundle in-reactor service.

To allow the future safe retrieval of the fuel from dry storage, the used fuel bundles need to remain structurally intact and retain sufficient strength to sustain the stresses associated with future handling operations, transportation and disposal. The integrity of the used fuel sheath is also a key requirement for radiological safety. As discussed in Section 4.1.1, the used fuel matrix and the Zircaloy sheath provide a primary barrier to prevent the release of radionuclides. Although the seal-welded lid of the DSC provides containment for any radionuclides released by the fuel, as per Section 4.1.2, the DSC has been designed to maintain used fuel integrity under normal and abnormal operating conditions, along with credible accident conditions. These design features ensure retrieval and other future operations will be simpler and safer. The processes considered which are relevant to fuel sheath integrity within the DSC are captured in subsequent subsections.

4.7.1.1 Processes that May Affect Used CANDU Fuel Integrity during Dry Storage

During the fuel bundle in-reactor service, the structural materials of the fuel bundle are affected by the bundle irradiation history, the bundle burnup and power rating and the release of radionuclides from the fuel pellets. The fission processes in the fuel result in the formation of radionuclides or fission products in the form of gases, volatiles and solid phase particles that remain contained in the fuel pellets and the fuel elements. Additionally, the high temperatures experienced by the fuel during irradiation contribute to thermo-mechanical stresses that may lead to mechanical deformations – albeit nearly imperceptible – of the fuel elements and the fuel assembly.

Due to the decay of the radionuclides and the resultant heat generated, a number of processes have been postulated that could have an impact on the long-term behaviour of the fuel. The following mechanisms can be listed:

(a) Oxidation of the sheath;

(b) Fast brittle fracture;

(c) Creep rupture;

(d) Stress Corrosion Cracking (SCC) by iodine and metal vapour embrittlement by cesium and cadmium;

(e) Effects of hydrogen/deuterium migration;

(f) Delayed hydride cracking; and

(g) UO2 oxidation of defected fuel.

The possibility that any of these processes might contribute to the degradation of the fuel has been studied for CANDU fuel since the early 1980's. Results from these studies, as discussed below, indicate that for the conditions of dry storage, these processes are not expected to have an impact on the long-term condition of the fuel.

Early studies by (BYRNE84) and (HUNT81) addressed the potential impact of these postulated mechanisms on the long-term integrity of the fuel. The (BYRNE84) analysis concluded that for fuel dry stored at temperatures below 200°C:

1. Oxidation of the Zircaloy sheath when the fuel is exposed to air was found to be negligible. A 0.04 percent thickness loss of the sheath thickness was predicted for the first 100 years of storage and 0.4 percent in 1000 years.
2. Failure by fast fracture occurs preferentially by plastic collapse rather than brittle fracture.
3. Failure of cladding due to fast fracture by plastic collapse will be insignificant. Typical bundles representative of the CANDU fuel population will only fail if cracks with a depth of 92 percent of the sheath thickness are present.
4. Typical CANDU fuel will be unaffected by stress rupture for at least 10^6 years.
5. Migration of hydrogen to the endplate/endcap welds is negligible and will not contribute significantly to a decrease in its mechanical strength.

(HUNT81) developed a database for both SCC by iodine and vapour metal embrittlement by cesium and cadmium of the Zircaloy sheath by stressing irradiated fuel sheath rings in atmospheres of these elements contained in glass vials at various concentrations relevant to the expected concentrations in the fuel element gap. The results of those tests indicated that unless a 95 percent through-wall crack is already present in the sheath, an unlikely event, both SCC and vapour metal embrittlement will not be operative at the temperatures of storage relevant to dry storage of CANDU fuel.

For intact fuel, creep in the sheath material could potentially be the limiting factor for fuel sheath degradation. However, when used fuel is stored in a helium atmosphere, temperatures of up to 300°C can be considered safe for the planned storage period for intact used fuel in DSCs (PEEHS91). The upper temperature limit ensures that creep strain remains within acceptable limits and the inert gas precludes oxidation processes. The temperature limit of 300°C offers a conservative safety buffer to CANDU fuel stored in the DSCs, which is normally below 150°C. (CANN02) also studied the impact of creep on fuel in dry storage and concluded that for at least 300 years, creep will be negligible in that environment.

The robustness of the CANDU fuel bundle as an assembly has also been demonstrated early on by shock and vibration studies (FOREST82) as well as repeated monitoring of fuel performance during transportation which has repeatedly confirmed the robustness of the welds.

To further confirm the integrity of the bundles when stored both under water (wet) and dry conditions, and their ability to be handled safely post storage for transportation and disposal, former Ontario Hydro and AECL initiated a comprehensive program of research and testing in 1977 for the wet storage of used fuel (Wet Storage Program), and, in 1980 for the dry storage of the fuel (Dry Storage Program). Commercial CANDU fuel with burnup and linear power ratings bounding the characteristics of the fuel population in wet storage at the nuclear stations were selected for the programs. Fuel elements of designated bundles were intentionally defected by drilling a pin-hole through the sheath to follow any potential degradation of the UO_2 pellets from oxidation.

The fuel was characterized initially in great detail prior to their storage and withdrawn at regular intervals for re-examination to detect any significant deterioration that could affect their condition in the long term. Since their last examination, the fuel remains available for further testing if required. Main tests used to characterize the fuel and determine if there was any degradation of the zircaloy sheath and the UO_2 pellets included:

- Visual examination;
- Profilometry;
- Metallographic examination;
- Scanning Electron Microscopy, neutron radiography, and x-ray analysis;
- Fission gas analysis;
- Hydrogen and deuterium analysis;
- Ring tensile tests to determine increase susceptibility to SCC from iodine; and
- Torque tests of the end cap/end plate welds to determine any changes in their structural strength.

After over 27 years in storage, results of the Wet Storage program indicated that no significant degradation of the fuel had taken place during its storage. It was concluded that fuel could remain in wet storage for at least 50 years without any significant adverse changes that could compromise its handling or integrity in subsequent steps of fuel management including dry storage, transportation and disposal (FROST84; WASYWICH91).

The Dry Storage Program consisted of three experiments: Controlled Experiment (CEX) CEX-1, CEX-2 and Easily Retrievable Basket (ERB). The fuel was stored in baskets located in designated concrete canisters at the AECL-Whiteshell facilities. The fuel was stored:

(a) In the CEX-1 experiment in air at 150°C;

(b) In the CEX-2 experiment in steam at 150°C; and

(c) In the ERB air at ambient temperatures.

The CEX-2 experiment is of minor relevance to the OPG dry storage system and will not be discussed any further.

The results of the CEX-1 experiment and ERB indicated that for undefected fuel, the storage conditions did not lead to any significant degradation of the fuel. In all instances, the torque tests confirmed the robustness of the end plate/end cap welds and did not detect any significant mechanical changes to the welds.

In the case of the intentionally defected fuel, the bulk of the UO_2 remained intact but some changes to the appearance of the UO_2 matrix were observed in the CEX-1 experiment with limited conversion to U_3O_7 after about 10 years in storage. Bulk oxidation of the UO_2 matrix to U_3O_8 was not observed and there were no diametral changes detected of either the UO_2 matrix or the sheath that could have led to the sheath splitting. In summary, the results of the Dry Storage Program corroborated the assessment studies and provided evidence about the safety of the dry storage of CANDU fuel.

The conditions of fuel storage in the DSC are more benign than the conditions of fuel storage in the Dry Storage Program since helium, an inert gas, is used as the storage medium. Further, no defected fuel is intentionally stored in the DSCs.

Since both sheath creep and fuel matrix oxidation are temperature-dependent processes, the temperature of the fuel in dry storage is an important factor in the assurance of fuel integrity and safety. The provisions used to maintain used fuel integrity during storage include seal welding of the DSC and the addition of an inert helium atmosphere in the DSC cavity. Low temperatures during DSC processing and storage keep the rate of sheath creep sufficiently slow to prevent sheath rupture over time, and minimize the rate of used fuel oxidation for elements with undetected sheath defects and available oxygen.

Further, at OPG facilities, fuel is placed in dry storage after a minimum of 10 years of storage in the IFBs. Analysis and measurements carried out at the NSS-PWMF indicate that the maximum fuel cladding temperature is not expected to exceed 150°C in dry storage. At this temperature, UO_2 oxidation is sufficiently slow for residual oxygen in the fuel cavity to be of any concern (FC16). Additionally, the DSC seal-weld boundary is helium leak-tested. This ensures that the helium atmosphere is retained in the DSC cavity throughout the entire storage period and that there is no in-leakage of oxygen into the DSC cavity.

A worst case scenario leading to fuel degradation of a fuel element with an undetected sheath defect has been analyzed (OPG99). This case assumed a fuel element in which the ~ 1 cc volume of the gap between the fuel pellets and the cladding was filled with water. Radiolysis of the water content was assessed to yield ~ 0.9 g of oxygen. If this oxygen were to combine with the uranium fuel matrix at low temperatures, the resulting increase in pellet diameter due to volumetric expansion should not result in damage to the fuel sheath.

The above considerations support the conclusion that under normal operating conditions DSCs provide safe and retrievable storage for OPG's used CANDU fuel.

4.7.2 Dry Storage Container Integrity

The DSC is designed to provide a design life of at least 50 years and to meet all shielding and containment integrity requirements over this period. Shielding is provided by the high-density concrete and containment is provided by the steel inner liner, the bottom plate of the lid, the base perimeter flange, the lid locating pin housings, the drain port plug, and all seal-welds.

4.7.2.1 High Density Concrete

The reinforced high-density concrete of the DSC has been assessed for potential degradation resulting from elevated temperature, low temperature, radiation fields, presence of water and chemical reactivity. The potential effects are a reduction in mechanical properties (compressive strength, modulus of elasticity), cracking and chemical attack. It has been concluded that the concrete of the DSC will provide at least 50 years of service even when subjected to conservative conditions; for example, higher heat load and thermal gradient than actually experienced during storage.

4.7.2.2 Steel Components and Welds

The steel of the DSC has been assessed for potential degradation resulting from temperature, radiation fields and corrosion. It has been concluded that there is no metallurgical factor or combination of factors which limit the expected service life for the DSC to less than its design life, provided the integrity of the surface coating is maintained. The surface coating is expected to provide at least 50 years of service when stored indoors and adequately maintained.

4.7.2.3 Aging Management Program

OPG has implemented an Integrated Aging Management Process (N-PROC-MP-0060), the purpose of which is to determine the life-limiting characteristics of the critical components and to provide timely detection and mitigation of significant aging effects. The goals of the Integrated Aging Management Program are achieved by establishing a set of programs and activities which ensure performance requirements of all critical equipment are met on an ongoing basis. Aging Management requires life cycle plans, to assess end of life and identify an aging management strategy, and ongoing condition assessments. This program provides assurance that the design service life will be achieved.

4.7.2.4 Inspection and Maintenance Program

The DSCs have been designed to remain maintenance-free for many years. Visible parts of containers are, however, checked for corrosion and paint deterioration on a regular basis as per I-PROG-AS-0001 (Conduct of Inspection and Maintenance Services). The weld area is specifically inspected for visible signs of degradation. If corrosion is observed on a DSC, the affected area is cleaned up and recoated with the specified touch-up paint or repaired as needed.

A limited number of DSC bottoms are inspected annually as per the DSC base inspection plan.

4.7.2.5 Used Fuel Retrievability

OPG recognizes that DSCs may have to be retrieved and the fuel may have to be placed back into the IFB. Therefore, the design of the DSC and processing and storage processes fulfill the requirement that the fuel is safely retrievable from storage.

Retrieval of fuel from a DSC would be a reversal of the loading sequence. The basic steps would be:

(a) Remove drain plug weld by grinding or milling.

(b) Sample interior atmosphere;

(c) Remove lid weld by grinding or milling;

(d) Clamp lid to base;

(e) Return DSC to IFB; and

(f) Unload used fuel from DSC.

The technology is readily available and detailed procedures would be developed when required after securing appropriate approvals. Availability of sufficient space at the IFB would be a pre-requisite to such a transfer.

5.0 RETUBE COMPONENTS STORAGE

5.1 Retube Components Storage Description

The purpose of the RCS area at the NSS-PWMF is to provide storage for components removed during retubing of the Pickering NGS A reactors. Reactor components become radioactive during their residence in the core, due to neutron activation and deposited contamination.

The retube components (including pressure tubes, end fittings, garter springs, shield plugs and miscellaneous identified components) have been loaded into specifically designed and shielded Dry Storage Modules (DSMs) for interim storage at the NSS-PWMF. The DSMs are large cylindrical casks (see Figure 5-1), made of reinforced high-density concrete and thick carbon steel inner and outer liners. A bolted and gasketed shield door is used to seal the fill port on each DSM. Saddle supports are used to hold each DSM in a horizontal position. The DSMs are stored outdoors in the fenced RCS area.

All four Pickering NGS A reactors were successfully retubed during the Large Scale Fuel Channel Replacement Program (LSFCRP). DSMs containing components from Pickering NGS Unit 1 (P1) and Unit 2 (P2) were loaded inside the station and placed into storage at the RCS area between 1985 and 1988. Unit 3 (P3) and Unit 4 (P4) DSMs were loaded and stored between 1990 and 1993.

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

5.1.1 General Lay-Out of the Retube Components Storage Area

The RCS area is located in the southeast corner of the Pickering NGS site and within the NSS-PWMF Phase I area. The physical location of the area is shown in Figures 1-1 and 2-1. The facility is designed to accommodate 38 DSMs. At present it has 16 DSMs from retubing Units 1 and 2 and 18 DSMs from Units 3 and 4, plus two empty DSMs and two vacant storage positions. A chart of DSM contents is provided in Appendix D.

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 supported on compacted crushed stone overlying fill, composed of coarse to fine sand and gravel.

The storage area is paved and further covered with a polymer membrane coating to provide an easy to maintain surface. A drainage system is provided to direct the runoff water from the storage area to the Pickering NGS B outfall, with catch basins permitting periodic sampling of the water.

The dose rates from the storage facility are monitored by Thermoluminescent Dosimeters (TLDs) placed at two locations on the perimeter fence.

5.1.2 Dry Storage Module Description and Performance

5.1.2.1 Dry Storage Module Description

The DSMs (the P1/P2 design is shown in Figure 5-1) are cylindrical casks made from reinforced high-density concrete (nominal density is approximately 3.5 Mg/m^3). A 6.4 mm thick carbon steel liner forms the outer shell. The inner liner, which acts as formwork during construction, is 6.4 mm thick carbon steel for P3/P4 DSMs. For P1/P2 DSMs, a 3.2 mm thick galvanized corrugated pipe forms the inner liner. DSMs are designed to be leak resistant, employing welded construction. The DSM is a safety-related structure because failure of the DSM to perform its design function under certain conditions may lead to radiation exposures to workers/public that exceed regulatory dose limits (OPG19a).

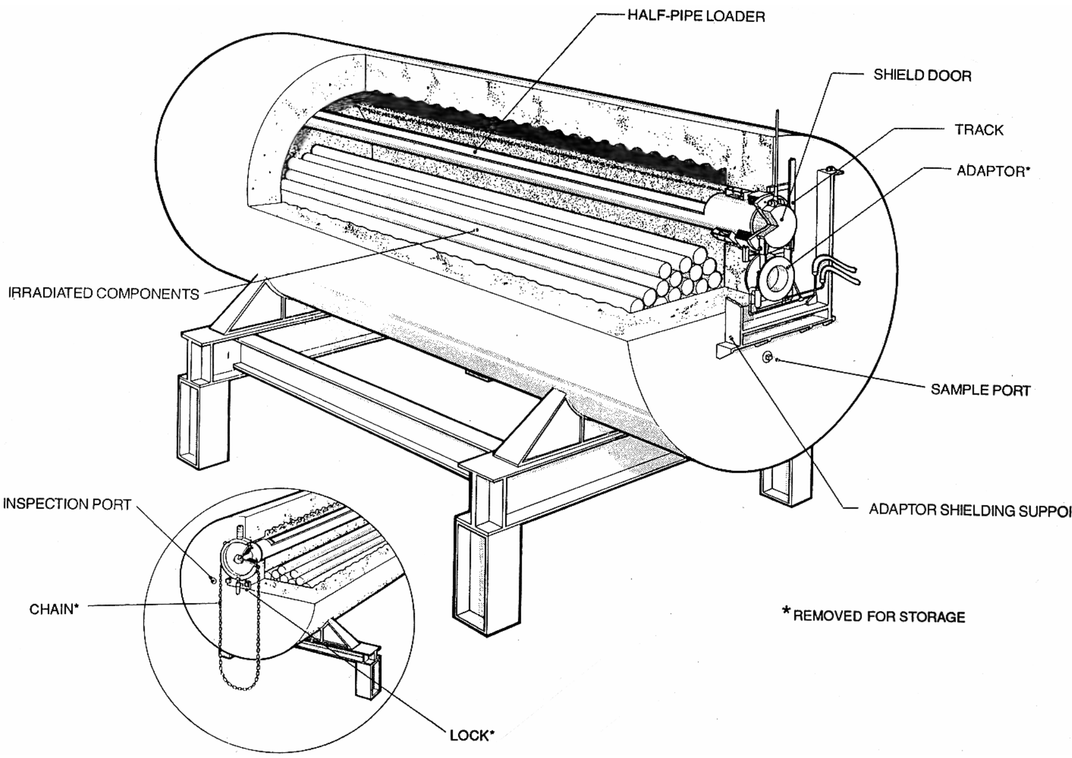

Figure 5-1: Dry Storage Module

In order to provide adequate shielding to meet dose rate requirements outside the facility and to keep worker dose rates ALARA, the perimeter and end walls of each DSM are made of 0.57 m thick high-density concrete.

Each DSM is 3.3 m in diameter and has an overall length of 7.6 m. To prevent deterioration and minimize maintenance, the outer steel shell is coated with two coats of ceramic elastomeric paint.

The DSMs are designed for horizontal loading, with a fill port and a loading mechanism running the full length of the module. Each DSM can hold about 90 pressure tubes or various combinations of pressure tubes, end fittings and miscellaneous components. After the loading operation is complete, the fill port is sealed with a bolted and gasketed shield door. The gasket is made of silicone rubber.

Each DSM additionally 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.

[redacted]

5.1.2.2 Module Integrity

The DSMs are designed for outdoor storage for a minimum of 50 years. The module is designed to withstand the following loads:

(a) Dead load of irradiated components up to 23 tons;

(b) A uniformly distributed wind load of 102 kg/m^2 and a uniformly distributed snow load of 214 kg/m^2;

(c) The forces due to dropping of irradiated components from the half-pipe unloader into the module; and

(d) The forces caused by the rotation of the module from the vertical to horizontal position when picked up at the centre.

The outer metal shell of the module prevents the ingress of moisture into the high-density concrete and, therefore, protects the high-density concrete from freeze-thaw action. In addition, rapid freeze-thaw cyclical testing of high-density concrete specimens (far more severe than the conditions to which DSMs would be actually exposed) has concluded that the DSMs have satisfactory resistance under such conditions (OH89).

The DSMs are capable of withstanding the thermal stresses arising from the heat load in the module. The heat load due to the radioactive decay of isotopes has been calculated to be less than 1,200 Watts, considering 150 pressure tubes and 300 shield plugs loaded in a module. The thermal stresses due to this heat load on a DSM are negligible.

The silicone rubber gasket used for sealing the DSMs is capable of withstanding an absorbed radiation dose of 1.0×10^5 Gy. Since the gasket is located outside the DSM shielding, the radiation dose over 50 years would be less than 100 Gy.

5.1.2.3 Irradiated Components

The irradiated components stored from Units 1, 2, 3, and 4, consist of pressure tubes, the inboard section of the end fittings, garter springs, shield plugs and miscellaneous identified components. In order to facilitate handling and to reduce the potential spread of radioactive contamination, the shield plugs were pushed into the pressure tubes and the pressure tubes, along with their garter springs, were inserted into waste containers.

The pressure tube waste containers are made of carbon steel and their dimensions are approximately 0.16 m in diameter and 6.2 m in length. The end fittings from P1/P2 were bagged but for P3/P4 these were also put in containers (0.2 m outer diameter) in order to control loose contamination. All the end fittings were cut into two parts, and only the irradiated inboard sections were packaged in containers and stored in the modules. The components are corrosion-resistant. Corrosion of the stored components is, therefore, expected to be negligible over the design life of the module.

During the P1/P2 retube, carbon-14 in particulate form was found to have adhered to the components removed from the reactor. However, specific procedures were in place to remove carbon-14 from the annulus gas system before component removal for the P3/P4 project. With these provisions, there should be less carbon-14 in the P3/P4 storage modules than in the P1/P2 storage modules. This is discussed further in Appendix C. A description of the components stored in the DSMs is given in Table 5-1.

Table 5-1: Irradiated Components for Storage[13]

Quantity per Reactor Unit	Component	Diameter (approximate)	Length (approximate)	Weight (approximate)
390	Pressure Tube	0.114 m	6.1 m	73 kg
780	Shield Plug	0.102 m	0.9 m	41 kg
780[14]	End Fitting (inboard section)	0.172 m	1.8 m	109 kg
780	Garter Springs			

[13] Some components were sent to Chalk River Nuclear Laboratories (CRNL) for research purposes. The actual number of stored components would therefore be less.

[14] 390 end fittings for P1/P2.

5.1.3 Long-Term Performance

5.1.3.1 On-Going Inspection and Maintenance

The DSM has been engineered and designed for minimum upkeep over its design life. Periodic inspection and maintenance is 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, as per I-PROG-AS-0001 (Conduct of Inspection and Maintenance Services), 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 touch-up paint or repaired as needed.

An aging management plan (OPG16) is in place, which includes detailed annual visual inspections of DSMs.

6.0 RETUBE COMPONENT STORAGE SAFETY ASSESSMENT

6.1 Introduction

This chapter provides a summary of the NSS-PWMF RCS area radiological safety assessment for the continued storage of DSMs. Conservative estimates of public dose rates due to releases resulting from hypothetic failures of DSMs for normal and abnormal operating conditions, and credible accident conditions are also presented.

6.1.1 Safety Assessment Approach

Under normal operating conditions during storage, DSMs are expected to provide reasonable assurance that the waste can be stored without undue risk to workers, members of the general public, or the environment.

RCS waste operations comply with OPG requirements to keep total radioactive emissions under normal operating conditions below regulatory limits and ALARA.

The safety assessment of normal and abnormal operating conditions and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions that are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions.

6.1.2 Acceptance Criteria

The radiation safety requirements under normal operation for NSS-PWMF are as follows:

- ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.

- ≤ 100 µSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years.

The public/worker dose acceptance criteria for credible abnormal events are the following:

- The dose limit for the public at or beyond the PNGS site boundary due to an abnormal event or credible accident is 1 mSv.
- The dose limit for a worker due to an abnormal event/accident is 50 mSv.

The limit of 50 mSv for abnormal event/accident refers to NEWs. The equivalent limit for non-NEWs/members of the public is 1 mSv.

6.2 Radiological Safety – Normal Operating Conditions

6.2.1 Public Dose

Dose to members of the public from normal operation of the NSS-PWMF has been determined based on the latest information on radionuclide emissions, representative group locations, and meteorological data.

The maximum dose rate calculated to an individual member of the public is 1.46×10^{-3} µSv/h at the site boundary. This is equivalent to an annual dose of 2.92 µSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 1.41×10^{-3} µSv/h. This is equivalent to an annual dose of 1.41 µSv based on 1,000 hours occupancy; this is a conservative assumption for boaters and fishermen.

These results indicate that the NSS-PWMF administrative dose target of ≤100 µSv/y at the station site boundary as set out in Section 1.6.1 is met during NSS-PWMF operations. Details of the assessment can be found in Appendix C.

6.2.2 Worker Dose

The actual worker doses received during normal operation of the NSS-PWMF are reported in the NSS-PWMF quarterly reports. Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported during the 2007-2021 period are 12.6 person-mSv and 1.6 mSv.

The maximum effective dose to NEWs working at the NSS-PWMF is well below the regulatory dose limits for NEWs; 50 mSv in a one-year dosimetry period and 100 mSv in a five-year dosimetry period.

6.2.3 Dose Rates from the Retube Components Storage

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 TLDs 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 2007-2021 TLD survey monitoring results, the maximum dose rates are 0.16 µSv/h at the south fence, 0.11 µSv/h at the east fence, 0.12 µSv/h at the west fence and 0.34 µSv/h at the north fence.

6.3 Radiological Safety – Abnormal Operations

The DSMs located in the RCS area are in interim storage. They are occasionally inspected, however, there is no regular handling of the waste. A detailed screening of external and internal events has demonstrated that there is no credible event that will result in a release of radioactive material. Therefore, there are no radiological consequences to workers or the public from continued storage of waste in the RCS area. Details of the hazard assessment for each malfunction or accident are given in Appendix C.

6.3.1 Potential Off-Site Consequences

A DSM failure is very unlikely during its proposed 50-year storage life because of 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.

7.0 SAFETY AND MONITORING PROGRAMS

7.1 Introduction

OPG has established comprehensive occupational, public, and environmental protection programs, including occupational dose monitoring, radiation and contamination monitoring in the workplace, environmental monitoring programs, and conventional safety programs in support of its nuclear facility operations to assure compliance with the Nuclear Safety and Control Regulations, applicable Provincial Legislation, and OPG requirements. Program administration is conducted on a site-wide basis, and encompasses all nuclear facilities at the Pickering NGS site including the NSS-PWMF.

The safety and environmental monitoring program elements applicable to the NSS-PWMF include the following:

- Radiation Protection (RP) Program;
- Effluent Monitoring;
- ALARA – Occupational Radiological Risks and Safety Management; and
- Occupational Non-Radiological (Conventional) Environment, Safety and Health Management.

Details of these program elements are provided below.

7.2 Radiation Protection Program

7.2.1 Program Overview

An RP program is in place at the NSS-PWMF. The program addresses occupational radiation safety and contamination control. It identifies the operations and materials that have the potential to contribute to occupational dose, and provides guidelines to monitor and minimize occupational dose and reduce the potential for contamination in the facility.

The RP program (N-PROG-RA-0013) is implemented through a series of standards and procedures for the conduct of activities within nuclear sites and with radioactive materials intended to achieve and maintain high standards of RP including the achievement of the following objectives:

(a) Controlling occupational and public exposure:

   (1) Keeping individual doses below regulatory limits.

   (2) Avoiding unplanned exposures.

   (3) Keeping individual risk from lifetime radiation exposure to an acceptable level.

   (4) Keeping collective doses ALARA. Social and economic factors are also taken into account.

(b) Preventing the uncontrolled release of contamination of radioactive materials from the nuclear sites through the movement of people and materials.

(c) Demonstrating the achievement of (a) and (b) through monitoring.

This program complies with the CNSC requirement that all licensees implement an RP program and establish a quality program that meets the specific Canadian Standards Association (CSA) standards for RP programs.

This program is designed to comply with the RP program requirements of the following acts and regulations as applied to licensed OPG facilities and licensed OPG activities:

- General Nuclear Safety and Control Regulations (SOR/2000-202)
- Radiation Protection Regulations (SOR/2000-203)
- Class II Nuclear Facilities and Prescribed Equipment Regulations (SOR/2000-205)
- Nuclear Substances and Radiation Devices Regulations (SOR/2000-207)
- Occupational Health and Safety Act, R.S.O. 1990, Chapter O.1
- Occupational Health and Safety Act, R.R.O. 1990, Regulation 861, X-Ray Safety
- Radiation Emitting Devices Act, R.S., 1985, c. R-1
- Radiation Emitting Devices Regulations, C.R.C., c. 1370

This program is applicable to OPG Nuclear Facilities such as the Pickering Waste Management Facility, including contract and consulting personnel.

The RP program takes its authority from N-CHAR-AS-0002, Nuclear Management System.

Control and measurement of releases of radioactive materials to the environment through nuclear systems is exercised through OPG-PROG-0005, Environment Health and Safety Managed Systems.

7.2.2 Occupational Dose Control

The dose received from routine NSS-PWMF operations are monitored and assessed against dose targets. TLD badges are worn as a minimum external dosimetry requirement by personnel involved in all tasks of NSS-PWMF operations. In addition, NSS-PWMF staff are required to wear Electronic Personal Dosimeters (EPDs) for all radioactive or potentially radioactive work, which are used to monitor dose received and alarm when dose or dose rates reach predetermined levels.

Internal and external occupational radiation exposures are kept ALARA through facility design, procedural controls, and the control of access to the NSS-PWMF.

Access to the NSS-PWMF is limited to designated personnel and those escorted by qualified personnel. Access to the RCS area is restricted via a locked gate. Only qualified personnel are allowed to access the area for the purpose of monitoring and maintenance of DSMs. Inadvertent or unplanned radiation exposures are avoided by means of clear warning signs within the facility for higher than expected radiation fields, when required. As discussed in Section 7.2.5.1, alarming gamma monitors are provided in the DSC processing building where sudden changes in gamma radiation fields could potentially occur.

7.2.3 Contamination Control

Surface contamination on DSCs is minimized by decontamination of the container surface after loading at the station IFBs. During storage, DSCs and DSMs are monitored for loose contamination as discussed below. Any case of loose contamination found is removed by manually wiping with a cloth, or by wet methods if necessary, taking appropriate measures for containment of contamination at the source and personnel protection. Wet decontamination of storage containers has not been required and is not expected to be required at the NSS-PWMF.

7.2.4 Zoning

An important means of contamination control is the division of a facility into zones. These are Zone 1, a clean area which may be considered as the equivalent of a normal public access area, and radiological zones of higher number (i.e., Zones 2 and 3) in which the potential for radioactive contamination or radiation exposure exists.

Zoning for the NSS-PWMF Phase I site is as follows: the DSC processing building and DSC Storage Buildings 1 and 2 are designated Zone 2; the storage room for helium bottles is accessible only from the outdoors and is designated an unzoned area. The RCS area is designated as a fenced, restricted, radioactive storage area within the unzoned radioactive work area.

DSC Storage Buildings 3 and 4 in Phase II are located outside the radiologically zoned Pickering NGS protected area, in a separate nearby protected area within the Pickering NGS property perimeter fence. DSC Storage Buildings 3 and 4 and the outdoor area within the NSS-PWMF Phase II site protected area are designated as a Zone 2 area. The entrance kiosk is designated Zone 1.

Radiological controls are in place in accordance with OPG RP Requirements.

7.2.5 Radiological Hazard Monitoring

RP requirements include area gamma radiation monitoring and routine radiological surveys, as well as surface and airborne contamination monitoring. The main objective of monitoring is the timely detection of changes in radiological hazard levels so that appropriate remedial actions can be taken.

Routine Radiological Surveys for the NSS-PWMF are documented in N-INS-09071-10025, Radiation Surveys and Visual Survey Data System.

7.2.5.1 Gamma Radiation Monitoring

Continuous alarming, semi-portable gamma monitors are provided in the DSC processing building workshop where sudden changes in gamma radiation fields could potentially occur.

Personnel initiated surveys and periodic monitoring of the NSS-PWMF are conducted in accordance with facility-specific procedures. Gamma surveys are performed upon each loaded DSC receipt at the NSS-PWMF, and during “hands on” work with a DSC or DSM.

Routine gamma radiation surveys are performed at appropriate points covering the entire sequence of UFDS operations to:

- Monitor for overall changes in radiation levels; and
- Initiate corrective action, if needed, as per approved RP procedures to maintain occupational safety standards.

Gamma radiation monitoring is conducted weekly inside the station fence along the DSC processing building, the DSC storage building walls, and at the RCS perimeter fence. Gamma radiation monitoring of the DSMs is performed twice per year.

7.2.5.2 Contamination Control and Monitoring

Surface contamination checks are performed at set frequencies on DSCs and DSMs in storage as well as in the DSC processing building (including offices, washrooms, and coffee shop), and DSC storage buildings. Loose contamination found in these areas is required to be isolated and cleaned up immediately.

7.2.5.3 Airborne Contamination Monitoring

Airborne contamination is monitored by a Continuous Air Monitor in the welding shop. A portable monitor is also used for weekly monitoring of the air inside the DSC storage buildings.

7.3 Environmental Monitoring

Environmental effects due to radioactive releases from the NSS-PWMF are monitored under the Pickering NGS Environmental Monitoring Program (EMP). The EMP is designed to measure environmental radioactivity in the vicinity of Pickering NGS from all site sources. Data from the EMP are used to assess off-site public dose consequences resulting from the operation of nuclear facilities at the Pickering NGS site.

Additionally to the site-wide environmental monitoring program, the following NSS-PWMF effluent streams are monitored:

- Active ventilation exhaust;
- Active liquid waste; and
- RCS surface area drainage.

7.3.1 Active Ventilation Exhaust Effluent Monitoring

The active ventilation exhaust from the DSC processing building is monitored for radioactive particulates. A continuous effluent sample is passed through a particulate filter that is replaced and analyzed on a weekly basis. No significant tritium, noble gas or radioiodine emissions are expected from the NSS-PWMF, and no fixed monitors are provided for these radioactive species. Although no significant particulate emissions are expected, the exhaust is monitored as a precautionary measure since there is a potential for airborne emissions as a result of DSC processing operations such as welding and vacuum drying.

DSC paint touch-up operations involve minimal paint quantities. Due to small quantities, painting methods, and the use of appropriate filtration, no significant emissions of paint materials are expected. Welding fumes from DSC seal welding operations are additionally exhausted through the HEPA filtered active ventilation system; therefore minimal welding contaminants are expected in NSS-PWMF emissions. The conventional air emissions are approved under the Pickering Nuclear MECP air approval.

7.3.2 Active Liquid Waste Effluent Monitoring

Liquid effluent is generated from water collected in the floor drains (e.g., from floor wash water, precipitation such as snow or rainwater that enters the building during DSC transfer and from condensate from the air conditioners). No contamination is expected in the sump water from the DSC operations, however, the sump water is analyzed for tritium and gross beta-gamma activity prior to being transferred to the station’s active liquid waste management system to ensure it meets the station’s acceptance criteria.

7.3.3 Retube Components Storage Surface Area Drainage Monitoring

Water is sampled from the RCS surface drainage system for gross beta-gamma activity for confirmation purposes. This provides assurance that any radioactive contamination of the surface water originating from the storage area is detected, however, no contamination of the surface water is expected from the DSMs.

7.4 ALARA – Occupational Radiological Safety Management Assessment

An ALARA – Occupational Radiological Risks and Safety Management Assessment (OH94) was originally carried out during the detailed design phase of the NSS-PWMF. The purpose of this pre-operational assessment was to identify radiological and non-radiological hazards associated with the UFDS operations, and thereby provide a baseline for radiological safety performance.

This assessment was subsequently updated (OH98b) after the first year in service for the NSS-PWMF, to compare actual operational experience against the selected ALARA targets. This resulted in adjustment in the collective and individual ALARA dose targets downwards.

An updated shielding analysis (OPG00c) was performed to re-evaluate dose rates from DSCs and to evaluate gamma radiation dose rates inside the DSC Storage Building 2 and outside the NSS-PWMF as a result of increased storage capacity.

In 2003, operating experience at the NSS-PWMF was again compared with the ALARA targets (OPG03a). The assessment concluded that doses had remained below the collective and individual worker dose targets and recommended that the same targets be extended to cover operations at both the NSS-PWMF Phase I and Phase II sites.

Another post-operational ALARA assessment for the NSS-PWMF was conducted in 2007 (OPG07b). The assessment concluded that the design, operational and procedural measures in place at the NSS-PWMF for radiation/contamination prevention and control, have been effective in meeting the existing ALARA targets set during its first post-operational ALARA assessment. Annual collective and individual worker doses received as a result of the facility's operations have been well below the ALARA targets. Operational controls and task procedures employed at the facility have been effectively and promptly revised when necessary to further reduce worker dose expenditure in line with the objectives of the RP program.

The 2007 assessment recommended that, in light of the low doses received by workers, the ALARA targets for individual workers be further reduced to 3 mSv/yr, however this value is not an internal OPG exposure limit or administrative level under the RP Program.

For collective doses received by workers, the 2007 assessment recommended that the ALARA targets be changed to describe doses received from a single DSC. This would permit the ALARA dose target to reflect DSC throughput, which can change over time. The assessment recommended that the ALARA target for the collective worker dose due to the loading, processing, and storing of a single DSC be 0.3 Person-millisievert (P-mSv), and that the accompanying ALARA target for collective worker dose due to all other activities at the NSS-PWMF be 1.4 P-mSv/year. Brought together, the recommended ALARA target for collective worker dose is calculated as follows:

$$(\text{annual DSC throughput} \times 0.3 \text{ P-mSv}) + 1.4 \text{ P-mSv/year}$$

Actual collective doses at NSS-PWMF are lower than the bounding ALARA assessments outlined above and are reported in the quarterly operations reports. Aggressive ALARA targets are prepared annually with input from Operations and Radiation Protection and reviewed at a division level on a monthly basis.

Collective ALARA dose targets are prepared annually based on DSC throughput and required work activities and are summarized in the quarterly operations report. Individual worker dose breakdowns are also provided and demonstrate that doses are well below OPG exposure control and administrative dose limits.

7.4.1 Occupational Radiological Hazards Identification and Quantification

Potential occupational radiological hazards associated with NSS-PWMF operations can be categorized as follows:

- Chronic radiological hazards associated with normal operations of the NSS-PWMF; and
- Acute radiological hazards associated with some normal and abnormal operating conditions of the NSS-PWMF.

7.4.1.1 Chronic Radiological Hazards Associated with Normal Operations of the Pickering Waste Management Facility

The following radiological hazards to NSS-PWMF personnel may exist throughout the DSC processing, including surveying and monitoring activities. These hazards would consist of potential external and internal exposures arising from (i), irradiated fuel inside the DSC, and (ii), surface contamination on the DSC. The breakdown of these hazards with respect to type, process and location is given below.

7.4.1.1.1 External Gamma Radiation Hazards

Personnel working on or in the vicinity of a loaded DSC are exposed to external gamma radiation from its surface, originating from the used fuel inside. Radiation fields from a loaded DSC have been discussed in Section 4.3.2.1.

7.4.1.1.2 Surface Contamination Emitting Beta-Gamma Radiation

Surface contamination in NSS-PWMF operations presents an external radiation hazard exposure potential for fixed contamination, and an internal radiation exposure potential for loose contamination. The primary source of surface contamination for NSS-PWMF operations is the contamination picked up by the DSC from the IFB water during the process of used fuel loading. Other potential sources of surface contamination are discussed below.

(a) Process-Generated Surface Contamination on Material and Equipment

Surfaces of loaded DSCs are decontaminated, and checked that no loose contamination is present, before leaving the station IFB. One exception is the DSC bottom that is decontaminated but for reasons of practicality and the minimal contamination hazard involved, no further checks for loose contamination are made to the DSC bottom after it has been successfully decontaminated at the IFB.

The welding and painting process does not generate significant levels of loose surface contamination. The overall extent of surface contamination in the NSS-PWMF workshop is very small.

(b) Surface Contamination on Personnel

Surface contamination on personnel would arise either directly from contact with DSC surface contamination, or indirectly from contact with material and equipment contaminated by the processes discussed above.

(c) Airborne Contamination

Airborne contamination hazards from NSS-PWMF operations may present a hazard if DSC loose surface contamination becomes airborne, or through leakage of the DSC internal volume gas (e.g., could contain krypton-85 as well as radioactive particulates). The processes that could potentially give rise to this airborne hazard (via leakage) are:

- DSC draining and drying;
- Transfer clamp and seal removal; and
- DSC back-filling with helium.

An airborne particulate monitor and gamma monitor is used to detect any abnormally high levels.

(d) Liquid Contamination

Liquid contamination in UFDS operations would be a low-level hazard with potential for internal and external exposure. The following sources could potentially generate liquid contamination in the form of the following:

- Gross particulate beta-gamma emitters due to:
    - DSC vacuum drying;
    - DSC decontamination;
    - DSC processing building drainage; and
    - DSC storage building floor drainage.
- Hot particles (localized, discrete sources of beta-gamma activity) due to:
    - DSC vacuum drying.

7.4.1.2 Acute Radiological Hazards during Normal Operating Conditions

7.4.1.2.1 Removal of Dry Storage Container Drain Plug

It was recognized in the early design phase of the DSC that upon removal of the DSC steel drain plug to connect and disconnect the drainage and vacuum line connection, a rise in the gamma radiation field would occur near the drain port. A design modification consisting of adding a 90° bend to the DSC drain pipe was implemented, and a subsequent shielding calculation indicated a significant reduction in the gamma dose rate at the opening of the unshielded drain port. This has been confirmed with operational experience.

7.4.1.3 Acute Radiological Hazards during Abnormal Operating Conditions

Abnormal or accident conditions have the potential for the release of radioactive material and loss of shielding. Section 4.4 provides a detailed assessment of the potential impact of postulated events both inside and outside the NSS-PWMF.

7.4.2 ALARA Targets

The following targets are used as operations performance measures for the NSS-PWMF RP Program towards achieving the objective of keeping radiation exposures and contamination hazard risks ALARA.

7.4.2.1 Occupational Individual and Collective Dose Targets

Individuals attached to NSS-PWMF operations must comply with OPG requirements for Dose Limits and Exposure Control.

The recommended ALARA target for individual worker dose is 3 mSv/yr. This is reduced from the previous target of 5 mSv/year to set a more effective target based on historically low doses received by workers.

7.4.2.2 Dose Rate Limits at Pickering Waste Management Facility Boundaries

A dose rate target of 0.5 μSv/h on a quarterly average basis has been set at the security fence adjacent to the NSS-PWMF, based on the current regulatory annual dose limit for non-NEWs (i.e., members of the general public), with a maximum occupancy factor of 2,000 hours per year. Operational experience has shown that this requirement is met. As per references (OPG02, OPG03d), since 2003, TLDs placed on the NSS-PWMF Phase I east perimeter wall have a target perimeter dose rate of 1.75 μSv/h which is equivalent to the dose rate target 0.5 μSv/h described above but calculated for their location at the east perimeter wall.

An administrative dose target of 100 μSv/year has been set at the NSS-PWMF site boundary. This target is ten percent of the CNSC annual public dose limit (Table 1-1) and covers any normal operation exposure standing at the site boundary.

7.4.2.3 Derived Release Limits

The Derived Release Limit (DRL) for a given radionuclide is the release rate to air or surface water during normal operation of a nuclear facility that would cause an individual of the most highly exposed group around the facility to receive and be committed to a dose equal to the annual regulatory dose limit over the period of the calendar year. The criteria for calculating release limits for members of the general public is the bounding dose of 1 mSv/year as set out in the Radiation Protection Regulations.

The NSS-PWMF uses the DRLs established for PNGS, and the associated methodology for determining their calculation is documented in P-REP-03482-00006.

7.4.2.4 Action Levels

7.4.2.4.1 Radiation Protection Action Levels

RP Action Levels have been developed and implemented according to license requirements for the NSS-PWMF. Action Levels for NSS-PWMF are captured in N-STD-RA-0045, Occupational Radiation Protection Action Levels for Nuclear Waste Management Facilities (OPG23b).

7.4.2.4.2 Environmental Action Levels

Environmental Action Levels for the NSS-PWMF, as well as the methodology for their calculation, are documented in P-REP-03482-00006.

7.4.2.5 Contamination Control Targets

No detectable loose surface contamination is permitted in accessible areas. If any contamination is found it is remediated promptly and reviewed for reporting requirements by the Facility Health Physicist. Fixed contamination is permitted under limited, controlled circumstances.

NSS-PWMF operations complies with OPG requirements for the monitoring, identification and control of fixed and loose contamination hazards.

7.5 Occupational Non-Radiological (Conventional) Safety and Environmental Management System

7.5.1 Occupational Non-Radiological Health and Safety Management System

NSS and its contractors are committed to the prevention of workplace injuries/illnesses and to continuous improvement in the management of health and safety. NSS has adopted and implemented the requirements of OPG's Employee Health and Safety Policy, OPG-POL-0001. The requirements of the policy are developed, implemented, maintained, continuously improved, and documented in the OPG Environment Health and Safety Managed Systems Program Document (OPG-PROG-0005). OPG-PROG-0005 is structured in accordance with the requirements of the ISO 45001 standard, which defines the following:

Planning of the work

- Hazard identification
- Risk assessment
- Determining controls
- Identification of legal and other health and safety requirements
- Setting objectives, targets and achievement plans

Implementing the Requirements of the Planning Process

- Identifying roles, responsibilities, accountability and authority
- Setting standards for competence, training and awareness
- Setting standards for communication, participation and consultation
- Setting standards for control of documents
- Defining controls for the hazards
- Defining an emergency preparedness plan

Monitoring and Measurement of Success of the Planning Process

- A systematic approach for monitoring and measuring occupational health and safety performance is an integral part of the OHSMS.
- The monitoring and measurement evaluates the extent to which OPG objectives are being met; the effectiveness of controls; reactive measures of performance that monitor ill health incidents (including accidents, near misses etc.) and other historical evidence of deficient occupational health and safety (OH&S) performance.
- An auditing process and procedure measures compliance to legal requirements and compliance to the OHSMS.
- A procedure is in place for the control of records.
- A procedure is in place for incident investigation/management, corrective action and for capturing operating experience from internal and external sources.

Management Review of the HSMS and OH&S Performance

OPG Nuclear management reviews the OHSMS on planned intervals to ensure its ongoing suitability, adequacy and effectiveness, and alignment with the strategic direction of the organization. The review includes assessing the opportunities for improvement, the need for changes to the OHSMS, including the OH&S policy and OH&S objectives and targets.

Inputs to review may include:

- Results of internal audits and evaluations of compliance to regulatory requirements.
- Relevant communications from internal and external interested parties.
- The OH&S performance.
- The extent to which objectives have been met.
- Status of investigations, corrective and preventive actions.
- Recommendations for continual improvement.

Outputs from top management reviews shall be consistent with the NSS commitment to continuous improvement and includes decisions and actions related to changes in:

- OH&S performance.
- OH&S policy and objectives.
- Resources.
- Other elements related to the OHSMS.

7.5.2 Environmental Management System

The Environmental Policy (OPG-POL-0021) sets out OPG's environmental performance and environmental management commitments. OPG strives to improve its environmental performance by committing to the following:

- OPG shall establish an environmental management system and maintain registration for this system to the ISO 14001 Environmental Management System standard.
- OPG shall work to prevent or mitigate adverse effects on the environment with a long-term objective of continual improvement in its environmental management system and its environmental performance.
- OPG shall execute its Climate Change Plan and strive to achieve the milestones and goals therein.

- OPG shall manage its sites in a manner that strives to maintain, or enhance where it makes business sense, significant natural areas and associated species of concern. OPG will work with its community partners to support regional ecosystems and biodiversity through science based habitat stewardship. Where disruption is required, OPG shall take reasonable steps to manage the residual impact to these areas and species.

- OPG shall set environmental performance objectives as part of its annual business planning process. Performance against these environmental objectives will be monitored and associated documented information will be maintained.

- OPG shall communicate its environmental performance to employees, governments, local communities, and other stakeholders.

The policy is supported by OPG-PROG-0005, Environment Health and Safety Managed Systems. The Environmental Management System (EMS) defines the requirements for:

Planning the work

- Environmental aspect identification associated with our activities, products and services
- Risk assessment
- Identification of compliance obligations
- Setting objectives and targets

Implementing the Requirements of the Planning Process

- Identifying roles, responsibilities, accountability, and authority
- Setting standards for competence, training and awareness
- Setting standards for control of documents
- Defining requirements for communications to internal and external stakeholders
- Defining procedures and operating criteria associated with OPG significant environmental aspects consistent with the policy, objectives and targets
- Defining an emergency preparedness plan

Monitoring, Measurement, Analysis, and Evaluation of the Planning Process

- A systematic approach for monitoring, measuring, analyzing, and evaluating environmental performance is an integral part of the EMS.
- The extent to which OPG objectives have been met is monitored and measured. The effectiveness of controls; environmental incident monitoring and response (including spills, near misses etc.) and other historical evidence of deficient environmental performance, feed into the analysis and evaluation of process effectiveness.
- An auditing process and procedure has been developed to measure compliance to legal requirements and compliance to the EMS.
- A procedure has been established for the control of records.
- A procedure has been established for incident investigation/management, corrective action and for capturing operating experience from internal and external sources.

Management Review of the EMS and Environmental Performance

OPG management reviews the EMS on planned intervals to ensure its continuing suitability, adequacy and effectiveness. The review includes assessing the opportunities for improvement, the need for changes to the EMS, including the Environmental policy and EMS objectives and targets.

Inputs to the reviews include:

- Actions from previous management review meetings
- Changes in internal and external issues, including compliance obligations, risks and opportunities
- Significant Environmental Aspects
- Environmental Performance and Objectives
- Results of audits and evaluations of regulatory compliance
- Status of Corrective Actions
- Adequacy of resources, including Environmental Policy review
- Feedback from interested parties
- Opportunities for continual improvement

Outputs from management reviews shall be consistent with OPG commitment to continuous improvement and includes decisions and actions related to changes to:

- Environmental policy
- Objectives and targets
- EMS documentation
- Other elements of the EMS

7.5.3 Hazardous Materials

OPG Nuclear programs are in place for the control and safe handling of hazardous materials. The handling of hazardous materials must meet provincial legislation, particularly the Occupational Health and Safety Act and the Environmental Protection Act. Material Safety Data Sheets for hazardous materials are readily available to NSS-PWMF staff as required by Workplace Hazardous Materials Information System (WHMIS) legislation.

The NSS-PWMF contains a variety of non-radiological hazardous materials typically found in industrial buildings, including the following:

(a) Compressed gases: The main gases used include an inert weld cover gas discussed in Section 3.4.4.3.1, and helium, discussed in Section 3.4.4.3.4.

(b) Paint: As described in Section 3.4.4.4, touch-up paint is applied to areas on the DSC that have been affected by the welding process and to scrapes or scuffs on the DSC that may have resulted from handling.

(c) Consumables for maintenance: These include items such as welding rods, adhesives, paints, abrasives, various solvents, and lubricants for equipment.

(d) Janitorial and cleaning supplies.

7.5.4 Fire Protection

OPG has a process in place to ensure adequate fire protection at the nuclear waste facilities. The purpose of this process is to minimize both the probability of occurrence and the consequences of fire at these facilities.

The objective of this process embodies the commitment to:

(a) Protect OPG personnel, contractors, and visitors from the hazards of a fire;

(b) Ensure that a fire does not significantly increase the risk of radiological releases or other hazardous substances to the environment and public. The specific protection of the environment objectives defined in N393 (CSA22) include the following;

  (1) Contain and control hazardous substances; and

  (2) Mitigating and control the releases and effects of hazardous substances, nuclear substances, and radioactive material

As part of design, each NSS facility maintains fire protection documentation regarding the physical details of fire prevention, detection and suppression systems and features of the facility. OPG will design and install all fire protection systems in accordance with CSA N393 (CSA22), NFCC (CCBFC20b), NBCC (CCBFC20a), and all applicable codes and standards referenced therein.

For fire protection during operations, Fire Safety Plans, as required by the NFCC, are prepared for each licensed facility to describe the fire protection measures at the facility and reviewed periodically to ensure they remain current. The Facility Fire Safety Plan identifies the specific type of fire detection and suppression systems located in various buildings or areas. Procedures and maintenance call-ups are in place to ensure the operations, maintenance, testing, and inspection of fire related activities and equipment meet the requirements of CSA N393, the NFCC, the NBCC and all applicable codes and standards referenced therein. All staff involved in the responsibilities of the fire protection procedure are qualified to the requirements identified in the relevant OPG governance.

Pre-Fire Plans and Evacuation plans are prepared for all NSS facilities, identifying the building floor layout, and location of fire related information (e.g., location of extinguishers, fire hoses, fire exits, etc.) These are reviewed periodically and used by the emergency responders in the event of a fire.

Emergency response services at the NSS-PWMF Phase I are provided by the Pickering NGS as identified in the Partnering Agreements between Pickering Nuclear and NSS-PWMF, as stipulated in the Memorandum of Understanding with the City of Pickering. NSS-PWMF Phase II has services provided by Pickering Fire Services, as stipulated in the Memorandum of Understanding with the City of Pickering.

Fire drills are regularly scheduled to test occupant responses and test the capabilities of the Pickering Fire Services and ERT as per CSA N393 and the NFCC.

8.0 NUCLEAR MANAGEMENT SYSTEM

OPG has an on-going program in place to assure that the required quality of products and services is properly defined and effectively achieved in activities associated with OPG's nuclear facilities. This program provides a disciplined approach in determining, communicating, and attaining the required level of safety, reliability, maintainability, environmental protection, and performance for OPG's nuclear waste facilities. The program defines requirements for the work to be done and provides for the integration and co-ordination of pertinent activities.

The Nuclear Management System (N-CHAR-AS-0002), in conjunction with Nuclear Waste Management program document (W-PROG-WM-0001), the Nuclear Safety and Security Policy (N-POL-0001), Environment Health and Safety Managed Systems (OPG-PROG-0005), as well as the Conduct of Engineering program document (N-STD-MP-0028), establish the overall managed system requirements for the management and operation of OPG's nuclear waste facilities.

The Nuclear Waste Management program (W-PROG-WM-0001), the Conduct of Engineering program (N-STD-MP-0028), and the Decommissioning program (W-PROG-WM-0003) encompass engineering and design, procurement, manufacturing, construction and installation, commissioning, operation, decommissioning, and records keeping. These programs provide overall direction in the administration of the Nuclear Sustainability Services (NSS) and establish the Nuclear Management System requirements with which employees must comply. These programs apply to all organizational units in NSS and external groups working on NSS items or areas. These programs also establish the overall systems for NSS and incorporate, directly or by reference, the controls necessary to meet the requirements of CSA N286-12 and ISO 14001, as appropriate to NSS facilities and activities.

The Engineering Change Control program (N-PROG-MP-0001) ensures all modifications to SSCs, including software and engineered tooling, are planned, designed, installed, commissioned, placed into service, or removed from service within the safe operating envelope (SOE) or safety and design envelope (SDE), design basis, and licensing conditions. The program also complies with CSA N285 and CSA N286 code editions as referenced in the facility operating licenses.

The Information Management program (OPG-PROG-0001) establishes a set of standards and procedures for the management of OPG's information throughout its life-cycle, to ensure consistent and appropriate use.

The Items And Services Management program (OPG-PROG-0009) supports Supply Chain activities across all of OPG. The program includes processes which ensure that procurement is planned, and that purchased, stored, and issued items, along with purchased services meet appropriate and applicable design and quality requirements.

The Independent Assessment program (N-PROG-RA-0010) provides independent assessment (internal and external) processes to perform a comprehensive and critical evaluation of all activities affecting OPG nuclear facilities, including the NSS-PWMF. These independent audits and assessments cover the overall Nuclear Management System requirements. The independent audits and assessments, performed by the Nuclear Oversight organization, monitor compliance with governing codes, standards and technical requirements, and confirm that program requirements are being effectively implemented. Independent audit and assessment results are documented, reported to and evaluated by a level of management having sufficient breadth of responsibility to assure actions are taken to address the findings thereof.

Additional oversight of OPG's nuclear waste activities is provided through self-assessment and the corrective action program (N-PROG-RA-0003). In particular, the corrective action program assures that adverse conditions are identified, documented, reported, evaluated, and corrected in a timely manner.

8.1 Employee Training

OPG has established and implemented a training program, N-PROG-TR-0005, following a systematic approach to training, to provide personnel with the knowledge and skills necessary to meet the performance requirements of their jobs (OPG17, OPG22h, OPG23a).

The objective of the training program is to provide sufficient qualified personnel to operate and maintain the nuclear waste facilities in a safe and efficient manner and to ensure compliance with applicable regulations, operating licences and established operational limits.

The skill and knowledge of personnel is developed through initial training and job performance is maintained and improved through periodic requalification and refresher training.

The Integrated Fleet Management is accountable for the training that the NSS personnel receive. The primary tool to manage the information related to the training program is the Training Information Management System.

The overall training program and its individual elements are periodically assessed. The training program is updated to reflect the results of program evaluations, internal and external operating experiences and changes to equipment, procedures, and regulations.

9.0 COMMUNITY RELATIONS AND PUBLIC INFORMATION PROGRAM OBJECTIVES

OPG is committed to being an ethical and credible company in its relationships with employees, suppliers, customers and the public with whom it does business and in the communities in which it operates.

In mid-2012, OPG reviewed the issuance of RD-99-3 *Requirements for Public Information and Disclosure.* To ensure full compliance, OPG conducted a review of existing communication protocols and polices in place and revised procedures to ensure compliance. To further guide our actions, OPG has developed a Nuclear Public Information Disclosure and Transparency Protocol. This protocol sets forth our commitment to high standards of information disclosure and reporting. It also ensures information is provided in a timely manner to the communities in which we operate, related to operations, health, safety, security and environment.

For operational status changes or unscheduled operations that may cause public concern or media interest, OPG follows a Public Interest Notification protocol to notify key community stakeholders in a timely manner. To support this protocol, Corporate Affairs maintains a duty on-call position 24/7 to manage this requirement.

OPG regularly and proactively provides information to the public on its on-going facility activities; public and environmental impact; and radioactive materials transportation program and consults with key stakeholders and the public on future planned activities.

OPG's community relations and public information program has been recognized as a strength by national and international utility peers. OPG benchmarks current practices amongst other industries to ensure continuous performance improvement.

9.1 Program Highlights and Accomplishments

Each year a community engagement and consultation plan is developed to support OPG's business strategy to build community awareness and support of OPG and site operations. The objective of the strategy is to:

a) Reinforce and maintain a positive reputation based on timely, open and transparent communications and information disclosure.

b) Ensure accessible clear, consistent, and accurate information is provided.

c) Ensure material information is disclosed in accordance with applicable legal and regulatory requirements.

OPG's community programming reflects our value and commitment to the key tenets of safety, performance, and environment and as a strong corporate citizen within our host community.

OPG's Corporate Affairs staff manages external communications and relationships between the waste facilities and our host communities by fostering healthy, open dialogue and sustainable partnerships with community stakeholders, including government, media, business leaders, educational institutions, interest groups, and community organizations. OPG ensures transparent disclosure of our operations and potential impacts, both positive and negative that may occur as a result of our operations.

Additionally, we target key audiences of the public with a variety of activities to encourage them to interact with OPG staff and to visit our Information Centres. We view public education as a key component of our program: Two-way dialogue with our community stakeholders and residents is facilitated through personal contact, community newsletters, speaking engagements, educational outreach, the Internet and many other products and programs.

Over the current licensing period, OPG has conducted an extensive and responsive community relations program for OPG/NSS as a whole and specifically for the NSS-PWMF.

OPG regularly provides milestones and regular waste management updates to key stakeholders. Presentations are regularly made at the Pickering Community Advisory Committee and Durham Nuclear Health Committee. In addition, presentations and informal meetings are held with local elected officials and community leaders a number of times each year to provide updates on performance and other activities taking place both at the stations and waste management facilities.

To increase the understanding of nuclear waste management, community presentations and facility tours have been provided at the NSS-PWMF, to key stakeholder groups, media and community and community-of-interest leaders, with strict oversight by OPG nuclear security. Any and all station tours also include information and discussion on the management of nuclear waste.

OPG's Corporate Citizenship Program, provides financial support for community-based programs to foster sustainable partnerships and to benefit the social fabric of a community.

Nuclear waste management is regularly discussed at community updates with the public as part of the nuclear generation station projects.

OPG also receives and manages inquires raised by stakeholders and the public and has a managed process in place to respond and track actions and resolution of issues. This record is used as a gauge to monitor public attitudes and potential issues.

A number of media communication products are produced. *Pickering Neighbours* is distributed quarterly to approximately 110,000 residents and businesses in the Municipality of Pickering. Periodically, nuclear waste management articles are highlighted in these editions.

Quarterly Performance Reports are produced for NSS's waste operations, which include performance of the NSS-PWMF, and are shared with key stakeholders and available on the OPG website.

The OPG corporate website www.opg.com provides online access to information on the Nuclear Waste Management program and projects. All waste information and videos are available on the website and provides an opportunity for users to email questions, comments and concerns to OPG staff. Updates have been made to the Nuclear Sustainability Services – Waste Management portion of the OPG website to identify opportunities for the community to meet us face to face.

Numerous presentations and discussions are held each year with interested municipalities and communities in which our waste transportation program travels. Presentations are made to first responders to explain the safety of our nuclear waste transportation program and emergency response procedures in place.

9.2 Indigenous Relations

OPG has an Indigenous Relations policy and a team of Indigenous Relations Advisors that support:

- Outreach and engagement with Indigenous Nations and communities;
- Capacity building within communities;
- Awareness of employment/business contracting opportunities; and
- Settlement of past grievances or legacy issues.

Building positive relationships with Indigenous Nations and communities is important to OPG nuclear with respect to current operations and the planning of new projects. We recognize close consultation with community members and leaders is an essential part of the process. OPG Nuclear continues to engage in active dialogue with Indigenous Nations and communities on a number of issues and decisions related to our nuclear operations. Discussions and information sharing are undertaken to build long-term mutually beneficial working relationships with Indigenous Nations and communities proximate to our facilities.

10.0 PRELIMINARY DECOMMISSIONING PLANNING

The NSS-PWMF is composed of two sites. The NSS-PWMF Phase I site is located within the Pickering NGS protected area, southeast of Pickering NGS Unit 8, adjacent to the east side of the station security fence. The NSS-PWMF Phase II site is located approximately 500 m north-east of the site in the East Complex, within a distinct “protected area”, but still within the existing Pickering NGS site boundary. After permanent shutdown of the Pickering NGS, all used fuel from the station will be loaded into DSCs and transferred to the NSS-PWMF until a long term used fuel management strategy is implemented. The NSS-PWMF will remain in operation after the cessation of power production at Pickering NGS, and decommissioned in a subsequent time frame.

The Preliminary Decommissioning Plan (PDP) [OPG22d] describes Ontario Power Generation’s (OPG’s) current plan for the decommissioning of NSS-PWMF. It has been prepared in accordance with the requirements of the Class I Nuclear Facilities Regulations (SOR/2000-204 – Clause 3(k) [CNSC00a]) as well as the requirements of the PWMF licence condition handbook [OPG20a]. It has also been written to meet the requirements of CNSC Guide G-219 - Decommissioning Planning for Licensed Activities [CNSC00], and CSA N294:19 – Decommissioning of Facilities Containing Nuclear Substances [CSA19]. OPG updates the PDP every five years (or sooner if required by the CNSC) in support of the Financial Guarantee submission. The requirements of current standards and regulatory documents, as well as any relevant domestic and international experience obtained in the previous five years will be incorporated into the revision.

The PDP covers decommissioning activities performed after the used fuel and radioactive waste have been removed from the site and the decision has been made to shut down the NSS-PWMF for decommissioning until the site is available for other OPG uses. The objective is to retire the facility in a manner that ensures the safety of workers, the general public, and the protection of the environment. The PDP demonstrates that decommissioning is feasible with existing technologies and it provides a basis for estimating the cost of decommissioning. The PDP also includes schedules and cost estimates based on the assumptions that form the basis for the plan.

Previously the waste generated from decommissioning of NSS-PWMF was assumed to be disposed at the Low and Intermediate Level Waste (L&ILW) Deep Geological Repository (DGR) in Kincardine. Early in 2020, the L&ILW DGR Project was cancelled. OPG is exploring options and remains committed to permanent and safe disposal of its operational waste as well as future decommissioning waste. In March 2023, Natural Resources Canada (NRCan) issued the final modernized Policy for Radioactive Waste Management and Decommissioning. The NWMO’s final Integrated Strategy for Radioactive Waste (ISRW) recommendations were submitted to the federal government in June 2023 for their consideration. The ISRW recommended that Low Level Waste (LLW) be disposed in Near Surface Disposal Facilities implemented by waste generators and waste owners, and that Intermediate Level Waste (ILW) be disposed in a central DGR implemented by the Nuclear Waste Management Organization (NWMO). In view of these recommendations, OPG is initiating activities

Under the NWMO's Adaptive Phased Management (APM) program, established by the federal government, the long-term facility for High level waste or Used Fuel is expected to be in service no earlier than 2043. After the APM long term facility is in service, used fuel will start to be transferred from the interim storage location at NSS-PWMF to the facility.

OPG continuously monitors and incorporates best practices from the industry and has a high degree of confidence that the current plans are appropriate and sufficient.

11.0 SAFETY AND ENVIRONMENTAL EFFECTS SUMMARY

11.1 Radiological Safety – Normal Operations

11.1.1 Gamma Radiation Dose Rates

11.1.1.1 Dose Rates Inside the Pickering Waste Management Facility

Detailed discussions of dose rates in the NSS-PWMF are set out in Chapters 4 and 6 of this report. Under the NSS-PWMF RP program, dose rates in work areas at the NSS-PWMF are maintained to levels that are ALARA.

Dose rates in routinely occupied areas inside the NSS-PWMF are at or near ambient background radiation levels. Although higher dose rates are normally found when walking between stored DSCs and DSMs, they are well below calculated estimates. Workers employ survey monitoring, and appropriate dosimetry is required to be carried out in areas having potentially elevated dose rates.

11.1.1.2 Dose Rates Outside the Pickering Waste Management Facility

As discussed in Section 4.3.2.3, the maximum doses from DSC Storage Buildings 1-4, when filled with 1692 DSCs, are estimated at about 2.92 µSv/year at the Pickering NGS site boundary based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the maximum dose is estimated to be 1.41 µSv/year based on 1,000 hours occupancy. This is below the facility administrative dose target at the Pickering NGS site boundary and is expected to be indistinguishable from the variations in natural background.

Compared with the average annual background dose rate of about 1,400 µSv/year in the vicinity of the Pickering NGS site (OPG22f) and the CNSC regulatory dose limit of 1,000 µSv/year, these dose rates are very small.

The total annual background doses to members of the public will vary considerably, from individual to individual, according to lifestyle, housing type and location, occupation, and medical requirements. The typical resident in the vicinity of the Pickering NGS site receives a yearly average effective dose commitment of 565 µSv from radon and thoron daughters, 152 µSv from terrestrial radiation, 306 µSv from internal radionuclides, and 293 µSv from cosmic radiation. These doses are received independent of the PNGS and NSS-PWMF facilities. In addition to naturally occurring radiation, the public also receives about 70 µSv effective dose from man-made sources such as nuclear weapon test fallout and exposures from technological processes and consumer products and services. About 1,100 µSv on a per capita basis is received from medical exposures (OPG03b).

11.1.2 Radiological Contamination

11.1.2.1 Surface Contamination

The external surfaces of DSCs and DSMs in storage are routinely demonstrated to be free from loose contamination. Similarly, there is no loose contamination in the DSC storage buildings or the RCS area.

There have been no contamination incidents in the NSS-PWMF office area or coffee shop.

Minor incidences of loose contamination have occurred related to hoses connected to the DSC vacuum drying system. Dedicated hoses have been assigned to DSC processing systems, to prevent cross-contamination between systems that are directly connected to the internal DSC cavity and the helium leak detection system.

Contamination monitoring is performed during UFDS operations.

11.1.2.2 Airborne Releases and Liquid Effluents

Active ventilation and active drainage systems are provided in the DSC processing building. Active ventilation exhaust is filtered using HEPA filtration to remove radiological particulates. Monitoring of the stack sampler has routinely demonstrated no significant levels of particulates in the active ventilation exhaust.

The contents of the holding tanks in the active drainage system are monitored and periodically pumped to the existing active liquid waste treatment system at Pickering NGS. Monitoring results show no significant levels of activity in active drainage effluent transferred to the station system.

Under normal operating conditions, no airborne emissions are expected from the seal-welded DSCs during on-site transfer to or storage in the DSC storage buildings. As each DSC is fully vacuum-dried, helium back-filled, seal welded, and leak-tested at the DSC processing building, there will be no liquid emissions expected from DSCs.

Monitoring of the RCS area catch basins has demonstrated that there are generally no detectable levels of activity in surface water runoff from the RCS area.

11.2 Radiological Safety – Abnormal Operating Conditions and Credible Accidents

Under postulated accident scenarios presented in this report, dose rates and emissions from the NSS-PWMF are within allowable limits and risk to the public, the workers and the environment is negligible. The combined effects of credible accident scenarios are not expected to exceed the dose rates resulting from bounding scenarios for single effects.

11.3 Cumulative Environmental Effects

Radiation dose resulting from the operation of the NSS-PWMF is very low in comparison to background levels and is expected to remain well below CNSC limits. The NSS-PWMF has been performing satisfactorily for storage of radioactive materials.

Estimated releases during normal operations, based on conservative assumptions, are negligible.

Environmental Assessments have been conducted for the DSC processing building and DSC Storage Building 1; DSC Storage Building 2; and DSC Storage Buildings 3 and 4. The assessments documented the results of the analysis of potential effects associated with the staged construction and operation of the NSS-PWMF as follows:

- Analyses of non-radiological effects on the natural environment;
- Assessment of the potential radiological environmental effects on humans and the non-human environment;
- Public health effects due to both radiological and non-radiological aspects of the facility;

- Socio-economic impacts;
- Environmental effects of alternative options;
- Cumulative effects due to facility construction and operation in combination with the effects of other existing and planned activities; and
- The significance and uncertainty of the assessed environmental effects.

The Environmental Assessments concluded that the staged construction and operation of the NSS-PWMF is not expected to result in any significant incremental effects on the local or regional environment. The assessment indicated no adverse cumulative effects due to conventional pollutants and radiation dose on the valued ecosystem components (VECs). It concluded that cumulative radiation doses to humans from all sources at the Pickering NGS site were expected to remain very low in comparison to background levels, and that the cumulative radiation effects on the most sensitive non-human species would be several orders of magnitude less than the threshold values documented in the literature.

The NSS-PWMF is, therefore, considered to have no significant adverse cumulative environmental effects. Also, adverse effects to the environment are not expected to occur during NSS-PWMF operations.

11.4 Human Performance and Human Factors

The human performance program establishes a systematic framework for human performance management. The program is specifically designed to achieve higher levels of Nuclear, industrial and environmental safety, higher unit reliability, and reduced operating costs through event-free operation. The goal of the program is to continually reduce human performance events and errors in pursuit of recognition as an event-free operator via consistent application of event prevention tools.

This program addresses human performance management and improvement by improving human performance through individual behaviours (all employees and contractors), organizational processes, and management and leadership behaviours.

Human Factors provides inputs and recommendations to the building and systems designs using a tailored approach that is based on industry accepted Human Factors Engineering processes, standards and guidelines. The aim is to ensure that the storage buildings, their equipment, tasks and work environment are compatible with the sensory, perceptual, cognitive and physical attributes of the people who are operating and maintaining the facility. Human Factors in design is applied to all facility modifications. Radiological and non-radiological issues are considered for improvement of the general operability, maintainability and safety of the facilities.

The training program and practices to support the provision of qualified staff is discussed in Section 8.1.

12.0 CONCLUSIONS

The information presented in this Safety Report addresses the health and safety of workers and the public, and the protection of the environment. It shows that the NSS-PWMF is a safe undertaking. Specifically:

- The public dose from all NSS-PWMF operations is less than the administrative dose target. The public dose includes over fifteen years of historical measurements of radiological releases and chronic releases from DSCs and DSMs. These doses are negligible compared to the total background dose of 1,400 µSv/year in the vicinity of the Pickering NGS site. NSS-PWMF's contribution to public dose is small.

- Accident analyses show that hypothetical public doses resulting from NSS-PWMF credible accident scenarios are below the 1 mSv public dose limit.

- No significant adverse effects on the natural or the social environment have been identified due to operation of the NSS-PWMF.

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14.0 GLOSSARY

APM	Adaptive Phased Management
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
BNPD	Bruce Nuclear Power Development
CANDU	Canada Deuterium Uranium (trademark of AECL)
CCW	Condenser Cooling Water
CEX	Controlled Experiment
CN	Canadian National (rail line)
CNEP	Consolidated Nuclear Emergency Plan
CNSC	Canadian Nuclear Safety Commission
CRNL	Chalk River Nuclear Laboratories
C/S	Containment/Surveillance
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DGR	Deep Geologic Repository
DNGS	Darlington Nuclear Generating Station
DRL	Derived Release Limit
DSC	Dry Storage Container
DSM	Dry Storage Module
EA	Environmental Assessment
ECA	Environmental Complication Approval
EMP	Environmental Monitoring Program
EMS	Environmental Management System
EPD	Electronic Personal Dosimeter
ERB	Easily Retrievable Basket
ERT	Emergency Response Team
GMAW	Gas Metal Arc Welding
GSS	Guaranteed Shutdown State
GTA	Greater Toronto Area
GTAW	Gas Tungsten Arc Welding
HAZ	Heat Affected Zone
HEPA	High Efficiency Particulate Air
HSMS	Health and Safety Management System
HVAC	Heating, Ventilation, and Air Conditioning
IAD	Inactive Drainage
IAEA	International Atomic Energy Agency
IFB	Irradiated Fuel Bay
IGLD	International Great Lakes Datum
ILW	Intermediate-Level Waste
ISO	International Organization for Standardization
LAN	Local Area Network
L&ILW	Low and Intermediate Level Waste
LSFCRP	Large Scale Fuel Channel Replacement Program
M	Magnitude

MDA	Minimum Detectable Activity
MECP	Ministry of the Environment Conservation and Parks
NBCC	National Building Code of Canada
NCSA	Nuclear Safety and Control Act
NEW	Nuclear Energy Worker
NFCC	National Fire Code of Canada
NFPA	National Fire Protection Agency
NGS	Nuclear Generating Station
NPT	The Treaty on the Non-proliferation of Nuclear Weapons
NRCan	Natural Resources Canada
NSS	Nuclear Sustainability Services
NWMO	Nuclear Waste Management Organization
NSS-DWMF	Nuclear Sustainability Services – Darlington Waste Management Facility
NSS-PWMF	Nuclear Sustainability Services - Pickering Waste Management Facility
NSS-WWMF	Nuclear Sustainability Services - Western Waste Management Facility
OH&S	Occupational Health and Safety
OPG	Ontario Power Generation
PA	Public Address
PAUT	Phased Array Ultrasonic Testing
PDP	Preliminary Decommissioning Plan
PHTS	Primary Heat Transport System
PM	Particulate Matter
P-mSv	Person-millisievert
PNGS	Pickering Nuclear Generating Station
POR	Points of Reception
RCS	Retube Components Storage
RP	Radiation Protection
SCC	Stress Corrosion Cracking
SSC	Structures, Systems, and Components
SOR	Statutory Orders and Regulations
TAB	Turbine Auxiliary Bay
TIBL	Thermal Inversion Boundary Layer
TLD	Thermoluminescent Dosimeter
TMB	Training and Mock-up Building
TSP	Total Suspended Particulates
UFDS	Used Fuel Dry Storage
UPS	Uninterruptible Power Supply
VEC	Valued Ecosystem Component
VOC	Volatile Organic Compound
WHMIS	Workplace Hazardous Materials Information System
WPCP	Water Pollution Control Plant

15.0 DRAWINGS

- NK30-D0A-10200-0001, Pickering NGS Building Development – Site Plan.
- 00104-DRAW-79171-10001, Dry Storage Container (DSC) – General Arrangement.
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- 92896-DRAW-29651-10075, Pickering Waste Management Facility Dry Fuel Storage Building #3: Architectural Site Plan.
- 92896-DRAW-29651-10065, Pickering Waste Management Facility Dry Fuel Storage Building #3 & 4: Overall Ground Floor Plan.

Appendix A: Illustrative Photographs

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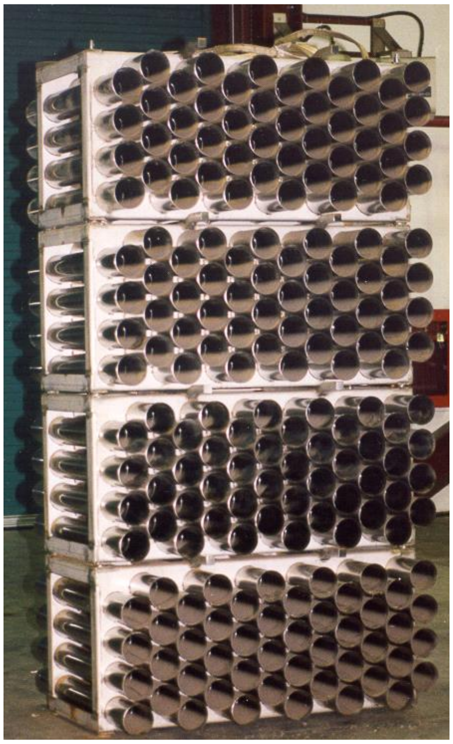

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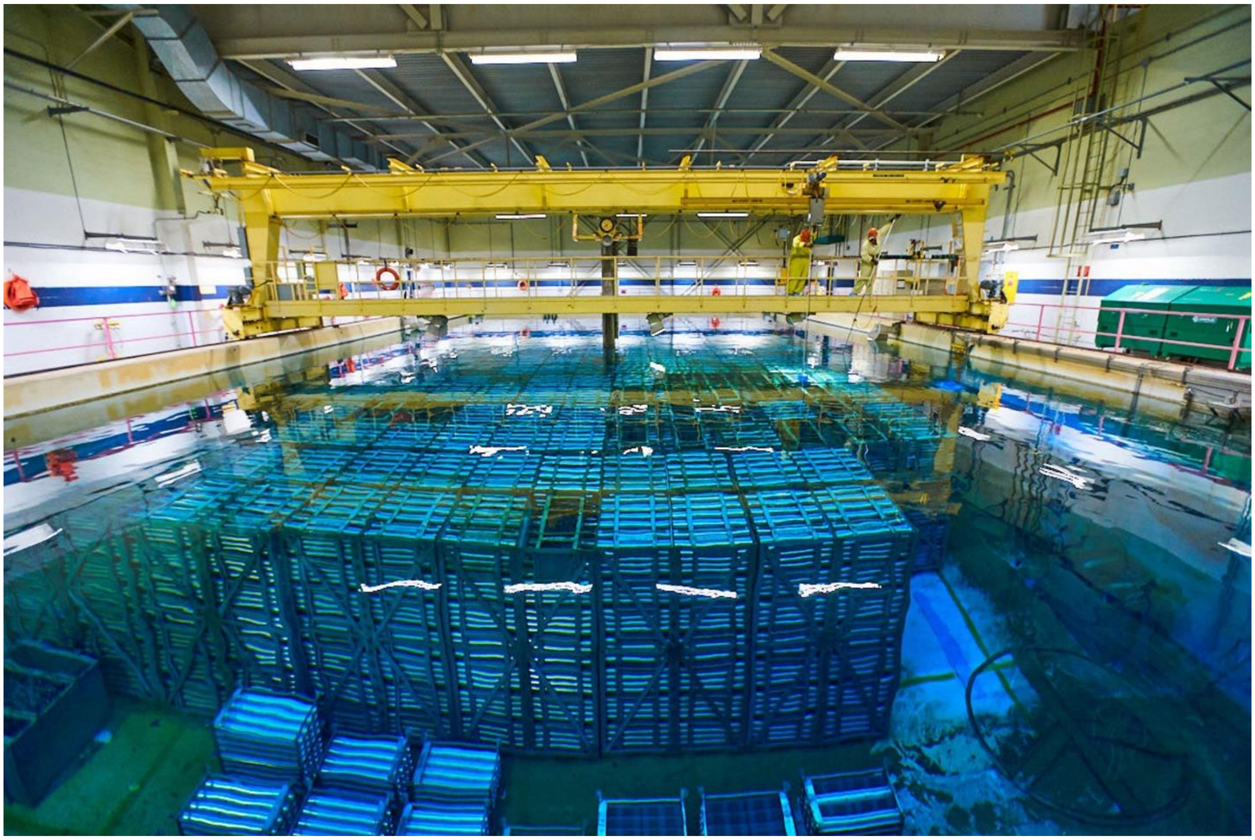

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B.1.0 PURPOSE

A safety assessment of the Pickering Waste Management Facility (NSS-PWMF) and the Used Fuel Dry Storage (UFDS) process is required to support the operating licence for the NSS-PWMF. The purpose of the safety assessment is to ensure the protection of the environment, and the health and safety of workers and the public by maintaining releases and exposures within acceptance criteria with the goal being the prevention of unreasonable risk.

This safety assessment encompasses:

- The on-site transfer of loaded Dry Storage Containers (DSCs) from the Irradiated Fuel Bays (IFBs) at the Pickering Nuclear Generating Station (NGS) A and B to the DSC processing building at the NSS-PWMF;
- The processing of DSCs inside the DSC processing building;
- The transfer of seal welded DSCs from the DSC processing building to a DSC storage building at either the NSS-PWMF Phase I site or Phase II site, the transfer of seal welded DSCs from DSC storage buildings at the NSS-PWMF Phase I site to NSS-PWMF Phase II site; and
- The storage of DSCs within the DSC storage buildings at either NSS-PWMF Phase I or Phase II sites.

The loading, decontamination, and vacuum drying of DSCs at the station IFBs prior to transfer are part of the Pickering NGS safety reports and licensing basis and are not covered by this document.

B.2.0 ASSESSMENT BASIS

B.2.1 Used Fuel from the Pickering Nuclear Generating Station Reactors

The Pickering NGS reactors use 28-element Canadian Deuterium Uranium (CANDU) fuel bundles. Approximately 3,000 bundles are discharged each year from each of the reactors at Pickering NGS. After a minimum of 10 years of cooling[15], fuel bundles may be transferred to DSCs for interim dry storage.

B.2.1.1 Bundle Design

The Pickering fuel bundles are comprised of 28 cylindrical fuel elements, arranged in concentric rings of 16, eight, and four elements and held in place between two zirconium-alloy end plates (see Figure B-1). Each fuel element is made up of natural uranium in the form of sintered pellets of high-density uranium dioxide (UO_2), sealed in zirconium-alloy tubes. The

[15] A minimum of 10 years of cooling can include residence time in fuel channels during GSS followed by IFB storage

sheath of the tubes is designed to minimize sheath strain during operation. The zirconium alloy (called “Zircaloy-4”, referred to herein as “zircaloy”) contains the elements tin, iron, chromium and sometimes nickel. In addition, a braze alloy containing beryllium is used to attach the appendages to the fuel sheath (i.e., bearing pads, inter-element spacers) and CANLUB, a commercial graphite coating, is applied to the inner surface of fuel elements. Table B-1 shows the nominal dimensions for a Pickering fuel bundle.

B.2.1.2 Bundle Burnup[16]

The core is divided into different fuelling regions which may employ different fuelling schemes: four-bundle or eight-bundle shifting. The statistical data available for the exit burnup of Pickering NGS A and B fuel bundles is shown in Table B-1.1. The historical maximum bundle discharge burnup is 595 MWh/kgU. Although the maximum exit burnup of some bundles could be as high as 595 MWh/kgU, statistically those bundles represent a very small percentage of the total number of discharged bundles.

The reference burnup for the reference fuel bundle used for the safety assessment was chosen as [redacted], .

[16] Burnup is the fission energy generated per unit mass of heavy element initially in the fuel (unit: MWh/kgU).

Table B-1.1: Statistical Data for Bundles Discharged from Pickering NGS (B-NWMO21)

Reactor	Period	Bundle Burnup (MWh/kgU) at Different Percentile				
		50th	90th	95th	99th	100th
Pickering A	1980s	192	231	244	286	518
	1990s	202	265	285	318	521
	2000s	198	266	279	295	438
	2010s	210	277	288	303	415
	All	206	273	286	305	521
Pickering B	1980s	188	218	228	249	479
	1990s	193	225	236	259	595
	2000s	190	225	235	253	548
	2010s	196	230	239	256	466
	All	192	226	236	255	595

Notes:
The 1980s period covers 1987-1989 (Pickering A) and 1983-1989 (Pickering B).
The 2010s period covers 2011 to 2018.

B.2.1.3 Reference Used Fuel Bundle

The primary factors that determine the characteristics of used fuel are physical attributes, power and burnup histories, and decay time. These factors are, in turn, influenced by fuelling strategies and reactor conditions. Therefore, for the purpose of performing the safety assessment of UFDS at the NSS-PWMF, a reference fuel bundle has been defined.

B.2.1.3.1 Reference Used Fuel Bundle Dimensions

The fuel bundle used at the Pickering NGS reactors is 495 mm in length, has an outer diameter of 100 mm, and has a nominal total bundle mass of 24.6 kg. The complete dimensions of the reference fuel bundle are given in Table B-1.

B.2.1.3.2 Reference Used Fuel Bundle Age

This safety assessment assumes that used fuel bundles are kept in the Pickering IFBs for 10 years before they are transferred to the NSS-PWMF. In practice, the age of fuel when loaded in DSCs generally exceeds 10 years as operational procedures are employed to select older fuel first. Therefore, the average age of used fuel bundles in dry storage at the NSS-PWMF is greater than 10 years.

B.2.1.3.3 Reference Used Fuel Bundle Burnup

Given statistical data of the fuel discharged from the Pickering NGS reactors, [redacted] has been retained as the burnup for the Pickering reference used fuel bundle. Due to fuelling strategies and reactor operating conditions, the distribution of the fuel bundles within the IFBs is such that the average burnup of the fuel loaded into each DSC is expected to be below the chosen reference used fuel bundle burnup.

B.2.1.3.4 Reference Used Fuel Bundle Power

The total core fission power for a Pickering NGS reactor is 1,744 MW(f). The average bundle power is [redacted] and the average fuel bundle residence time[17] is [redacted] Full Power Days[18].

B.2.2 Nuclide Inventory

The radionuclide inventory for the defined Pickering reference fuel bundle was calculated using the computer code ORIGEN-S (B-ORNL11). Table B-2, Table B-3 and Table B-4 show the radionuclide inventories for the actinides[19],fission products[20] and light elements found in a reference fuel bundle.

B.2.3 Source Term

Table B-5 gives the gamma spectrum for a 10-year-cooled Pickering reference fuel bundle. The computer code ORIGEN-S was used to obtain this gamma spectrum.

B.2.4 Decay Heat

The energy produced by radioactive decay is released from the fuel bundle in the form of heat and radiation. The decay heat released from a Pickering reference fuel bundle for different cooling periods, as calculated using the ORIGEN-S computer code, is shown in Figure B-2. The decay heat for the 10-year-cooled reference fuel bundle is approximately 5.8 W.

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17 Fuel Bundle Residence Time is the time for which a fuel bundle resides in the reactor.
18 Full Power Day is a 24 hours of reactor operation at nominal 100 percent reactor power.
19 The term Actinides is used by the computer code ORIGEN-S to group fuel nuclides and their decay products, plus ^{4}He (since it results from alpha decay).
20 Fission Products are the isotopes produced by fission and their decay products.

B.2.5 Chemical and Physical Characteristics of Radionuclides in Used Fuel

The location of radionuclide species in a fuel element depends on their chemical and physical behaviours and where they were produced. The majority of new radionuclides, such as the actinides (Table B-2) and fission products (Table B-3) in 10-year-cooled used fuel are embedded within the lattice of uranium and oxygen atoms, very close to where they were produced. They substitute for uranium in the uranium dioxide lattice. Activation products that are produced in the zircaloy sheath are primarily trapped by the zirconium alloy and cannot diffuse any significant distance from the site of their formation.

Cracking and grain growth occurs in the ceramic fuel pellets at the high temperatures and temperature gradients in the reactor. A diagram of the cross section of a ceramic fuel pellet is shown in Figure B-3. About 2 percent of the gaseous radionuclides, or those that are volatile at fuel irradiation temperatures, are released to the cracks in the pellets and to the gap between the pellets and the fuel sheath. A further 6 percent segregates to the grain (crystal) boundaries within the uranium dioxide pellets. Laboratory studies have been used to predict the quantities of radionuclides in the fuel-sheath gap and at the grain boundaries.

The release of radionuclides from failed fuel depends on the volatility of the chemical forms found in the fuel at the maximum fuel temperature and their ability to migrate through the fuel grains. Radionuclides born in the fuel may remain in elemental form or combine with other nuclides, the uranium dioxide fuel, the zircaloy sheath, or excess oxygen. The chemical forms of fission products have been studied in the past to determine their respective contributions to releases from failed fuel (B-OH86).

The maximum sheath temperature of a used fuel bundle with a decay heat of 6.4 W/bundle is not expected to exceed 150°C in dry storage in a helium atmosphere (B-OPG04a). For Pickering, it is expected to be less due to smaller decay heat load (5.8 W/bundle compared to 6.4 W/bundle assumed in the analysis). At these low temperatures, only the volatile fission products would be released should the fuel sheath become damaged. These volatile fission products include krypton-85, tritiated water vapour (HTO), and tritiated hydrogen gas (HT) should there be any tritium in the fuel-sheath gap that has evaded hydriding of the zirconium sheath. Release of these radionuclide vapours upon sheath failure is used for this safety assessment.

As discussed above, actinide series radionuclides along with most of the remaining fission and activation products will be embedded within the uranium dioxide grains or in the zircaloy. A small number of radionuclides are potentially volatile; therefore assessment of their characteristics relative to release is appropriate. The following is a brief review of the most likely chemical forms of the potentially volatile, radiologically significant fission and activation products found in used fuel at 10 years after discharge from the reactor.

Krypton-85 ($T_{1/2}$ = 10.7 y)

Krypton-85 is a member of the noble gas family, and as such is characterized by high ionization potential and small Van der Waals forces. Consequently, the element is chemically inactive, shows low solubility in water and requires very low temperatures and/or high pressures for liquefaction. Krypton has a melting point of -156.5°C, and a boiling point of -153°C.

Krypton-85 exists as elemental gas in the fuel. Due to its high volatility, krypton-85 residing in the fuel-sheath gap and other open voids in the fuel will be available for release almost instantaneously upon sheath failure.

Tritium ($T_{1/2}$ = 12.3 y)

The tritium generated during irradiation of the fuel is located in one of four regions: the fuel matrix (fuel grains), fuel grain boundaries, the fuel-sheath gap, and the zircaloy sheath. During irradiation, the tritium in the fuel grains migrates to the grain boundaries as tritium or HT, or combines with oxygen as oxygen-T (TO) or HTO (tritiated water vapour). The measured diffusion coefficient for tritium in UO_2 is orders of magnitude lower than the corresponding diffusion coefficient for molecular hydrogen (B-SCAR78). A slight increase in oxygen availability in the fuel due to the fission process may shift the equilibrium towards the formation of TO or HTO, both of which are volatile at used fuel temperatures in a DSC. When gas bubbles at the grain boundary reach saturation, tritium, together with other species stored in the bubbles, will be released to the fuel-sheath gap region. The final major chemical forms of tritium in the gap are HT and HTO, both of which are volatile.

These species will diffuse and/or react with the zircaloy sheath resulting in the increase of the sheath inner surface oxide layer thickness and a large tritium pickup by the zircaloy. On reaction with the sheath, the less volatile compound zirconium tritide (ZrT_2) will form, thereby holding up the free inventory of tritium in the cladding. The fractional tritium inventory resident in the fuel-sheath gap is the balance between two competing processes: the fuel-to-gap release process that increases the gap fraction, and the sheath pickup that decreases the gap fraction.

Fuel element puncture tests (B-GOODE70) have determined that the gap fraction is only 10^{-5} of the total tritium in fuel elements. Therefore, the majority of tritium atoms released to the gap will be held up in the cladding (B-GOODE70). Very little tritium will be available for release to the environment from the gap in case of fuel sheath failure.

The final tritium grain-boundary fraction and gap fraction are postulated to be similar to the values for the non-reacting species krypton-85, even though krypton-85 has a cumulative fission yield 100 times higher and a gap fraction about 1,000 times larger.

Carbon-14 ($T_{1/2}$ = 5,730 y)

The activation product carbon-14 is non-volatile in elemental form, and volatile when combined with oxygen as carbon dioxide ($^{14}CO_2$). The majority of carbon-14 atoms produced in the fuel remain within the uranium dioxide grains, but some segregate to grain boundaries within the UO_2 pellets and to the gap between the pellets and the fuel sheath. Much less than 0.1 percent of the carbon-14 is released to the gap and grain boundary regions (B-AECL94). Carbon-14 is also generated in the fuel sheath as a result of activation of the nitrogen impurity in the sheath.

Iodine-129 ($T_{1/2} = 1.6 \times 10^7$ y)

Iodine is one of the most widely studied fission products. Release experiments have demonstrated that only a very small fraction of iodine (i.e., ranging from a few thousandths of one per cent to less than 0.5 percent of the total amount available for release from the fuel) can be in volatile form at the temperature conditions of used fuel when it is discharged from the reactor core.

These experimental results, along with numerous other investigations have concluded that almost all of the iodine in the fuel-sheath gap exists as cesium iodide (CsI) in the condensed and vapour phases. Since the free energy of formation of CsI is strongly negative, the reaction will proceed to completion and essentially all of the iodine will participate in the reaction. Cesium iodide is semi-volatile with a boiling point of 1,280°C. At lower temperatures, condensable fission products such as CsI are shown to undergo complete deposition onto metal surfaces in the immediate proximity of the fuel. If iodine is in contact with the zircaloy sheath, the volatiles ZrI_3 and ZrI_4 may form in very small concentrations. Diatomic iodine (I_2), a stable and volatile form of the element, will not be found in fuel as the CsI reaction dominates.

Cesium-134 ($T_{1/2} = 2.1$ y)
Cesium-135 ($T_{1/2} = 2.3 \times 10^6$ y)
Cesium-137 ($T_{1/2} = 30$ y)

Several chemical forms of cesium in fuel are possible as there is a high concentration of cesium isotopes relative to other fission products. The most volatile species of cesium are the monatomic and the diatomic forms, Cs and Cs_2, together with CsI. However, these species behave as non-volatile at 10-year-cooled fuel temperatures.

Strontium-89 ($T_{1/2} = 51$ d)
Strontium-90 ($T_{1/2} = 29$ y)

Strontium has a very stable oxide and is predicted to be entirely in this form in the fuel. Strontium has a high affinity for oxygen, and will readily oxidize to strontium oxide (SrO), a non-volatile compound with a boiling point of about 3,000°C. The oxide is insoluble in the fuel matrix but will form compounds with zircaloy. Because of the extremely low volatility of SrO, no releases of strontium are expected in the event of fuel element damage.

B.2.5.1 Deposits on the Exterior of Fuel Elements

Corrosion products in the Primary Heat Transport System (PHTS) are present in low concentrations in the reactor coolant, and subsequently become radioactive as a result of neutron activation in the reactor core. These are removed from the PHTS coolant by purification systems; however, some of these corrosion products will deposit on the surfaces of the fuel bundles.

Fission products, which originate from defective fuel or the fission of trace levels of uranium present in the reactor coolant, also deposit on the surfaces of fuel bundles.

An experimental study has characterized surface deposits on fuel elements, for the purpose of assessing the radioactive handling hazards associated with the dry storage of fuel bundles (B-CHEN86). The study had two parts: first, surface deposits were removed from nineteen Bruce and Pickering fuel elements using aggressive chemical treatment, and the chemical solutions were analyzed for corrosion products. In the second part of the study, four Pickering used fuel bundles were stored in individual pressure vessels in moist air at 150°C (moisture was provided by a small quantity of water in the bottom of each pressure vessel). In two of the bundles, all of the outer elements except a control element in each were intentionally defected (i.e., the bundles were damaged under relatively cool fuel temperatures, some time after discharge from the reactor core), to determine the effect of storage conditions on defected versus undamaged bundles. After a 30-month storage period, the water from the pressure vessels was analyzed.

Conclusions from analyses are summarized as follows:

(a) Visual examination of all the elements showed no detectable evidence of surface deposits due to corrosion products.

(b) Corrosion deposits removed by chemical treatment of fuel elements were found to consist mainly of iron (concentration range of 11 to 300 mg/m^2). Nickel, copper, and chromium were found in lower concentrations on fuel element outer surfaces. Cobalt-60 was detected in concentrations of about a factor of 1000 lower than iron. Cesium-137 and strontium-90 were the major fission products detected as corrosion products on fuel element outer surfaces.

(c) Analysis of the residues from the pressure vessel solutions after storage indicated the presence of iron as its major constituent. The major fission product on the outer sheath surface, due to corrosion product deposits, was cesium-137. The levels of surface deposit activities measured were consistent with those obtained by chemical treatment of fuel elements prior to storage. Higher levels of fission products were leached from defected fuel than intact bundles (due to leaching of the fuel matrix inside the element).

(d) The activities of cesium-137, cesium-134, strontium-90, and cobalt-60 corrosion products on the outer surfaces of fuel elements after storage are smaller by at least four orders of magnitude compared with the activities of krypton-85 or tritium available for release (from the fuel matrix inside a damaged element). The measured activities of cesium-137 surface deposits on a used fuel bundle were smaller by about six orders of magnitude compared with the total cesium-137 inventory in a 10-year-cooled reference fuel bundle.

(e) No significant activities of iodine-129 were measured in pressure vessel solutions after 30 months of storage.

The study showed that particulate contamination is present on used fuel bundles in the form of corrosion and fission product surface deposits. However, the corrosion products detected in the study are not volatile at the temperatures of 10-year cooled used fuel during handling in the station IFB or during dry storage in a helium atmosphere.

The study used aggressive chemical techniques or 30 months of leaching to remove surface deposits; hence, the conclusions of the study are conservative indications of the relative levels of particulate contamination that could be released into IFB water or be made airborne during dry storage operations.

Activated corrosion and fission products that have been deposited on used fuel external surfaces while in the reactor core, and that have remained adhered under the flow of PHTS coolant and during subsequent storage in the IFB, will be fixed to the outer surfaces of fuel elements and will require either physical abrasion or leaching to become released.

B.2.6 Defective and Damaged Fuel

When a sheath fails in the reactor core at high temperatures, the free inventory of volatile radionuclides residing in the fuel-sheath gap and other open voids in the fuel is released almost instantaneously. Leaching of water-soluble radionuclides from the fuel matrix occurs slowly over a longer-term, while the fuel element remains submerged in the fuel bay.

In the event that reference used fuel (i.e., 10-year cooled) should become damaged during dry storage operations, the only significant radionuclide species that are volatile are krypton-85, tritium and carbon-14. For a fuel element damaged under abnormal operating conditions, it is postulated that 100 percent of the krypton-85 and tritium present in the gap will be released, along with 10 percent of the krypton-85 and tritium present in the grain boundary. 0.1 percent of the carbon-14 activated in the sheath is assumed to be released as $^{14}CO_2$.

Used fuel with visible or known defects is not transferred to DSCs.

B.2.7 Free Inventories Available for Release

For a fuel element within a DSC that is damaged under abnormal operating conditions, it is postulated that the free inventory of tritium and krypton-85 is the radionuclide inventory in the gap between the fuel matrix and the zircaloy sheath, plus 10 percent of the inventory in the grain boundary. The gap fraction is assumed to be 0.0095 for tritium and for krypton-85. The grain-boundary fraction is assumed to be 0.123 for tritium and for krypton-85. These numbers are based on the discussion in Section B.2.5. This is an appropriate assessment of the grain boundary release, at the lower grain-boundary gas pressures that would be associated with the lower temperatures of 10-year-cooled used fuel. For carbon-14, 0.1 percent of the inventory is assumed to be available for release.

Conservatively, for the assessment of airborne emissions, the calculation of tritium , krypton-85 and carbon-14 inventories takes into account the activation in the reactor core of small quantities of impurities present in the used fuel. The adjusted tritium inventory for a Pickering reference used fuel bundle is [redacted] Bq per bundle. The krypton-85 inventory remains essentially unchanged by impurities at [redacted] Bq per bundle. The carbon-14 inventory is [redacted] Bq per bundle.

For each radionuclide, the release from failed fuel can be written as:

$$\mathbf{R_e = (f_{gap} + 0.1f_{gb})I_e}$$

Where,

R_e = radionuclide released per failed used fuel element (Bq/element),
f_{gap} = fraction of radionuclide inventory in the gap,
f_{gb} = fraction of radionuclide inventory in the grain boundary, and
I_e = radionuclide inventory per used fuel element (Bq/element).

For carbon-14, the amount released to the gap and grain boundary is set to 0.1 percent.

B.2.8 Breathing Rates

Breathing rates for the public dose calculations were based on the 95th percentile of the breathing rate for the representative individual. This is consistent with the guidance given in the CSA N288.2-19 standard (B-CSA19a). The breathing rate value from the CSA N288.2-19 standard is 0.31 m^3/h (8.61 × 10^{-5} m^3/s) for an infant and 0.96 m^3/h (2.67 × 10^{-4} m^3/s) for an adult.

The breathing rate for a NEW is based on the ICRP 89 male adult breathing rate performing light exercise (1.5 m^3/h or 4.17 × 10^{-4} m^3/s) (B-ICRP02).

B.2.9 Dose Conversion Factors

As discussed in Section B.2.5, only krypton-85,tritium and carbon-14 associated with the used fuel in the DSC can contribute to airborne emissions. Therefore, the dose assessment methodology and the derivation of an emission source term are focused on releases of krypton-85, tritium in the form of HTO and carbon-14.

The inhalation dose conversion factors for tritium (HTO) used in this assessment are based on the recommended values from the ICRP 119 (B-ICRP12).

- 1.8 × 10^{-11} Sv/Bq for adult; and
- 6.4 × 10^{-11} Sv/Bq for infant.

The immersion skin absorption factor for tritium is taken as 1.5 (B-ICRP95).

The cloudshine dose conversion factors for krypton-85 used in this assessment are based on the recommended values from the Health Canada publication (B-HC99):

- 2.55 × 10^{-16} $Sv.s^{-1}.Bq^{-1}.m^3$ for adult; and
- 3.83 × 10^{-16} $Sv.s^{-1}.Bq^{-1}.m^3$ for infant.

The inhalation dose conversion factors for carbon-14 used in this assessment are based on the recommended values from the ICRP 119 (B-ICRP12).

- 6.2×10^{-12} Sv/Bq for adult; and
- 1.9×10^{-11} Sv/Bq for infant.

The cloudshine dose conversion factors for carbon-14 used in this assessment are based on the recommended values from the Health Canada publication (B-HC99).

- 2.6×10^{-18} $Sv.s^{-1}.Bq^{-1}.m^3$ for adult; and
- 3.9×10^{-18} $Sv.s^{-1}.Bq^{-1}.m^3$ for infant.

B.2.10 Dry Storage Container

B.2.10.1 Dimensions

The DSC dimensions and design details used for this assessment are given in the following drawing:

00104-DRAW-79171-10024, *Used Fuel Dry Storage Ontario Power Generation Long Module Dry Storage Container General Arrangement*.

An illustration of the modified DSC design can be found in 00104-DRAW-79171-10051. The safety assessment results are also applicable to the original long module design, as the MKII constitutes the bounding DSC design.

The direct radiation fields from a loaded DSC are discussed in Section B.6.1.1.

B.2.10.2 Concrete Density

The concrete used as radiation shielding in the DSC design is reinforced high-density concrete. For extra support, a series reinforcing steel bars are inserted into the high-density concrete area. The concrete has a nominal density of 3.5 Mg/m^3; the presence of the reinforcing bars increases the homogenized concrete density to a nominal density of 3.57 Mg/m^3.

B.2.10.3 Fuel Module

Used fuel bundles are placed into storage modules. The modules are made of [redacted] with dimensions [redacted] Each empty fuel module weighs approximately [redacted] and can hold 96 bundles, two bundles in each of 48 [redacted] storage tubes. The fuel storage module design is shown in Figure B-5. Each DSC has the capacity to store four modules.

Freedom of Information and Protection of Privacy Act (FIPPA) S. 18 and Access to Information and Privacy (ATIP) S.13.

B.2.11 Pickering Used Fuel Dry Storage

UFDS at the NSS-PWMF currently consists of industrial-type buildings that are designed and constructed to provide facilities for the safe processing and storage of DSCs. The DSC processing building and storage buildings are not safety-related structures at the NSS-PWMF since they are not credited in the safety assessment as a barrier to the release of radiation.

The NSS-PWMF Phase I site consists of two stages. The NSS-PMWF Phase I Stage I site contains a DSC processing building, including a DSC workshop, offices and utilities, and a DSC storage building to accommodate 185 DSCs; NSS-PWMF Phase I Stage I buildings are a single-story structure with a two-story office area in the DSC processing building. The NSS-PWMF Phase I Stage II is a single-story building that shares the north wall of the NSS-PWMF Phase I Stage I building to form a single structure. The NSS-PWMF Phase I Stage II can accommodate 469 loaded DSCs and includes an area for receiving new empty DSCs. The NSS-PWMF Phase I is located within the Pickering NGS protected area.

The NSS-PWMF Phase II site consists of two DSC storage buildings. The Phase II site is in the eastern half of the Pickering NGS property and is situated on level ground adjacent to the shoreline of Lake Ontario directly east of Pickering NGS B, in the general area of the East Complex and well inside the Pickering NGS property boundary fence.

The NSS-PWMF has been designed to:

- Handle empty DSCs, which includes:
    - Receiving and offloading new empty DSCs and their lids from the DSC delivery vehicle;
    - Performing receipt inspections/preparation of DSCs and preparation/ cleaning the DSC and the lid seal-weld area as required;
    - Housing and support of DSC handling equipment; and
    - Preparing empty DSCs for transfer to the Pickering IFBs.
- Handle DSCs loaded with fuel, including housing equipment and providing support for:
    - Receiving a loaded DSC from the DSC transporter vehicle;
    - DSC lid seal-weld pre-heating;
    - Seal welding the DSC lid to the DSC body;
    - DSC cool down;
    - Inspection of the lid seal-weld;

- Repairing welds as necessary;
- DSC vacuum drying and helium backfilling;
- Welding of the drain plug;
- Inspection of drain plug weld;
- Helium leak detection;
- Painting of scuffs and welds; and
- Application of International Atomic Energy Agency (IAEA) safeguards seals.

- And to provide services and space for personnel and equipment, such as:
    - OPG administrative offices;
    - IAEA personnel office;
    - Guard station;
    - Personnel amenities;
    - Gas container storage areas;
    - Janitor's rooms;
    - Tool storage area;
    - Electrical and mechanical rooms;
    - Equipment maintenance area; and
    - Welding rooms.

The NSS-PWMF Phase I building walls are made of 8 inch-thick (20 cm) ordinary concrete. This is sufficient to meet the dose rate limits at the station security fence, which is located 5 m from the building, and the dose rate target at the station site boundary, which is located 850 m east of the building inland and 420 m east on the lakeside.

Inside the NSS-PWMF Phase I DSC storage buildings, DSCs are placed with a minimum spacing of 0.6 m north to south and 0.2 m in the east to west directions. The narrower side of the containers face the east and west directions. The peripheral rows of containers are 1 m away from the building wall. In the Stage 2 building, a corridor of approximately 10 m width divides the floor layout into east and west blocks. Each block of containers is divided into north and south sections that are separated by a much narrower corridor of approximately

2.1 m wide. For the shielding analysis, the NSS-PWMF Phase I DSC storage buildings were assumed to be filled to nominal design capacity with stored DSCs carrying an average of 10-year-cooled used fuel (occupational dose rate calculations were based on DSCs carrying 10-year-cooled used fuel, since a worker can directly handle a DSC or work near DSCs containing used fuel of this minimum age). Operating experience has shown that the calculated dose rates for the assumed fuel ages were conservative and that actual dose rates are lower. The narrower sides of the DSCs form the peripheral row facing the eastern wall.

The NSS-PWMF Phase II DSC storage building 3 walls are designed with 12 inch-thick (30 cm) ordinary concrete. Storage Building 4 is located immediately south of the Storage Building 3 with a shared wall. Unlike the Storage Building 3, Storage Building 4 does not have precast concrete panels as shielding walls. The Storage Building 3 shielding wall and the DSC storage arrangement in Storage Building 3 and 4 are sufficient to meet the dose rate target at the perimeter fence and to meet the administrative dose target at the site boundary, which is located about 330 m east of the building inland. The dose target at the lakeside exclusion zone boundary, which is located about 340 m south-east, is also met.

Inside NSS-PWMF Phase II DSC Storage Building 3, a corridor of approximately 13 m width divides the floor layout into east and west blocks. For the shielding analysis, the NSS-PWMF Phase II DSC storage building was assumed to be filled with 480 DSCs consisting of 84 DSCs filled with 30-year cooled used fuel and 396 DSCs filled with 25-year cooled used fuel. The analysis assumed the younger DSCs were placed on the east side of the building to provide a conservative dose rate calculation at the site boundary. Operating experience has shown that the calculated dose rates are conservative with respect to normal operation. Similar to DSC Storage Building 3, in DSC Storage Building 4 a corridor divides the floor layout into east and west blocks. For the shielding analysis, Storage Building 4 was assumed to be filled with 624 DSCs. The youngest DSCs are placed on inner area of the Storage Building and the older DSCs are placed on peripheral locations to minimize dose rates from the building.

B.3.0 NORMAL OPERATION ASSESSMENT METHODOLOGY

The methodology used to calculate the dose rates inside the NSS-PWMF and outside the facility are described below.

The computer code ORIGEN-S (B-ORNL11) together with a burnup-dependent library for natural uranium oxide fuelled CANDU reactor 28-element bundle design (B-OPG22a) was used to determine the radioactivity content of the irradiated fuel. After calculating the buildup of actinides and fission products during a specified irradiation period, the code then calculates their inventory as a function of the time after discharge from the reactor (i.e., cooling time). The photon and neutron spectrum as a function of the cooling time is also calculated by the code. Verification and validation of the ORIGEN-S code has been performed (B-OPG01).

The gamma dose rate calculations for a single long module DSC (MKII) and for dose points inside and outside of the NSS-PWMF were carried out using the Monte Carlo code (MCNP

6.1[21]) (B-LANL11). This code is capable of rigorously simulating the stochastic nature of gamma, neutron and electron transport by explicitly modelling the physical nature of their travel through space and their interactions with matter. The MCNP code captures gamma dose rate contributions from irradiated fuel in large arrays of storage containers in different storage buildings to provide an accurate, integrated shielding analysis, taking into account all gamma radiation dose pathways. This type of model has been used in the shielding analysis for the NSS-PWMF, NSS-DWMF and NSS-WWMF.

The MCNP code applies the 'Monte Carlo' method of analysis, simulating photon histories explicitly in the modelled geometry. A characteristic of this method is that all of the results ('detector tally' results of dose rate values at the specific locations) are always statistical quantities in the form of an estimated mean value and an estimated standard deviation. MCNP utilizes a sufficient number of 'photon histories' that the statistical uncertainty is very small. In cases where the estimated statistical uncertainties in the MCNP results are greater than five percent, special care has been taken in the consideration of the results. In such cases, the margin between the target dose rate and the computed value was considered related to the estimated uncertainty, and a judgement was made as to whether the values can be accepted as not exceeding the targets. This methodology is documented in Reference B-OPG17b.

The methodology used to develop the MCNP model of UFDS buildings and DSCs at waste facilities has been validated/benchmarked (B-COG20a) through simulation of TLDs surrounding the UFDS buildings. The results show that the predictions using the MCNP model, which is based on the reference methodology, is conservative by 35-60%. In the normal operation assessment, a bounding fuel source term is used for DSCs such that the decay of the fuel is minimized and the burnup is maximized. The overall outcome of the normal operations safety assessment is that the predicted dose rates from the UFDS building loaded to full nominal design capacity at the NSS-PWMF site and Pickering NGS site boundary are considered to be conservative and below acceptance criteria.

This conservative outcome justifies the application of this methodology and the use of MCNP to model large arrays of DSCs loaded with irradiated fuel in normal operations safety assessments.

Doses to individual members of the public from releases due to the normal operation of NSS-PWMF were calculated using the IMPACT code (B-EcoMetrix18), which follows the guidance given in the CSA N288.1-14 standard. Although the code was based on the 2014 revision of the standard, the calculation results for the NSS-PWMF safety assessment are in compliance with the latest revision of the standard CSA N288.1-20 (B-CSA20).

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[21] MCNP® and Monte Carlo N-Particle® are registered trademarks owned by Los Alamos National Security, LLC, manager and operator of Los Alamos National Laboratory

B.4.0 MALFUNCTIONS AND ACCIDENTS ASSESSMENT METHODOLOGY

Based on the distinct stages of the Pickering UFDS process, this assessment was divided into the following operational stages:

(a) On-site transfer operations;

(b) Operations inside the DSC processing building; and

(c) Storage.

B.4.1 Identification of Initiating Events

For each stage of the UFDS operations, release of radiation can occur due to the failure of the systems and components being used.

There are two general categories of initiating events that can result in abnormal conditions or accidents: internal events and external events.

- **Internal events** are abnormal conditions generated within the area of operation as a result of equipment failure or human error.

- **External events** are natural and man-made phenomena originating outside the area of operation that may, as a result, cause an internal event or perhaps even multiple, wide-spread events.

The list of initiating events identified for the on-site transfer stage of UFDS operations are given in Table B-6; at the DSC processing building in Table B-7; and at the DSC storage building in Table B-8.

B.4.2 Event Frequency

As per the definition of a credible abnormal event in CSA N292.0, if the frequency of occurrence estimated for any postulated accident scenario is less than 10^{-6} events per year (B-CSA19b), it is considered incredible.

B.4.3 Screening of Events

Each event was screened following the OPG screening criteria for internal hazards (B-OPG18b) and external hazards (B-OPG18a) to establish if it could result in any radiological impact to the public, the workers and/or the environment. Design provisions and procedural measures that could prevent the event or mitigate its consequences were also considered.

B.4.4 Dose Consequences

The guidance documented in CSA N288.1-20 (B-CSA20) and CSA N288.2-19 (B-CSA19a) was used in performing the safety assessment[22].

B.4.4.1 Public

Doses to individual members of the public from the postulated chronic release from DSCs during processing were calculated using the IMPACT code (B-EcoMetrix18), which follows the guidance given in the CSA N288.1-14 standard. Although the code was based on the 2014 revision of the standard, the calculation results for the PWMF safety assessment are in compliance with the latest revision of the standard CSA N288.1-20 (B-CSA20).

The potential doses to individual members of the public were calculated using the ADDAM code (B-COG11). The ADDAM code considers the inhalation, cloudshine, and groundshine pathways. The 95th percentile individual dose for an exposure period of 30 days was calculated consistent with the guideline given in the CSA N288.2-19 standard (B-CSA19a).

ADDAM is a safety analysis computer program developed by the Atomic Energy of Canada Limited (AECL) for use by the CANDU Owners Group (COG) community. ADDAM calculates doses to the public due to a postulated accidental release of radioactive material to the atmosphere from a nuclear facility. Radionuclides being released can be in the form of gases, vapours or small particles. The radionuclides will disperse as a result of the effects of atmospheric turbulence. The dispersion of the release is affected by the characteristic of the release, the prevailing meteorological conditions, the surrounding terrain and the nearby buildings. The concentrations in the cloud and on the ground take into account factors such as the nature of the releases, decay, build-up and deposition. Doses are calculated for various age groups and receptor locations, and categorized by release pathways (stack, inlet, leakage, or hole) and exposure pathways (inhalation, cloudshine, groundshine). The calculations of atmospheric dispersion and doses are based on CSA N288.2-M91.

The potential doses to an adult and an infant from airborne tritium, krypton-85 and carbon-14 emissions for all exposure pathways are determined using ADDAM. The bounding doses associated with the malfunction or accident scenario for used fuel dry storage involving used fuel bundles in the DSC are 2.43×10^{-3} mSv (adult) and 2.90×10^{-3} mSv (infant). These dose consequences are associated with the malfunction/accident scenario of a DSC drop during on-site transfer where 100 percent of the fuel elements are assumed to be damaged.

B.4.4.2 Occupational

The worker is assumed to be present in the vicinity of the accident location wearing no protective clothing or respiratory protection at the time of the accident. The worker's response time to remove himself or herself from the accident location (i.e., under emergency back-out conditions) is assumed to be two minutes.

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[22] Pending the completion of a gap assessment and subsequent implementation plan, REGDOC–2.4.4 (Safety analysis for Class IB Nuclear Facilities) will be referenced for the guidance of future safety assessments.

The resulting dose rate is assessed using the semi-infinite cloud model. The cloud volume is assumed to be 500 m^3 and the exposure time 120 seconds.

The inhalation dose from exposure of tritium is calculated using the following equation:

$$D_{inhalation} = \sum_{n=1} \left(R_n \times BR \times sk_{a,n} \times DCF_{inhalation,n}\right) \times T/V$$

where;

$D_{inhalation}$ = worker dose from inhalation (Sv);

R_n = release amount (Bq) of nuclide *n* during the exposure time;

BR = worker breathing rate = 4.17×10^{-4} m^3/s;

$sk_{a,n}$ = skin absorption factor for nuclide *n*. $sk_{a,n}$ = 1.5 for tritium and = 1 for other nuclides;

$DCF_{inhalation,n}$ = inhalation dose coefficient (Sv/Bq) of nuclide n. 1.8×10^{-11} Sv/Bq for HTO and 6.2×10^{-12} Sv/Bq for carbon-14;

T = exposure time (120 s); and

V = contaminated cloud volume (500 m^3)

The cloudshine dose from krypton-85 exposure is calculated using the following equation:

$$D_{cloudshine} = \sum_{n=1} \left(R_n \times DCF_{cloudshine,n}\right) \times T/V$$

where;

$D_{cloudshine}$ = worker dose from cloudshine (Sv);

$DCF_{cloudshine,n}$ = cloudshine dose coefficient ($Sv.m^3.Bq^{-1}.s^{-1}$) of nuclide n. 2.55×10^{-16} $Sv.m^3.Bq^{-1}.s^{-1}$ for Kr-85 and 2.60×10^{-18} $Sv.m^3.Bq^{-1}.s^{-1}$ for carbon-14;

The bounding occupational dose associated with the malfunction or accident scenarios involving used fuel bundles in the DSC is 4.7 mSv, which came from the following contributors:

- 4.15 mSv from inhalation (including skin absorption) of tritium;
- 0.56 mSv from cloudshine of Kr-85

Carbon-14 was included for completeness, however, the contribution from carbon-14 was not significant. The calculated worker dose is 9.4% of the worker dose limit.

B.5.0 ACCEPTANCE CRITERIA

The radiation safety requirements under normal operation for the NSS-PWMF are the following:

- The OPG administrative dose target to the public at the Pickering site boundary shall be less than 100 µSv/y, which is ten percent of the regulatory limit of 1 mSv/y for members of the public.
- The dose rate to non-NEWs, derived from the regulatory limit of 1 mSv/y for members for the public and on 2,000 hours of work per year, is 0.5 µSv/h. This dose rate is applied at the NSS-PWMF Phase I site security fence (located 5 m from the building wall), at the NSS-PWMF Phase II site perimeter fence, and anywhere between this fence and the Pickering NGS fence (where non-NEWs could be working).
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over five years.

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose limit for the public at or beyond the Pickering NGS site boundary due to an abnormal event/credible accident is 1 mSv.
- The dose limit for a worker due to an abnormal event/credible accident is 50 mSv.

B.6.0 ASSESSMENT OF DRY STORAGE CONTAINER ON-SITE TRANSFER

New DSCs are received from the manufacturer and are inspected and checked for component fit at the NSS-PWMF before being sent to the stations for loading.

At the stations, each DSC is wet-loaded with four used fuel storage modules in the fuel bay, decontaminated, drained, and vacuum dried. The transfer clamp and seal are installed to secure and seal the lid during on-site transfer. The loaded DSC is then transported to the DSC processing building using a transporter.

When a loaded DSC is scheduled to be transferred from an IFB to the DSC processing building, the transporter picks up the loaded DSC with a transfer clamp installed. The transfer clamp is designed to maintain the lid secured to the DSC base during all normal operations and abnormal events/credible accidents. The process uses OPG radiation protection and security procedures. Security is provided in accordance with the approved security plan. IAEA monitoring and surveillance is performed in accordance with the safeguards requirements.

The DSC transporter is a multi-wheeled vehicle for on-site transfer of DSCs. The transporter is powered by a diesel engine, and is self-loading; it does not require the assistance of a crane when picking up or depositing a DSC. The tires on the transporter are designed not to deflate if punctured.

The DSC is lifted and transferred via lifting trunnions mounted on the upper vehicle frame. A mechanical lock prevents the DSC from being inadvertently lowered to the ground upon hydraulic failure.

While traveling at full speed of up to 12 km/h, the transporter is capable of stopping within 4.8 m when emergency stop buttons are depressed, front or rear bumpers are displaced by impact, sensors are activated, or when the seat occupancy switch is activated. Emergency stop when traveling at minimal speeds (for example, when moving DSCs within the NSS-PWMF) is achieved almost instantaneously.

A transfer route assessment has been carried out to assess the route by which loaded and seal welded DSCs are transferred from the NSS-PWMF Phase I site to the NSS-PWMF Phase II DSC Storage Buildings 3 and 4. The assessment included an evaluation of the road condition for the expected load and frequency of transfer, assessment of the layout of the roads in relation to buildings, obstacles and intersections which the transfer will have to negotiate, and identification of culverts and ditches along the route to confirm that an accidental roll over of the transporter would not exceed a 2 m drop.

The assessments also looked at possible hazards and obstructions that the transporter may encounter along the route, including hazards such as oil tanks. The assessment has shown that there are no hazards identified that jeopardize the safe transfer of DSCs. The assessment has confirmed that there are no grade changes or obstructions where the bottom of the DSC would scrape the ground due to its low (6 in) clearance, and the comparatively long wheelbase of the transporter.

B.6.1 Normal Operating Conditions during Dry Storage Container On-Site Transfer

B.6.1.1 Direct Radiation Fields

Figure B-6 shows the gamma dose rates calculated as a function of distance from the top, front, and side surfaces of the reference DSC design (that is, the modified design – see Section B.2.10.1). The calculations assume the DSC was filled with 384 Pickering reference fuel bundles as described in Table B-1. The label ‘front’ corresponds to the wider face of the DSC, the face bearing the lift plates.

B.6.1.2 Radioactive Emissions

Under normal operating conditions, no airborne emissions are expected from DSCs during transfer from the Pickering IFBs to the NSS-PWMF. This is because the uranium dioxide matrix, the used fuel sheath and the transfer elastomeric seal provide multiple barriers towards preventing the release of radioactive materials:

- The UO_2 matrix effectively contains the radionuclides present in 10-year cooled used fuel (either under wet or dry storage conditions), except for the free fractional inventory of tritium (in vapour form) and krypton-85 (which is a gas);

- The fuel sheath additionally contains the free fractional inventory of tritium, krypton-85 and carbon-14 that would otherwise be available for release; and

- The transfer clamp and elastomeric seal on the lid provide an additional barrier against the release of tritium, krypton-85 and carbon-14 in the event of fuel sheath failure during transfer.

B.6.1.3 Thermal Assessment

The thermal power for a reference fuel bundle was calculated using the computer code ORIGEN-S. Figure B-2 shows the thermal power per reference fuel bundle as a function of the fuel age.

The thermal analysis carried out for the DSC, for 10-year-cooled used fuel with a heat load of 6.4 W/bundle, demonstrated that the fuel would be adequately cooled. Ten-year-cooled Pickering reference used fuel has a lower heat load of 5.8 W/bundle (see Figure B-2); therefore the thermal analysis is considered to envelope the conditions for storage of Pickering used fuel in DSCs. The experimental measurements obtained during the thermal performance verification program at the NSS-PWMF in summer 1998 are consistent with the results of the thermal analysis (B-OPG04a)

A structural integrity assessment for DSCs (B-OPG14b) considered fuel bundles with a significantly higher decay heat of 7.4 W, which is conservative with respect to the NSS-PWMF conditions. The resulting thermal gradient in the concrete base of the DSC was estimated to be 54°C (B-OPG14a). The predicted stresses generated in the concrete by the thermal gradient of 54°C indicate that through wall cracking will not occur and thermal expansion does not compromise the structural integrity of the DSC.

B.6.2 Malfunctions and Accidents Assessment for Operations during On-Site Transfer

B.6.2.1 Dry Storage Container Drop during On-Site Transfer

Transporter design features and administrative control requirements are expected to ensure that the transporter will not collide with another vehicle during DSC transfer. However, a bounding assessment has been carried out to envelope the radiological consequences resulting from the drop of a DSC due to an unforeseeable accident during on-site transfer.

B.6.2.1.1 Drop of a Dry Storage Container during Transfer from an Irradiated Fuel Bay to the Dry Storage Container Processing Building

Consider the case where the transporter collides with another vehicle during a DSC transfer from an IFB to the DSC processing building. The transfer clamp has been designed to withstand the impact resulting from collision with another vehicle, and will ensure that the lid will stay on the DSC. Therefore, only the airborne release of tritium, krypton-85 and $^{14}CO_2$ from the DSC cavity is considered for this assessment.

Failure of 100 percent of a DSC's used fuel content is assumed, i.e., 100 percent of the fuel elements in all the 384 fuel bundles, for a total of 10,752 failure fuel elements. The free inventory of tritium, krypton-85 and $^{14}CO_2$ in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barriers provided by the transfer clamp, the elastomeric seal and the sub-atmospheric pressure inside the DSC cavity, it is assumed that these radionuclides are released immediately into the environment.

Based on the methodology presented in Section B.4.4.1, the total dose to the public due to this event was assessed to be 1.60×10^{-3} mSv for an adult and 1.96×10^{-3} mSv for an infant at the Pickering site boundary. Based on the methodology presented in Section B.4.4.2, the dose to the worker due to this event was assessed to be 4.7 mSv.

B.6.2.1.2 Drop of a Dry Storage Container during Transfer from the Pickering Waste Management Facility Phase I Site to the Pickering Waste Management Facility Phase II Site

Another possible scenario involving collision of the DSC transporter with another vehicle could take place during DSC transfer between the NSS-PWMF Phase I and Phase II sites. Along this transfer route, a Transporter would carry a seal welded DSC. Although the seal-weld is extremely robust, the collision is postulated to compromise the seal-weld.

Failure of 100 percent of a DSC's used fuel content is assumed i.e., 100 percent of the fuel elements in all the 384 fuel bundles, for a total of 10,752 failed fuel elements. The free inventory of tritium, krypton-85 and $^{14}CO_2$ in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barriers provided by the seal-weld and the sub-atmospheric pressure inside the DSC cavity, it is assumed that the radionuclides are released at once into the environment.

Based on the methodology presented in Section B.4.4.1, the total dose to the public due to this event was assessed to be 2.43×10^{-3} mSv for an adult and 2.90×10^{-3} mSv for an infant at the Pickering site boundary. Based on the methodology presented in Section B.4.4.2, the dose to the worker due to this event was assessed to be 4.7 mSv.

B.6.2.2 Transporter Failure

In the event of transporter failure, the transfer clamp and elastomeric seal is assumed to fail as a result of the longer than expected time taken to transfer the DSC from the IFBs to the DSC processing building.

Conservatively, it is assumed that the free inventory of tritium, krypton-85 and carbon-14 in four damaged fuel elements is released into the DSC cavity (if one percent of all bundles contain one damaged element, there would be approximately four damaged elements in each DSC). The barrier provided by the transfer clamp and elastomeric seal are ignored and these radionuclides are considered to be released at once into the environment.

Based on the NSS-PWMF and NSS-WWMF operating experience, the frequency for this event is three events per year. The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.6.2.3 Fire

The potential for an accident involving DSC contact with a source of combustible material during on-site transfer has been considered. Fire sources directly along the transfer route of the DSC following the north side of the station units could include hydrogen cylinders, hydrogen trailers and stationary tank set, compressed gas bottles, oil storage tanks and the fuel tanks of other vehicles. The combustible materials that could be contributed by the transporter itself are the diesel fuel in the tank, engine lubricating oil and hydraulic oil. It is expected that such a fire would be of short duration. The duration of the fire would be further limited as a result of the fire detection and suppression systems in the transporter design and the expected response of the Pickering NGS Emergency Response Team.

The effect of a fire could potentially be to increase the temperature of the DSC and the used fuel bundles inside the DSC. Given the large thermal inertia of the DSC and the limited duration of the event, it is concluded that a fire along the transfer route could not cause overheating or damage to the used fuel.

No releases of radioactivity are expected from this scenario and, as such, there would be no public or occupational dose consequences.

In the event of vehicle failure, the dose consequence to the public and workers has been described in Section B.6.2.2.

B.6.2.4 Criticality

As per (B-CSA22), Nuclear criticality safety objectives are achieved when the following criteria have been met:

a) Sources of fissile materials are contained within vessels or structural boundaries which can maintain containment or pressure boundary function in the event of fires.

b) Accidental criticality is prevented by defence-in-depth measures including one or more control parameters, such as mass, volume, concentration, geometry, moderation, reflection, interaction, isotopic composition and density, and with account taken of neutron production, leakage, scattering, and absorption.

Criticality considerations for used fuel stored in DSCs can be based on the criticality assessments previously carried out for the NSS-PWMF (B-OH98) and the NSS-WWMF (B-OPG04b). The internationally accepted criterion for assuring subcriticality in such storage facilities is that k_{eff} should be less than 0.95. Consistent with expectations for irradiated natural uranium fuel, the earlier analyses and assessments have yielded adequate subcriticality margin and have demonstrated that there will be no criticality of used CANDU fuel, even in DSCs filled with light water.

The specific cases previously analyzed using the WIMS-AECL code for a DSC containing Pickering used fuel are given below:

Environment Inside DSC	k∞	keff
Dry inert atmosphere (187 MWh/kgU)	0.4734	0.2114
Flooded with H20 (187 MWh/kgU)	0.7815	0.7327
Flooded with H20 (27 MWh/kgU)	0.8119	0.7610

[redacted]

The reference fuel burnup for NSS-PWMF shielding analysis purposes is [redacted]. The level of conservatism in dose rate calculations increases as the reference burnup is increased. For criticality considerations, it is more conservative to assume a lower burnup, since CANDU fuel reactivity reaches a maximum around the plutonium peak (~ 27 MWh/kgU) and decreases with further fuel irradiation. The criticality analysis used a nominal discharge burnup of 187 MWh/kgU and also considered, as a limiting case, the plutonium peak burnup of 27 MWh/kgU. These assumptions in the criticality assessments provide a conservative upper bound for the potential for criticality in Pickering reference used fuel stored in DSCs. It is therefore concluded that 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under normal conditions or under any postulated accident scenario.

B.6.2.5 Common Mode Incidents

B.6.2.5.1 Adverse Road Conditions

Procedural controls are in place to prohibit DSC transfer under poor road conditions or until potentially slippery conditions can be corrected by, for example, sanding or salting of the transfer route. Even if the transfer vehicle were to lose traction on a slippery surface resulting in the vehicle leaving the road, a release of radioactivity from a clamped or seal-welded DSC is not expected given the robust design of a DSC, which is intended to withstand transportation accident loads.

In the event the on-site transfer of a DSC takes longer than expected as a result of adverse road conditions, the radiological consequences would be within the envelope of those in Section B.6.2.1.

B.6.2.5.2 Earthquake

The Pickering B Design Basis Earthquake (DBE) is defined as an earthquake with a peak ground acceleration (PGA) of 0.05 g and a frequency of reoccurrence once in 1,000 years, using the 84th percentile seismic hazard curve for the Pickering site.

Since the transporter with a DSC is not on the road 100 percent of the time, the combined occurrence of having a DBE and the transporter on the road simultaneously was calculated. The following assumptions were taken into account:

(a) A maximum of 150 DSCs are transferred each year between the station IFBs and the NSS-PWMF DSC processing building.

(b) The greatest distance the transporter needs to travel between the IFBs and the DSC processing building is approximately 1 km.

(c) The transporter is conservatively assumed to take a much longer time during transfer and be on the road for 1 hour.

With these assumptions, the probability of finding a loaded DSC in transit during a 1-year period would be:

$$150 \text{ transfers x } \frac{1\,hour}{transfer} \text{ x } \frac{1\,year}{8760\,hours} = 1.71 \text{ x } 10^{-2}$$

The frequency of a DBE occurring at a time when a DSC is being transferred is:

$$\frac{1\,event}{1{,}000\,years} \text{ x } (1.71 \text{ x } 10^{-2}) = 1.71 \text{ x } 10^{-5} \text{ events per year}$$

If the on-site transfer of the DSC from the IFBs to the DSC processing building occurs during an earthquake, the DSC will not topple over due to the forces from the DBE for horizontal and vertical PGA of 0.12 g, which is higher than the 0.05 g corresponding to a postulated DBE at Pickering B.

In the event the on-site transfer of the DSC from the IFBs to the DSC processing building may take longer than expected as a result of an earthquake, the consequences would be within the envelope of those in Section B.6.2.1.

DSCs transferred from the NSS-PWMF Phase I site to the NSS-PWMF Phase II site will be seal-welded. This scenario is, therefore, not expected to result in any radioactivity dose consequence to workers or to the public.

B.6.2.5.3 Tornadoes

Tornadoes normally occur in unstable atmospheric conditions when warm moist air comes into contact with cold air. A tornado is a rotating thunderstorm with a vortex of air extending downward from a thundercloud. The strong updraft in a thunderstorm interacts with strongly sheared winds causing rotation of the updraft that intensifies to become a tornado. The bounding Design Basis Tornado (DBT) defined for the Darlington nuclear site (B-OPG22e) is defined as follows:

- Rotational wind speed of 322 km/h,
- Translational wind speed of 96 km/h,
- Pressure drop of 9.6 kPa,
- Rate of pressure drop of 5.6 kPa/s and
- Radius of maximum rotational wind speed of 46 m.

These parameters are considered to be large enough to envelope any credible tornadoes in Southern Ontario. Based on the Pickering NGS site wind speed frequencies, the DBT-definition rotational wind speeds correspond to a mean frequency of 3.13×10^{-6} events per year.

During tornado winds, objects can be picked up by the wind forces and accelerated to high velocities. Reference (B-OPG22e) has established a spectrum of tornado-generated missiles considered in the Darlington NGS design as part of the DBT:

(a) Woodplank,102 mm × 305 mm × 3.7 m, mass 91 kg, velocity 335 km/h (80 percent of total tornado velocity, rotational plus translational).

(b) Steel pipe, 76 mm diameter, schedule 40, 3 m long, mass 35.4 kg, velocity 168 km/h (40 percent of total tornado velocity).

(c) Steel rod, 25 mm diameter × 914 mm long, mass 3.6 kg, velocity 251 km/h (60 percent of total tornado velocity).

(d) Steel pipe, 152 mm diameter, schedule 40, 4.6 m long, mass 129 kg, velocity 168 km/h (40 percent of total tornado velocity).

(e) Steel pipe, 305 mm diameter, schedule 40, 4.6 m long, mass 337 kg, velocity 168 km/h (40 percent of total tornado velocity).

(f) Utility pole, 343 mm diameter, 10.7 m long, mass 676 kg, velocity 168 km/h (40 percent of total tornado velocity).

(g) Automobile, frontal area 1.9 m2, mass 1,800 kg, velocity 84 km/h (20 percent total tornado velocity).

The effect of tornado-generated missiles listed by the Southern Ontario DBT impacting on the DSC has been evaluated (B-AECL03). The analysis showed that the transfer clamp will keep the lid in place, the DSC containment will not be breached, and the DSC will not overturn under the impact of postulated missiles during on-site transfer. The safety factor against overturning was found to be 5.41 for the DSC loaded with used fuel.

The probability of finding a loaded DSC in transit from the IFBs to the processing building is 1.71×10^{-2} in a year. Therefore, the frequency of a tornado occurring at a time when a DSC is being transferred is $(1.71 \times 10^{-2}) \times (3.13 \times 10^{-6}) = 5.36 \times 10^{-8}$ events per year, which is below the cut-off frequency of 10^{-6} per year. This event is therefore considered incredible.

B.6.2.5.4 Thunderstorms

Thunderstorms can potentially involve lightning striking a loaded DSC on the transporter during on-site transfer. The impact of a lightning strike on the DSC was evaluated (B-OPG22) and it was concluded that in an unlikely event of a direct lightning strike to the DSC during transfer, arcing will occur between the vehicle and the ground, dissipating the lighting energy and the container will not be compromised. However, the lightning may be hazardous for the driver or the electrical/electronic components of the vehicle. Even if operator incapacity were to result in the transporter leaving the road, a release of radioactivity from a DSC is not expected given the design of the DSC and transfer clamp or the seal-weld. Therefore, the dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

Given the results from the analyses carried out to study the effects of the DBT and DBE on the DSC during on-site transfer, it is expected that the integrities of both the DSC and DSC transfer clamp would not be affected if subjected to thunderstorms.

In case the on-site transfer of a DSC takes longer than expected as a result of thunderstorms, the consequences would be within the envelope of those in Section B.6.2.1.

For the transfer of DSCs from the NSS-PWMF Phase I site to the NSS-PWMF Phase II site, DSCs will be seal-welded and, consequently, no radiological dose consequence is expected to result from this accident scenario.

B.6.2.5.5 Floods

The only possibility for flooding at the Pickering site would be as a result of extreme local meteorological events. A Review Level Condition (RLC) for Probable Maximum Precipitation (PMP) has been developed to be used at OPG sites, which represents a rainfall of 420 mm in a 12-hour period, of which 51% (214 mm) falls within a one-hour period.

Transfer procedures require that loaded DSCs not be transferred during anticipated extremely adverse weather conditions. In addition, sufficient warning time is available for site staff to prevent this scenario from occurring.

If transport of a DSC during an extreme rainfall were to occur, extensive flooding would likely affect the operation of the transporter. However there would be no detrimental effect on the DSC. The DSCs are designed to tolerate water immersion at 2 MPa (B-OPG22g), so the temporary water levels would not be of a concern to the radiological safety. In this event any potential radiological consequences would be within the envelope of those in Section B.6.2.1.

B.6.2.5.6 Explosions Along the Transfer Route during Dry Storage Container Transfer

There are several sources of explosion along the on-site transfer route of the DSCs from the IFBs to the Phase I processing building, such as acetylene cylinders and compressed gas bottle storage facility. Explosions originating from handling accidents of acetylene cylinders, compressed gas bottle explosion and pressure vessel burst leading to missiles due to normal wear and tear of oxygen, nitrogen or air cylinder have been assessed and the combined hazard frequency has been calculated to be 9.74×10^{-8} events per year, which is lower than the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

Explosion hazards along the onsite transfer route of the DSCs from the Phase I processing building to the Phase II storage building have been assessed. The following hazard scenarios may have a potential to damage a passing DSC:

- Acetylene cylinder detonation
- Propane storage tank Boiling Liquid Expanding Vapour Explosion (BLEVE)
- Vapour Cloud Explosion (VCE) due to a propane storage tank rupture.

The combined explosion hazard frequency has been determined to be 5.2×10^{-8} events per year, assuming 20 DSC shipments a week. Even based on this extremely conservative approach, the explosion hazard frequency is lower than the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

B.6.2.5.7 Turbine Missile Strike

The frequency of turbine missiles impacting structures, systems and components (SSC) is about 6×10^{-6} events per year.

The probability of transferring a loaded DSC between the IFBs and the processing building is 1.71×10^{-2} over a year.

The frequency of turbine missiles impacting a loaded DSC while the DSC is being transferred from either the IFB to the processing building or from the processing building to Storage Building 3 is: $(6 \times 10^{-6}) \times (1.71 \times 10^{-2}) = 1.03 \times 10^{-7}$ events per year, which is below the cut-off frequency of 10^{-6} events per year. This hazard is considered incredible.

B.6.2.5.8 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area. Using total crash rates determined for the Pickering NGS site, the aircraft impact frequency impacting the transporter has been determined, considering the limited time that a loaded DSC transporter will be in transit and taking into account that the transporter is a small moving target.

There will be a maximum of 150 DSC transfers within the NSS-PWMF in one year. The LiftKing transporter dimensions, slightly larger than the GEN IV transporter, were used to assess this hazard. The transporter has an overall length of 8.5 m, an overall width of 3.3 m and an overall height of 4.7 m.

The frequency of an aircraft crash impacting the transporter carrying the DSC during on-site transfer is 6.17×10^{-10} events per year (B-OPG22b), and therefore this event is considered incredible.

B.6.2.5.9 Toxic Gas Release – Chlorine Originating from Ajax Water Treatment Plant

The Ajax Water Treatment Plant uses chlorine cylinders for water treatment. The facility is located at approximately 4.1 km from the Phase II Storage Buildings 3 and 4 and the route of the DSC transfer from the processing building. The Screening Distance Value (SDV) for chlorine is 4.4 km, hence this hazard cannot be screened out based on distance.

Chlorine leakage from the Ajax Water Treatment may have an impact on the transporter operator’s ability to keep the transporter safely on the road. Even if the operator illness were to result in the transporter leaving the road, a release of radioactivity from a seal-welded DSC is not expected. The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.6.2.5.10 Soil Failures/Slope Instability

In the event the on-site transfer of a DSC takes longer than expected as a result of adverse road conditions due to soil failure or slope instability, the radiological consequences would be within the envelope of those in Section B.6.2.1.

B.7.0 ASSESSMENT OF THE DRY STORAGE CONTAINER DURING PROCESSING OPERATIONS

The transporter takes DSCs from the IFBs to the DSC processing building. DSCs are then taken through the following steps before they are ready for storage:

(a) The DSC lid is welded and the inspection of the lid seal-weld is completed.

(b) The DSC is subjected to vacuum drying and helium backfill.

(c) The drain port plug is installed and seal welded in place.

(d) The DSC is helium leak-tested by using a vacuum chamber (bell jar).

(e) A radiological survey is performed on the DSC and spot cleaning is carried out, as required.

(f) Paint is applied to the welded areas and to any scrapes or scuffs on the DSC exterior.

(g) IAEA safeguards seals are applied.

(h) The transporter moves the DSC to the DSC storage building for final storage.

B.7.1 Normal Operating Conditions during Dry Storage Container Processing

B.7.1.1 Dose Rates Inside the Dry Storage Container Processing Building

The DSC processing building is a single-story structure with a two-story amenities area. The processing building's workshop space is approximately 830 m², divided into the workspaces that support the various DSC processing operations.

Conservatively, one can assume that there are approximately 20 loaded DSCs occupying the designated staging and processing stations in the workspaces. With approximately 20 DSCs in the building, the occupational dose rate of individual workers is dominated by their proximity to a single loaded DSC in any of the designated stations. Workers may routinely handle DSCs containing 10-year-cooled used fuel, the minimum age of used fuel permitted to be loaded into a DSC for routine storage.

Figure B-6 shows the gamma dose rates calculated as a function of distance from the top, bottom, front and side surfaces of a DSC loaded with reference used fuel (10-year-cooled). In the Figure, the label "front" corresponds to the wider face of the container (i.e., the face bearing the lift plates), and the label "side" indicates its narrower face.

Dose rates inside the NSS-PWMF have been demonstrated to be acceptably low (most working areas are normally at or near ambient background radiation levels). The actual dose rates from working with DSCs have consistently been found to be much lower than predicted. Results of historical radiation monitoring at the NSS-PWMF indicate that contact dose rates on the front and side surfaces of DSCs loaded with used fuel that has cooled for 10 years or more have varied between 9 µSv/h and 13 µSv/h.

As shown in the drawing 92896-D0A-29660-1001, there are some offices in the building on the second floor. Workers in offices are not likely to be in line of sight with any one of the approximately 20 DSCs that can be present in the workshop area.

Analysis has shown that, due to the high-density concrete used as shielding material in the container, the contribution of neutrons to dose rate is negligible compared to that of gamma radiation (B-OPG22f). Neutron dose rate contributions, therefore, were not calculated.

B.7.1.2 Radioactive Emissions during Dry Storage Container Processing

There is a small potential for airborne emissions resulting from DSC processing operations such as welding and vacuum drying. Surface contamination on DSC exterior surfaces is effectively controlled through prevention measures and decontamination at the Pickering NGS IFBs. Nevertheless, small quantities of fixed surface contamination may become airborne during welding operations. Such airborne particulate contamination, if present, would be effectively removed by the High Efficiency Particulate Air (HEPA) filters in the active ventilation system.

Radioactive contamination may be present on the outside of the fuel cladding. Although this contamination is expected to adhere to the fuel during storage, there is some potential for it to become airborne during vacuum drying of the DSC cavity. A dedicated hose is used for DSC vacuum drying operations to prevent the spread of such contamination to other workshop systems. Vacuum skid discharge is directed to the active ventilation system, where particulate contamination is removed by HEPA filters. Fuel temperatures during vacuum drying have been assessed and are expected to remain well within safe temperatures for maintaining the long-term integrity of the used fuel cladding.

Additionally, NSS-PWMF experience demonstrates that particulate emissions in exhaust from DSC processing operations have been typically below the Minimum Detectable Activity (MDA) level.

However, because DSCs are unclamped and unwelded at some stages during processing, an assessment of the chronic radioactive emissions during processing is presented below.

A very small quantity of fuel elements may have minor defects in their fuel cladding. OPG fuel performance experience has demonstrated that cladding defects are present in less than 0.01 percent of fuel bundles (representing < 0.001 percent of fuel elements). Residual releases from these defective elements have been assessed and described below.

For the purpose of evaluating the potential emissions under normal operating conditions, the following conservative assumptions are used to obtain an estimate for chronic airborne emissions:

(a) One fuel element in 1 percent of fuel bundles is damaged during handling (four elements per DSC) and for each failed fuel element the free inventory of tritium, krypton-85 and carbon-14 is released into the DSC cavity.

(b) The barrier provided by the DSC lid seal-weld is ignored and these radionuclides are released into the environment.

These assumptions are deemed conservative for the following reasons:

- The postulated defect rate is more than 20 times higher on a per element basis than OPG fuel performance experience.

- Fuel element defects occur primarily in the bundle manufacturing process or as a result of debris fretting in the reactor core. At high fuel temperatures during irradiation, the free inventory of tritium, krypton-85 and carbon-14 in elements with cladding defects would have been released within the reactor core.

- Used fuel is stored for at least 10 years in the Pickering IFBs prior to transfer to a DSC. Leaching of grain-boundary inventory and release of gap inventory would have additionally occurred over this period for bundles with minor cladding defects.

- Should free inventory remain in the fuel-sheath gap or grain boundaries subsequent to in-bay storage, its release would have occurred during initial vacuum drying inside the Pickering IFBs.

- From the historical data, the maximum number of DSCs being processed in a year is 70 DSCs. To accommodate the decommissioning of the Pickering NGS, the number of DSCs to be processed per year is expected to be 110. For conservatism, a value of 150 is applied for the safety assessment. A total of 600 fuel elements is assumed to fail during 1 year under normal operating conditions (a very conservative scenario).

The chronic off-site dose consequences from this scenario for the most exposed age group and location at the Pickering NGS site boundary and beyond are estimated to be less than 1.0×10^{-3} µSv/year.

Since the above assumptions are very conservative, the assessed consequence is considered an upper bound for any possible chronic emissions during normal operating conditions.

B.7.2 Malfunctions and Accidents Assessment for Operations during Dry Storage Container Processing

B.7.2.1 Drop of a Dry Storage Container during Handling

Failure of the lifting beam, lift plates, the crane, or the DSC trunnions could potentially result in dropping a loaded DSC while it is being lifted during operations at the DSC processing building. Fuel sheath failure is not expected from a DSC drop from the low lift height of DSC processing building operations.

The failure probability of crane lifting very heavy loads, based on US nuclear plant operating experience (B-NRC03), is estimated to be 5.6×10^{-5} per demand. During processing, a DSC is lifted eight times. For DSCs that fail the weld inspection, there are four additional lifts to bring the DSC to and from the weld bay and PAUT bay, i.e., 12 lifts in total. Assuming 150 DSCs are processed per year and ten percent require a weld repair, the number of lifts to be carried out at the DSC processing building using the crane is (135 DSCs) x (8 lifts per DSC) + (15 DSCs) x (12 lifts per DSC) = 1,260 lifts per year. Therefore, the total frequency of crane failure, potentially resulting in dropping the DSC, would be (1,260 lifts per year) x (5.6×10^{-5}) = 7.1×10^{-2} events per year.

A handling accident involving the dropping or tip over of multiple DSCs is not considered to be a credible event. Should a crane accident result in the drop of a clamped DSC or seal welded DSC, the low lift height inside the DSC processing building would prevent the container from tipping over and striking a second DSC. In two instances, for a very short duration, the seal-welded DSC is lifted above low-lift height: when it is loaded into the bell jar (1.5 m) and when it is placed on the drain port welding stands. In these instances, it is assumed that the DSC could drop and tip over as a result of a crane failure. A release from this event is not expected.

However, even in the worst-case scenario, dropping a DSC during handling is not expected to result in failure of more than 30 percent of a DSC's used fuel elements, a total of 3,226 failed fuel elements. Realistically, fuel sheath failure is not expected to result from an accidental DSC drop from the low lift height of the transporter or from the crane in the DSC processing building.

The free inventory of tritium, krypton-85 and carbon-14 in the damaged fuel elements 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 dose consequences for this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.2 Equipment Drop onto a Dry Storage Container

The crane auxiliary hoist is used to handle other processing equipment, including the DSC transfer clamp, lid welding equipment, and vacuum bell jar lid. A structural failure of any lifting/rigging equipment such as slings, shackles, or other specialty equipment lifting points or lifting beams while suspended by the auxiliary crane could result in a drop of equipment onto the lid of a loaded DSC. These accident scenarios are unlikely given that the rated load capacity of the auxiliary hoist and the lifting/rigging equipment will not be exceeded, and routine inspections and pre-operational checks will be performed.

However, to calculate the event frequency of equipment dropping onto a DSC, the following assumptions were made:

(1) A maximum of 150 DSCs could be processed each year;

(2) Even though three pieces of equipment have the potential to collide with a loaded DSC while suspended from the crane auxiliary hoist (transfer clamp, lid welding equipment, and vacuum bell jar lid), only two pieces of equipment, the transfer clamp and the lid welding equipment, have the potential to drop onto an unwelded DSC lid;

(3) Each of the above two structures (transfer clamp and the lid welding equipment) is lifted once over the unwelded DSC; a) when removing the transfer clamp and b) when installing the welding equipment, which means a total of two lifts.

Assuming that the probability of the equipment failure is 5.6×10^{-5} (B-NRC03), the total frequency of a drop of equipment onto a loaded DSC lid would be: (5.6×10^{-5} failure event per movement) x (150 DSCs per year x 2 movements per DSC) = 1.68×10^{-2} events per year.

The dose consequences for this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.3 Dry Storage Container Collision during Craning

A DSC craning accident due to operator error could result in a loaded DSC colliding with another DSC (loaded or empty) on the DSC processing building floor or other process building equipment or structure. The assumed frequency of the operator error probability is 1.0×10^{-3} (B-Sandia83).

Assuming that 150 DSCs are processed per year, the total number of times a loaded and unwelded DSC is lifted would be approximately 165 (one lift per DSC plus 10 percent of them are assumed to have weld failure and required weld repairs). The postulated frequency of a loaded and unwelded DSC craning collision accident is (1.0×10^{-3}) × (165 lifts per year) = 1.65×10^{-1} events per year.

Given that the overhead crane bridge and trolley maximum speeds are limited by design, the dose consequences from this event are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.4 Transporter Collision with a Loaded Dry Storage Container or Another Transporter

Operator error during transporter vehicle operations could result in a collision with a loaded DSC on the DSC processing building floor or with another transporter in the DSC processing building. The transporter collision could occur while it is carrying a loaded or empty DSC.

It is assumed that a maximum of 150 DSCs are loaded each year and that the transporter is used four times for each DSC during processing (pick up of an empty DSC for transfer to the IFBs, transfer of a loaded DSC to the DSC processing building, transfer of a seal-welded DSC to the paint station at the DSC Storage Building 1, and transfer to storage). The assumed frequency of the operator error probability is 1.0×10^{-3} (B-Sandia83). Therefore, the probability of a transporter collision with a loaded DSC or with another transporter in the processing building due to operator error would be (1.0×10^{-3} event per movement) × (150 DSCs per year) × (4 movements per DSC), which equals 6×10^{-1} events per year.

Given that the transporter vehicle speed is limited by design and that it is equipped with front and rear bumper emergency stops or sensors, the dose consequences from this scenario would be bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.5 Equipment Collision with a Loaded Dry Storage Container during Craning

A craning accident due to operator error could result in process equipment colliding with a loaded DSC while suspended from the auxiliary hoist. To calculate the frequency of this event, the following assumptions were made:

(1) A maximum of 150 DSCs could be processed each year;

(2) Even though three pieces of equipment have the potential to collide with a loaded DSC while suspended from the crane auxiliary hoist (transfer clamp, lid welding equipment, and vacuum bell jar lid), only the transfer clamp and the lid welding equipment have the potential to collide with an unwelded DSC;

(3) Each of the above two structures is lifted once over the unwelded DSC (remove the transfer clamp and install the welding equipment), which means a total of two lifts;

(4) The operator error probability is 0.001 per movement (B-Sandia83).

The total postulated frequency of this event due to operator error is (1.0×10^{-3} event per lift) × (150 DSCs per year) × (2 lifts per DSC), which equals 3×10^{-1} events per year.

Given that the overhead crane bridge and trolley maximum speeds are limited by design, the dose consequences from this scenario are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.6 Criticality

See Section B.6.2.4.

B.7.2.7 Dry Storage Container Processing Building Fire

The DSC processing building has been designed in accordance with CSA N393, NFCC, NBCC, and all applicable codes and standards referenced therein.

The Fire Hazard Assessment (FHA) for NSS-PWMF (B-OPG22d) demonstrated that the bounding fire scenario for the processing building was a fire involving one DSC transporter. The DSC processing building contains a limited quantity of combustible materials: for example, the transporter contains a small quantity of diesel fuel. The welding cover gases used at the workshop are inert and will not burn or explode. As a result, a DSC transporter fire would likely ignite the adjacent combustibles located in the workshop. The fire would be of short duration due to the limited quantity of combustible material, fire detection, protection systems in the building, and the expected prompt arrival of emergency response personnel.

The effect of a fire would be to increase the ambient temperature in the proximity of the DSC. As outlined in the FHA (B-OPG22d), given the large thermal inertia of the DSC and the limited duration of the event, a fire inside the processing building is not expected to cause fuel overheating or fuel damage. Therefore, no releases are expected from this scenario and therefore there would be no public or occupational dose consequences as a result of this event.

B.7.2.8 Common Mode Incidents

B.7.2.8.1 Earthquake

The DSC processing building has been designed to NBCC-1990 seismic requirements; it would not be expected to collapse in the event of an earthquake with a ground motion equal to or small than 0.05 g.

An analysis was performed to determine the impact of a collapsing DSC processing building on an unclamped and un-welded DSC lid (B-OPG13a). It was concluded that there is some potential for damage to the DSC lid and base, although the DSC outer liner remains intact. There is no likelihood of the contained fuel being exposed due to the complete removal of the lid or tipping of the DSC container.

Calculations were performed (B-OPG03) to assess the DSC seismic stability for the lower probability Pickering 'A' DBE ground motion parameters of 0.12 g horizontal PGA and 0.08g vertical PGA. These parameters bound both the Pickering 'B' DBE and the NBCC ground motion parameters for the Pickering site. The calculations demonstrated that the safety factor against overturning of the DSC is 3 and the safety factor against sliding over concrete floor is 1.54 and the safety factor against sliding between unwelded lid and the base of the DSC is 1.5 (B-OPG03). While the safety factors determined based on the Pickering A seismic parameters are lower than the ones for the Pickering B DBE, the values are greater than 1, which means that the DSC will not overturn or slide during an earthquake scenario using the more stringent Pickering A DBE.

The hazard for the DSC overturning or tipping under the loads described for an earthquake scenario is bounded by the case when the processing building collapses.

Nonetheless, in the event that an earthquake causes a DSC to be dropped during handling, the dose consequences are bounded by those of the DSC drop during on-site traffic accident in Section B.6.2.1.

B.7.2.8.2 Tornadoes

As discussed in Section B.6.2.5.3, a DSC can resist overturning in tornado winds of up to 425 km/h. This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the interceding building structures.

The effect of tornado-generated missiles on a clamped DSC has also been considered in Section B.6.2.5.3; the transfer clamp will keep the lid in place, the DSC containment will not be breached, and the DSC will not overturn under the impact of postulated missiles.

No radiological dose consequence is expected to result from either the DSC being subjected to the full force of the tornado horizontal wind, or the DSC being impacted by a tornado-generated missile.

It has been postulated that the processing building is subject to a tornado at the time that an unclamped DSC stands in preparation of seal-welding, and that an unclamped DSC is struck by a tornado-generated missile.

The DSC has already been shown to be able to withstand the direct impact of a tornado missile without any release; therefore, a missile striking the DSC is expected to have negligible consequences, even if the DSC is un-welded. In addition, the likelihood of both the tornado (3.13×10^{-6} events per year) and preparation of the DSC for seal welding scenarios occurring simultaneously is below the cut-off frequency of 10^{-6} events per year. This event is therefore considered incredible.

B.7.2.8.3 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC processing building. As per design requirements, the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during processing. The impact of a lightning strike on the DSC was evaluated (B-OPG22c). It was concluded that DSCs can withstand a direct lightning strike without any negative impact on their functionality. The shielding function of the DSC will not be compromised and the DSC containment will not be breached. Thunderstorms would result in no releases from DSCs in the processing building and there would be no public or occupational dose consequences.

B.7.2.8.4 Floods

Water entry into the facility originating from a PMP event is possible but the consequences are negligible (B-OPG22b). The simulated water depth is too shallow to be anywhere near the level of the DSC lid. As well, no loose contamination is permitted on the exterior DSC surfaces or on accessible surfaces within the processing building, therefore there is no potential for waterborne contamination.

B.7.2.8.5 Turbine Missile Strike

The NSS-PWMF Phase I is located southeast of Unit 8, with Storage Building 2 situated the closest at a distance of 30 m. The DSC Storage Building 2 is attached to the north wall of the processing building and the Storage Building 1.

The break-up (failure) of the Pickering NGS turbines could lead to a high-trajectory missile strike on the dry storage facility. The frequency of a turbine failure is 1×10^{-4} turbine failures per year (B-OPG21). At rated speed, the frequency of a missile from Pickering NGS striking the NSS-PWMF Phase I facility is 4.8×10^{-4} strikes per missile. The most significant missile is a large fragment of Disc 3 from a low-pressure turbine. There are three low-pressure turbines per unit and two Disc 3s in each low-pressure turbine. Considering six large, high-energy missiles are produced following a turbine failure, the frequency of a strike is (4.8×10^{-4} strikes per missile) × (6 missiles), which is approximately 3×10^{-3} strikes per turbine failure.

The frequency of turbine missiles impacting SSCs at Phase I site is (3×10^{-3} strikes per turbine failure) × (1×10^{-4} turbine failures per year) = 3×10^{-7} impacts per year. Based on the location of the DSC processing building with reference to the Unit 8 turbine, a turbine missile striking the processing building and then the DSC is considered to be an incredible event.

B.7.2.8.6 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of NSS-PWMF Phase I and Phase II. The details on the total aircraft crash frequency calculations are documented in B-OPG22b.

The aircraft crash frequency calculated for the NSS-PWMF Phase I area is 2.37×10^{-7} events per year. The aircraft crash frequency calculated for the NSS-PWMF Phase II area is 2.94×10^{-7} events per year. The aircraft crash frequency for the area occupied by the DSMs was calculated to be 1.62×10^{-7} events per year.

The total aircraft crash frequency for the NSS-PWMF holding safety related waste containers, such as DSCs and DSMs, was determined by summation of the frequency of an aircraft crash impacting the DSC processing and storage buildings and RCS area where the DSMs are stored. A qualitative aircraft impact assessment, conducted for the DSC against light general aviation aircraft crashes concluded that the Category 1 aircraft (light aircraft) will not cause damage to a DSC expect for slight concrete cracking or scabbing. Therefore, for DSCs, the aircraft crash frequency calculations were performed for aircraft categories 2 to 5 for Phase I and Phase II areas occupied by the DSC storage buildings 1-4 and the DSC processing building. For the RCS area aircraft crash frequency calculation, all aircraft categories (Category 1 to 5) were considered.

The summation of the above aircraft crash frequencies calculated for the NSS-PWMF site where safety-related containers are stored or processed is:

$$(2.37\times10^{-7}) + (2.94\times10^{-7}) + (1.62\times10^{-7}) = 6.93\times10^{-7} \text{ events per year}$$

Therefore, the event of aircraft crashing into DSCs during processing operation is considered incredible.

B.7.2.8.7 Release of Oxidizing, Toxic, Corrosive Gases and Liquids Stored in the Processing Building

For a chemical release to have an impact on nuclear safety, the chemical must fall into one of the following categories under Part IV of the Canadian Controlled Products Regulations (B-CAN18):

- Acute Toxicity;
- Corrosive;
- Oxidizing/Reactive; and
- Asphyxiant.

All chemicals held within the Processing Building are categorized and screened based on quantities held on site. Direct exposure to the categorized chemicals may cause an operator to become incapacitated, leading to container mishandling errors, which at worst, would not result in a dose consequence worse than the drop of a dry storage container.

B.8.0 DRY STORAGE CONTAINER STORAGE BUILDINGS ASSESSMENT

Once the DSC processing is completed, the transporter moves the DSC to one of the DSC storage buildings for storage.

B.8.1 Normal Operating Conditions

B.8.1.1 Direct Radiation Fields

B.8.1.1.1 Dose Rates Outside the Pickering Waste Management Facility NSS-PWMF Phase I Site

Dose rates at the perimeter fence east of Phase 1 are calculated to be less than 0.24 µSv/h. The predicted dose rates are less than 50% of the dose rate target of 0.5 µSv/h.

When the NSS-PWMF DSC storage buildings 1 to 4 are filled to nominal design capacity, the dose rate at the site boundary is calculated to be 1.46×10^{-3} µSv/h. This is equivalent to an annual dose of 2.92 µSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 1.41×10^{-3} µSv/h. This is equivalent to an annual dose of 1.41 µSv based on 1,000 hours occupancy; this is a conservative occupancy assumption for boaters and fishermen.

These results indicate that the NSS-PWMF dose rate target of ≤0.5 µSv/h at the station security fence on a quarterly average basis and the administrative dose target ≤100 µSv per year at the station site boundary (ten percent of the CNSC regulatory dose limit for members of the public), as set out in Section B.5.0, are met during UFDS operations.

B.8.1.1.2 Dose Rates Outside the Pickering Waste Management Facility NSS-PWMF Phase II Site

The calculated dose rates at the north and east perimeter fences of the Storage Buildings 3 and 4 are within the criterion of ≤0.5 µSv/h established for limited occupancy (i.e., up to 2,000 hours per year) by non-NEW personnel.

The calculated maximum dose rates at the west and south perimeter fences of the Storage Buildings 3 and 4 are 0.69 and 0.64 µSv/h, respectively, which is above the 0.5 µSv/h criterion. These calculations are known to be conservative, and actual site measurements, which are recorded in quarterly reports and issued to the regulator, all remain below the 0.5 µSv/h criterion. Other, more appropriately placed, site fences can also be used as boundaries at which the target of 0.5 µSv/h can be applied if necessary. OPG will take every precaution to ensure that administrative dose rate targets are met. The dose consequences for personnel walking or driving on the Pickering site roads (i.e., at further distances and for short time durations) will be well within acceptance criteria.

As discussed in Section 3.4.3, 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 TMB are below the dose rate target of 0.5 µSv/h (see Section B.5.0). The dose rate at the TMB was estimated to be 7.57×10^{-2} µSv/h.

When the NSS-PWMF DSC storage buildings 1 to 4 are filled to nominal design capacity, the dose rate at the site boundary is calculated to be 1.46×10^{-3} µSv/h. This is equivalent to an annual dose of 2.92 µSv based on 2,000 hours occupancy. At the eastern lakeside exclusion zone boundary, the calculated dose rate is 1.41×10^{-3} µSv/h. This is equivalent to an annual dose of 1.41 µSv based on 1,000 hours occupancy. This is a conservative occupancy assumption for boaters and fishermen.

B.8.1.1.3 Dose Rates Inside the Dry Storage Container Storage Buildings

The predicted dose rates from a row of DSCs facing the corridor in the middle of Phase I storage building 2 loaded with 10-year-cooled fuel in storage, are presented in Figure B-7. The results show that the dose rates on the west side are approximately 30 µSv/h, this drops to 12.5 µSv/h in the middle of the corridor and increases to 28 µSv/h on the east side. Note that due to the increased number of DSCs, the dose rates presented for Storage Building 2 will bound those of Storage Building 1.

The predicted dose rates from a row of DSCs facing the corridor in the middle of Phase II Storage Building 3 loaded with an average of 26-year-old fuel in storage, are presented in Figure B-8. The DSC configuration is symmetric on the east and west side. The results show that the dose rates on the west or east side are approximately 18 µSv/h and this drops to 5 µSv/h in the middle of the corridor.

The predicted dose rates from a row of DSCs facing the corridor in the middle of Storage Building 4 loaded with an average of 18-year-old fuel in storage, are presented in Figure B-9. The dose rates on the west and east side are approximately 34 µSv/h and this drops to 9 µSv/h in the middle of the corridor.

B.8.1.2 Radioactive Emissions during Storage

There are no mechanisms for airborne releases to occur under normal operating conditions during storage of seal welded DSCs.

B.8.2 Malfunctions and Accidents Assessments for Operations during Dry Storage Container Storage

B.8.2.1 Seal-Weld Failure during Storage

Both the fuel sheath and the DSC lid 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.

The DSC lid and base are sealed with a full-penetration groove weld. Once the weld has cooled sufficiently for inspection, a 100 percent inspection of the weld is performed to check for any weld defects. The DSC is subsequently filled with inert helium and leak tested prior to storage.

As the seal-welds are inspected and pressure tested, and there is no external force acting upon DSCs in storage, it is concluded that a random weld failure is not a credible event. No instances of random-weld failures have been identified.

B.8.2.2 Dry Storage Container Drop during Transfer to Storage

Failure of the transporter or the DSC lift plates while the DSC is lifted by the transporter during transfer to placement of a loaded DSC in a DSC storage building could result in a DSC drop. This scenario is unlikely given the independent mechanical locking mechanism on each side of the transporter to prevent DSC drop.

The failure probability of one independent mechanical locking mechanism of the transporter is 1.0×10^{-4} (B-OPG18c). Assuming that 150 DSCs are processed per year, the probability of both mechanical locking mechanisms failing simultaneously while carrying a loaded DSC in the storage building would be $(1.0 \times 10^{-4}) \times (1.0 \times 10^{-4}) \times 150 = 1.5 \times 10^{-6}$ events per year. This event is credible.

Given that the transporter is equipped with front and rear bumper emergency stops or sensors, the low-lift height of the DSC while in the Transporter, and that the loaded DSC has already been seal-welded at this stage of the process, no releases would result from this scenario. There would be no public or occupational dose consequences as a result of this event.

B.8.2.3 Transporter Collision with a Dry Storage Container or Another Transporter

Operator error during transporter operations could result in a collision with a loaded DSC on the DSC storage building floor or with another transporter in the DSC storage building.

It is assumed that a maximum of 150 DSCs are loaded and transferred each year. The transporter is used once in the DSC storage building for each DSC. The assumed frequency of the operator error probability is 1.0×10^{-3} (B-Sandia83). The probability of a DSC drop or collision event in the DSC storage building due to operator error would be $(1.0 \times 10^{-3}) \times 150 = 1.5 \times 10^{-1}$ events per year. This event is credible.

Given that the transporter is equipped with front and rear bumper emergency stops or sensors, the low lift height of the DSC while in the transporter, and that the loaded DSC has already been seal-welded at this stage of the process, no releases would result from this scenario. There would be no public or occupational dose consequences as a result of this event.

B.8.2.4 Criticality

See Section B.6.2.4.

B.8.2.5 Dry Storage Container Building Fire

The DSC storage buildings have been designed in accordance with CSA N393, NFCC, NBCC, and all applicable codes and standards referenced therein.

The Fire Hazard Assessment (FHA) for NSS-PWMF (B-OPG22d) evaluated two design basis fires for the Phase I and Phase II storage buildings: a DSC transporter fire and a fire scenario involving up to four Rhino tile pallets.

The DSC transporter fire scenario was assessed crediting the available fire detection, suppression and emergency response. Due to the fire detection and alarm systems in all Phase I and Phase II storage buildings, and the expected prompt arrival of the emergency response personnel, the fire would be of short duration and localized. The effect of the fire would be to increase the temperature in the proximity of the DSC. Given the large thermal inertia of the DSC and the limited duration of the event, the fire inside the DSC storage building will not cause fuel overheating or fuel damage. Therefore, no releases would result from DSCs in storage and there would be no public or occupational dose consequences as a result of this event.

The Rhino tile pallet fire was assessed without crediting the fire detection and suppression systems. The Rhino tile pallets fire (up to four pallets) could produce localized high temperatures. However, the fire will not damage the storage building structures and will not damage DSCs. Therefore, there would be no public or occupational dose consequences as a result of this event.

B.8.2.6 Common Mode Incidents

B.8.2.6.1 Earthquake

The earthquake scenario was described previously in Section B.6.2.5.2.

The DSC has a safety factor of 7 against overturning and 4 against sliding under the loads described for the earthquake scenario (B-OH92). The structure of the container is adequately strong to ensure the integrity of the DSC in case of an earthquake.

Additionally, the impact of a building truss collapse (in different orientations) onto an array of DSCs has been assessed (B-OPG22g). It was concluded that the seal-welded DSCs would maintain their integrity if an earthquake caused a DSC storage building to collapse directly on top of them.

An earthquake would result in no releases from DSCs in storage and there would be no public or occupational dose consequences.

B.8.2.6.2 Tornadoes

As discussed in Section B.6.2.5.3, a DSC can resist overturning in tornado winds of up to 425 km/h. This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the interceding building structures.

The effect of tornado-generated missiles on a seal-welded DSC is bounded by a similar discussion regarding clamped DSCs in Section B.6.2.5.3; the DSC containment will not be breached and the DSC will not overturn under the impact of postulated missiles.

A tornado would result in no releases from DSCs in storage and there would be no public or occupational dose consequences.

B.8.2.6.3 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC storage buildings. As per design requirements, the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during storage. The impact of a lightning strike on the DSC was evaluated (B-OPG22c). It was concluded that DSCs can withstand a direct lightning strike without any negative impact on their functionality. The shielding function of the DSC will not be compromised, and the DSC containment will not be breached. Thunderstorms would result in no releases from DSCs in storage buildings and there would be no public or occupational dose consequences.

B.8.2.6.4 Floods

As discussed in Section B.6.2.5.5, given the characteristics of the Pickering NGS site, extensive flooding affecting a DSC storage building is not a credible event.

Water entry into the NSS-PWMF storage buildings originating from a PMP event is possible. However, the consequences are negligible. The DSCs are seal-welded and designed to tolerate water immersion at 2 MPa (B-OPG22g), so the temporary water levels would not be of a concern to the radiological safety.

B.8.2.6.5 Toxic Materials Stored in Storage Building 3

Toxic chemicals stored in Storage Building 3 are minimally toxic when inhaled. Strict safety procedures and processes are in place for storage and handling of the hazardous chemicals within Storage Building 3. The handling of hazardous materials must meet provincial legislation, particularly the Occupational Health and Safety Act and the Environmental Protection Act.

The dose consequences from this postulated scenario would be within the envelope of those in Section B.6.2.1.

B.8.2.6.6 Turbine Missile Strike

The NSS-PWMF Phase I is located southeast of Unit 8, with Storage Building 2 situated the closest at a distance of 30 m. The DSC Storage Building 2 is attached to the north wall of the processing building and the Storage Building 1.

The break-up (failure) of the Pickering NGS turbines could lead to a high-trajectory missile strike on the dry storage facility. The frequency of a turbine failure is 1×10^{-4} turbine failures per year (B-OPG21). At rated speed, the frequency of a missile from Pickering NGS striking the NSS-PWMF Phase I facility is 4.8×10^{-4} strikes per missile. The most significant missile is a large fragment of Disc 3 from a low-pressure turbine. There are three low-pressure turbines per unit and two Disc 3s in each low-pressure turbine. Considering six large, high-energy missiles are produced following a turbine failure, the frequency of a strike is (4.8×10^{-4} strikes per missile) × (6 missiles), which is approximately 3×10^{-3} strikes per turbine failure.

The frequency of turbine missiles impacting SSCs at Phase I site is (3×10^{-3} strikes per turbine failure) × (1×10^{-4} turbine failures per year) = 3×10^{-7} impacts per year.

The Phase II DSC Storage Building 3 and 4 are located approximately 500 m northeast of the NSS-PWMF Phase I and from Unit 8. These storage buildings are separated not only by distance from the Unit 8 turbine, but they are also shielded by various buildings located between the two facilities.

Based on the low frequency of a turbine missile impacting a SSC and taking into account the location of the Phase I and Phase II DSC Storage Buildings with reference to the Unit 8 turbine, this event is considered to be an incredible event.

B.8.2.6.7 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of NSS-PWMF Phase I and Phase II. The details on the total aircraft crash frequency calculations are documented in B-OPG22b.

The aircraft crash frequency calculated for the NSS-PWMF Phase I area is 2.37×10^{-7} events per year. The aircraft crash frequency calculated for the NSS-PWMF Phase II area is 2.94×10^{-7} events per year. The aircraft crash frequency for the area occupied by the DSMs was calculated to be 1.62×10^{-7} events per year.

The total aircraft crash frequency for the NSS-PWMF holding safety related waste containers, such as DSCs and DSMs, was determined by summation of the frequency of an aircraft crash impacting the DSC processing and storage buildings and RCS area where the DSMs are stored. A qualitative aircraft impact assessment, conducted for the DSC against light general aviation aircraft crashes concluded that the Category 1 aircraft (light aircraft) will not cause damage to a DSC except for slight concrete cracking or scabbing. Therefore, for DSCs, the aircraft crash frequency calculations were performed for aircraft categories 2 to 5 for Phase I and Phase II areas occupied by the DSC Storage Buildings 1 – 4 and the DSC processing building. For the RCS area aircraft crash frequency calculation, all aircraft categories (Category 1 to 5) were considered.

The summation of the above aircraft crash frequencies calculated for the NSS-PWMF site where safety-related containers are stored or processed is:

$$(2.37 \times 10^{-7}) + (2.94 \times 10^{-7}) + (1.62 \times 10^{-7}) = 6.93 \times 10^{-7} \text{ events per year}$$

Therefore, the event of aircraft crashing into DSCs during storage is considered incredible.

B.9.0 CONCLUSIONS

The above sections describe the safety assessment of the NSS-PWMF. The radiological dose consequences to the public and the workers during normal operations and under abnormal and credible accident conditions were evaluated.

It is concluded that under normal operation the dose consequences to the public would be well below 100 µSv per year, an OPG administrative dose target at the Pickering site boundary and ten percent of the regulatory dose limit. The dose rates will also be less than 0.5 µSv per hour, an OPG target based on a quarterly average of 0.5 µSv per hour over a maximum occupancy of 2,000 hours per year at the NSS-PWMF Phase I and Phase II site fences. Occupational doses were also found to be below the dose limit of 50 mSv per year in any single year and 100 mSv over 5 years.

The radiological dose consequences to the public as a result of abnormal and credible accident conditions were concluded to be well below the dose limit of 1 mSv at or beyond the OPG property boundary; the associated dose consequences to a Nuclear Energy Worker were concluded to be well below the limit of 50 mSv. Tables B-6, B-7 and B-8 show the dose consequences for the postulated abnormal and accident conditions during on-site transfer, DSC processing, and DSC storage, respectively.

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B-OPG22g Ontario Power Generation, “Used Fuel Dry Storage, Ontario Power Generation, Dry Storage Container”, 00104-DR-79171-10000 R004, April 2022.

B-ORNL11 Oak Ridge national Laboratory, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design”, ORNL/TM-2005/39, Version 6.1, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

B-Sandia83 Sandia national Laboratory, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”, NUREG/CR-1278, SAND80-0200, August 1983.

B-SCAR78 D. Scargill, “Diffusion of Fission Product Tritium in Irradiated UO2”, J. Nucl. Matr. 74, 1978.

B.11.0 GLOSSARY

ADDAM	Atmospheric Dispersion and Dose Analysis Method
AECL	Atomic Energy of Canada Limited
BLEVE	Boiling Liquid Expanding Vapour Explosion
CANDU	Canada Deuterium Uranium (a registered trademark of AECL)
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners' Group
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DCF	Dose Conversion Factor
DSC	Dry Storage Container
DSM	Dry Storage Module
FHA	Fire Hazard Assessment
HazMat	Hazardous Materials
HEPA	High Efficiency Particulate Air
HT	Tritiated Hydrogen Gas
HTO	Tritium Oxide or Tritiated Water Vapour
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IFB	Irradiated Fuel Bay
MCNP	Monte-Carlo N-Particle
MDA	Minimum Detectable Activity
MSDS	Material Safety Data Sheet
NBCC	National Building Code of Canada
NEW	Nuclear Energy Worker
NFCC	National Fire Code of Canada
NGS	Nuclear Generating Station
NSS-DWMF	Nuclear Sustainability Services – Darlington Waste Management Facility
NSS-PWMF	Nuclear Sustainability Services - Pickering Waste Management Facility
NSS-WWMF	Nuclear Sustainability Services - Western Waste Management Facility
OPG	Ontario Power Generation
ORIGEN-S	Oak Ridge Isotope Generation Code
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
RLC	Review Level Conditions
SDV	Screening Distance Value
SORO	Simulation of Reactor Operation
SSC	Structures, Systems and Components
TMB	Training and Mock-Up Building
UFDS	Used Fuel Dry Storage
VCE	Vapour Cloud Explosion
WIMS	Winfrith Improved Multigroup Scheme

B.12.0 TABLES AND FIGURES

Table B-1: Pickering Reference Used Fuel Bundle Properties

Number of fuel elements	28
Length	495 mm
Mass of UO_2	22.87 kg
Mass of Zircaloy	1.67 kg
Mass of U	20.16 kg
Mass of the bundle	24.54 kg
Average bundle power	[redacted]
Exit burnup	[redacted]
Time after discharge	10 years

Table B-2: Pickering Waste Management Facility Reference Fuel Bundle Actinides and Heavy Metals Inventory (10-year-cooled fuel)

[redacted]

[redacted]

Table B-3: Pickering Waste Management Facility Reference Fuel Bundle Fission Products Inventory (10-year-cooled fuel)

[redacted]

[redacted]

Table B-4: Pickering Waste Management Facility Reference Fuel Bundle Light Elements Inventory (10-year-cooled fuel)

[redacted]

[redacted]

Table B-5: Gamma Spectrum for a 10-year-cooled Fuel Pickering Reference Used Fuel Bundle

			Photons/Second/Bundle			
Energy Band	**Lower MeV**	**Upper MeV**	**Ring 1**	**Ring 2**	**Ring 3**	**Bundle**
00	0.30	0.65	1.538E+11	3.565E+11	9.586E+11	1.469E+12
01	0.65	1.00	1.858E+12	4.155E+12	1.041E+13	1.643E+13
02	1.00	1.25	9.788E+09	2.323E+10	6.533E+10	9.835E+10
03	1.25	1.50	1.345E+10	3.231E+10	9.279E+10	1.385E+11
04	1.50	2.00	1.075E+09	2.495E+09	6.723E+09	1.029E+10
05	2.00	3.00	7.648E+07	1.721E+08	4.403E+08	6.888E+08

Note: Values under the "Bundle" column are the sum of values from fuel elements in Ring 1, Ring 2, and Ring 3.

Table B-6: Postulated Malfunctions or Accidents during Dry Storage Container On-Site Transfer

Malfunction or Accident	Potential for Occurrence (events per year)	Credible Event (Y/N) See Note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Transporter failure	3	Y	< 1.60×10^{-3}	< 1.96×10^{-3}	< 4.7
Transporter Operator Health-Related Emergency	See Note 2	Y	< 1.60×10^{-3}	< 1.96×10^{-3}	< 4.7
DSC drop during on-site transfer from AIFB/IFB to DSC Processing Building	See Note 2	Y	1.60×10^{-3}	1.96×10^{-3}	4.7
DSC drop during on-site transfer between the NSS-PWMF Phase I and Phase II sites	See Note 2	Y	2.43×10^{-3}	2.90×10^{-3}	4.7
Fire	1.04×10^{-7}	N	—	—	—
Criticality	See Note 3	N	—	—	—
Adverse road conditions	See Note 2	Y	< 2.43×10^{-3}	< 2.90×10^{-3}	< 4.7
Earthquake	1.71×10^{-5}	Y	0	0	0
Tornado	5.36×10^{-8}	N	—	—	—
Thunderstorm	See Note 2	Y	0	0	0
Flooding	See Note 2	Y	0	0	0
Explosion along transfer route	5.2×10^{-8}	N	—	—	—
Turbine missile strike	1.03×10^{-7}	N	—	—	—
Aircraft crash	6.17×10^{-10}	N	—	—	—
Toxic gas releases – chlorine from Ajax water treatment plant	See Note 2	Y	0	0	0
Soil failure/slope instability	See note 2	Y	< 2.43×10^{-3}	< 2.90×10^{-3}	< 4.7

Notes:

1. The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.
2. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature, or it is bounded by an assessed credible hazard.
3. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.

Table B-7: Postulated Malfunctions or Accidents during Dry Storage Processing

Malfunction or Accident	Potential for occurrence (events per year)	Credible event (Y/N) See note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Drop of a DSC during handling	3.36×10^{-2}	Y	$< 1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
Equipment drop onto a DSC	1.68×10^{-2}	Y	$<1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
DSC collision during craning	1.50×10^{-1}	Y	$<1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
Transporter collision with a loaded DSC or another Transporter	6×10^{-1}	Y	$<1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
Equipment collision with a loaded DSC during craning	3×10^{-1}	Y	$<1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
Criticality	See Note 2	N	—	—	—
DSC Processing building fire	See Note 3	Y	0	0	0
Earthquake	See Note 3	Y	$<1.60 \times 10^{-3}$	$<1.96 \times 10^{-3}$	<4.7
Tornado	See Note 4	N	—	—	—
Thunderstorm	See Note 3	Y	0	0	0
Flood	See Note 3	Y	0	0	0
Turbine missile strike	See Note 5	N	—	—	—
Aircraft crash	6.93×10^{-7} See Note 6	N	—	—	—
Release of oxidizing, toxic, corrosive gases and liquids stored in the Processing Building	See Note 3	Y	0	0	0
Asphyxiants	See Note 3	Y	0	0	0

Notes:

1. The term incredible is used for those events with frequency of occurrence less than 10^{-6} events per year.
2. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.
3. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature or it is bounded by an assessed credible hazard.
4. Simultaneous occurrence of a tornado (3.13×10^{-6} events per year) and preparation of the DSC for seal-welding scenarios is sufficiently low to be considered an incredible scenario (frequency of occurrence less than 10^{-6} events per year).
5. Considering the location of the DSC processing building with reference to the Unit 8 turbine, a turbine missile striking the processing building and the DSC is considered an incredible event.
6. The calculated frequency of occurrence considers the Phase I, Phase II sites and DSM storage area together.

Table B-8: Postulated Malfunctions or Accidents during Dry Storage Container Storage

Malfunction or Accident	Potential for Occurrence (events per year)	Credible Event (Y/N) See note 1	Potential Maximum Dose Consequence to the Public (mSv)		Potential Maximum Occupational Dose Consequence (mSv)
			Adult	Infant	
Seal-weld failure during storage	See Note 2	N	—	—	—
DSC drop during transfer to storage	1.5×10^{-6}	Y	$<2.43 \times 10^{-3}$	$<2.90 \times 10^{-3}$	<4.7
Transporter collision with a DSC or another Transporter	1.5×10^{-1}	Y	$<2.43 \times 10^{-3}$	$<2.90 \times 10^{-3}$	<4.7
Criticality	See Note 2	N	—	—	—
DSC storage building fire	See Note 3	Y	0	0	0
Earthquake	See Note 3	Y	0	0	0
Tornado	See Note 3	Y	0	0	0
Thunderstorm	See Note 3	Y	0	0	0
Flood	See Note 3	Y	0	0	0
Toxic Material stored in SB3	See Note 3	Y	0	0	0
Turbine missile strike	3×10^{-7}	N	—	—	—
Aircraft crash	6.93×10^{-7} See Note 4	N	—	—	—

Notes:

1. The term incredible is used for those events with frequency of occurrence less than 10^{-6} events per year.
2. 10-year-cooled Pickering used fuel stored in DSCs cannot achieve criticality under any postulated accident scenario.
3. Quantitative screening was not conducted (hazard frequency was not calculated) for this scenario. The event is considered credible based on its nature, or it is bounded by an assessed credible hazard.
4. The calculated frequency of occurrence considers the Phase I, Phase II sites and DSM storage area together.

Table B-9: Calculated Dose Rates from a Dry Storage Container of the Modified Long Module Design, Fully-Loaded with Pickering 10-year-cooled Used Fuel Bundles

Distance from DSC	Position	Radiation Fields (μSv/h)
Contact	Side	33.10
	Front[23]	37.93
	Top	27.22
1 m	Side	15.89
	Front	20.01
	Top	13.09
2 m	Side	8.10
	Front	10.57
	Top	4.06

23 The label 'front' corresponds to the wider face of the DSC and 'side' indicates the narrower face.

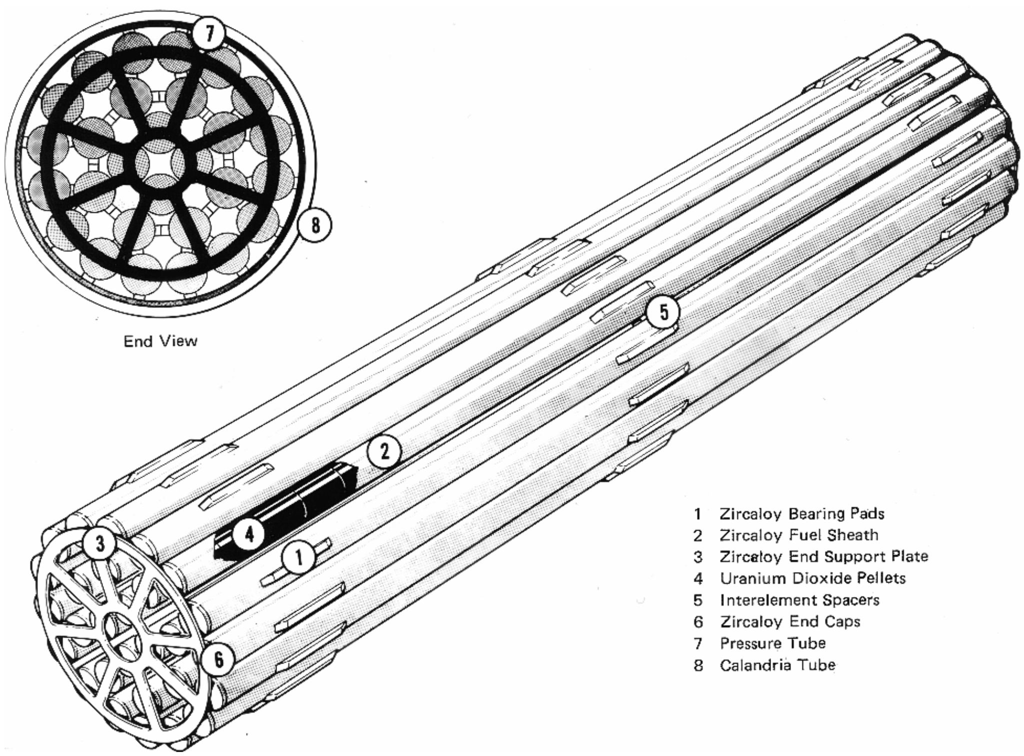

Figure B-1: 28-Element Fuel Bundle

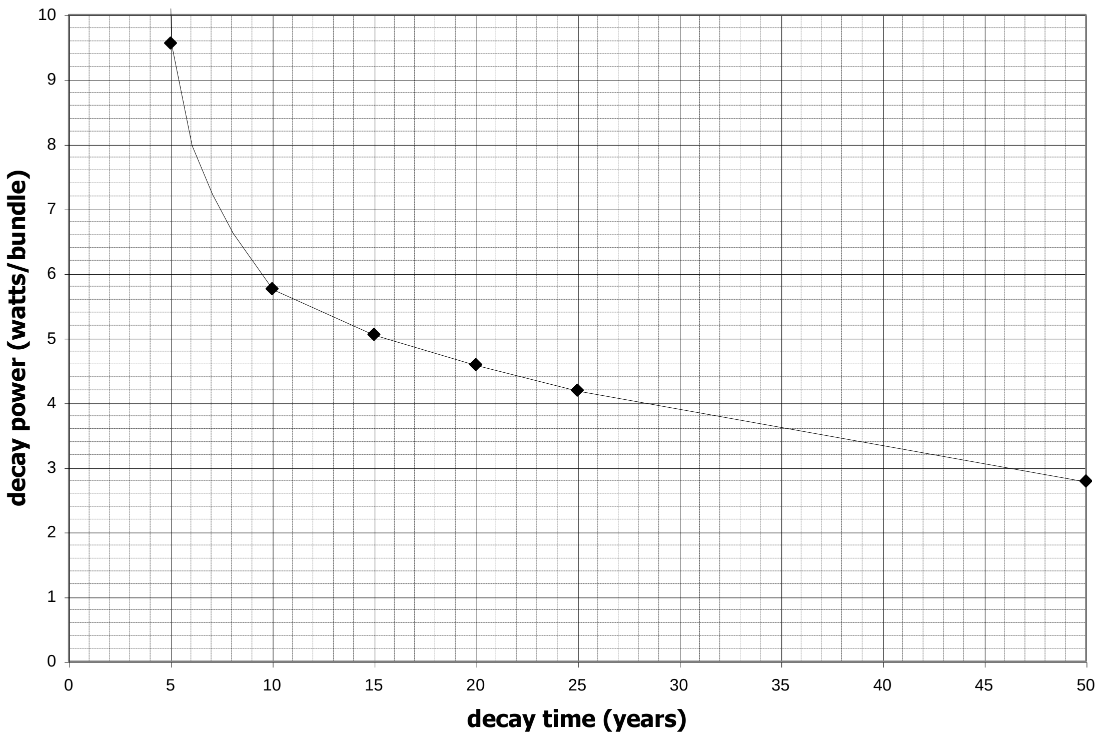

Figure B-2: Total Thermal Power per Pickering Waste Management Facility Reference Fuel Bundle as a Function of Decay Time After Discharge from the Reactor

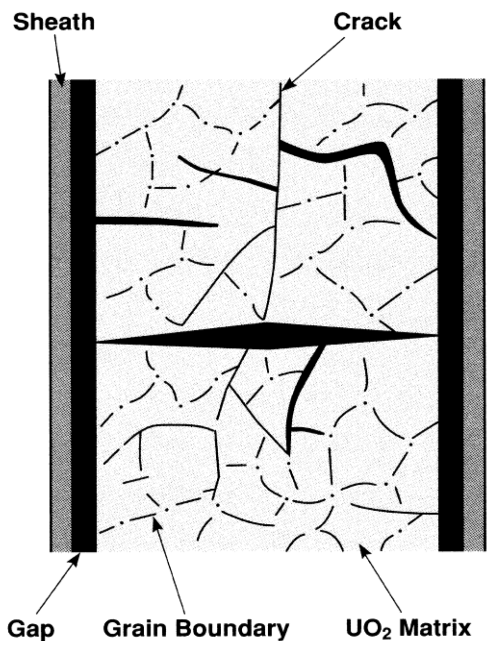

Figure B-3: Locations of Radionuclides in Used Fuel (B-AECL94)

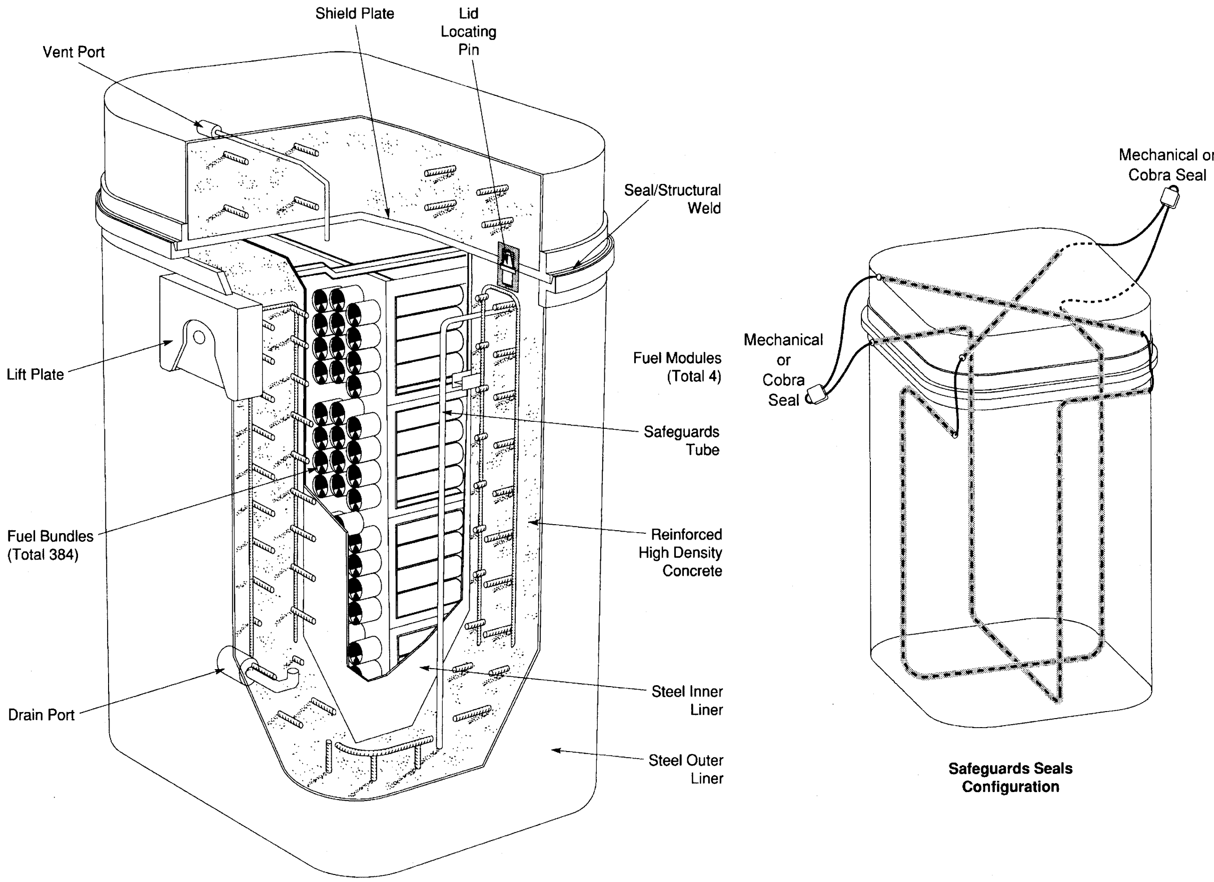

Figure B-4: Dry Storage Container

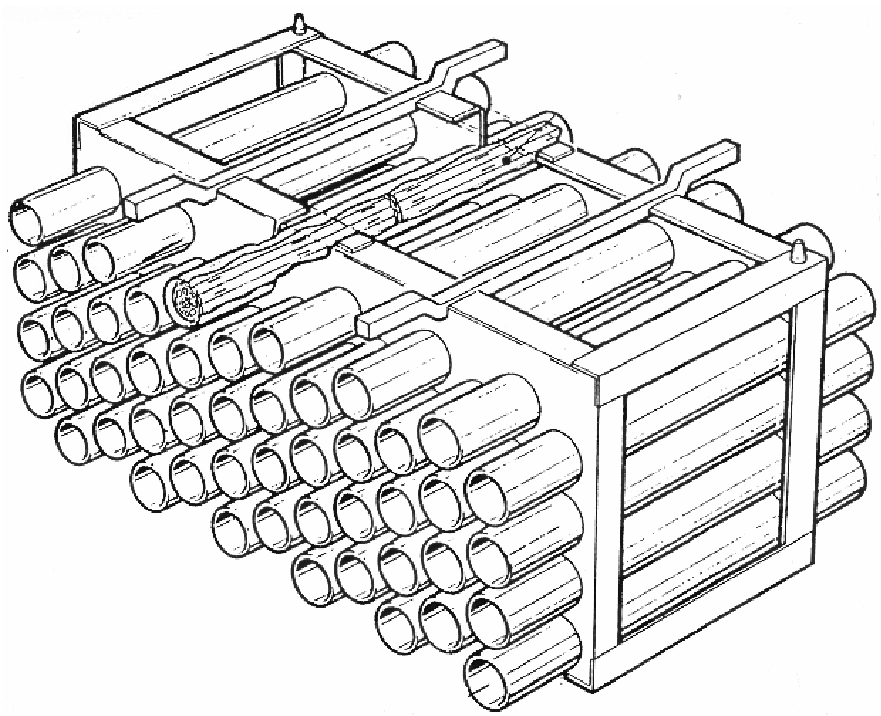

Figure B-5: Storage Module

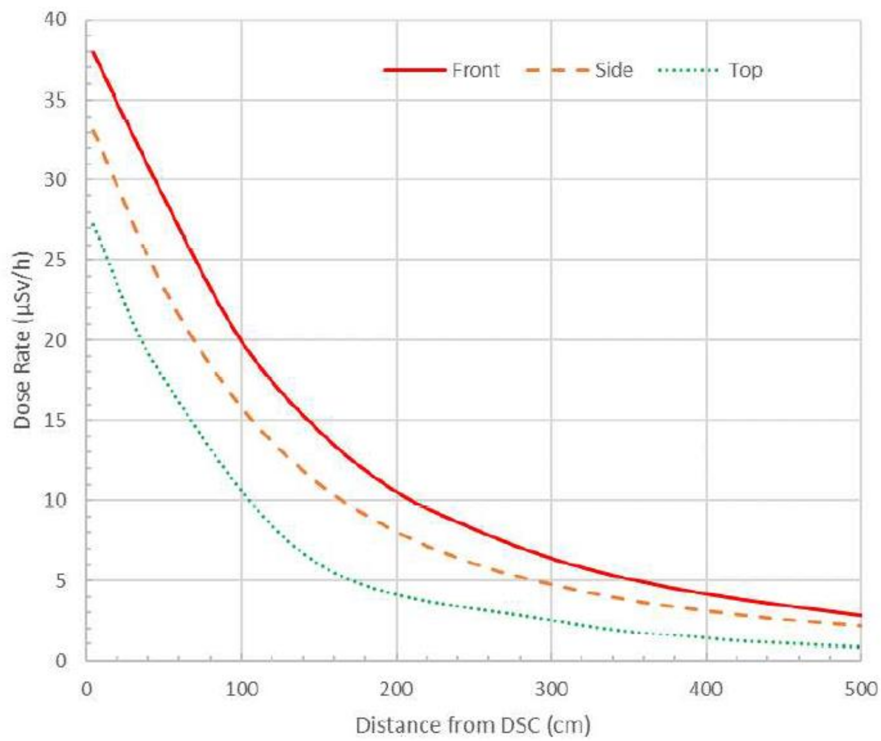

Figure B-6: Dose Rate versus Distance from the Surface of a Single Dry Storage Container with 384 10-year-cooled Pickering Reference Fuel Bundles. The Front Corresponds to the Wider Face of the Dry Storage Container.

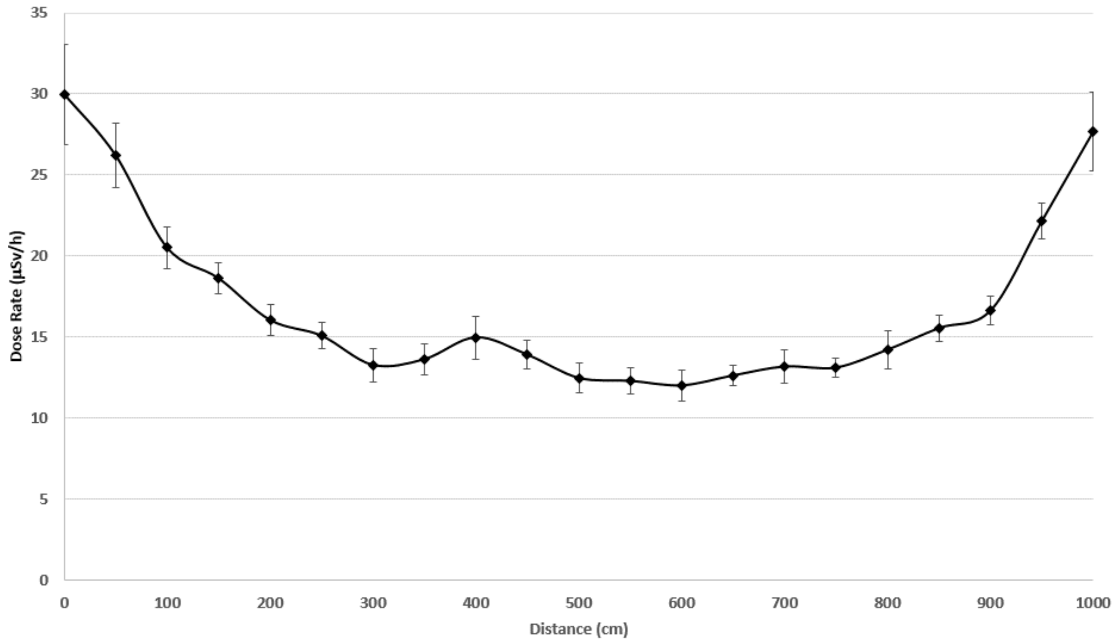

Figure B-7: Calculated Dose Rates Across the Width of the North-South Corridor of Phase I Dry Storage Container Storage Building 2

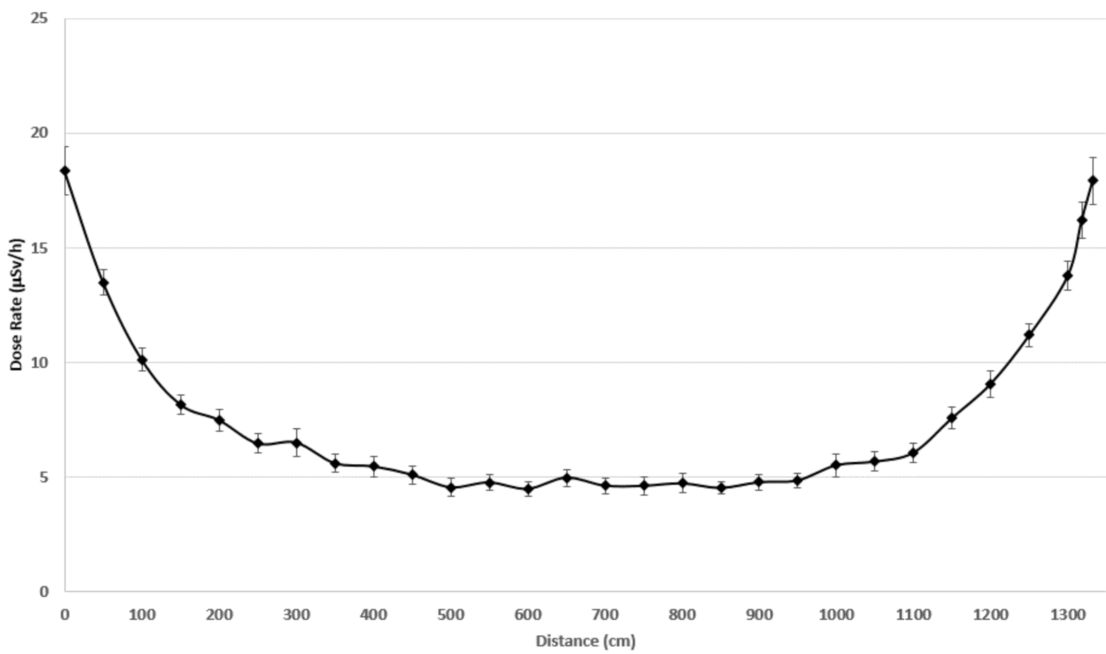

Figure B-8: Calculated Dose Rates Across the Width of the North-South Corridor of Phase II Dry Storage Container Storage Building 3

Dose Rates at SB4 Main Hallway

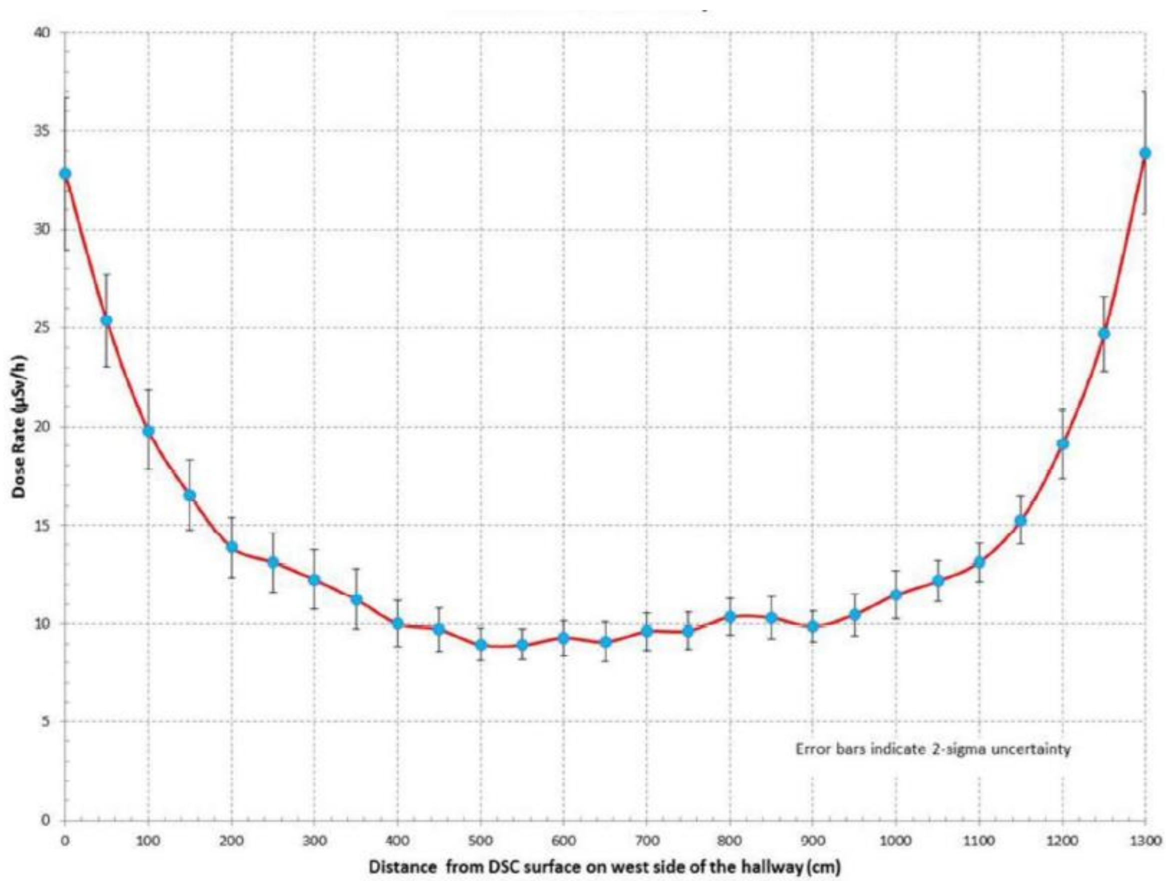

Figure B-9: Calculated Dose Rates across the Width of the North-South Corridor of Dry Storage Container Building 4

Appendix C: Retube Component Storage Area Safety Assessment

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C.1.0 INTRODUCTION

This appendix provides a summary of the NSS-PWMF Retube Component Storage (RCS) area radiological safety assessment during storage. Conservative estimates of public dose rates due to normal operation conditions, abnormal operating conditions, and credible accident conditions are also presented.

C.1.1 Safety Assessment Approach

Under normal operating conditions during storage, DSMs are expected to provide reasonable assurance that the waste can be stored without undue risk to workers, members of the general public, or the environment.

The guidance document in CSA N288.1-20 (C-CSA20) and CSA N288.2-19 (C-CSA19a) was used in performing the safety assessment[24].

RCS waste operations comply with OPG requirements to keep total radioactive emissions under normal operating conditions below regulatory limits and As Low As Reasonably Achievable (ALARA).

The safety assessment of normal and abnormal operating conditions and credible accident conditions is discussed below. In many cases, the scenarios represent bounding abnormal or postulated accident conditions that are improbable or highly unlikely to occur. Design provisions and procedural measures have been introduced as necessary to prevent, mitigate and accommodate the assessed consequences of these conditions

C.1.1.1 Safety Assessment Hazard Screening

Potential hazards for the RCS area were identified and the events were screened in or out based on the OPG hazard screening process for internal hazards (C-OPG18b) and external hazards (C-OPG18a). Events that were screened out were deemed to have a negligible contribution to risk.

First a qualitative screening was conducted and hazards that were judged to have minimal or negligible impact on public or occupational dose were screened out without the need to perform any detailed quantitative assessments.

Quantitative screening criteria, based on event frequency, was then applied to the remaining events. CSA N292.0-19 (C-CSA19b), which provides guidance for the management of radioactive waste and irradiated fuel, defines a credible abnormal event as a naturally occurring or human generated event or event sequence that has a frequency of occurrence equal to or greater than 10^{-6} per year. Using this definition of a credible event, an event screening frequency of 10^{-6} per year was applied to quantitative screening.

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[24] Pending the completion of a gap assessment and subsequent implementation plan, REGDOC–2.4.4 (Safety analysis for Class IB Nuclear Facilities) will be referenced for the guidance of future safety assessments.

C.1.1.2 Computer Codes used in Safety Assessment

The potential doses to individual members of the public were calculated using the ADDAM code (C-COG11). The ADDAM code considers the inhalation, cloudshine and groundshine exposure pathway. An exposure period of 30-days was applied, consistent with CSA N288.2-14.

ADDAM is a safety analysis computer program developed by Atomic Energy of Canada Limited (AECL). ADDAM calculates doses to the public due to a postulated accidental release of radioactive material to the atmosphere from a nuclear facility. Radionuclides being released can be in the form of gases, vapours or small particles. The dispersion of the release is affected by the characteristic of the release, the prevailing meteorological conditions, the surrounding terrain and the nearby buildings. The concentrations in the cloud and on the ground take into account factors such as the nature of the releases, decay, build-up and deposition. Doses are calculated for various age groups and receptor locations, and categorized by release pathways (stack, inlet, leakage, or hole) and exposure pathways (inhalation, cloudshine, groundshine). The calculations of atmospheric dispersion and doses are compliant with CSA N288.2-14.

The IMPACT code (C-EcoMetrix18), which is in compliance with CSA N288.1-14, was utilized to analyse the public dose resulting from routine airborne and waterborne releases from NSS-PWMF. Pathways include in the assessment follow the recommendations given in the CSA N288.1-14 standard. Doses were reported by age group. Although the code was based on the 2014 revision of the standard, the calculation results for the PWMF safety assessment are in compliance with the latest revision of the standard CSA N288.1-20 (C-CSA20).

C.1.2 Acceptance Criteria

The radiation safety requirements under normal operation for NSS-PWMF are as follows:

- ≤ 0.5 μSv/h outside the RCS and UFDS areas, on a quarterly average basis, based on the CNSC dose limit of 1 mSv/year for a member of the public, over a maximum of 2,000 hours per year occupancy for non-NEWs.
- ≤ 100 μSv/year at the Pickering site boundary. This is an administrative dose target of ten percent of the CNSC dose limit of 1 mSv/year for a member of the public.
- The dose limit for NEWs is 50 mSv in any single year, and 100 mSv over 5 years

The radiation requirements considered under an abnormal event or credible accident are the following:

- The dose limit for the public at or beyond the OPG site boundary due to an abnormal even/accident is 1 mSv.
- The dose limit for a worker due to an abnormal event/accident shall be 50 mSv.

The limit of 50 mSv for abnormal event/accident refers to NEWs. The equivalent limit for non-NEWs/members of the public is 1 mSv.

C.2.0 RADIONUCLIDE INVENTORY

Radionuclide inventories were estimated at the time of loading, for increasing decay times from unit shutdown through 50 years. To date, radionuclide inventories in DSMs have decayed over periods ranging from 20 to 27 years (i.e., since the completion of the Pickering NGS A retubing of the respective Units), and will continue to undergo radioactive decay. Estimates of the radionuclide contents and activities for different decay periods are presented in this section.

Dose rate monitoring at the RCS area fence has confirmed that the original estimates remain conservative. The calculated radionuclide content due to the activation of the Pickering Unit 1 (P1), Pickering Unit 2 (P2), Pickering Unit 3 (P3) and Pickering Unit 4 (P4) retube components are given in Table C-1 and Table C-3. The estimated total activities due to activation of the P1/P2 and P3/P4 components are given in Tables C-2 and C-4, respectively and were calculated using the computer code ORIGEN (C-BELL73).

Table C-1: Radionuclide Content of each Pressure Tube, Shield Plug, and End Tube Fitting for P1/P2

[redacted]

Table C-2: Total Activity from P1/P2 Components (TBq)

[redacted]

Table C-3: Radionuclide Content of Each Pressure Tube, End Fitting and Shield Plug for P3/P4

[redacted]

Table C-4: Total Activity from P3/P4 Components (TBq)

[redacted]

The activities for P1/P2 and P3/P4 components differ because of differences in composition of the components. For example, the P3/P4 pressure tubes are made of Zr-2.5 percent Nb whereas the P1/P2 pressure tubes are made of Zircaloy-2 and do not contain niobium. In addition, the degree of conservatism used in the calculations for P1/P2 was greater than that for P3/P4 calculations, which were also based on conservative assumptions. These assumptions are given in Section C.3.3.1.

The values given in Tables C-1 to C-4 do not include loose contamination.

Prior to removal of components, the CANDECON process was used to remove adherent activated corrosion products (crud). On P1/P2, the reactor cooling system was then drained and flushed with demineralized water to remove heavy water. On P3/P4, vacuum drying was used without flushing the reactor cooling system. For Pickering Units 3 and 4, the annulus gas system was oxygenated to convert carbon-14 dust to $^{14}CO_2$, which is removed as a gas.

The following residual contaminants may be present in small amounts associated with the stored components:

(a) Carbon-14 particulate, in the case of Pickering Units 1 and 2, present on the outside of the pressure tubes, end fitting inboard stubs and garter springs.

(b) Activated corrosion products deposited on the inside of the pressure tubes and end fittings, and on the outside of the shield plugs (in the form of loose crud).

(c) Metallic swarf generated during cutting of the pressure tubes and end fittings. Due to the use of roll cutting, the swarf was in the form of slivers, with very little dust.

(d) Tritium in the form of zirconium tritide, which is a very stable compound requiring temperatures of several hundred degrees Celsius for its dissociation.

The amounts of these contaminants are conservatively estimated in Section C.2.1.1. Each year the remaining contaminants are less radioactive due to decay.

C.2.1 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/bags will keep the loose contamination from mixing in the free atmosphere of the DSM.

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.

A conservative estimate of the inventory of loose radioactivity in DSMs is given below. Assessments are for inventories at the time of initial loading. Short-lived radionuclides will have significantly decayed since then.

C.2.1.1 Carbon-14

During the course of retubing, it was discovered that some reactor systems had become contaminated with carbon-14, produced in the annulus gas as a result of the reaction ^{14}N (n,p) ^{14}C. There has been carbon-14 contamination of the P1/P2 components stored in DSMs. For P3/P4 retubing, oxygen was added to the annulus gas system to oxidize the carbon-14. Therefore, only a relatively small amount of carbon-14 is expected in the DSMs loaded with components from P3/P4 retubing. At this time, any remaining carbon-14 is expected to be present predominantly in particulate form.

Carbon-14 measurements on pressure tubes and end fittings for P1/P2 have indicated a maximum of 5.6 kBq/cm^2 (C-AECL91). Using a surface area of 2 m^2 for a pressure tube and for 90 pressure tubes per DSM, this gives a total inventory of 10^{10} Bq per DSM for loose carbon-14 activity.

C.2.1.2 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.

C.2.1.3 Loose Crud

Data on loose crud (defined as crud that can be removed ultrasonically) were obtained for Pickering Units 1, 2 and 3. These data indicate a maximum initial activity of 2.6×10^7 Bq cobalt-60 per m^2 for zirconium alloy surfaces, and 6.7×10^7 Bq cobalt-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.

C.2.1.4 Activated Swarf

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.

C.3.0 NORMAL OPERATION CONDITIONS

C.3.1 Public Dose

Dose to members of the public from normal operation of the NSS-PWMF has been determined based on the latest information on radionuclide emissions, representative group locations, and meteorological data (C-OPG22a). The calculated doses are listed in Table C-5 below. Gamma radiation dose rates under normal operating conditions are discussed further in Section C.3.3. The potential for dose due to hypothetical chronic radioactive emissions is discussed in Section C.3.3.4.

Table C-5: Annual Individual Dose from NSS-PWMF Normal Operation

Radiation Source	Max. annual individual dose (μSv/year)	Dose Receptor Location
External gamma radiation from DSCs and DSMs	2.92	Landside boundary
	1.41	Lakeside boundary
Chronic particulate emission from NSS-PWMF measurements reported in quarterly reports	2.0×10^{-4}	Landside boundary
	7.85×10^{-7}	Lakeside boundary
Postulated volatile releases from DSC processing	4.03×10^{-4}	Landside boundary
	9.29×10^{-4}	Lakeside boundary
Postulated release from DSMs	1.82×10^{-4}	Landside boundary
	7.40×10^{-5}	Lakeside boundary
Total annual individual dose	2.92	Landside boundary
	1.41	Lakeside boundary

C.3.2 Worker Dose

The actual worker doses received during normal operation of the NSS-PWMF are reported in the NSS-PWMF quarterly reports. Pickering operating experience has demonstrated that employee dose is much lower than conservatively estimated through ALARA assessment. The highest total collective dose and individual dose reported during the 2007- 2021 period are 12.6 person-mSv and 1.6 mSv.

The maximum effective dose to NEWs working at the NSS-PWMF is well below the regulatory dose limits for NEWs; 50 mSv in a one-year dosimetry period and 100 mSv in a five-year dosimetry period.

C.3.3 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 RCS normal operation are reported below.

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 in Section C.3.3.1. Results of dose rate monitoring at the RCS area and predicted dose rates at protected area perimeter fences are presented in Section C.3.3.2 and C.3.3.3, respectively. These dose rates measurements are representative of over more than a decade decay time. Dose rates continue to decrease with increasing facility life.

C.3.3.1 Estimated Dose Rates from Dry Storage Modules

The estimated gamma dose rates from a DSM (containing irradiated components), at various distances as a function of time after removal from the reactor are given in Figure C-1 for DSMs with P1/P2 components and Figure C-2 for DSMs with P3/P4 components. A worst case payload, i.e., the DSM contents giving the highest dose rate, was used.

These dose rates were calculated using the state-of-the-art radiation transport code, MCNP 6.1[25] (C-LAN11).

This code is capable of rigorously simulating the stochastic nature of gamma, neutron and electron transport by explicitly modelling the physical nature of their travel through space and their interactions with matter. The MCNP code captures gamma dose rate contributions from irradiated fuel channel components in large arrays of storage containers to provide an accurate, integrated shielding analysis, taking into account all gamma radiation dose pathways.

The MCNP code applies the ‘Monte Carlo’ method of analysis, simulating photon histories explicitly in the modelled geometry. A characteristic of this method is that all of the results

[25] MCNP® and Monte Carlo N-Particle® are registered trademarks owned by Los Alamos National Security, LLC, manager and operator of Los Alamos National Laboratory

(‘detector tally’ results of dose rate values at the specific locations) are always statistical quantities in the form of an estimated mean value and an estimated standard deviation. MCNP utilizes a sufficient number of ‘photon histories’ that the statistical uncertainty is very small. In cases where the estimated statistical uncertainties in the MCNP results are greater than five percent, special care has been taken in the consideration of the results. In such cases, the margin between the target dose rate and the computed value was considered related to the estimated uncertainty, and a judgement was made as to whether the values can be accepted as not exceeding the targets. This methodology is documented in Reference C-OPG17b.

The methodology used to develop the MCNP model of DSMs at NSS-PWMF has been validated/benchmarked (C-OPG17a, C-COG20a) through simulation of TLDs surrounding the Used Fuel Dry Storage (UFDS) buildings. The results show that the predictions using the MCNP model, which is based on the reference methodology, is conservative by 35-60%. As mentioned, a worst case payload is used in the normal operation assessment to maximize dose rate. The overall outcome of the normal operations safety assessment is that the predicted dose rates from the DSMs are considered to be conservative and below acceptance criteria.

This conservative outcome justifies the application of this methodology and the use of MCNP to model large arrays of heavily shielded waste containers in normal operations safety assessments.

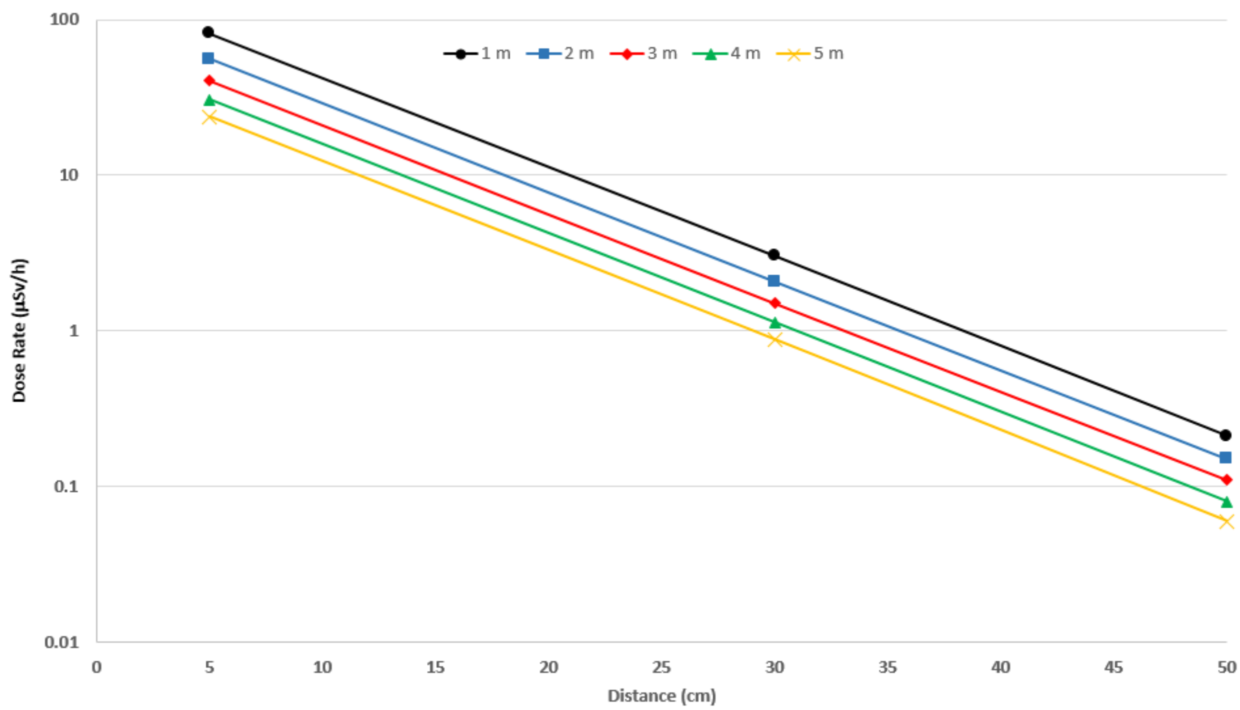

Figure C-1: Estimated Dose Rates from a Dry Storage Module Loaded with P1/P2 Components (C-OPG18)

Notes: For Figure C-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.
- Radiation fields from storage modules containing end fittings are very similar to those from modules containing pressure tubes.
- These dose rates were calculated using conservative early estimates of the activation product inventories.
- The earliest time after shutdown that a DSM was initially loaded was 18 months.

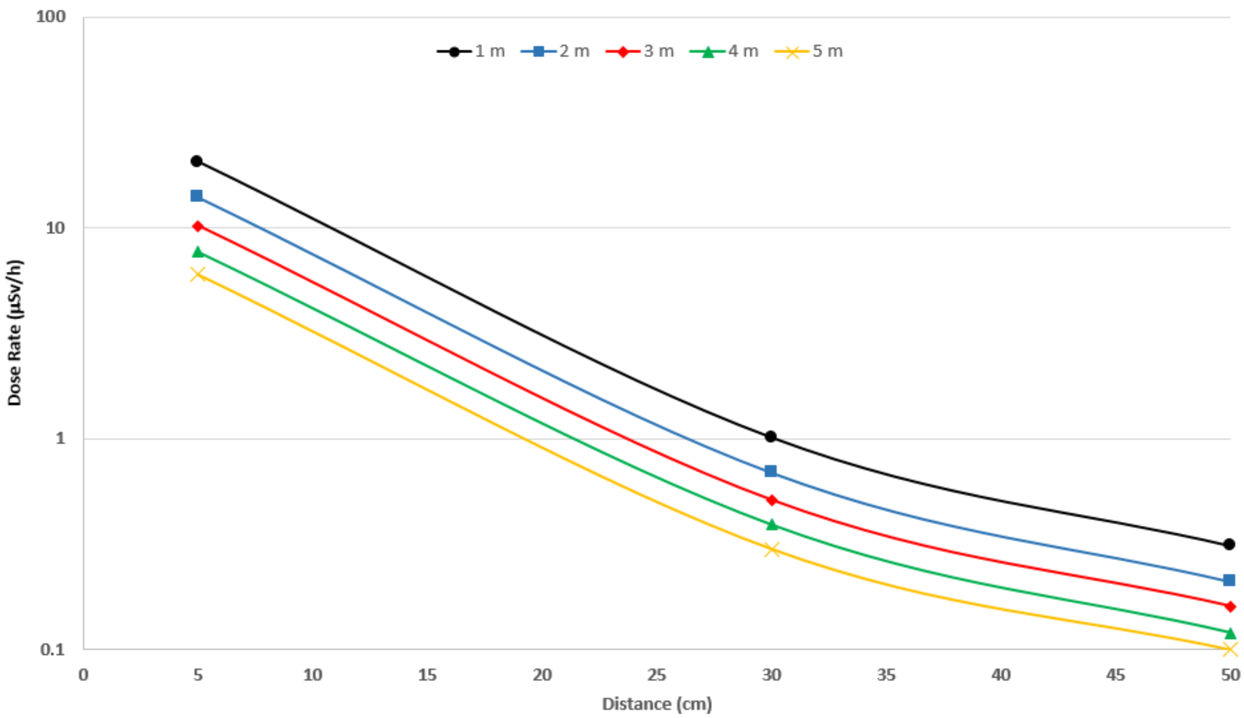

Figure C-2: Estimated Dose Rates from a Dry Storage Module Loaded with P3/P4 Components (C-OPG18)

Notes for Figure C-2:

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

C.3.3.2 Results from Dry Storage Module Dose Rate Monitoring

Comparisons between available measurements and the corresponding calculated quantities have historically yielded good agreement. The activity content and radiation fields associated with pressure tubes and end fittings have been shown to be well predicted by the analytical methods used.

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.

In ten DSMs containing P3/P4 pressure tubes with shield plugs, at some specific locations, notably toward the underside, measurements of contact dose rates exceed calculated estimates by a factor of 3 to 4 over limited areas of the surface. These locations of higher than expected dose rates occur at about 1.8 m from the loading end, towards the saddle area, within a 0.5 m wide band.

Due to correlation with position, as well as evidence from other shielded flasks, these elevated dose rates are believed attributable to the shield plugs pushed in toward one end of each pressure tube. Higher than expected activity levels are suspected to be the result of greater cobalt impurity content in the shield plugs' stainless steel than was assumed in the calculations.

High contact dose rates (up to 6.5 mSv/h) have been measured in localized spots near the top of the shield doors of six DSMs. These areas of elevated dose rate are only a few centimetres in width at the DSM surface, and are sufficiently scattered such that there are no discernible beams near the DSM face. Maximum dose rates were assessed to be below about 700 µSv/h at 30-cm distance, and below about 100 µSv/h at 1 m. The highest contact dose rate from routine surveys of the designated survey locations on the DSMs is currently 79 µSv/h (7.9 mrem/hr).

The localized spots probably correspond to small gaps in the DSM shielding wall, perhaps at the gasket seal near the top of the door. The dose rates drop to ambient background at ground level, and are therefore relevant only for employees using a ladder or platform to perform close work such as painting the shield door. Where contact work near the shield doors is required, controls such as dose rate survey monitoring, and minimizing exposure time in elevated radiation fields, are employed to keep occupational dose ALARA.

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 because 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.

C.3.3.3 Dose Rates from the Retube Components Storage

(a) 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 TLDs 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 2007 - 2021 TLD survey monitoring results, the maximum dose rates are 0.16 µSv/h at the south fence, 0.11 µSv/h at the east fence, 0.12 µSv/h at the west fence and 0.34 µSv/h at the north fence.

(b) Calculated Dose Rates

The dose rate at the limiting location where non-NEW may be present was calculated (C-OPG16) and found to be well within the 0.5 µSv/h dose rate target. The highest dose rate at locations east of the DSC storage buildings 1 and 2 is 0.24 µSv/h. The dose rate at locations east of the RCS is 0.22 µSv/h.

C.3.3.4 Chronic Radioactive Emissions

Annual public doses from releases during normal operation were calculated at the Pickering NGS site boundary and at the critical group locations, which include urban residents, dairy farm and farm residents, industrial/commercial worker, correctional institute resident and sport fisher.

The public dose estimate was performed using the methodology outlined in the CSA N288.1-14 standard. The methodology covers releases to the atmosphere and to surface water (both fresh and marine).

C.3.3.4.1 Potential Airborne Emissions

Conservative estimates were made previously for emissions of particulate contamination. However, routine contamination survey monitoring conducted since the RCS area at the NSS-PWMF 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 canisters;
- The gasketed seal of the DSMs; and

- 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 container).

Carbon-14 particulate 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 percent per year or about 0.2 percent per week (C-OH84). This conservative estimate assumes that moisture may be present in the DSM atmosphere.

Based on conservative assumptions (C-OH84), 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 P1/P2 components were considered. Since carbon-14 is expected to be in stable form after over 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 carbon-14 as $^{14}CO_2$ is:

$$1.0 \times 10^{10} \text{ (Bq/DSM)} \cdot 0.2\%/\text{wk} \cdot 16 \text{ DSMs} = 3.2 \times 10^{8} \text{ Bq/wk (8.6 mCi/wk)}$$

The chronic off-site dose consequence resulting from this postulated scenario, for a member of the public at the most exposed age group/location is estimated to be 1.82×10^{-3} µSv/year.

C.3.3.4.2 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 RCS area basins confirms that contamination levels are generally below the Minimum Detectable Activity (MDA) of 14 Bq/L (3.8×10^{-7} µCi/mL).

The storage modules 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 storage 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 storage 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. Routine monitoring of surface water in RCS area basins confirms that contamination levels are generally below the MDA.

C.3.3.4.3 Comparison with the Derived Release Limits

Compared with the Derived Release Limits (DRLs) of carbon-14 for Pickering NGS B, the emission of carbon-14 (as $^{14}CO_2$) from the RCS is only 6×10^{-4} percent of the Pickering NGS carbon-14 operational DRL for airborne releases.

C.4.0 RADIOLOGICAL SAFETY – ABNORMAL OPERATIONS

C.4.1 Abnormal Events

C.4.1.1 Earthquakes

The DSMs were designed in accordance with CSA N289 (Seismic Design for CANDU Nuclear Power Plants). The DSM supports have been designed to withstand a horizontal ground acceleration of 10 percent of gravity, and the DSM design is qualified to an acceleration of 3 percent of gravity and for a 1.2 m drop.

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 NSS-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 B Design Basis Earthquake (DBE) ground motion as the design basis event. It was found that the required safety function of the DSMs was met.

The reinforced concrete footings of the DSM supports are founded in a compacted gravel "working surface", installed when the area was used for construction storage and laydown. The loading from the DSMs has not resulted in detectable settlement. The foundations are considered adequate to support the DSMs under normal static loads. However, the variable fill deposit underlying the DSM foundations could amplify earthquake ground motion. In order to address potential effects of dynamic loads on the foundations, a detailed seismic capacity assessment of the DSMs was undertaken, and is described below.

The DSM storage site foundation conditions are similar to those for the adjoining UFDS buildings and are described in Section 2.8.1 of this report. Amplified ground motion resulting from the soft foundation conditions was accounted for in the seismic analysis. A finite element model of the DSM, its support frame, concrete footings and the soil foundation was subjected to the Pickering NGS B DBE ground motion using the response spectral analysis method (C-LEE94). Conservative soil and structural component damping values were used in the analysis.

The structural members of the support frames were found to have adequate seismic capacity, and the DSM container itself to have low seismic stresses. The assessment indicated seismic stresses in anchor bolts at the base of the support pedestals and in field welds at the top of pedestal connections to the underside of the DSM support frame which were at, or marginally exceeded, code allowable seismic stress levels. If these connections were to fail, structural stability of the DSM supports could be jeopardized, allowing the DSM container to fall to grade.

The structural implications of the DSM container falling to grade as a consequence of an earthquake were examined. ABAQUS/Explicit finite element computer code (C-HIBBIT94) was used to simulate dynamic impact of the DSM container on the concrete footing and to analyze non-linear material/geometry effects at the point of impact (C-LEE95). It was found that plastic deformation of the outer steel shell would occur with crushing of the adjacent concrete of the DSM container in a limited zone at the point of impact. Crushing of concrete occurred to a depth less than one-half the concrete wall thickness. The damage was very limited in extent and would be repairable. The inner liner and concrete surrounding it remained intact and in the elastic range. The loading port area also was not distorted. No path for radiological release to the environment would, therefore, be created by the impact.

Based on the above assessments it was concluded that there are no radiological implications for DSM storage in the NSS-PWMF for the seismic design basis event.

C.4.1.2 Flooding

The RCS area is located at an elevation of 77.4 m. Each DSM is set on 1 m high pedestals, atop a 1.8 m x 5.5 m x 0.6 m concrete foundation.

Review Level Conditions (RLC) were developed for natural hazards, and the RLC for Probable Maximum Precipitation (PMP) was determined to be 12-hour precipitation, equivalent to 420 mm of total rainfall with 51% occurring in the 6^{th} hour. The RCS area shall have a maintenance free, well drained non-water-ponding surface and the storage saddles shall not trap water at the surface where they contact the DSM. Taking into account the location of the RCS and the elevated storage of the DSMs, the probability of flooding-induced failures from PMP is negligible.

The 500-year maximum still water level is 76.6 m and the maximum recorded wave uprushes is 2.20 m, so the maximum wave run-up heights would reach 78.8 m. This means that the DSMs are still 0.2 m above the maximum recorded lake levels, including wave uprushes. In addition, a 3.3 m high embankment protects the RCS area from wave action in case of a storm. The probability of flood water coming in contact with a DSM is considered 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.

C.4.1.3 Fires and Explosions

Two fuel oil tanks located in a dyked area approximately 50-60 m west of the NSS-PWMF are the only potential fire and explosion hazard near the site. A detailed assessment of this hazard has been carried out (C-NATHWANI84) 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 storage modules, it was assumed that the initiating event caused the oil tanks to rupture and all the fuel draining into the dyked area was involved in the fire. It was judged that a fire which 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, have 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 as a result of the analysis 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 under storage at the NSS-PWMF.

In addition, based on the latest NSS-PWMF Fire Hazard Assessment (FHA) (C-OPG22c), a fire involving the fuel oil storage tanks would result in a maximum heat flux of 6.1 kW/m^2. Therefore it was concluded that based on the magnitude of the heat flux, the fire originating from the fuel oil storage tanks would not impact the integrity of the DSMs.

The RCS area is protected from the lake by a 3.3 m high embankment. Also, boats are not permitted to dock on the shore near the NSS-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 under storage at the NSS-PWMF.

C.4.1.4 Aircraft Crash

The probability of an aircraft striking a DSC or DSM is proportional to the target area of NSS-PWMF Phase I and Phase II. The details on the total aircraft crash frequency calculations are documented in C-OPG22a.

The aircraft crash frequency calculated for Phase I is 2.37×10^{-7} events per year.

The aircraft crash frequency calculated for Phase II is 2.94×10^{-7} events per year.

The aircraft crash frequency for the area occupied by the DSMs was calculated to be 1.62×10^{-7} events per year.

The total aircraft crash frequency for the NSS-PWMF holding safety related waste containers, such as DSCs and DSMs, was determined by summation of the frequency of an aircraft crash impacting the DSC processing and storage buildings and RCS area where the DSMs are stored. For the RCS area aircraft crash frequency calculation, all aircraft categories (Category 1 to 5) were considered.

The summation of the above aircraft crash frequencies calculated for the NSS-PWMF site where safety-related containers are stored or processed is:

$$(2.37 \times 10^{-7}) + (2.94 \times 10^{-7}) + (1.62 \times 10^{-7}) = 6.93 \times 10^{-7} \text{ events per year}$$

Therefore, aircraft crashing into a DSC/DSM at the NSS-PWMF is considered incredible.

C.4.1.5 Extreme Climatic Conditions

(a) Thunderstorms

Lightning would be the only consequence of a thunderstorm which could pose concern for the RCS, from the safety viewpoint. The lightning strike impact on the DSM has been evaluated (C-OPG22b). Direct lightning flashes striking a DSM will ablate the metal surface at the attachment point where lightning touches the DSM metal surface. The DSM safely conducts the lightning current to its base. The metal-reinforced concrete pad supporting the DSM provides a suitable ground electrode and will carry lightning current safely from the base of the DSM to ground without damaging other DSMs nearby.

The lightning hazard is credible; however, since lightning strikes are within the design basis of the DSM (C-OPG02), this hazard can be screened out.

(b) Tornadoes

The effect of DBT-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 skin could be penetrated, but that the concrete would stop the missile. The maximum penetration calculated for a variety of missiles was 145 mm, compared with the DSM wall thickness of 570 mm. The maximum scabbing thickness calculated (i.e., the thickness required to prevent spalling of concrete from the inner surface) was about 570 mm, however, the inner steel liner would prevent loss of concrete.

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

C.5.0 CONCLUSION

This appendix describes the safety assessment of the RCS area at the NSS-PWMF. The dose consequences to the public and the workers under normal operations and considering potential accidents and malfunctions were evaluated.

It is concluded that under normal operation the dose consequences to the public are well below the OPG administrative dose target at the site boundary (ten percent of the regulatory dose limit) and at perimeter fences. Occupational dose were also found to be below the acceptance criteria.

All potential malfunctions and accidents that could impact the DSMs in storage have been found to be non credible given the event frequency cut off of 10^{-6} events per year and therefore there are no radiological consequences to the public.

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C.7.0 GLOSSARY

ADDAM	Atmospheric Dispersion and Dose Analysis Method
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DRL	Derived Release Limit
DSC	Dry Storage Container
DSM	Dry Storage Module
FHA	Fire Hazard Assessment
MDA	Minimum Detectable Activity
NSS-PWMF	Nuclear Sustainability Services - Pickering Waste Management Facility
PMP	Probable Maximum Precipitation
PNGS	Pickering Nuclear Generating Station
RCS	Retube Component Storage
RLC	Review Level Condition
TLD	Thermoluminescent Dosimeter
UFDS	Used Fuel Dry Storage

Appendix D: Dry Storage Module Storage Area Utilization Summary

[redacted]

ATTACHMENT 3

CD# 92896-CORR-00531-01510 P

Summary of Enclosure 1 and Enclosure 2 submitted under CD# 92896-CORR-00531-01430, "*OPG Response to CNSC Staff Comments on OPG's Proposal to Store Minimum 6- Year Old Cooled Used Fuel at the Pickering Waste Management Facility*"

ATTACHMENT 3

Summary of Enclosure 1 & Enclosure 2 for CD# 92896-CORR-00531-01430

Summary of Enclosure 1: Thermal Modelling of OPG Used Fuel Dry Storage Container (2004)

This report presents the results of the thermal analysis of the OPG Used Fuel Dry Storage Container (DSC) during interim storage, i.e. short-term DSC storage scenario for standard and long module types. This analysis looks at a DSC containing 10-Year-Old fuel.

The DSC, either the long module design or the standard module design, is a free standing reinforced concrete container with an inner steel liner and outer steel shell. For the purpose of this analysis, the fuel bundles inside the DSC were assumed to be 10 year old. It was also assumed that the outer steel shell was exposed to the surrounding air under ambient storage conditions.

The natural convection plays an important role as a heat transfer mechanism, together with radiation. However, under total vacuum, the convection contribution is equal to zero. As a consequence, the region of higher temperatures is found around the center of the DSC rather than near the top.

Thermal analysis was performed at two pressures: 93 kPa which is the internal pressure of a standard sealed DSC and 0 kPa which simulates a total vacuum. The thermal analysis results reported are as follows:

At 93 kPa the maximum and average temperature on the inner liner are approximately 71 degrees Celsius and 62 degrees Celsius respectively. The maximum and average temperature on the outer liner are approximately 45 degrees Celsius and 43 degrees Celsius respectively. The fuel sheath maximum temperature is approximately 115 degrees Celsius.

At 0 kPa the maximum and average temperature on the inner liner are approximately 77 degrees Celsius and 67 degrees Celsius respectively. The maximum and average temperature on the outer liner are approximately 46 degrees Celsius and 44 degrees Celsius respectively. The fuel sheath maximum temperature was approximately 145 degrees Celsius.

It was found that if the cavity is under total vacuum, the maximum fuel sheath temperature was approximately 30 degrees higher than if the cavity pressure was 93 kPa. It was also demonstrated that other interior temperatures are affected, but to a lesser degree.

Summary of Enclosure 2: Thermal Modelling of OPG Used Fuel Dry Storage Container Sensitivity Analysis (2006)

Enclosure 2 was a Computational Fluid Dynamics (CFD) analysis aimed at exploring the sensitivity of particular parameters to predicted thermal results. This analysis looks at a DSC containing 10-Year-Old fuel.

The value for module tube emissivity was varied with two values being tried: a low value and high value. These represented the boundaries of the range reported in technical literature.

The value of thermal conductivity for the high density concrete was also varied and two values were tested; a low value and a high value.

In this study the assumption that heat generation is uniformly distributed on all tube surfaces in all four modules, was corrected with new values representing a non-uniform distribution of heat generation in all four modules.

The main conclusions were as follows:

- Thermal conductivity of the concrete shows a stronger impact on predicted inner liner temperatures than the emissivity of module tubes.
- By increasing the thermal conductivity, the correlation between predicted results and measurements was improved by approximately two degrees.
- An increase or decrease in the module tubes emissivity from the nominal value had no significant impact on the predicted results.
- By decreasing the value of the overall heat generation inside the DSC, the correlation between the predicted results and measurements was improved.
- It can be expected that correct values of heat input together with an increased value of thermal conductivity of concrete would further reduce the over prediction of computational results.
- There was not a significant difference in predicted inner liner temperatures when using either the ‘actual’ or ‘average’ distribution of bundle power. Only when there was a highly non-uniform heat distribution were there marked changes in predicted temperatures.
- The changes in thermal conductivity of the concrete, emissivity of the module tubes, and heat input had similar effects on predicted temperatures, wall heat flux, and radiative wall heat flux of the module tubes, as they did for the predicted temperatures on the inner liner.