ONTARIO POWER GENERATION

Kapil Aggarwal, M. Eng., P. Eng.
Vice President
Nuclear Sustainability Services

889 Brock Road, Pickering, ON, L1V 1W2 Tel: (416) 402-6484 kapil.aggarwal@opg.com

OPG Proprietary

June 20, 2023

CD # 92896-CORR-00531-01478

MR. DENIS SAUMURE
Commission Registrar

Canadian Nuclear Safety Commission
280 Slater Street
P.O. Box 1046, Station B
Ottawa, Ontario
K1P 5S9

Dear Mr. Saumure:

OPG – Change Request Application for Amendment to the Pickering Waste Management Facility (PWMF) Waste Facility Operating Licence WFOL W4-350.00/2028

The purpose of this letter is to submit to the Canadian Nuclear Safety Commission, herein referred to as "the Commission", a change request application for Pickering Waste Management Facility (PWMF) under Waste Facility Operating Licence (WFOL) WFOL-W4-350.00/2028, to be able to store younger than 10-year cooled fuel from the Pickering Nuclear Generating Station (PNGS). The younger fuel would have a minimum 6-year cooling period. The change request application has been drafted per the CNSC's direction in Reference 1. OPG had previously communicated the operational need for this activity in Reference 2.

To support the OPG Safe Storage Project for Pickering Nuclear Generating Station (PNGS), additional space in the PNGS-B Irradiated Fuel Bay (IFB-B) is required in order to accept the discharged used fuel from the required core dumps. As PWMF is currently waiting for IFB-B used fuel to mature to the 10-year required period before transferring, there is a need to accept younger fuel to allow for the additional space. At this time, however, OPG is only licensed to process minimum 10-year cooled fuel at all its Nuclear Waste Facilities.

Attachment 1 describes how the licensing basis for the proposed activity, as defined in OPG's application, will accommodate the pertinent clauses of relevant regulatory requirements.

Attachment 2 provides a description and key attributes of the storage of minimum 6-year cooled fuel and documents the licensing impact assessment on all 14 Safety and Control Areas of PWMF's WFOL. Enclosures 1-3 are OPG documents that are being provided to support this assessment.

Enclosure 4 contains an assessment of the findings of a previously trialed Dry Storage Container containing 6-year cooled fuel in 1998. The CNSC requested this evaluation in Reference 3. Prior to that, OPG had submitted technical assessments to the CNSC related to the storage of minimum 6-year cooled fuel (References 4, 5) (Enclosures 1, 2 and 3).

The design considerations of the storage of minimum 6-year cooled fuel complies with all applicable regulatory requirements. The safety assessment, which is referred to as the "safety case", demonstrates that the storage of minimum 6-year cooled fuel will have no significant impact on the continued safe operation of the PWMF, and on public, employee and environmental safety, as is defined in the following elements:

- **Design:** OPG has and will continue to follow its established Engineering Change Control (ECC) process for ensuring the design complies with applicable regulatory requirements and that configuration management for the station will be maintained

- **Continued Safe Operation of PWMF:** Safety analysis submitted to CNSC staff demonstrates that the storage of minimum 6-year cooled fuel will have negligible effect on the safe operation of the PWMF, and on worker and public safety.

- **Environmental Protection**: the conclusions of the PWMF Phase II EA are still considered fully valid with the storage of minimum 6-year fuel.

- **Licensing Basis**: As documented in Attachments 1 and 2, the storage of minimum 6-year cooled fuel will have a negligible impact on PWMF's licensing basis, governance, and its well-established programs and processes.

OPG is targeting to start loading DSC's with minimum 6-year cooled fuel in July 2024. After an initial loading of two to four DSC's to confirm temperature and dose measurements, a full campaign of loading younger fuel will commence.

In summary, OPG remains committed to the safe operation of the PWMF and re-affirms that younger than 10-year cooled fuel can be stored safely as presented in the associated safety case.

If you have any questions, please contact Mr Cliff Barua, Senior Advisor, Regulatory Programs and Support Strategies, at (416) 526-5075 or cliff.barua@opg.com.

Sincerely,

Kapil Aggarwal

Kapil Aggarwal, M. Eng., P. Eng.
Vice President
Nuclear Sustainability Services

Enc.

cc: N. Petseva - CNSC (Ottawa)
T. Kalindjian - CNSC (Ottawa)
S. Watt - CNSC (Ottawa)
R. van Hoof - CNSC (Ottawa)

References:

1. CNSC letter, N. Greencorn to J. Van Wart, “CNSC Staff Response to OPG’s Submission – Notice of Intent to Store Minimum 6-Year Old Used Fuel at the Pickering Waste Management Facility”, April 5, 2022, 2018, CD# W-CORR-00531-01819, e-Doc 6755368.

2. OPG letter, J. Van Wart to N. Greencorn, “Notice of Intent to Store Minimum 6-Year Old Used Fuel at the Pickering Waste Management Facility”, February 1, 2022, CD# W-CORR-00531-01801.

3. CNSC letter, T. Kalindjian to K. Aggarwal, “CNSC Staff Review of OPG Responses to CNSC Staff Comments - Proposal to Store Minimum 6-Year Old Cooled Used Fuel at the Pickering Waste Management Facility”, December 20, 2021, CD#92896-CORR-00531-01443, e-Doc 6687357

4. OPG letter, K. Aggarwal to T. Kalindjian, “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”, June 14, 2021, 92896-CORR-00531-01430.

5. OPG letter, K. Aggarwal to G. Steedman, “Proposal to Store Minimum 6-Year Old Used Fuel at the Pickering Waste Management Facility, November 5, 2020, CD# 92896-CORR-00531-01397, e-Doc 6416392.

List of Enclosures:

1. OPG technical memo, “Storage of Dry Storage Containers (DSCs) containing less than 10 year old used fuel bundles at the Pickering Waste Management Facility (PWMF)”, July 27, 2020, W-CORR-00531-01662

2. OPG report, “Safety Assessment Storing Lower Aged Fuel in PWMF SB3”, 92896-REP-01320-00012

3. OPG report, “Dose Rate Assessment Considering Lower Aged Fuel in PWMF SB3”, 92896-REP-03200-00009

4. OPG letter, P. Dinner to Y. Mroueh, “Additional Information Concerning: Thermal Gradients Pertaining to Dry Storage Containers (DSCs)”, May 4, 2005, 00104-CORR-79171-0139942

ATTACHMENT #1

Licence Amendment Matrix
Applicable Regulations

ATTACHMENT 1

Licence Amendment Matrix - Applicable Regulations

This Attachment, along with the accompanying letter and Attachment 2 of this submission, provides the information required by the Nuclear Safety and Control Act and the applicable Nuclear Regulations made pursuant to the Act, and constitutes an application by OPG to amend the current Pickering Waste Management Facility (PWMF) Waste Facility Operating Licence (WFOL) WFOL-W4-350.00/2028.

The tables below are divided by applicable Regulation and demonstrate how OPG has addressed each applicable regulatory requirement of the subject Regulation.

Nuclear Safety and Control Act		
Section	**Requirement**	**OPG Response**
	Licences	
24(2)	***Application*** *The Commission may issue, renew, suspend in whole or in part, amend, revoke, or replace a licence, or authorize its transfer on receipt of an application:* *(a) in the prescribed form;*	This submission (letter and attachments) provides the information required by the Nuclear Safety and Control Act (referred to as the Act) and the Regulations made pursuant to the Act and provides supplemental information in support of OPG's application for licence amendment. This requirement has been met.
	(b) containing the prescribed information and undertakings and accompanied by the prescribed documents; and	See response above under clause 24 (2) (a).
	(c) accompanied by the prescribed fee.	OPG is in good standing with respect to the provision of CNSC licensing fees and will provide any additional fees associated with this WFOL amendment request, if requested.
24(4)	***Conditions for issuance, etc.*** *No licence may be issued, renewed, amended or replaced - and no authorization to transfer one given - unless, in the opinion of the Commission, the applicant:*	OPG understands that qualification will be determined through consideration by the Commission of this application and the associated supporting material, as well as deliberation through the Commission decision-making process.

Nuclear Safety and Control Act		
Section	**Requirement**	**OPG Response**
	(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and	OPG is qualified to safely undertake the additional activities associated with the storage of minimum 6-year cooled fuel at PWMF.
	(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.	Attachment 2 of this submission documents the assessments and provisions in support of the licence amendment request. Specifically: • documents worker health and safety provisions. • documents assessments and impact on environmental protection. • documents the security considerations. • documents the impact on Canada's international obligations related to safeguards and non-proliferation.
25	***Renewal, etc.*** *The Commission may, on its own motion, renew, suspend in whole or in part, amend, revoke or replace a licence under the prescribed conditions.*	OPG understands this requirement and will continue to comply.

Nuclear Safety and Control Act		
Section	**Requirement**	**OPG Response**
26	***Prohibitions*** *Subject to the regulations, no person shall, except in accordance with a licence:* *(a) possess, transfer, import, export, use or abandon a nuclear substance, prescribed equipment or prescribed information;* *(b) mine, produce, refine, convert, enrich, process, reprocess, package, transport, manage, store or dispose of a nuclear substance;* *(c) produce or service prescribed equipment;* *(d) operate a dosimetry service for the purposes of this Act;* *(e) prepare a site for, construct, operate, modify, decommission or abandon a nuclear facility; or* *(f) construct, operate, decommission or abandon a nuclear-powered vehicle or bring a nuclear-powered vehicle into Canada.*	OPG staff understand these requirements and will continue to comply.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	Licences - General Application Requirements	
3 (1)	*An application for a licence shall contain the following information:* *(a) the applicant's name and business address;*	Applicant's name and business address: **Ontario Power Generation, Inc** **700 University Avenue,** **Toronto, Ontario, M5G 1Z5** Official Language: **English** Contact person, signing authority and licence holder: **Kapil Aggarwal** **Vice President** **Nuclear Sustainability Services,** **Ontario Power Generation** **Telephone: 416-402-6484**
	(b) the activity to be licensed and its purpose;	OPG requests an amendment to the PWMF WFOL, WFOL-W4-350.00/2028, to authorize the storage of minimum 6-year cooled fuel from Pickering NGS.
	(c) the name, maximum quantity and form of any nuclear substance to be encompassed by the licence;	100 Dry Storage Containers (DSC's) containing less than 10-year cooled used fuel from Pickering NGS. These 100 DSC's are included in the current approved total for PWMF (and are not considered additional to the inventory).
	(d) a description of any nuclear facility, prescribed equipment or prescribed information to be encompassed by the licence;	A description of the PWMF is provided in Attachment 2 of this submission.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	(e) the proposed measures to ensure compliance with the Radiation Protection Regulations, the Nuclear Security Regulations and the Packaging and Transport of Nuclear Substances Regulations, 2015;	OPG understands this requirement and will remain in compliance with the current licence conditions documented in WFOL-W4-350.00/2028 and with the Radiation Protection Regulations, the Nuclear Security Regulations, and the Packaging and Transport of Nuclear Substances Regulations as described in Attachment 2 of this submission.
	(f) any proposed action level for the purpose of section 6 of the Radiation Protection Regulations;	The requested WFOL amendment will not require changes to the radiation protection action levels.
	(g) the proposed measures to control access to the site of the activity to be licensed and the nuclear substance, prescribed equipment or prescribed information;	The requested WFOL amendment will not require changes to the measures to control PWMF site access, the nuclear substance, prescribed equipment or prescribed information.
	(h) the proposed measures to prevent loss or illegal use, possession or removal of the nuclear substance, prescribed equipment or prescribed information;	The requested WFOL amendment will not require changes to the measures to prevent loss or illegal use, possession or removal of the nuclear substance, prescribed equipment or prescribed information.
	(i) a description and the results of any test, analysis or calculation performed to substantiate the information included in the application;	The requested PROL amendment to authorize the storage of minimum 6-year cooled fuel at PWMF is supported by a robust safety case that is included with this submission (Enclosures 1-5) and is summarized in Attachment 2 of this submission.

	(j) the name, quantity, form, origin and volume of any radioactive waste or hazardous waste that may result from the activity to be licensed, including waste that may be stored, managed, processed or disposed of at the site of the activity to be licensed, and the proposed method for managing and disposing of that waste;	This waste will be managed in accordance with OPG's current programs and processes. No hazardous waste will be generated from the storage of minimum 6-year cooled fuel.
	(k) the applicant's organizational management structure insofar as it may bear on the applicant's compliance with the Act and the regulations made under the Act, including the internal allocation of functions, responsibilities and authority;	The organizational management structure will not change as a result of the requested licence amendment.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	(l) a description of any proposed financial guarantee relating to the activity to be licensed; and	OPG understands the regulatory requirements for a financial guarantee. The financial guarantee for PWMF will not change as a result of the requested PROL amendment.
	(m) any other information required by the Act or the regulations made under the Act for the activity to be licensed and the nuclear substance, nuclear facility, prescribed equipment or prescribed information to be encompassed by the licence.	OPG understands this requirement and will continue to comply.
(1.1)	*The Commission or a designated officer authorized under paragraph 37(2)(c) of the Act, may require any other information that is necessary to enable the Commission or the designated officer to determine whether the applicant* *(a) is qualified to carry on the activity to be licensed;* *(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.*	OPG understands this requirement and will continue to comply.
Application for Amendment, Revocation or Replacement of Licence		
6	*An application for the amendment, revocation or replacement of a licence shall contain the following information:* *(a) a description of the amendment, revocation or replacement and of the measures that will be taken and the methods and procedures that will be used to implement it;*	Attachment 2 of this submission documents the description of the amendment and of the measures that will be taken and the methods and procedures that will be used to implement it.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	(b) a statement identifying the changes in the information contained in the most recent application for the licence; *(c) a description of the nuclear substances, land, areas, buildings, structures, components, equipment and systems that will be affected by the amendment, revocation or replacement and of the manner in which they will be affected; and* *(d) the proposed starting date and the expected completion date of any modification encompassed by the application.*	Attachment 2 of this submission documents the changes that will be required to any licensing basis documents. The minimum 6-year cooled fuel will be stored within a specified array in PWMF Storage Building (SB) #3, a shielded building. The younger fuel will be stored in the same DSC's that are being used to store minimum 10-year cooled fuel. Initial loading of 2-4 DSC's containing minimum 6-year cooled fuel is proposed to commence in July 2024. After obtaining indicators related to temperature and dosage, the full campaign of storing 6-year cooled fuel will commence.
Incorporation of Material in Application		
7	*An application for a licence or for the renewal, suspension in whole or in part, amendment, revocation or replacement of a licence may incorporate by reference any information that is included in a valid, expired or revoked licence.*	OPG understands and has provided applicable references to information contained in the existing licence and Licence Conditions Handbook.
Obligations		
12 (1)	***Obligations of Licensees*** *Every licensee shall*	OPG understands the requirements and will continue to comply. Specifically:
	(*a) ensure the presence of a sufficient number of qualified workers to carry on the licensed activity safely and in accordance with the Act, the regulations made under the Act and the licence;*	The regulatory requirement will not change as a result of the requested licence amendment.

	(b) train the workers to carry on the licensed activity in accordance with the Act, the regulations made under the Act and the licence;	OPG staff will be trained on operation and maintenance activities associated with the requested licence amendment.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	(c) take all reasonable precautions to protect the environment and the health and safety of persons and to maintain the security of nuclear facilities and of nuclear substances;	Refer to section LC 9.1 in Attachment 2 of this submission for details on environmental protection. Refer to section LC 12.1 in Attachment 2 of this submission for further details on the impact to security.
	(d) provide the devices required by the Act, the regulations made under the Act and the licence and maintain them within the manufacturer's specifications;	OPG understands this requirement and will continue to comply.
	(e) require that every person at the site of the licensed activity use equipment, devices, clothing and procedures in accordance with the Act, the regulations made under the Act and the licence;	OPG understands this requirement and will continue to comply.
	(f) take all reasonable precautions to control the release of radioactive nuclear substances or hazardous substances within the site of the *licensed activity and into the environment as a result of the licensed activity;*	OPG understands this requirement and will continue to comply. Refer to section LC 9.1 in Attachment 2 for further details on security.
	(g*) implement measures for alerting the licensee to the illegal use or removal of a nuclear substance, prescribed equipment or prescribed information, or the illegal use of a nuclear facility;*	OPG understands this requirement and will continue to comply. Refer to section LC 13.1 in Attachment 2 of this submission for further details on security.
	(h) implement measures for alerting the licensee to acts of sabotage or attempted sabotage anywhere at the site of the licensed activity;	OPG understands this requirement and will continue to comply.
	(i) take all necessary measures to facilitate Canada's compliance with any applicable safeguards agreement;	OPG understands this requirement and will continue to comply. Refer to section LC 13.1 in Attachment 2 of this submission for further details on safeguards.
	(j) instruct the workers on the physical security program at the site of the licensed	OPG understands this requirement and will continue to comply.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	activity and on their obligations under that program;	Refer to section LC 12.1 in Attachment 2 of this submission for further details on security.
	(k) keep a copy of the Act and the regulations made under the Act that apply to the licensed activity readily available for consultation by the workers.	OPG understands this requirement and will continue to comply.
12 (2)	*Every licensee who receives a request from the Commission or a person who is authorized by the Commission for the purpose of this subsection, to conduct a test, analysis, inventory or inspection in respect of the licensed activity or to review or to modify a design, to modify equipment, to modify procedures or to install a new system or new equipment shall file, within the time specified in the request, a report with the Commission that contains the following information:* *(a) confirmation that the request will or will not be carried out or will be carried out in part;* *(b) any action that the licensee has taken to carry out the request or any part of it;* *(c) any reasons why the request or any part of it will not be carried out;* *(d) any proposed alternative means to achieve the objectives of the request; and* *(e) any proposed alternative period within which the licensee proposes to carry out the request.*	OPG understands this requirement and will continue to comply. Testing and commissioning procedures and reports associated with the storage of minimum 6-year cooled fuel will be made available to facilitate the regulatory role of CNSC staff.
Transfers		
13	*No licensee shall transfer a nuclear substance, prescribed equipment or prescribed information to a person who does not hold the licence, if any, that is required to possess the nuclear substance,*	OPG understands this requirement and will continue to comply.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	prescribed equipment or prescribed information by the Act and the regulations made under the Act.	
Notice of Licence		
14	*(1) Every licensee other than a licensee who is conducting field operations shall post, at the location specified in the licence or, if no location is specified in the licence, in a conspicuous place at the site of the licensed activity,* *(a) a copy of the licence, with or without the licence number, and a notice indicating the place where any record referred to in the licence may be consulted; or* *(b) a notice containing* *(i) the name of the licensee,* *(ii) a description of the licensed activity,* *(iii) a description of the nuclear substance, nuclear facility or prescribed equipment encompassed by the licence, and* *(iv) a statement of the location of the licence and any record referred to in it.* *(2) Every licensee who is conducting field operations shall keep a copy of the licence at the place where the field operations are being conducted.* *(3) Subsections (1) and (2) do not apply to a licensee in respect of* *(a) a licence to import or export a nuclear substance, prescribed equipment or prescribed information;* *(b) a licence to transport a nuclear substance; or* *(c) a licence to abandon a nuclear substance, a nuclear facility,*	OPG understands this requirement and will continue to comply with this requirement.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	prescribed equipment or prescribed information.	
Publication of Health and Safety Information		
16	*(1) Every licensee shall make available to all workers the health and safety information with respect to their workplace that has been collected by the licensee in accordance with the Act, the regulations made under the Act and the licence.* *(2) Subsection (1) does not apply in respect of personal dose records and prescribed information.*	OPG understand this requirement and will continue to comply. OPG's Health and Safety Policy is posted on the OPG intranet website.
Obligations of Workers		
17	*Every worker shall:* *(a) use equipment, devices, facilities and clothing for protecting the environment or the health and safety of persons, or for determining doses of radiation, dose rates or concentrations of radioactive nuclear substances, in a responsible and reasonable manner and in accordance with the Act, the regulations made under the Act and the licence;* *(b) comply with the measures established by the licensee to protect the environment and the health and safety of persons, maintain security, control the levels and doses of radiation, and control releases of radioactive nuclear substances and hazardous substances into the environment;* *(c) promptly inform the licensee or the worker's supervisor of any situation in which the worker believes there may be* *(i) a significant increase in the risk to the environment or the health and safety of persons,*	OPG understands this requirement and will continue to comply.

General Nuclear Safety and Control Regulations		
Section	**Requirement**	**OPG Response**
	(ii) a threat to the maintenance of the security of nuclear facilities and of nuclear substances or an incident with respect to such security, *(iii) a failure to comply with the Act, the regulations made under the Act or the licence,* *(iv) an act of sabotage, theft, loss or illegal use or possession of a nuclear substance, prescribed equipment or prescribed information, or* *(v) a release into the environment of a quantity of a radioactive nuclear substance or hazardous substance that has not been authorized by the licensee;* *(d) observe and obey all notices and warning signs posted by the licensee in accordance with the Radiation Protection Regulations; and* *(e) take all reasonable precautions to ensure the worker's own safety, the safety of the other persons at the site of the licensed activity, the protection of the environment, the protection of the public and the maintenance of the security of nuclear facilities and of nuclear substances.*	

Radiation Protection Regulations		
Section	**Requirement**	**OPG Response**
4	*Every licensee must implement a radiation protection program and must, as part of that program,* *(a) keep the effective dose and equivalent dose received by and committed to persons as low as reasonably achievable, taking into account social and economic factors, through the implementation of* *(i) management control over work practices,* *(ii) personnel qualification and training,* *(iii) control of occupational and public exposure to radiation, and* *(iv) planning for unusual situations; and* *(b) ascertain the quantity and concentration of any nuclear substance released as a result of the licensed activity* *(i) by direct measurement as a result of monitoring, or* *(ii) if the time and resources required for direct measurement as a result of monitoring outweigh the usefulness of ascertaining the quantity and concentration using that method, by estimating them.*	OPG has a well-established radiation protection program that complies with all elements of the Radiation Protection Regulations. Further details are provided in Section LC 7.1 on OPG's Radiation Protection considerations for the loading of minimum 6-year cooled fuel.

Nuclear Security Regulations
OPG will continue to adhere to all facets of the Nuclear Security Regulations and keep in place all current security processes in the handling and storage of used fuel from Pickering NGS.

ATTACHMENT #2

Licence Amendment Application for the Storage Of Minimum 6-Year Cooled Fuel at Pickering Waste Management Facility

ATTACHMENT 2

LICENCE AMENDMENT APPLICATION FOR

THE STORAGE OF MINIMUM 6-YEAR COOLED FUEL

AT THE PICKERING WASTE MANAGEMENT FACILITY

Prepared by: Cliff Barua

Checked by: Heather Innis

Introduction

Background

The purpose of this attachment is to provide information in support of OPG's request for amendment to the Pickering Waste Management (PWMF) Waste Facility Operating Licence (WFOL), WFOL-W4-350.00/2028, to allow for the storage of minimum 6-year cooled used fuel from Pickering Nuclear Generating Station (PNGS).

Description of PWMF – Used Fuel Dry Storage

The PWMF is located within the traditional territory of the Michi Saagiig Anishinaabe people. These lands are covered by the Williams Treaty between Canada and the Mississauga and Chippewa Nations. The PWMF operates under a Waste Facility Operating Licence (WFOL). At the PWMF, OPG processes and stores dry storage containers (DSCs) containing used nuclear fuel (high-level radioactive waste) generated at the PNGS, that has cooled for a minimum of ten years in the fuel bays at PNGS.

The dry storage of used fuel at the PWMF spans over 2 physically separate areas - Phase I and Phase II - within the overall boundary of the Pickering site. Phase I is located within the protected area of the PNGS and consists of the DSC Processing Building and two DSC storage buildings (Storage Buildings #1 and #2). Phase II of the PWMF is located northeast of Phase I and is contained within its own protected area, but within the boundary of the Pickering site. Phase II contains Storage Building #3 and #4. The PWMF currently has the capacity to store 1,778 DSCs. The transfer route of the loaded DSCs from the PWMF Phase I to the PWMF Phase II is solely on OPG property.

The information provided in this Attachment is divided into three sections as follows:

Section 1: Provides the need to store minimum 6-year cooled fuel at PWMF to support the Safe Storage Project at PNGS Units 5-8, and operational considerations for this activity.

Section 2: Summarizes regulatory compliance for the storage of younger than 10-year cooled fuel at PWMF and impact on OPG's governance, programs and processes for each of PWMF's WFOL's fourteen (14) Safety and Control Areas (SCA).

Section 3: Summarizes public, Indigenous and Metis engagement related to the application of licence amendment.

OPG is responsible for continued safe operation of the PWMF and confirms that the storage of minimum 6-year cooled fuel will be implemented based on a robust safety case and proven engineering methods.

OPG has concluded that the proposed activities to support the storage of minimum six-year cooled fuel will not compromise continued safe operation of the PWMF. OPG has and will continue to follow a robust and well-established Engineering Change Control (ECC) process and will continue to provide information to CNSC staff to assist in fulfillment of their regulatory oversight role.

The storage of minimum 6-year cooled fuel at PWMF is an important initiative to support the OPG Safe Storage Project for PNGS Units 5-8. The objective is to **only** accept minimum 6-year cooled fuel at PWMF from PNGS Units 5-8 (and **not** PNGS Units 1 and 4).

Section 1: Summary of Proposed Activity Requiring Licence Amendment

To support the OPG Safe Storage Project for PNGS, additional space in the PNGS-B Irradiated Fuel Bay (IFB-B) is required in order to accept the discharged used fuel from the required core dumps. As PWMF is currently waiting for IFB-B used fuel to mature to the 10-year required period before transferring, there is a need to accept younger fuel (as young as 6-year cooled fuel) to allow for the additional space. However, OPG is currently licensed to only process minimum 10-year cooled fuel at all Nuclear Waste Facilities. In order to store younger (i.e., fuel as young as 6-year cooled), OPG must apply for a License Amendment for the PWMF's Operating License (WFOL-W4-350.00/2028).

Master EC# 154806, "Loading, Processing and Storing a Maximum of 100 Dry Storage Containers (DSCs) (at one time) that Contain Used Fuel with a Minimum Cooling Period of 6 Years of Age in PWMF" was initiated to support the PNGS-B and PWMF operational need. The modification includes loading, transferring, processing and storage of up to 100 DSCs from the IFB-B that contain used fuel with a minimum cooling period of 6 years as well as the rearrangement of a number of the existing DSCs in Storage Building 3 (SB3) to accommodate the incoming DSCs. The younger used fuel will be loaded into DSCs in the IFB-B, transferred to the PWMF for processing, moved into the IAEA Surveillance Area in Storage Building 1 (SB1) for the application of safeguard seals and transferred for storage into SB3. Once all fuel in a DSC reaches the minimum cooling period of 10 years, the DSC can be treated the same as existing DSCs in the Used Fuel Storage Buildings at PWMF (and would not be considered part of the inventory of 100 DSC's containing younger than 10-year cooled fuel). Based on the analysis performed, it was determined that no design changes are required to the DSC to accept the storage of younger fuel within the DSC and stored in the PWMF storage buildings.

OPG proposes to start commissioning DSC's containing younger cooled fuel in July 2024, with the aim to initially gather predictive indicators around temperature and dosage. If this initial campaign proves successful (indicators are agreeable with modelling predictions) and doesn't present unforeseen challenges, the full campaign to store the younger cooled fuel would commence in Q1 of 2025.

Safety Case

Safety is OPG's number one priority, proven over many years of both reactor operation and radioactive waste management and storage. OPG is responsible for continued safe operation of the PWMF and confirms that the minimum 6-year cooled fuel modifications at PWMF will be implemented based on a robust safety case and in accordance with OPG's Engineering Change Control process, which is supported by safety assessments (Enclosures 2-3) that demonstrate continued safe facility operation, public and worker safety, and environmental protection.

The safety case for the storage of younger than 10-year cooled fuel at PWMF can be defined based on the following elements:

1) Design: OPG has and will continue to follow its Engineering Change Control process, as described in N-PROG-MP-0001, "*Engineering Change Control*", for ensuring the design complies with applicable PWMF Licence Condition Handbook, LCH-W4-350.00/2028, regulatory requirements and that configuration management for the station is maintained.

2) Continued Safe Operation: Safety analysis (Enclosure 1) demonstrates that the storage of younger than 10-year cooled fuel at PWMF will have a negligible effect on safe operation of PWMF, and on public and worker safety.

3) Environmental Protection: An assessment of existing environmental-related submissions to the CNSC (environmental assessments, environmental risk assessment and predictive environmental effects assessment) (Enclosure 6) concludes that the storage of younger than 10-year cooled fuel at PWMF will have negligible impact on the environment.

4) Licensing Basis: The storage of younger than 10-year cooled fuel at PWMF will have negligible impact on PWMF's licensing basis, governance, programs and processes. Attachment 1 documents the impact on the "*Nuclear Safety Control Act*" and applicable regulations.

Overall, there are no notable safety or operational issues that result from storing younger than 10-year cooled fuel at PWMF.

Section 2: Safety and Control Areas

The purpose of this section is to document the impact of the storage of minimum 6-year cooled fuel on OPG's (and hence PWMF's) governance, programs and processes. A review of the impact on the PWMF WFOL's fourteen (14) Safety and Control Areas (SCA's) was completed and is summarized in the following sections.

OPG is responsible for the continued safe operation of the PWMF and confirms that all modifications made with respect to the storage of younger than 10-year cooled fuel, will be implemented based on a robust safety case and in accordance with OPG's ECC process and that is supported by safety assessments, which demonstrate continued safe operation of the PWMF, public safety, worker safety and environmental protection.

LC 1.1 Management System

Licence Condition 1.1 states "*the licensee shall implement and maintain a management system*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

OPG's proven Nuclear Management System provides a framework that establishes the processes and programs required to ensure OPG achieves its safety objectives, continuously monitors its performance against these objectives, and fosters a healthy safety culture.

List of Management System Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Management System Requirements for Nuclear Facilities	CSA N286 (2012)	Continued compliance as applied to all aspects of operation and modifications at PWMF.

Quality Assurance, CSA Standard N286-12 Compliance

PWMF is compliant with CSA Standard N286-12, "*Management system requirements for nuclear facilities*". The Nuclear Charter, N-CHAR-AS-0002, "*Nuclear Management System*", establishes the Nuclear Management System for OPG Nuclear. The Nuclear Management System will not change as a result of the proposed storage of minimum 6-year cooled fuel at PWMF.

Impact of the Storage of Minimum 6-Year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Management System and identifies the impact of storing younger than ten-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Management System Licensing Basis Documents

OPG Management System Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Items and Services Management	OPG-PROG-0009	No Change
Health and Safety Managed Systems	OPG-PROG-0005	No Change
Nuclear Management Systems Organization	N-STD-AS-0020	No Change
Nuclear Safety Culture Assessment	N-PROC-AS-0077	No Change
Nuclear Safety Oversight	N-STD-AS-0023	No Change
Nuclear Safety Policy	N-POL-0001	No Change
Nuclear Management System	N-CHAR-AS-0002	No Change

LCH 1.2 Management of Contractors

Licence Condition 1.2 states "*the licensee shall ensure that every contractor at the facility complies with this licence*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Vendors and contractors are qualified by OPG Supply Chain Quality Services under a process that ensures that the contractors have developed and implemented a management system that meets the applicable requirements outlined in the CSA Standard N286 series of standards. OPG is ultimately responsible for ensuring that all on-site contractor activities comply with OPG's safety requirements. Day-to-day operations at the PWMF are generally maintained by full-time staff of OPG.

LCH 2.1 Human Performance Program

Licence Condition 2.1 states "*the licensee shall implement and maintain a human performance program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Human performance relates to reducing the likelihood of human error in work activities. It refers to the outcome of human behaviour, functions and actions in a specified environment, reflecting the ability of workers and management to meet the system's defined performance under the conditions in which the system will be employed.

List of Human Performance Management Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Fitness for Duty: Managing Worker Fatigue	CNSC REGDOC-2.2.4 (2017)	Continued compliance, no impact.
Fitness for Duty, Volume II: Managing Alcohol and Drug Use, Version 2	CNSC REGDOC- 2.2.4 (2017)	Continued compliance, no Impact.
Safety Culture	CNSC REGDOC-2.1.2 (2018)	Continued compliance, no impact.

Human Performance Program

The objective of OPG's Human Performance program, N-PROG-AS-0002, "*Human Performance*" is to reduce human performance events and errors by managing defences in pursuit of zero events of consequence.

The Human Performance program integrates proactive (prevention) and reactive (detection and correction) human performance initiatives, which includes the following:

- Providing oversight and mentoring of department human performance.
- Identifying emerging human performance issues and determining strategies for related improvement.
- Approving site-wide human performance improvement initiatives and measures and overseeing implementation progress.
- Use of the human performance toolbox, prevent event tools.
- Identifying and implementing human performance improvement communication, education, and training opportunities.

The site strategic plan provides guidance to the leadership team on the requirements for the development and implementation of an integrated site and department human performance strategic plan. Department managers and supervisors develop a human performance plan that sets clear direction and priorities to achieve common goals.

Fitness for Duty

As part of OPG's fitness for duty program, OPG has in place a Continuous Behaviour Observation Program which trains supervisors and managers to monitor workers for signs of fatigue or other factors which could adversely impact worker performance.

OPG has in place hours of work requirements that are documented in N-PROC-OP-0047*, "Hours of Work Limits and Managing Worker Fatigue"* that sets limits for the number of hours within a specified time period that station staff can work. The limits, which are in place to guard against fatigue in the workplace, are very strict in comparison to other jurisdictions.

The storage of minimum 6-year cooled fuel will not impact OPG's fitness for duty program or compliance to hours-of-work requirements.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Human Performance Program Licensing Basis Documents

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Human Performance program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of Younger than 10-year Cooled Fuel on PWMF's Human Performance Management Licensing Basis Documents

OPG Human Performance Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Human Performance	N-PROG-AS-0002	No Change
Hours of Work Limits and Managing Worker Fatigue	N-PROC-OP-0047	No Change

LC 2.2. Training Program

Licence Condition 2.2 states "*the licensee shall implement and maintain a training program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Personnel at the PWMF will be fully trained on the loading of minimum 6-year year cooled fuel and also on mitigative measures for backout when required. All required staff will be fully trained before the first DSC containing younger than 10-year cooled fuel is commissioned.

List of Training Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Personnel Training	CNSC REGDOC-2.2.2 (2014)	Continued compliance, no impact.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Training Program Licensing Basis Documents

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Training program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of Minimum 6-year Cooled Fuel on PWMF's Training Program Licensing Basis Documents

OPG Human Performance Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Systematic Approach to Training	N-PROC-TR-0008	No Change
Training	N-PROG-TR-0005	No Change

LC 3.1 Operating Performance

Licence Condition 3.1 states "*the licensee shall implement and maintain an operating program, which includes a set of operating limits*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Operational Analysis

Processing minimum 6-year cooled fuel is essentially the same as processing 10-year cooled fuel. There will be minor changes required to operational documentation and procedures. DSCs that contain as young as 6-year cooled fuel have been analyzed for the anticipated temperatures throughout the DSC. Based on conservative bounding scenario assumptions, it has been conservatively identified that contact temperatures could potentially reach approximately 85 degrees Celsius (°C), which impacts worker safety in handling the DSC. The increased temperatures potentially impact interfacing equipment such as Advanced Inspection and Maintenance (AIM) equipment and International Atomic Energy Agency (IAEA) equipment including seals and NDE profiling.

Based on OPEX from 1998, DSC 0024 contained four full modules (384 bundles) of 6-year cooled fuel and had temperature probes fixed to the DSC. Temperature measurements were much lower than the conservative design analysis from 2022. These temperatures are documented in OPG Controlled Document 00104-CORR-79171-0139942 "Additional Information Concerning: Thermal Gradients Pertaining to Dry Storage Containers (DSCs)" (2005) and summarized in Figure 1 below. Based on this OPEX, it is anticipated that contact temperatures will not be as high as analyzed.

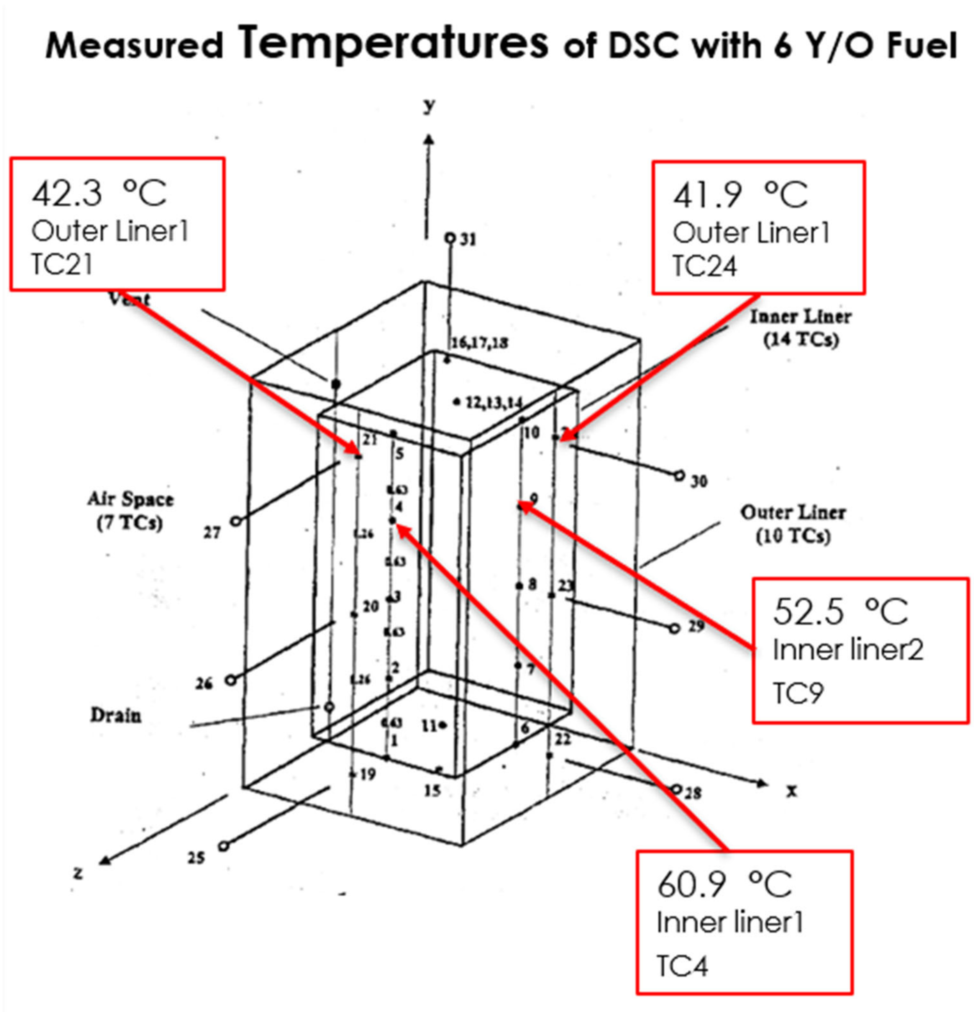

Figure 1: Measured Temperatures in DSC 0024

<u>Commissioning Plan</u>

Operationally, only one DSC is required to be loaded with 6-year cooled fuel to commission the modification. However, to avoid reverse loading (see Reverse Loading below), a conservative approach is recommended to be used. A potential option would be to load and vacuum dry the commissioning DSCs, starting with 9-year cooled fuel and working down to 6-year cooled fuel while measuring temperatures and dose rates. Based on OPG report with Controlled Document I-REP-79171-00001, the time taken for the outer liner of the DSC to reach equilibrium temperature is of the order of three weeks on average. Therefore, this option will take several months to complete the commissioning.

Acceptable temperatures are driven by Advance Inspection and Maintenance (AIM) and IAEA equipment at specific DSC locations outlined in Table 1 below:

Table 1 - IAEA/AIM Equipment Temperature Limits

Container Location	Equipment	Temp. Limit	Analyzed Temp.	OPEX Temp. DSC 0024
Weld Flange	AIM – Phased Array Ultrasonic Testing – (PAUT)	50 °C	~ 85 °C	~ 45 °C
Weld Flange	IAEA - Laser Container Mapping Verification	Deformation ~60 °C (Estimate)	~ 85 °C	~ 45 °C
Seal Tubes	IAEA -Fiber Optic Seals	Degradation ~70 °C (Estimate)	~100 °C	< 62 °C

IAEA temperature limits in Table 1 are estimates since they are not allowed to be identified. Estimates listed in Table 1 are based on discussions with IAEA.

AIM Equipment

As part of the commissioning, the intent is to ensure that the temperatures meet AIM equipment requirements before proceeding with the welding and continuation of processing the DSC to interim storage. The AIM equipment has a temperature limitation 50°C, shown in Table 1 above. If temperatures are measured less than 50°C, then nothing changes except conventional and Radiation Protection (RP) safety aspects. The AIM Acquisition Procedure would remain unaltered and there would be no issue.

Options have been considered for cooling the DSC flange if temperatures are measured in excess 50°C. Details on flange cooling are discussed below. If temperatures exceed 50°C, and the flange cooling methods are ineffective then the DSC will be Reverse loaded (discussed below).

Flange cooling: options for cooling the DSC flange are available if temperatures are measured in excess 50°C. Having an effective means to cool the DSC temperatures reduces the risk of having to resort to the back-out option (Reverse Loading).

IAEA Equipment

Temperature limits for IAEA are listed in Table 1 above. The impact of higher temperatures on IAEA safeguards and security interfacing equipment is being evaluated and discussion with the IAEA and CNSC is in progress. There is a risk that some IAEA equipment used for the sealing processes is not designed for the increased temperatures that could be observed.

Current proposal to the IAEA is to:

- Load and Vacuum Dry commissioning DSC with 6-year old fuel.
- Within Camera View – Allow DSC to reach maximum temperature (not welded).
- Allow for residency time of three weeks to allow for DSC to reach equilibrium temperature, measure temperatures and doses.

- If temperatures are conducive for Fiber Optic seals: complete DSC processing. Confirm weld flange temperatures before sealing with IAEA.
- If temperatures exceed limits outlined in Table 1 above, OPG suggest tri-seals to be applied (i.e., LMCV, FBOS, & Metallic). Monitor health seals during regular IAEA visits. OPG will also explore using mixed age modules.
- If the above commissioning DSC is excessively hot, then the DSC will need to be reverse loaded (see below, Reverse Loading). DSC temperatures will be controlled operationally – for example through the mixed age module loading.

Based on discussions, the IAEA have agreed to support the commissioning DSC test case to see if the actual temperatures are similar to the calculated temperatures or more similar to OPEX of DSC 0024. An Operating Memo is currently being prepared to provide the changes to documents required to operationalize the change. This will be completed prior to commissioning of the first DSC.

Reverse Loading

If the temperatures are higher than the limits required as discussed above, there will need to be a backout option to reverse load the DSC back to the IFB-B. A reverse loading plan is being developed to outline the steps required to reverse load a DSC loaded with 6-year cooled fuel. This is being developed using OPEX from 2012 to address an issue with a partially loaded DSC 1538. This DSC had to be emptied (SCR N-2012-00289). The reverse loading plan will be issued before the loading of any DSC's containing minimum 6-year cooled fuel.

Impact of the Storage of Minimum 6-Year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Operating Performance and identifies the impact of the storage of minimum 6-year cooled fuel on these programs. Identified changes in new revisions of licensing basis documents will be submitted by written notification to the CNSC per the requirements of the PWMF LCH LC G2.

Impact of the Storage of Minimum 6-Year Cooled Fuel on PWMF's Operating Performance Related Licensing Basis Documents

OPG Document Title	OPG Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Application for Renewal of Pickering Waste Management Facility Operating Licence	92896-CORR-00531-01031	No Change
Additional Information to Support the Application for Renewal of Pickering	92896-CORR-00531-01075	No Change

Waste Management Facility Operating Licence		
Nuclear Waste Management	W-PROG-WM-0001	No Change
Operating Policies and Principles, Pickering Waste Management Facility	92896-OPP-01911.1-00001	To be updated by March 15, 2024.
Pickering Waste Management Facility – Safety Report	92896-SR-01320-10002	Changes will be reflected in the next update of the PWMF Safety Report scheduled for 2028.

Impact of the Storage of Minimum 6-Year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Operating Performance and identifies the impact of the storage of younger than 10-year cooled fuel on these programs.

Impact of the Storage of Minimum 6-Year Cooled Fuel on PWMF's Operating Performance Related Licensing Basis Documents

OPG Document Title	OPG Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Conduct of Regulatory Affairs	N-PROG-RA-0002	No Change
Performance Improvement	N-PROG-RA-0003	No Change
Preliminary Event Notification	N-PROC-RA-0020	No Change
Operating Policies and Principles, Pickering Waste Management Facility	92896-OPP-01911.1-00001	To be updated prior to commissioning of the first DSC with younger fuel.

The following documents related to operations (but not included in the licensing basis) will also be updated prior to the commissioning of the first DSC containing minimum 6-year cooled fuel:

Document Number	Document Title
92896-MAN-79171-00001	IFB Loading
W-WOEP-79171-000010	Dry Storage Container Reverse Loading
W-PROC-WM-0082	Eastern Waste Acceptance Criteria for Used Fuel Dry Storage Containers
92896-OP-35540-00001	Pickering Waste Management Facility (PWMF) General
92896-OP-79171-00001	Pickering Waste Management Facility Operating Procedure Dry Storage Container Processing
92896-SR-01320-10002	Pickering Waste Management Facility – Safety Report
92896-OP-35570-00001	International Atomic Energy Agency Safeguards (*If required based on commissioning results)
92896-OP-79171-00003	DSC Loading PNGS 058 Irradiated Fuel Bay (IFB-B)
92896-OP-79171-00004	DSC Loading Auxiliary Irradiated Fuel Bay

LC 3.2 Reporting Requirements

Licence Condition 3.2 states "*the licensee shall implement and maintain a program for reporting to the Commission or a person authorized by the Commission*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Reporting Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Public Information and Disclosure	CNSC REGDOC-3.2.1 (2018)	Continued compliance, no impact.
Reporting Requirements, Volume I: Non-Power Reactor Class I Nuclear Facilities and Uranium Mines and Mills	CNSC REGDOC-3.1.2 (2018)	Continued compliance, no impact.

Operational Quarterly and Annual Reporting

Quarterly and Annual operational reporting will continue as currently conducted and will account for the DSC's containing minimum 6-year cooled fuel.

LC 4.1 Safety Analysis Program

Licence Condition 4.1 states "*the licensee shall implement and maintain a safety analysis program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Safety Assessment

As concluded in the safety analysis provided in Enclosure 2 to this submission: the safety assessment demonstrates compliance with the radiation safety requirements during normal operation of the PWMF when SB3 is in service. With the addition of the 100 DSCs containing 6-year decayed used fuel, the annual public dose estimates have increased compared to that of the existing PWMF configuration. The maximum annual dose to individual member of the public with the addition of these 100 DSCs is still a small percentage of the 1 mSv limit. Due to the specialised array of storing the DSC's containing minimum 6-year cooled fuel, the target dose rate to the public of 0.5mSv will also be met. With respect to malfunction and accident scenarios, the estimated bounding doses to members of the public are less than the 1 mSv acceptance criterion. The dose to workers following a postulated accident scenario is found to be much less than the 50 mSv limit. It is concluded that the dose consequences to workers and members of the public as a result of credible postulated malfunction / accident scenarios meet all acceptance criteria.

Enclosures 2 and 3 were previously provided to CNSC staff in 2020 and 2021 before it was determined that a Licence Amendment would be required to store minimum 6-year cooled fuel. Enclosure 4 contains information (previously requested by the CNSC) regarding the trialing of a single DSC containing 6-year cooled fuel in 1998.

List of Safety Analysis Program Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
General principles for the management of radioactive waste and irradiated fuel	CSA N292.0 (2014)	Minimum 6-year cooled fuel safety assessments were

		conducted in compliance with applicable requirements
Interim dry storage of irradiated fuel	CSA N292.2 (2013)	Minimum 6-year cooled fuel safety assessments were conducted in compliance with applicable requirements
Management of low- and intermediate-level radioactive waste	CSA N292.3 (2014)	Minimum 6-year cooled fuel safety assessments were conducted in compliance with applicable requirements
Quality assurance of analytical, scientific, and design computer programs	CSA N286.7 (2016)	Minimum 6-year cooled fuel safety assessments were conducted in compliance with applicable requirements

Impact of the Storage of Minimum 6-Year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Safety Analysis program and identifies the impact of storing minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Safety Analysis Licensing Basis Documents

OPG Safety Analysis Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Pickering Waste Management Facility – Safety Report	92896-SR-01320-10002	Changes will be reflected in the next update of the PWMF Safety Report scheduled for 2028.

LC 5.1 Design Program

Licence Condition 5.1 states "*the licensee shall implement and maintain a design program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Design Program Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Fire protection for facilities that process, handle, or store nuclear substances	CSA N393 (2013)	This code is not impacted by the storage of younger than 10-year cooled fuel.
National Building Code of Canada (2020)	NRC	The PWMF SB's design complies with the requirements in this national code.
National Fire Code of Canada (2020)	NRC	The PWMF SB's design complies with the requirements in this national code.

Facility and DSC Design

The storage of minimum 6-year cooled fuel will not require any change to the facility design. The DSC currently used for minimum 10-year cooled fuel will also be used for the storage of minimum 6-year cooled fuel.

Impact of the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Design Program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Design Program Related Licensing Basis Documents

OPG Physical Design Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Conduct of Engineering	N-STD-MP-0028	No Change
Configuration Management	N-STD-MP-0027	No Change
Design Management	N-PROG-MP-0009	No Change
Engineering Change Control	N-PROG-MP-0001	No Change

LC 5.2 Pressure Boundary

Licence Condition 2.2 states "*the licensee shall implement and maintain a training program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Pressure Boundary Program Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Power Piping	ASME (2010)	This code is not impacted by the storage of minimum 6-year cooled fuel.
Boiler, pressure vessel, and pressure piping code	CSA B51 (2009 and Update No. 1)	This code is not impacted by the storage of minimum 6-year cooled fuel.
General requirements for pressure-retaining systems and components in CANDU nuclear power plants	CSA N285.0 (2012 and Updates No. 1 and 2; and Annex N of N285.0-12 and Update No. 1)	This code is not impacted by the storage of minimum 6-year cooled fuel.

Impact of the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Design Program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-Year Cooled Fuel on PWMF's Design Program Related Licensing Basis Documents

OPG Physical Design Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Index to OPG Pressure Boundary Program Elements	N-LIST-00531-10003	No Change
Pressure Boundary Program Manual	N-MAN-01913.11-10000	No Change
Authorized Inspection Agency Service Agreement	N-CORR-00531-20012	No Change
Design Registration	N-PROC-MP-0082	No Change
Pressure Boundary	N-PROC-MP-0004	No Change
System and Item Classification	N-PROC-MP-0040	No Change

LC 6.1 Fitness for Service Program

Licence Condition 6.1 states "*the licensee shall implement and maintain a fitness for service program*" and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Fitness for Service Program Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Aging Management	CNSC REGDOC-2.6.3 (2014)	The storage of minimum 6-year fuel will be incorporated into the aging management program as applicable as part of the ECC process.

Impact of the Storage of Minimum 6-Year Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Fitness for Service and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-Year Cooled Fuel on PWMF's Aging Management Program Related Licensing Basis Documents

OPG Fitness for Service Licensing Basis Document Title	OPG Document Number	Impact from Storage of Minimum 6-Year Cooled Fuel
Conduct of Engineering	N-STD-MP-0028	No Change
Design Management	N-PROG-MP-0009	No Change
Equipment Reliability	N-PROG-MA-0026	No Change
Integrated Aging Management	N-PROG-MP-0008	No Change
Nuclear Waste Management	W-PROG-WM-0001	No Change

Ontario Power Generation Dry Storage Container – Base (Underside) Inspection Plan	00104-PLAN-79171-00002	No Change
Used Fuel Dry Storage Container Aging Management Plan	00104-PLAN-79171-00001	No Change

LC 7.1 Radiation Protection

Licence Condition 7.1 states "*the licensee shall implement and maintain a radiation program, which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

As per OPG's N-PROG-RA-0013, "*Radiation Protection*", the overriding objective of the Radiation Protection (RP) program at OPG is the control of occupational and public exposure to radiation. For the purposes of controlling radiation doses to workers and the public, this program has five implementing objectives:

- Keeping individual radiation doses below regulatory limits
- Avoiding unplanned radiation exposures
- Keeping individual risk from lifetime radiation exposure to an acceptable level
- Keeping collective radiation doses ALARA, social and economic factors taken into account
- Keeping public exposure to radiation well within regulatory limits.

Higher Dose Rates

Higher dose rates from the minimum 6-year cooled fuel DSCs directly impacts workers and equipment that interface with the DSC. It has been analyzed that the anticipated dose rates would be approximately 2.5 times higher in comparison to the storage of 10-year cooled fuel. This is manageable with a different Radiation Exposure Permit (REP) to address worker safety; and no meaningful impact on OPG equipment. Dose rates will be managed with the As Low as Reasonably Achievable (ALARA) principles associated with an updated REP.

New REP's for workers interfacing with the younger cooled fuel will be developed and implemented prior to commissioning of any DSC's containing younger cooled fuel.

Estimated Public Dose

Estimated public doses have been analyzed in Enclosure 2 (section 5.3.3) and in Enclosure 3 (section 4.3.2). Both analyses assess that the dose to public, as a result of the storage of minimum 6-year cooled fuel in SB3, remains far below regulatory limits.

Based on previous correspondence with the CNSC, and reaffirmed in this application, dose rates will be measured during the initial placement of 6-year-old fuel and actions will be taken are taken prior to the dose rate criterion being exceeded.

Dose Rates and Temperature Impact on the Public and Environment

Analysis has been conducted on the indirect impact that dose rates and temperatures would have on OPG equipment and the public/environment. The transfer of the DSC from IFB-B to the processing building, then to interim storage in SB1 and lastly to its final destination in SB3. This increase in dose and temperature has been analyzed to be within the regulatory limits for the public and environment (including all Action Levels stated in the PWMF LCH).

The existing TLDs around PWMF Phase I and Phase II will measure the dose rates, which are reported quarterly to the CNSC in the facility Operations Report. Monitoring of these results will confirm the impact on the regulatory dose rates. However, as SB3 is a shielded building, it is not anticipated to be a concern.

Thermal Analysis for PWMF SB3 storing 6-year cooled fuel has been completed during design. DSC's containing 6-year cooled fuel will be placed in the middle of SB3. An increase in dose and temperature has been analyzed to be within the regulatory limits to the public and environment. Temperature monitoring inside SB3 will be in place prior to the commissioning of any DSC's containing minimum 6-year cooled fuel.

List of Radiation Protection Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Radiation Protection Regulations	SOR/2000-203	Continued compliance as documented in Attachment 1

Impact of the Storage of Minimum 6-Year Cooled Fuel on PWMF's Radiation Protection Program Related Licensing Basis Documents

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Radiation Protection and identifies the impact of the storge of minimum 6-year cooled fuel on these programs.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Radiation Protection and ALARA Licensing Basis Documents

OPG Radiation Protection Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Occupational Radiation Protection Action Levels for Nuclear Waste Management Facilities	N-REP-03420-10011	No Change
Radiation Protection	N-PROG-RA-0013	No Change

LC 8.1 Conventional Health and Safety

Licence Condition 8.1 states "*the licensee shall implement and maintain a conventional health and safety program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Regulatory Requirements Related to Conventional Health and Safety

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
General Nuclear Safety and Control Regulations	SOR/2000-202	Continued compliance as documented in Attachment 1

Ensuring Conventional Safety Performance

The foundation of OPG's Health and Safety Management System is OPG-POL-0001, "*Employee Health and Safety Policy*" which describes the approach and commitments to conventional health and safety for the organization, and the requirements and accountabilities of all employees.

OPG's program document OPG-PROG-0005, "*Environment Health and Safety Managed Systems*" governs the design and execution of OPG's Health and Safety Managed Systems in accordance with OPG-POL-0001. The Health and Safety Managed System program and supporting governing documents establish process requirements that protect employees by ensuring they are working safely in a healthy and injury-free workplace. It also outlines the responsibilities of various levels in the organization to ensure activities are performed to meet the requirements of OPG's Health and Safety Policy.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Conventional Safety program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Conventional Safety Program Licensing Basis Documents

OPG Conventional Safety Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Employee Health and Safety Policy	OPG-POL-0001	No Change
Health and Safety Management System Program	OPG-PROG-0010	No Change

LC 9.1 Environmental Protection

Licence Condition 9.1 states "*the licensee shall implement and maintain an environmental protection program, which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Environmental Protection Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Environmental Protection: Environmental Principles, Assessments and Protection Measures	REGDOC-2.9.1, Section 4.6 (2017)	Environmental-related assessments were conducted in accordance with requirements
Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities	CSA N288.1 (2014)	Environmental-related assessments were conducted in accordance with requirements

Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	CSA N288.3.4 (2013)	Environmental-related assessments were conducted in accordance with requirements
Environmental monitoring program at class I nuclear facilities and uranium mines and mills	CSA N288.4 (2015)	Environmental-related assessments were conducted in accordance with requirements
Effluent monitoring programs at class I nuclear facilities and uranium mines and mills	CSA N288.5 (2011)	Environmental-related assessments were conducted in accordance with requirements
Environmental risk assessments at class I nuclear facilities and uranium mines and mills	CSA N288.6 (2012)	Environmental-related assessments were conducted in accordance with requirements
Groundwater protection programs at Class I nuclear facilities and uranium mines and mills.	CSA N288.7 (2015)	Environmental-related assessments were conducted in accordance with requirements

Effluent and Emissions Control (Releases)

OPG is committed to complying with the requirements of the CSA Standard N288 series documents, as required in the PWMF LCH. The licensee shall control radiological releases to ALARA, thereby minimizing dose to the public resulting from PWMF operation.

The PWMF reports against approved Derived Release Limits (DRLs), which are defined in CSA Standard N288.1 as the release rate that would cause an individual of the most highly exposed group to receive and be committed to a dose equal to the regulatory annual dose limit, due to release of a given radionuclide to air or surface water during normal operation of a nuclear facility over the period of a calendar year.

Because radiological releases are very small in comparison with the Derived Release Limits (DRLs) and Action Levels, lower Internal Investigation Levels (IILs) are used to demonstrate and maintain adherence to the ALARA principle. There will be no changes to the DRLs, Action Levels or IILs as a result of the storage of younger than 10-year cooled fuel. Consistent with current performance, the cumulative public dose resulting from the storage of the younger cooled fuel will remain well below 1% of the regulatory public dose limit of 1,000 μSv per year.

Environmental Management System (EMS)

OPG's OPG-POL-0021, "*Environmental Policy*" requires that OPG maintain an Environmental Management System (EMS) consistent with the ISO 14001, "*Environmental Management System Standard*".

Operation of the PWMF will continue to be in accordance with OPG's EMS as described in OPG-PROG-0005, "*Environment Health and Safety Managed Systems*" and OPG-POL-0021. The EMS provides specific direction on how the Environmental Policy is implemented while meeting the expectations of OPG-POL-0032, "*Safe Operations Policy*", N-POL-0001, "*Nuclear Safety & Security Policy"*, and N-CHAR-AS-0002, "*Nuclear Management System*".

Continued Validity of Prior Submissions to the CNSC/Licensing Documents

Enclosure 1 contains an assessment that reviewed the following current licensing documents:

Environmental Assessments (EAs):

- Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002
- Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment. December 2007. NK30-REP-07701-00002

Environmental Risk Assessment (ERA) and Predictive Effects Assessment (PEA):

- ERA for Pickering Nuclear. Feb 2018. P-REP-07701-00001 R001
- PEA for Pickering Nuclear Safe Storage. April 2017. P-REP-07701-00002 R000

Operating Licences and Handbooks:

- Nuclear Power Reactor Operating Licence. Pickering Nuclear Generating Station. PROL 48.00/2028.
- Pickering Nuclear Generating Station Nuclear Power Reactor. Licence Conditions Handbook. LCH-PR-48.00/2028-R000.
- Waste Facility Operating Licence. Pickering Waste Management Facility. WFOL-W4-350.0/2028.
- Pickering Waste Management Facility. Licence Conditions Handbook. LCH-W4-350.00/2028.

As a result, OPG concluded that a stand-alone environmental submission to CNSC is not required since loading, transporting, and storage of cooled used fuel, 6-year or older, is considered to be within the scope of the relevant project EAs and falls within the conditions of the Pickering Nuclear and PWMF Waste Operating Licences.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Environmental Protection and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Environmental Protection Licensing Basis Documents

OPG Environmental Protection Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Environment Health and Safety Managed Systems	OPG-PROG-0005	No Change
Environment Policy	OPG-POL-0021	No Change
Management of the Environmental Monitoring Program	N-PROC-OP-0025	No Change
Monitoring of Nuclear and Hazardous Substances in Effluents	N-STD-OP-0031	No Change
Environmental Risk Assessment Report for Pickering Nuclear	P-REP-07701-00001	No Change
Derived Release Limits and Environmental Action Levels for Pickering Nuclear	P-REP-03482-00006	No Change

LC 9.2 Environmental Assessment Follow-Up Program

Licence Condition 9.2 states "*the licensee shall implement an environmental assessment follow-up plan"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Enclosure 1 contains an assessment of the continued validity of the PWMF Phase II Site Environmental Assessment (EA) (December 2003) with the storage of minimum 6-year cooled fuel. As a result, the EA Follow-Up Plan also remains valid and will continue to be conducted as originally committed for SB3.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Environmental Assessment Follow-Up Plan and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Environmental Assessment Follow-Up Plan Licensing Basis Documents

OPG Environmental Protection Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Pickering Waste Management Facility Phase II – Environmental Assessment Follow-Up Plan	92896-REP-07701.8-00001	No Change

LC 10.1 Emergency Preparedness Program

Licence Condition 10.1 states "*the licensee shall implement an emergency preparedness program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Emergency Management Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Nuclear Emergency Preparedness and Response, Version 2	CNSC REGDOC-2.10.1 (2017)	No change

Nuclear Emergency Preparedness and Response

OPG's Emergency Preparedness program N-PROG-RA-0001, "*Consolidated Nuclear Emergency Plan*", requires OPG staff to implement and maintain its emergency response capability to protect the public, employees, and the environment in the event of a nuclear emergency."

Impact of the Storage of Minimum 6-Year Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Emergency Management and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Emergency Management Licensing Basis Documents

OPG Emergency Management and Fire Protection Licensing Basis Document Title	OPG Document Number	Impact
Radioactive Materials Transportation Emergency Response Plan	N-STD-RA-0036	No Change
Consolidated Nuclear Emergency Plan	N-PROG-RA-0001	No Change

LC 10.2 Fire Protection Program

Licence Condition 10.2 states "*the licensee shall implement a fire protection program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Fire Protection Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Fire protection for facilities that process, handle, or store nuclear substances	CSA N393-13 (2013)	No change
National Building Code of Canada (2020)	NRC	No Change
National Fire Code of Canada (2020)	NRC	No Change

Fire Emergency Preparedness and Response

OPG's Fire Protection program, N-PROG-RA-0012, "*Fire Protection"* establishes provisions to prevent, mitigate and respond to fires such that fire risk to OPG Nuclear workers, public, environment, nuclear physical assets, and power generation, is acceptably low and controlled. There will be no changes to N-PROG-RA-0012 as a result of the storage of minimum 6-year cooled fuel.

Impact of the Storage of Minimum 6-Year Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Fire Protection and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Fire Protection Licensing Basis Documents

OPG Emergency Management and Fire Protection Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Fire Protection	N-PROG-RA-0012	No Change

LC 11.1 Waste Management Program

Licence Condition 11.1 states "*the licensee shall implement a waste management program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Waste Management Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
General principles for the management of radioactive waste and irradiated fuel	CSA N292.0 (2019)	The storage of younger than 10-year cooled fuel complies with the requirements in this CSA Standard.
Interim dry storage of irradiated fuel	CSA N292.2 (2013)	The storage of younger than 10-year cooled fuel complies with the requirements in this CSA Standard.

Management of low and intermediate-level radioactive waste	CSA N292.3 (2014)	The storage of younger than 10-year cooled fuel complies with the requirements in this CSA Standard.

Impact of the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Waste Management program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Waste Management Licensing Basis Documents

OPG Waste Management Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-Year Cooled Fuel
Segregation and Handling of Radioactive Wastes	N-PROC-RA-0017	No Change
Management of Waste and Other Environmentally Regulated Materials	OPG-STD-0156	No Change
Nuclear Waste Management	W-PROG-WM-0001	No Change
Radiation Protection	N-PROG-RA-0013	No Change

LC 11.2 Decommissioning Plan

Licence Condition 11.2 states "*the licensee shall maintain a decommissioning plan"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Decommissioning Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Decommissioning of facilities containing nuclear substances	CSA N294-09 (2009)	The storage of minimum 6-year cooled fuel complies with the requirements in this CSA Standard.
Decommissioning of facilities containing nuclear substances	CSA N294-19 (2019)	The storage of minimum 6-year cooled fuel complies with the requirements in this CSA Standard.

Preliminary Decommissioning Plan

As the DSC used to store minimum 6-year cooled fuel remains the same, there is no requirement to update the Preliminary Decommissioning Plan (PDP). The current PWMF PDP does not stipulate the age of the fuel being stored within the DSC.

Impact of the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Decommissioning Plan and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact of the Storage of Minimum 6-year Cooled Fuel on PWMF's Decommissioning Licensing Basis Documents

OPG Waste Management Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Decommissioning Program	W-PROG-WM-0003	No Change
Preliminary Decommissioning Plan Pickering Waste Management Facility	92896-PLAN-00960-00001	No Change

LC 12.1 Security Program

Licence Condition 12.1 states "*the licensee shall implement and maintain a security program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Security Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Nuclear Security Regulations	SOR/2000-209	Compliance documented in Attachment 1
Fitness for Duty, Volume III: Nuclear Security Officer Medical, Physical, and Psychological Fitness	CNSC REGDOC-2.2.4 (2018)	Continued compliance.
High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices	CNSC REGDOC-2.12.1 (2018)	Continued compliance.
Site Access Security Clearance	CNSC REGDOC- 2.12.2 (2013)	Continued compliance.

Facilities and Equipment

The storage of minimum 6-year cooled fuel will not require changes to security related facilities, equipment or staffing levels at PWMF.

Response Arrangements

The storage of minimum 6-year cooled fuel will not require changes to security response arrangements or processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Security program and identifies the Impact from the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Security Program Licensing Basis Documents

OPG Security Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Pickering Waste Management Facility Phase II Security Report	92896-REP-08160-00001	No Change
Pickering Waste Management Facility Security Report Addendum	92896-REP-08160-00001 ADD 001	No Change
Transport Security Plan	TRAN-PLAN-03450- 10000	No Change
Nuclear Security	N-PROG-RA-0011	No Change
Cyber Security	N-PROC-RA-0135	No Change
Nuclear Waste Management Cyber Essential Assets	W-LIST-08161-00001	No Change

LC 12.2 Construction

Licence Condition 12.2 states "*the licensee shall not carry out the activities referred to in paragraph (iii) of Part IV of this licence that relate to completed construction activities in paragraph (iv) of Part IV of this licence until the submission of the proposed security arrangements and measures for the new building, or any potential modifications to the protected area that may be associated with this new building, that is acceptable to the Commission or a person authorized by the Commission"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

No construction activities will be required as a result of the storage of minimum 6-year cooled fuel at PWMF.

LC 13.1 Safeguards Program

Licence Condition 13.1 states "*the licensee shall implement and maintain a safeguards program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

List of Safeguards and Non-Proliferation Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Safeguards and Nuclear Material Accountancy	CNSC REGDOC-2.13.1 (2018)	Continued compliance

Nuclear Material Accountancy and Control

All reports and information necessary for safeguards implementation and compliance will continue to be provided to the IAEA and CNSC on a timely basis.

Access and Assistance to the IAEA

Canadian facilities are selected at random by the IAEA for physical inspections to confirm compliance with international non-proliferation requirements. The storage of less than ten-year cooled fuel will have no impact on IAEA inspections or access to IAEA equipment.

Safeguards Equipment, Containment and Surveillance

The storage of minimum 6-year cooled fuel may have some impact on existing IAEA safeguards surveillance monitoring equipment (with respect to temperatures and sealing processes). This is discussed in section LC 3.1 Operating Performance. Analysis in this area continues and OPG continues to work with both the IAEA and CNSC to reach an agreeable outcome.

NuFlash

NuFlash is a system used for tracking nuclear fuel location and storage history. Currently, NuFlash does not allow the preparation of DSC packages for younger than 10-year cooled fuel. The changes required to update the NuFlash database to allow for 100 DSCs to be processed with 6-year to 10-year old fuel will be completed prior to the commissioning of the first DSC containing younger cooled fuel.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Safeguards program and identifies the impact from the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Safeguards Program Licensing Basis Documents

OPG Safeguards and Non- Proliferation Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Nuclear Safeguards	N-PROG-RA-0015	No Change
Nuclear Safeguards Implementation	N-STD-RA-0024	No Change

LC 14.1 Packaging and Transport Program

Licence Condition 14.1 states "*the licensee shall maintain a packaging and transport program"* and the details in the PWMF Licence Conditions Handbook (LCH) outline the regulatory requirements. The information provided in the last PWMF licence application is still valid.

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Packaging and Transport program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Packaging and Transport Licensing Basis Documents

OPG Transportation and Packaging Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Radioactive Material Transportation	W-PROG-WM-0002	No Change
Radioactive Materials Transportation Emergency Response Plan	N-STD-RA-0036	No Change
Radiation Protection	N-PROG-RA-0013	No Change

Section 3: Other Matters of Regulatory Interest

Public Information and Engagement

OPG believes in timely open and transparent communication to maintain positive and supportive relationships and confidence of key stakeholders. OPG's Corporate Relations and Communications organization adheres to the principles and process for external communications as governed by the nuclear standard N-STD-AS -0013, "*Nuclear Public Information and Disclosure*".

List of Public Information and Disclosure Related Regulatory Requirements

Licensing Basis Document Title	Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Public Information and Disclosure	CNSC REGDOC-3.2.1 (2018)	Continued compliance

Impact from the Storage of Minimum 6-year Cooled Fuel on OPG Governance, Programs and Processes

The table below provides the list of OPG governance, programs and processes that form the licensing basis for PWMF's Public Information and Disclosure program and identifies the impact of the storage of minimum 6-year cooled fuel on these programs and processes.

Impact from the Storage of Minimum 6-year Cooled Fuel on PWMF's Public Information and Disclosure Licensing Basis Documents

OPG Transportation and Packaging Licensing Basis Document Title	OPG Document Number	Impact from the Storage of Minimum 6-year Cooled Fuel
Nuclear Public Information Disclosure	N-STD-AS-0013	No Change

OPG provides responses to issues and questions raised by stakeholders and the public, and tracks issues and questions to identify trends in order to further refine proactive communications. Two-way dialogue with community stakeholders and residents is facilitated through personal contact, community newsletters, speaking engagements, advertising and educational outreach.

Through this regular outreach of an on-going nature, OPG continues to provide members of the public and interested parties with information regarding activities at the Pickering Waste Management Facility.

Community Committees

The Pickering Community Advisory Council (CAC) meets to exchange information and provide advice to senior station management on station activities as they relate to the adjacent community and public use of the waterfront trail and adjacent lands. Feedback for the waste management facility is obtained through this venue.

OPG also has a representative on the Durham Nuclear Health Committee (DNHC). OPG Nuclear staff make regular presentations to the DNHC on a variety of environmental, community outreach and operational issues. The committee is chaired by the Durham Region Medical Officer of Health.

Community Publications

OPG provides a community newsletter called "Neighbours" on a quarterly basis that are circulated by mail to residents throughout Durham Region (specific to the proximity of the respective nuclear power reactor stations). This provides an update of activities and events that occur at the respective stations.

These forums provide an opportunity for public engagement and information exchange regarding the storage of minimum 6-year cooled fuel at PWMF. Once the Licence Amendment application has been submitted, OPG will communicate the need to store minimum 6-year cooled fuel and status updates to the public through these communication tools.

Indigenous Community Engagement

OPG acknowledges the Aboriginal and Treaty Rights of Indigenous communities as recognized in the *Constitution Act, 1982*. Under its Indigenous Relations Policy, OPG regularly undertakes engagement with Indigenous communities with established or asserted rights and/or interests.

Based on work undertaken through Indigenous engagement, OPG believes the following specific Indigenous Nations and communities continue to have a primary Aboriginal and/or treaty rights and interests with respect to OPG's waste operations at the PWMF:

- Williams Treaties First Nations
- Mohawks of the Bay of Quinte
- Métis Nation of Ontario Region 8

OPG has engaged with these Indigenous communities throughout 2022 and 2023 in order to provide them with information regarding activities at the PWMF (such as the in-service of SB4 in 2021) and to discuss any identified issues and concerns.

Once the Licence Amendment application to store minimum 6-year cooled fuel is submitted to the CNSC, OPG will engage with the Indigenous communities identified above during regular scheduled meetings and briefing to share details on the need and scope of this proposal.

Conclusion

The need to store minimum 6-year cooled fuel at PWMF is an important initiative within OPG to support the Safe Storage Project at PNGS-B. OPG is requesting an amendment of the PWMF WFOL to add a new licensed activity to possess, transfer, package, manage and store minimum 6-year cooled fuel.

OPG is responsible for continued safe operation of the PWMF and confirms that the storage of minimum 6-year cooled fuel will be implemented based on a robust safety case. The proposed activities to support the storage of minimum 6-year cooled fuel will not compromise continued safe operation at PWMF, public and employee safety, and environmental protection.

The safety case for this project can be summarized as follows:

- Design: OPG has and will continue to follow its Engineering Change Control process, to ensure the design complies with applicable PWMF Licence Condition Handbook W4-350.00/2028 regulatory requirements and that configuration management for the facility is maintained.

- Continued Safe Operation: Safety analysis demonstrates that the storage of minimum 6-year cooled fuel will have a negligible effect on safe operation of PWMF, and on public and worker safety.

- Environmental Protection: An assessment of existing environmental-related submissions to the CNSC (environmental assessments, environmental risk assessment and predictive environmental effects assessment) concludes that the storage of younger than 10-year cooled fuel at PWMF will have negligible impact on the environment.

- Licensing Basis: The storage of younger than 10-year cooled fuel at PWMF will have negligible impact on PWMF's licensing basis, governance, programs and processes.

ENCLOSURE #1

OPG Technical Memo
"Storage of Dry Storage Containers (DSCs) containing less than 10 year old used fuel bundles at the Pickering Waste Management Facility (PWMF)"
W-CORR-00531-01662

ONTARIO POWER GENERATION

Raphael McCalla
Director
Environment Nuclear

889 Brock Road, 3rd Floor, Pickering, ON L1W 3J2 (905) 839-6746 x5118 raphael.mccalla@opg.com

OPG Proprietary

Date: July 27, 2020

File No.: W-CORR-00531-01662

Lise Morton
VP Nuclear Waste Management
177 Tie Road, B21
Tiverton, On
N0G 2T0

Dear Lise Morton

Subject: Storage of Dry Storage Containers (DSCs) containing less than 10 year old used fuel bundles at the Pickering Waste Management Facility (PWMF)

References:

1. Letter from D. Howard to K. Talbot, "Pickering Waste Management Facility Thermal Performance Verification Program", dated June 10, 1997. NA44-CORR-N0014035.
2. OPG. Thermal Analysis of an Ontario Power Generation Dry Storage Container Containing Six-Year-Old 28 Or 37 - Element Fuel. Mar 20, 2014. 00104-REP-02308-00007 R00.
3. OPG. Structural Integrity Assessment of a Dry Storage Container Containing Six-Year-Old 28 - Element Fuel. Mar 5, 2014. 00104-REP-79171-00060 R00.
4. OPG. Structural Integrity Assessment of Dry Storage Container (DSC) Containing Six-Year-Old 28-element Fuel Under Postulated On-site Accident Scenarios. Sept 3, 2014. 00104-REP-79171-00061 R00.
5. OPG. Dose Rate Assessment Considering Lower Aged Fuel in PWMF SB3. Jun 15, 2020. 92896-REP-03200-00009 R00.
6. OPG. Email from C. Barua (OPG) to G. Steedman (CNSC). OPG Response To CNSC Question On OPG Submission Cd# 92896-CORR-00531-01355 In Support Of PWMF Safety Report 92896-SR-01320-10002 R006. Jun 29, 2020. 92896-CORR-00531-01381.

Introduction:

The purpose of this memo is to document OPG Environment's recommendation that a stand-alone environmental submission to the CNSC is not needed in order for OPG to perform loading, transfer, and interim storage of used fuel, that has observed a cooling period for a minimum of 6 years, from the Irradiated Fuel Bays (IFBs) to the existing PWMF Used Fuel Storage Building 3 (SB3) (a PWMF Phase II building), until a permanent storage solution becomes available.

The rationale for this decision was based on a review of the following documents:

Environmental assessments (EAs):

- Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002
- Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment. December 2007. NK30-REP-07701-00002

Environmental Risk Assessment (ERA) and Predictive Effects Assessment (PEA):

- ERA for Pickering Nuclear. Feb 2018. P-REP-07701-00001 R001
- PEA for Pickering Nuclear Safe Storage. April 2017. P-REP-07701-00002 R000

Operating Licences and Handbooks:

- Nuclear Power Reactor Operating Licence. Pickering Nuclear Generating Station. PROL 48.00/2028.
- Pickering Nuclear Generating Station Nuclear Power Reactor. Licence Conditions Handbook. LCH-PR-48.00/2028-R000.
- Waste Facility Operating Licence. Pickering Waste Management Facility. WFOL-W4-350.0/2028.
- Pickering Waste Management Facility. Licence Conditions Handbook. LCH-W4-350.00/2028.

Record of Proceedings and Record of Decision:

- Record of Proceedings, Including Reasons for Decision. May 28, 2004. Subject: Environmental Assessment Screening Report on the proposed expansion of the Pickering Waste Management Facility (Phase II). Available online: http://www.nuclearsafety.gc.ca/eng/the-commission/pdf/Decision-OPG-PWMF-e.pdf

- Record of Proceedings, Including Reasons for Decision. December 10, 2008. Subject: Screening Environmental Assessment of the Pickering Nuclear Generating Station B Refurbishment and Continued Operations Project, Pickering, Ontario. Available online: http://www.suretenucleaire.gc.ca/eng/the-commission/pdf/2008-12-10-Decision-PickeringB-e-Edocs3330500.pdf

- Record of Decision. April 13, 2017. Subject: Application to Renew the Waste Facility Operating Licence for the Pickering. Available online: http://nuclearsafety.gc.ca/eng/the-commission/pdf/2017-04-13-Decision-OPG-PickeringWasteManagementFacility-e.pdf

Relevant sections considered in the above documents are presented in Attachment A (section A1 – A6).

Background:

As a common practice, used fuel from operating units at the Pickering Nuclear Generating Station (PNGS) is cooled in the IFBs for a minimum of 10 years before being transferred into DSCs, and placed into interim storage buildings in the PWMF. This practice is described in the PWMF Phase II EA, Pickering B Refurbishment and Continued Operation EA, Pickering Nuclear ERA, Pickering Nuclear Safe Storage PEA, and the Record of Proceedings associated with the EA of the Pickering B Refurbishment and Continued Operations**.**

Loading and interim storage of a DSC containing four modules of 6-year-old used fuel was successfully completed in May 1998 at the PWMF. Authorization at the time was given by Atomic Energy Control Board (AECB) (Reference 1). Repeating this infrequent practice in the future will enable OPG to create additional space in the IFB-B to allow for storage of fuel from Unit 5 to 8 to support permanent shutdown of the PNGS and planning of Pickering Safe Storage.

Summarized below is the outcome of the review.

Project scope

There is no change in project scope as described in the EA's. The project scope includes used fuel transfer and interim storage in the PWMF. It does not specify the age of the fuel allowed for transfer and storage. Refer to attachment A section A1 for more details.

Licensed activities and conditions:

There is no change to licensed activities and conditions. The Pickering Operating Licence and the WWMF Operating Licence together covers the transport, packaging, management and interim storage of the nuclear fuel. Loading and storing younger used fuel will not deviate from any of the licence conditions. Refer to attachment A section A2 for more details.

Cooling period of used fuel:

Although various documents (i.e. EA's, ERA, PEA, Record of Proceedings) describe how used fuel is cooled in the IFBs for a minimum of 10 years before being loaded, transferred, and stored (see attachment A section A3), it is still possible to initiate and implement a change to this current practice via existing OPG processes (e.g., Engineering Change and Control (ECC)).

Since the change to reduce the cooling period is not considered a 'Designated Project' under the Canadian Impact Assessment Act (IAA), there is no requirement to conduct an EA (or now referred to as an IA) under the Impact Assessment Act. Past EA's were completed as part of the licence application process to support their respective licensing decisions. Once the licensing decisions are made, EAs are not revised.

The 2017 ERA was provided to CNSC to support the PWMF and Pickering operating licence renewal application and an ERA is required to be routinely updated every 5 years as per RegDoc 2.9.1 (Environmental Principles, Assessment and Protection Measures). The routine ERA updates for PN will consider any impacts from recent changes in operational activities including fuel loading, transfer and storage. A decision is expected to be made in 2021 on whether there is a need to update the PEA based on any known activities that may potentially invalidate the bounding scenarios or assumptions made in the PEA. The change identified in this memo will be assessed as part of that decision.

Fuel integrity:

Younger used fuel is expected to have a higher thermal temperature than older used fuel. It is mentioned in the EA (92896-REP-07701-00002) that the temperature of the fuel in dry storage is an important factor in the assurance of fuel integrity and safety and a temperature of up to 300ºC can be considered safe. A maximum and conservative fuel sheath temperature of about 272ºC is predicted based on a thermal analysis of a DSC containing 6-year-old fuel (Reference 2), which is less than the 300ºC limit mentioned in the EA. Thermal stresses produced from 6 year old fuel stored in a DSC is also predicted not to compromise the containment and radiation shielding functions of the DSC under processing and storage accident conditions based on structural integrity analysis completed (Reference 3 and Reference 4).

As long as the fuel sheath temperature remains under the upper limit of 300ºC, there should be no additional environmental risks associated with fuel integrity. Refer to Attachment A section A4 for more details on the fuel integrity related descriptions found in the EAs.

Dose rates:

DSCs containing younger used fuel may have higher dose rates compared to those without depending on the average age and arrangement of used fuel bundles inside in the DSCs. The predicted dose rates and annual doses from SB3 (from a bounding scenario that includes storage of 100 DSCs containing only 6 year old decayed used fuel in SB3) are still well within the regulatory limit (i.e., 1 mSv/y for a member of the public). Dose rates at the existing protected area fence for the SB3 bounding scenario are expected to remain well within the radiation dose rate targets of ≤ 0.5 μSv/h at the PWMF II perimeter fence and ≤ 100 μSv/y at the PNGS site boundary, as proposed in recent communication with the CNSC (Reference 6). Dose rates at the Phase II protected area fence will continued to be measured and monitored and mitigating actions taken if required.

The storage of younger fuel will not pose an unacceptable risk to workers or members of the public nor will it likely to result in adverse effects on the environment provided that the ECC process and the ALARA principle are followed and that all the relevant conditions under the Pickering Nuclear and PWMF Operating Licences (e.g., to implement and maintain the radiation protection program, environmental protection program, waste management program, and packaging and transport program) continue to be met.

For more details on the dose rate predictions, see shielding assessment for PWMF using lower fuel age in SB3 (Reference 5). For more details on the dose rate related assessments completed in the past EA's and the relevant regulatory limit and targets, see attachment A section A5. For more details on the licence conditions, see attachment A section A1

Conclusion:

A stand-alone environmental submission to CNSC is not required since loading, transporting, and storage of used fuel, 6 year or older, is considered to be within the scope of the relevant project EAs and falls within the conditions of the Pickering Nuclear and PWMF Waste Operating Licences.

Prior to the implementation of the plan to load, transfer, and store younger used fuel, the PWMF Safety Report will be updated and the OPG ECC process will be followed to demonstrate that OPG will be able to maintain an adequate level of safety. Changes to existing governance stemming from the plan to load, transfer and store used fuel with a shorter cooling period than 10 years will also be managed through the ECC process.

Sincerely,

RMcCalla

Raphael McCalla
Director
Environment Nuclear

RM/sl

cc. Cammie Cheng
Jason Wight
Paul Crowley
Rafi Asadi
Kapil Aggarwal
Mark Priest
Steve Bagshaw
Mark Ferry
Ram Kalyanasundaram
Cameron Spence

Attachment A

Supporting Information

Section A1

Scope of the Project:

SOURCE: Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002

1.3.1 Scope of the Project

The physical works involved in this project are the storage buildings to be built for the dry storage containers; all facilities, systems and activities required for the construction and operation of PWMF II; and the facilities, systems and activities required for the construction and operation of PWMF Phase II; and the facilities, systems, and activities involved in the transfer of loaded welded DSCs from PWMF I to the storage buildings in PWMF II.

Associated operations and activities that are within the scope of the project include:

- Preparation of systems and facilities involved in the transfer of loaded welded DSCs
    - Transfer of loaded welded DSCs from the Processing Workshop or Storage Buildings 1 and 2 in PWMF I to Storage Buildings 3 and 4 in PWMF II.

SOURCE: Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment. December 2007. NK30-REP-07701-00002

1.4.2 Scope of the Project

The physical works for the Project are the PNGS B Units 5, 6, 7 and 8 and ancillary systems necessary for their operation through to about 2060.

As outlined in the EA Guidelines (Section 7.0, p.5), the scope of project will consider the following activities related to the continued operation of the refurbished reactors until about 2060, including:

- continued interim storage of used fuel at the Pickering Used Fuel Dry Storage Facility (PUFDSF) within the PWMF;
- interim storage for the additional used nuclear fuel and the refurbishment waste at the PWMF;

Section A2

Licensed activities and conditions

SOURCE: Nuclear Power Reactor Operating Licence Pickering Nuclear Generating Station PROL 48.00/2028

IV) LICENSED ACTIVITIES:

This licence authorizes the licensee to:

(i) operate the Pickering Nuclear Generating Station (hereinafter “the nuclear facility”) at a site located in the City of Pickering, in the Regional Municipality of Durham, in the Province of Ontario;

(ii) possess, transfer, use, package, manage and store the nuclear substances that are required for, associated with, or arise from the activities described in (i);

(vi) transport Category II nuclear material by road vehicle from the nuclear facility spent fuel bay to the onsite waste storage facility;

VI) CONDITIONS:

4. Safety Analysis
4.1 The licensee shall implement and maintain a safety analysis program.

7. Radiation Protection
7.1 The licensee shall implement and maintain a radiation protection program, which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days.

9. Environmental Protection
9.1 The licensee shall implement and maintain an environmental protection program which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days.

11. Waste Management
11.1 The licensee shall implement and maintain a waste management program.

14. Packaging and Transport
14.1 The licensee shall implement and maintain a packaging and transport program.

SOURCE: Licence Conditions Handbook (LCH-PR-48.00/2028-R000) - Pickering Nuclear Generating Station Nuclear Power Reactor Operating Licence

Licence Condition G.1: Nuclear Substances

Activity (ii) in the licence authorizes the licensee to possess, transfer, use, package, manage and store nuclear substances.

Activity (vi) in the licence authorizes the licensee to transport Category II nuclear material i.e. fuel by road from Pickering NGS spent fuel bay to the onsite waste storage facility, The Pickering waste storage facility is licensed separately from the Pickering NGS licence (WFOL-W4-350.02/2018 – e-Doc 4002929). This activity is addressed as part of LC 14.1, which describes the packaging and transport program.

Licence Condition 14.1: Packaging and Transport Program

The licensee shall implement and maintain a packaging and transport program.

Preamble:

Every person who transports radioactive material, or requires it to be transported, shall act in accordance with the requirements of the Transportation of Dangerous Goods Regulations (TDGR) and the Packaging and Transport of Nuclear Substances Regulations, 2015 (PTNSR 2015).

The PTNSR 2015 and the TDGR provide specific requirements for the design of transport packages, the packaging, marking and labeling of packages and the handling and transport of nuclear substances.

The packaging and transport SCA includes the following specific areas (SpAs):
• Package design and maintenance;
• Packaging and transport; and
• Registration for use.

Compliance Verification Criteria:

Licensee Documents that Require Notification of Change		
Document #	**Title**	**Prior Notification**
W-PROG-WM-0002	Radioactive Material Transportation	No
N-STD-RA-0036	Radioactive Materials Transportation Emergency Response Plan	No

Package Design and Maintenance:
PTNSR 2015 apply to the packaging and transport of nuclear substances, including the design, production, use, inspection, maintenance and repair of packages, and the preparation, consigning, handling, loading, carriage and unloading of packages. Where necessary, OPG package designs are certified by the CNSC

Packaging and Transport (Program):
The licensee shall implement and maintain a packaging and transport program that will ensure compliance with the requirements of the TDGR and the PTNSR 2015 for all shipments of nuclear substances to and from the Pickering NGS site. Shipments of nuclear substances within the nuclear facility where access to the property is controlled are exempted from the application of TDGR and PTNSR 2015.

Registration and Use:
OPG's packaging and transport program also covers the registration for use of certified packages as required by the regulations.

Guidance:

Org / Document #	Title	Version
CNSC / REGDOC 2.14.1	Information Incorporated by Reference in Canada's Packaging and Transport of Nuclear Substance Regulations, 2015	2016

SOURCE: Waste Facility Operating Licence Pickering Waste Management Facility – WFOL-W4-350.0/2028

IV) LICENSED ACTIVITIES:

This licence authorizes the licensee to:

(i) operate the Pickering Waste Management Facility ("the facility") located at the Pickering Nuclear Generating Station, City of Pickering, Regional Municipality of Durham, Province of Ontario;

(ii) possess, transfer, use, process, package, manage, and store nuclear substances that are required for, associated with or arise from the activities described in (i);

(iii) transport Category II nuclear materials that are associated with the activities described in (i) on the site of the Pickering Nuclear Generating Station;

VI) CONDITIONS:

4 Safety Analysis
4.1 Safety Analysis Program
The licensee shall implement and maintain a safety analysis program.

7 Radiation Protection
7.1 Radiation Protection
The licensee shall implement and maintain a radiation protection program, which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days.

9 Environmental Protection
9.1 Environmental Protection

The licensee shall implement and maintain an environmental protection program, which includes a set of action levels. When the licensee becomes aware that an action level has been reached, the licensee shall notify the Commission within seven days.

10 EMERGENCY MANAGEMENT AND FIRE PROTECTION
10.1 Emergency Preparedness Program
The licensee shall implement and maintain an emergency preparedness program.
10.2 Fire Protection Program
The licensee shall implement and maintain a fire protection program.

11 Waste Management
11.1 Waste Management Program
The licensee shall implement and maintain a waste management program.

14 Packaging and Transport
14.1 Packaging and Transport Program
The licensee shall implement and maintain a packaging and transport program.

SOURCE: Pickering Waste Management Facility Licence Conditions Handbook LCH-W4-350.00/2028

Licence Condition 4.1 Safety Analysis Program

Compliance Verification Criteria
Licensee Documents that Require Notification of Change

Doc#	Title	Prior Notice
92896-SR-01320-10002	Pickering Waste Management Facility – Safety Report	Y

Licensing Basis Publications

Org	Doc#	Title
CSA Group	N292.0-14	General principles for the management of radioactive waste and irradiated fuel
CSA Group	N-292.2-13	Interim dry storage of irradiated fuel

The safety analysis report is to confirm that the consequences of a range of events are acceptable. It includes an integrated assessment of the facility to demonstrate, among other things, adequate safety for external events such as fires, floods, and tornados, and adequate protective features to ensure the effects of an event do not impair safety related systems, structures, and components (SSC).

Every 5 years, OPG shall submit a revised safety analysis report for the facility. CNSC staff review the safety analysis report to verify that OPG employs appropriate assumptions, applies adequate scope, and demonstrates acceptable results. The safety analysis report must demonstrate that the radiological consequences of accident scenarios do not exceed public dose limits.

Licensees shall carry out safety analyses to confirm that facility design changes will not result in a reduction of safety compared to the licensing basis, as per LC G.1. The safety analysis report shall:

- demonstrate compliance with public dose limits, the dose-related criteria, structural-integrity related criteria, the limits on process and safety parameters, and safety or safety-related system requirements;

- justify appropriateness of the technical solutions employed in the supporting justification of safety requirements; and,

- complement other analyses and evaluations in defining a complete set of design and operating requirements.

Licence Condition 7.1 Radiation Protection

The Radiation Protection Regulations require that the licensee implement a radiation protection program and also ascertain and record doses for each person who perform any duties in connection with any activity that is authorized by the NSCA or is present at a place where that activity is carried on. This program must ensure that doses to persons (including workers) do not exceed prescribed dose limits and are kept As Low As Reasonably Achievable (the ALARA principle), social and economic factors being taken into account.

The regulatory dose limit to workers and the general public are explicitly provided in sections 13, 14 and 15 of the Radiation Protection Regulations.

Licence Condition 11.1 Waste Management Program

With respect to the storage and management of spent nuclear fuel, the waste management program should reflect the fundamental safety concerns related to criticality, exposure, heat control, containment, and retrievability. That is, the systems that are designed and operated should assure subcriticality, control of radiation exposure, assure heat removal, assure containment, and allow retrievability.

Licence Condition 14.1 Packaging and Transport Program

Compliance Verification Criteria
Licensee Documents that Require Notification of Change

Doc#	Title	Prior Notice
W-PROG-WM-0002	Radioactive Material Transportation	N
N-STD-RA-0036	Radioactive Materials Transportation Emergency Response Plan	N
N-PROG-RA-0013	Radiation Protection	Y

Section A3

Description on the cooling duration of used fuel in IFBs:

SOURCE: ERA for Pickering Nuclear. Feb 2018. P-REP-07701-00001 R001

2.2.2.1.1 Used Fuel

Used fuel bundles are initially stored in the irradiated fuel bays for at least 10 years and then transferred to DSCs for interim storage in the PWMF. In the irradiated fuel bay, used fuel bundles are placed into 96-bundle storage modules. Modules with used fuel at least 10 years or older may be loaded into a DSC, which has the capacity to hold four storage modules. The DSC is loaded with the storage modules and the lid is secured while the DSC is submerged in water. The DSC is then removed from the water, drained, the exterior decontaminated, and then the DSC is prepared for on-site transfer to the PWMF for further processing and subsequent interim storage

SOURCE: PEA for Pickering Nuclear Safe Storage. April 2017. P-REP-07701-00002 R000

1.0 Introduction

Following shutdown, the activities at PN Generating Station would involve the four distinct phases outlined below.

1) A 2-3 year ***Stabilization Phase*** per unit to transition each unit, and the station as a whole, from their current operating states to their respective safe storage states. Stabilization activities will include defuelling and dewatering reactor units.

2) A 25-30 year ***Storage with Surveillance Phase*** to allow for natural decay of radioactivity. Activities during this phase include the ongoing operation of the irradiated fuel bays (IFBs) and the continued transfer of spent fuel to dry storage containers (DSCs). Current planning anticipates that used fuel transfer to DSCs will be completed within 10 years of the last unit transitioning to its safe storage state

1.1 Project Overview

Many of the specific details of the Stabilization activities are not finalized; however, assumptions have been made to provide a conservative (i.e., worst case) assessment of effects resulting from the transition and safe storage state.

Activities specific to the Stabilization Phase include:

- removal of all nuclear fuel from the reactor units and transfer to the IFBs and auxiliary irradiated fuel bay (AIFB);

Activities during the Storage with Surveillance Phase include:

- continued operation/surveillance of the IFBs, including transfer of used fuel from the IFBs to DSCs for storage on the PWMF site. It is anticipated that the irradiated fuel bays will be required for up to 10 years of cooling;

1.3 PEA Goals, Approach and Scope
The PEA report does not include the operations at the PWMF as it operates separately under the Waste Facility Operating Licence issued by the CNSC. The PEA report does, however, discuss the waste operation to the extent there are inter-relationships with the Stabilization and Storage with Surveillance activities.

3.0 Stabilization and Storage with Surveillance Activities
The main elements of the Stabilization and Storage with Surveillance Phases include the following.

- Removal of all nuclear fuel from the reactor units and transfer of the fuel to an IFB for approximately up to 10 years of cooling. Continued operation/surveillance of the IFBs and AIFB are required until all irradiated fuel and other components stored in the fuel bays are transferred into DSCs for safe interim storage at the PWMF.

3.13 Pickering Waste Management Facility
Used fuel bundles will continue to be stored in an IFB up to 10 years and then transferred to DSCs for interim storage in the PWMF.

SOURCE: Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002

Section 2.2.2.1 Development Background
Since 1996, used fuel that has been cooled for at least ten years in PN's IFBs has been routinely transferred into DSCs for dry storage at PWMF I.

Appendix C – Community and Stakeholder Consultation
C-6 Newsletters
PWMFII EA NEWS - May 2002, Issue One

When used fuel bundles are removed from the reactors at Pickering Nuclear, they are still highly radioactive. They have to be managed safely and responsibly for a long time. The first step is to cool the fuel bundles under water for up to 10 years in specially engineered used fuel bays. As the Pickering fuel bays become full, it is necessary to transfer the used fuel from the fuel bays to robust concrete and steel containers for dry storage in a specially designed facility on the station site.

C-6 Newsletters
PWMFII EA NEWS - September 2003, Issue Three

The initial used fuel dry storage facility, PWMF I, has been in operation since 1996. The facility uses a dry storage process that is a proven, safe and regulated technology, widely used by other nuclear facilities in Canada, the USA and other countries. The process involves removing used fuel bundles from the water-filled used fuel storage bays (after a minimum of 10 years in those bays) at PN and placing them in specially designed robust steel and concrete containers called "Dry Storage Container" or DSCs. The DSCs are then processed, sealed and transferred to the Used Fuel Dry Storage buildings.

C-7 Project Information Package

When used fuel bundles are removed from the reactors at Pickering Nuclear, they are still highly radioactive. They have to be managed safely and responsibly for a long time. The first step is to cool the fuel bundles under water for up to 10 years in specially engineered used fuel bays. As the Pickering fuel bays become full, it is necessary to transfer the used fuel from the fuel bays into robust concrete and steel containers and store them in a specially designed storage facility on the station site. The containers – called “Dry Storage Containers” (DSCs) - are engineered to last at least 50 years and will provide safe, interim storage until a long-term management program is in place.

Used fuel is stored for at least 10 years under water in fuel bays at Pickering Nuclear. The water keeps the fuel bundles cool and provides an effective radiation shield. This is normal practice at all OPG nuclear stations and elsewhere.

C-11 Presentation to the PN Community Advisory Council (CAC)

Presentation to Community Advisory Council - February 19, 2002

> After 10 years, the used fuel may be moved to dry storage, on site but separate from station operations.

Pickering Nuclear Generating Station Community Advisory Council
Pickering Nuclear Information Centre - March 18, 2003
Meeting Highlights

> Council Comment and Questions
> John Peters and Don Gorber responded to Council comments and questions:

- How did the EA address the effect of radiation over time?

  John: The contribution of PWMF II to gamma radiation over time depends on the age of the used fuel when it is loaded into the container. The EA took the worst case for calculating PWMF II contribution per year, fuel that is only 10 years old and put into the facility all at once.

Appendix D - Open House Information Panels

Phase II of the Pickering Waste Management Facility will:

- Be used to store only Pickering used fuel and only after it has spent at least ten years in the existing fuel bays within the stations (wet storage)

Appendix G – Review comments on draft EA Study Report and OPG’s Responses

Comments from IER & Scimus Inc. in association with North-South Environmental on behalf of the City of Pickering, July 2003 on the PWMF II Draft EA Study Report

IER comment:

> The total capacity of the storage buildings is 1654 Dry Storage Containers (DSC’s), only 7% more than the total number of DSC’s expected. This does not appear to provide sufficient contingency against unforeseen problems (Section 2.2.1, page 2-1).

OPG response:

> OPG maintains an overall nuclear waste system plan which includes all waste streams that it manages. Part of the plan addresses contingency plans for all phases of used fuel management. The dry storage step is only for used fuel that has been cooled for at least 10 years in wet storage, so there is a long lead time in determining requirements for additional storage capacity. If additional storage capacity was needed in the future another storage building could be proposed after 2016 when SB #4 was commissioned, but before 2025 when all the SBs at PN are filled to capacity. No change in the EA Study Report is required.

SOURCE: Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment. December 2007. NK30-REP-07701-00002

2.12 Basis for the Environmental assessment

Table 2.12-1, referred to as the “Basis for Environmental Assessment”, provides a listing and description of each of the works and activities associated with the Project. This information provides the basis for the assessment of the effects on each of the environmental components.

Table 2.12-1 Basis for EA Study

Project Phase / Works and Activities - Interim Storage of Used Fuel at PWMF:

Irradiated fuel is stored in the irradiated fuel storage bay for a minimum period of 10 years before being transferred to Dry Storage Containers (DSCs) for interim storage at PWMF until a long-term storage facility is available.

SOURCE: Record of Proceedings, Including Reasons for Decision. December 10, 2008. Subject: Screening Environmental Assessment of the Pickering Nuclear Generating Station B Refurbishment and Continued Operations Project, Pickering, Ontario

107. To address concerns raised by several intervenors on waste management, the Commission requested that OPG elaborates on the design of the dry-storage container used for used fuel storage and on the fuel cycle after the removal of fuel from the reactor…… To answer the fuel cycle portion of the question, OPG added that the fuel removed from the reactor is stored in water pools at the stations for a minimum of 10 years to allow the fuel to cool to about 0.1 % of the radioactivity levels present at the time of its removal from the reactor. The fuel is then transferred to dry-storage containers for storage until a disposal facility is available.

Section A4

Description on integrity of used fuel:

SOURCE: Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002

2.3.1.1 DSC Design and Operating Conditions

The DSC provides the necessary radiation shielding and containment of radioactive materials. It is designed to provide a storage life of at least 50 years and to meet all shielding and containment integrity requirements over this period.

To permit future retrieval, used fuel bundles in dry storage need to remain structurally intact and retain sufficient strength to sustain the stresses associated with future handling and transport. This requires limiting cladding deformation by creep or other degradation processes such as oxidation in the uranium dioxide fuel pellets. The integrity of used fuel cladding is also a key requirement for radiological safety. The pellet and the zircaloy sheath provide a primary barrier to prevent the release of radionuclides. The DSC provides secondary containment for any radionuclides released by the fuel, in the event that the fuel cladding integrity was compromised.

Both cladding creep and fuel matrix oxidation, the processes that could lead to splitting of the fuel cladding, resulting in release of radionuclides into the DSC cavity, are temperature dependent processes. Therefore, 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 welded closure of the DSC and the addition of an inert helium atmosphere in the DSC cavity. Oxidation is also limited due to helium.

Analysis and measurements carried out at PUFDSF indicate that the maximum fuel cladding temperature does not exceed 175ºC in dry storage. When used fuel is stored in a helium atmosphere, temperatures of up to 300ºC can be considered safe for the planned storage period for intact used fuel in DSCs. The upper temperature limit ensures that creep strain remains within acceptable limits. The inert gas precludes oxidation processes. These storage conditions are also considered safe for dry storage of used fuel with minor cladding defects. The above considerations support the conclusion that under normal operating conditions, DSCs provide safe and retrievable storage for OPG’s used nuclear fuel.

2.3.1.2 Factors Influencing Long-Term Integrity of the DSC and Used Fuel

The DSC has been designed to provide a storage life that will meet all shielding and containment integrity requirements over a minimum 50 year service life. Investigations were performed during DSC design regarding the integrity and stability of the DSC for different load cases over the 50 year service life. The DSC design is based on analyses of a range of considerations concerning the following:

- decay heat removal
- shielding
- containment
- structural integrity

2.3.1.3 DSC and Fuel Integrity under Credible Malfunction and Accident Scenarios

As part of the design process for the DSC, load scenarios approximating a range of potential accidents and malfunctions were studied. The scenarios included DSCs in a range of dry storage scenarios, and in a range of transfer methods.

Section A5

Dose, radiation, environmental effects, and mitigation:

SOURCE: Pickering Waste Management Facility Phase II Final Environmental Assessment Study Report. December 2003. 92896-REP-07701-00002

2.3.2.2 Radiation and Radioactivity Considerations in PWMF II Design

Radiation Shielding

The radiation dose rate targets for PWMF II, derived for a member of the general public, are as follows:

• ≤ 0.5 µSv/h at the PWMF II perimeter fence, based on maximum 2000 hours per year occupancy for non-Nuclear Energy Workers (non-NEWs),

• ≤ 10 µSv/y contribution at the PNGS exclusion zone boundary; this dose rate target is 1% of the CNSC dose rate limit of 1 mSv/y for a member of the public.

5.2.4.2 Regional/Local Study Area - Workers at PNGS and the PWMF I

The average individual doses to Nuclear Energy Workers (NEWs) at PNGS from both internal (i.e., inhaled or ingested) and external exposure sources were reported at 1.1 mSv/y, and the maximum individual dose was reported at 10 mSv in 2001. T hese doses are consistent with OPG's Exposure Control Level (ECL) of 10 mSv/y per calendar year, and are well below the CNSC regulatory limit of 50 mSv in any calendar year and 100 mSv over five calendar years (Canada Gazette 2000).

The baseline annual individual doses to workers (NEWs) at the PWMF I were taken from monitoring data. During 2001, nine operators at the PWMF I received an average individual dose of 0.64 mSv with a maximum of 1.94 mSv. Six mechanical maintainers who worked in the PWMF I reported measurable doses, with an average of 0.14 mSv and a maximum of 0.45 mSv (OPG 2002d). These occupational doses are consistent with OPG's ECL, and are well below the regulatory limit of 50 mSv in any calendar year and 100 mSv over five calendar years.

5.2.5.2 Site Study Area

The baseline dose from the existing environment to non-human biota in the Siting Area is attributable to two sources: i) natural background radiation and radioactivity (described in Section 5.2.5.1), and ii) licensed nuclear activities on the site.

Dose rates to biota in the Siting Area from radioactivity releases from the PNGS are attributable to external gamma radiation from radioactive noble gases, and from uptake and internal exposure to tritium and carbon-14; these dose rates were estimated at 0.11 µGy/d. The gamma dose rate (direct and skyshine gamma radiation) in the Siting Area from PWMF I was estimated at 0.03 µGy/d. The total dose rate from these sources was estimated at 0.14 µGy/d in this assessment.

In conclusion, the baseline dose to terrestrial fauna in the Siting Area was calculated to be 4.1 µGy/d, with over 90% of that contributed by natural background radiation and radioactivity. The corresponding dose to terrestrial flora was estimated in the range 1.8 to 20 µGy/d, also predominantly from natural background

7.3.1 Radiation and Radioactivity: Atmospheric Environment

7.3.1.2 Operations Phase – Likely Environmental Effects

The design of the Storage Buildings will provide for sufficient concrete shielding in the walls up to 30 cm (12") such that the gamma radiation level at the perimeter of the PWMF II site is predicted to be 0.13 µSv/h (Nuclear Safety Solutions 2003). This level meets the OPG target of < 0.5 µSv/h, corresponding to a dose < 1,000 µSv/y for 2000 h/y occupancy, the CNSC public dose limit (applicable for non-NEWs).

A dose rate of up to 50 µSv/h was predicted at the roof (Nuclear Safety Solutions 2003). This value was adopted in this assessment as a conservative estimate (i.e., overestimate) of the corresponding dose rate at the PWMF II. The predicted gamma radiation levels from full Storage Buildings located at Site Area B provides a dose rate of < 10 µSv/y at the PN east property boundary. This includes both direct and skyshine contributions. This is expected to increase the levels by less than three percent above a baseline of 350 µGy/y and will be indistinguishable from the temporal and spatial variations in natural background radiation levels at this location.

Identified Mitigation Measures

The gamma radiation from the DSCs was determined to be indistinguishable from background radiation levels at the PN east property boundary. The calculated dose rate meets OPG's dose targets and is well within CNSC's regulatory limit. Therefore, no mitigation measures are required.

7.3.2 Radiation and Radioactivity: Terrestrial Environment

7.3.2.2 Operations Phase – Likely Environmental Effects

The terrestrial environment will be affected by gamma radiation from DSCs. The effect of gamma radiation on the terrestrial environment from the operation of two Storage Buildings containing the full complement of loaded DSCs (approximately 1000) serves as the upper bound for both Project Works and Activities, including transfer of DSCs between PWMF I and PWMF II. The potential effects on birds perching on the roof, on flora and fauna at the exterior walls, and on flora and fauna at the perimeter of the PWMF II site boundary are described below.

Birds may perch on the roof of the Storage Buildings for brief periods, and be exposed to absorbed dose rates of approximately 0.05 mGy/h from gamma radiation. Exposure periods of one or two hours per day would result in dose rates of up to 0.2 mGy/d. This dose rate is less than the no-effect-level of 1 mGy/d reported by UNSCEAR (1996). Based on observations since the beginning of operation of PWMF I in 1996, birds have not nested on the roof of the PWMF I Storage Buildings 1 and 2, and therefore, are not expected to nest on the roof of the PWMF II.

The effects of gamma radiation on flora and on fauna with a limited range (e.g., field mouse) that live in the vicinity of the perimeter of the PWMF II site was evaluated by comparing estimated doses to no-effect levels reported by UNSCEAR. Fauna with a large range spend some of their time at distance from the Storage Buildings in lower radiation fields, and are expected to receive lower daily doses than the biota confined to the areas adjacent to the perimeter of the PWMF II site. Based on the assessment of the gamma radiation levels from loaded DSCs in the PWMF II Storage Buildings at the perimeter of the PWMF II site, the estimated daily dose rate to flora and fauna at that location is approximately 0.004 mGy/d. This is a small fraction of the no-effects level of 1 mGy/d reported by UNSCEAR (1996). This dose rate is expected to be within the range of natural background levels. Thus, the additional dose will be indistinguishable from the temporal and spatial variations in natural background radiation levels at this location.

Identified Mitigation Measures

Since the doses to flora and fauna are expected to be less than no-effect levels reported by UNSCEAR, no mitigation levels are required.

7.3.3 Radiation Doses to Members of the Public

7.3.3.2 Operations Phase - Likely Environmental Effects

Members of the public living and working outside the PN property boundary could potentially be affected by gamma radiation from DSCs during both of the Project Works and Activities listed above. The effect of gamma radiation on members of the public from two Storage Buildings containing the full complement of loaded DSCs (i.e., approximately 1000) serves as an upper bound for effects of both Project Works and Activities under normal operations. To ensure that all members of the public living, working or undertaking recreational activities beyond the PN property boundary are protected, a conservative estimate of the radiation dose to a hypothetical individual located year round at the PN east property boundary was compared to regulatory limits.

At the closest point on the site boundary to the PWMF II on Site Area B, the estimated annual dose from PWMF II, to a hypothetical individual located year round at the PN east property boundary was < 10 µSv (Nuclear Safety Solutions 2003), which is less than 1% of the CNSC regulatory limit of 1000 µSv/y. This annual dose is also below the level of regulatory concern of 10 µSv/y as recommended by the ACRP/ACNS (1988), and meets OPG's target of < 10 µSv/y for a member of the public.

The baseline dose to the hypothetical individual, as described in Section 5.2.4, is approximately 1,300 µSv/y from natural background radiation, and approximately 4.8 µSv/y from existing licensed operations at PN (OPG 2003d); therefore, the additional dose from the PWMF II project (< 10 µSv/y) is expected to be a very small fraction of the dose from natural background radiation, and will be indistinguishable from the temporal and spatial variations in radiation levels at this location (Figure 7.3-1).

Identified Mitigation Measures

Since it was determined that the additional dose from PWMF II to members of the public living, working or undertaking recreational activities outside the PN property boundary is expected to be a very small fraction of the dose from background radiation, it will be indistinguishable from the temporal and spatial variations in radiation levels. Therefore, no mitigation measures are required.

7.3.4 Radiation Doses to Workers

7.3.4.2 Operations Phase - Likely Environmental Effects

Workers (NEWs) Directly Involved with PWMF II

Doses to workers during normal operation at PWMF II were conservatively estimated on the basis of measured doses to workers at the PWMF I where similar activities to those identified for the PWMF II are carried out.

The individual doses to operators at PWMF II are expected to average 0.64 mSv/y with a maximum of 1.9 mSv/y. Individual doses to mechanical maintainers at PWMF II are expected to average 0.14 mSv/y with a maximum of 0.45 mSv/y. These annual doses to workers at PWMF II from normal operation are expected to be a small fraction of the regulatory limits, and well below OPG's ECL of 10 mSv/y.

PNGS Workers (NEWs)

The additional dose to individual PNGS workers from normal operation of the PWMF II (i.e., < 0.64 mSv/y) will be in addition to the baseline average annual dose received by PNGS workers of 1.1 mSv/y with a maximum of 10 mSv/y. Therefore, the average individual dose to a PN worker is predicted to be approximately 1.7 mSv/y (i.e., the sum from both activities). This is considered to be an over-estimate as the additional dose to PNGS workers from the PWMF II is expected to be much less than the average dose to workers at PWMF II. The doses from normal operation at PWMF II to PNGS workers (NEWs) are a small fraction of the CNSC's regulatory limit and OPG's ECL. Internal and external doses received by PNGS workers are monitored and reported as part of their cumulative annual dose.

<u>PN Workers (non-NEWs)</u>
PN workers (non-NEWs) who work outside the protected areas of the PWMF II and the PN will be exposed to low levels of gamma radiation from the PWMF II activities listed above and are subject to CNSC's regulatory limits on an annual dose of 1 mSv. The gamma dose rate at the security fence of the PWMF II will be maintained at levels below the OPG target of < 0.5 µSv/h (1,000 µSv for a 2,000 hour work year) (Nuclear Safety Solutions 2003). Therefore, the effects of normal operation of PWMF II on PN workers (non-NEWs) are expected to be below the regulatory limit.

Identified Mitigation Measures
Because the estimated doses to workers (both NEWs and non-NEWs) during normal operations at PWMF II were determined to be below CNSC's regulatory limits and below OPG's ECLs, no mitigation measures are required.

8.4 Radiation Dose Related to Radiological Malfunctions and Accidents

The assessment of the effects of radiological malfunctions and accidents focused on the two events during DSC on-site transfer and during DSC storage that have the potential to release radioactivity into the environment. The assessment of the effects of the release of tritium and krypton-85 following the bounding accident is based on releases of 1.4 x 1012 Bq of tritium and 7.8 x 1012 Bq of krypton-85 and is evaluated in a conservative manner.

Likely Environmental Effects
<u>Non-Human Biota</u>
The estimated dose from tritium and krypton-85 released following a bounding accident was calculated to be 0.0094 Gy which is less than 1% of the no-effect level (1 Gy) reported by UNSCEAR (1996).

<u>Members of the Public and PN Non-NEWs</u>
A preliminary estimate of the dose to members of the public at the PN property boundary was conservatively calculated at 1 µSv, based on PWMF I Safety Report methodology assumptions. This is a small fraction (0.1%) of the regulatory limit on annual dose to members of the public (Canada Gazette 2000). The estimated dose is below the level of regulatory concern as recommended by the ACRP/ACNS (1988), and the OPG dose target for malfunctions and accidents (i.e., that radiation doses to the public at the PN site boundary, following a postulated abnormal event or credible accident shall not exceed the annual public dose limit of 1,000 µSv). Also, the baseline annual dose to members of the public from licensed activities at the PN site is approximately 7 µSv/y, and from natural background radiation is approximately 1,300 µSv/y.

<u>PWMF II Workers (NEWs)</u>
The dose to workers at the PWMF II from the bounding malfunction and accident was estimated to be < 6 mSv, based on PWMF I Safety Report methodology assumptions. As discussed previously, the assumptions stated for the accident scenario are very conservative and extremely unlikely to occur. Nevertheless, if the bounding accident was postulated to occur near the end of a dosimetry year, the estimated dose to a worker at PWMF II could be in addition to a typical annual dose of approximately 0.64 mSv/y from normal operation. The total postulated dose for the year would be approximately 7 mSv, less than OPG's ECL of 10 mSv/y and a small fraction of the regulatory limit of 50 mSv in a calendar year, to a maximum of 100 mSv over a five-year period.

<u>PN Workers (NEWs)</u>
The dose to a worker at PN in proximity to a malfunction or accident is expected to be equal to or less than the corresponding dose to a PWMF II worker, i.e., 6 mSv as discussed above.

The individual dose to workers at PN from the bounding accident and malfunction would be in addition to the baseline dose received (average of 1.1 mSv/y). All internal and external doses received by workers at PN are monitored and are reported as part of their cumulative annual dose. If the accident was postulated to occur at the end of a dosimetry year, the average individual dose to a worker at PN is expected to be less than 7 mSv in that year, a small fraction of the regulatory limit. In some years, the annual dose to a few PN workers may approach the ECL of 10 mSv. If one of these workers were assumed to be exposed to the bounding malfunction or accident near the end of a dosimetry year, the total dose in the year could approach 16 mSv. This maximum postulated dose to a worker is also below the regulatory limit of 50 mSv in a calendar year, to a maximum of 100 mSv over a five-year period.

Identified Mitigation Measures
Radiation doses to workers and the public from radiological malfunctions and accidents are expected to be below CNSC's regulatory limits and OPG's ECLs. Also, radiation doses to nonhuman biota are expected to be below no-effects levels reported by UNSCEAR. Therefore, no mitigation measures are required.

9.4.3.2 Other Projects and Activities

<u>PWMF I</u>
The dose rate at the PN east property boundary from PWMF I operations has been estimated at 6 x 10-5 µSv/h (OPG 2002d), or a dose of 0.05 µSv/y to a member of the public assuming full occupancy at this location. This is a very small fraction of the CNSC regulatory limit of 1,000 µSv/y and is well below the level of concern recommended by the ACRP/ACNS.

9.5.1 Members of the Public

9.5.1.2 Identified Mitigation Measures

The estimated cumulative doses to the most exposed members of the public are expected to be small fractions of the CNSC regulatory limits; therefore, no mitigation measures are warranted.

9.5.2 Workers on the PN Property

9.5.2.1 Dose Levels

<u>PWMF II Workers (NEWs)</u>
In conclusion, cumulative radiation doses to PWMF II workers will be carefully controlled and monitored to ensure that OPG's ECL (< 10 mSv/y), which is well below regulatory dose limits, will not be exceeded.

<u>PN Workers (NEWs and Non-NEWs)</u>
In conclusion, cumulative radiation doses to PN workers will be carefully controlled and monitored to ensure that OPG's ECLs (< 1000 µSv/y to non-NEWs, and < 10 mSv/y to NEWs), which are below regulatory dose limits, will not be exceeded

9.5.2.2 Identified Mitigation Measures

The estimated cumulative dose to NEWs at the PWMF II and NEWs and non-NEWs at the PN, are expected to be less than CNSC regulatory limits; therefore, no mitigation measures are warranted or required.

9.5.3 Cumulative Dose to Non-Human Biota

The estimated cumulative dose to non-human biota is a small fraction (i.e., 5%) of the no-effects level (1 mGy/d) reported by UNSCEAR (1996) and is less than the corresponding values recommended by CNSC staff in a paper presented at the 2002 Conference on Ecological Risk Assessment in Australia (Bird et al. 2002)

9.5.3.2 Identified Mitigation Measures

The estimated cumulative dose to non-human biota is expected to be less than the no-effects levels reported by UNCSEAR; therefore, no mitigation measures are warranted or required.

SOURCE: Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment. December 2007. NK30-REP-07701-00002

5.9.2.3 Evaluation of Effects for Continued Operation

The predicted gamma radiation levels from full Storage Buildings provides a dose rate of ≤ 10 µSv/y at the PN property boundary, based on full occupancy 100% of the year. This includes both direct and skyshine contributions. The effect of gamma radiation on the terrestrial environment at the PN property boundary from the Storage Buildings is expected to be ≤ 10µSv/y. This effect will be indistinguishable from the temporal and spatial variations in natural background radiation levels at this location.

A dose rate of 50 µSv/h was predicted on the roof of the DSC Storage Buildings from an array of loaded DSCs completely filling the buildings. Nesting of birds on the roofs of storage buildings at PWMF I and PWMF II is discouraged by the very nature of the roof design. However, birds may perch on the roof of the Storage Buildings for brief periods, and be exposed to (absorbed) dose rates of approximately 0.05 mGy/h from gamma radiation. Exposure periods of one or two hours per day would result in dose rates of up to 0.1 mGy/d. This dose rate is less than the no effects level of 1 mGy/d reported by UNSCEAR (1996).

The gamma radiation levels from loaded DSCs in PWMF II Storage Buildings are predicted to produce a dose rate less than 0.5 µSv/h at the perimeter fence of the PWMF II site. Therefore, the corresponding absorbed dose rates to flora and fauna were estimated at 0.0005 mGy/h. The estimated daily dose rate to flora and fauna at the perimeter of the PWMF II site is approximately 0.012 mGy/d, and is a small fraction of the no-effects level of 1 mGy/d reported by UNSCEAR (1996). Also, the predicted dose rate is expected to be within the range of natural background, which is 0.004 to 0.02 mGy/d.

5.9.2.4 Identified Mitigation Measures

Similar to the Refurbishment Phase, the storage of the refurbishment waste is expected to have locally elevated gamma radiation levels which are predicted to be less than 0.5 µSv/h. This dose rate was established by OPG to ensure that even for 2000 h/y occupancy, the dose to a human would not exceed 1 mSv. In addition, however, a dose rate of 0.5 µSv/h is far below any relevant dose-rate criteria for non-human biota. Moreover, these levels are within the range of levels previously experienced at the PN site. Therefore, with the access to these storage areas closely controlled, there is no additional mitigation needed.

5.9.5.3 Evaluation of Effects for Continued Operation

The annual doses to individual NEWs during normal operation are well below the regulatory limits, a maximum of 50 mSv in a one-year dosimetry period and an average of 20 mSv in a one year dosimetry period (i.e., a cumulative dose of 100 mSv in five one-year dosimetry periods). In addition, doses will be controlled to ALARA using internal dose control limits, such as the ADL and ECL.

Doses to NEWs due to continued operation of the waste management facility will be the same as encountered presently at the PWMF (i.e., an average individual dose of approximately 0.64 mSv per year per worker). After completing the placement of the refurbishment waste into storage, there will only be maintenance and caretaking activities inside the storage buildings, and thus, future doses to workers at PWMF are expected to be comparable to existing doses.

5.9.5.4 Identified Mitigation Measures

Radiation doses to NEWs in the Regional and Local Study Areas from the Continued Operation of PNGS B following refurbishment are expected to be indistinguishable from the baseline doses from the PNGS in the Regional and Local Study Areas. Furthermore, the Continued Operation of PNGS B following refurbishment is expected to result in radiation doses to NEWs in the Site Study Area that are well below the corresponding regulatory limits, and within OPG dose targets and ECLs.

As no distinguishable changes in dose levels from baseline conditions are expected during refurbishment or continued operation, additional mitigation measures are not required.

5.9.6.3 Evaluation of Effects for Continued Operation

As mentioned previously, the access and movement of visitors and non-NEW workers on the PN site is controlled by OPG, and the radiation doses to these individuals from licensed activities on the PNGS site are controlled by OPG to ensure that they do not exceed 1 mSv/y, the regulatory limit on annual dose to non-NEWs (Canada Gazette 2000). At the perimeter fence of the PWMF II site, the dose rate is predicted to be less than 0.5 µSv/h which corresponds to a dose rate of < 1,000 µSv/y for 2,000 h/y occupancy, the CNSC public dose limit for non-NEWs (Canada Gazette 2000). It is highly unlikely that a non-NEW would spend appreciable time in this area and thus, the doses to non-NEWs are expected to be well below the CNSC public dose limit. Therefore, the radiation doses to non-NEWs from the continued operation are expected to be indistinguishable from the radiation doses from normal operation of the reactors and well below the regulatory limit of 1 mSv/y for non-NEWs.

5.9.6.4 Identified Mitigation Measures

Radiation doses to members of the public in the Regional and Local Study Areas from the continued operation of PNGS B following the refurbishment are expected to be indistinguishable from the baseline doses from the PNGS in the Regional and Local Study Areas. Furthermore, the continued operation following refurbishment is expected to result in radiation doses to visitors and non-NEW workers on the PN site (i.e., in the Site Study Area) that are less than the corresponding regulatory limit for members of the public of 1 mSv/y (Canada Gazette 2000). As no distinguishable changes in dose levels from baseline conditions are expected during refurbishment or continued operation, additional mitigation measures are not required.

Section A6

Description from Record of Proceedings and Record of Decision:

SOURCE: Record of Proceedings, Including Reasons for Decision. May 28, 2004.
Subject: Environmental Assessment Screening Report on the proposed expansion of the Pickering Waste Management Facility (Phase II)

4. Conclusion

The Commission concludes that the environmental assessment Screening Report attached to CMD 04-H7 (as amended) is complete and meets all of the applicable requirements of the Canadian Environmental Assessment Act.

The Commission concludes that the project, taking into account the appropriate mitigation measures identified in the Screening Report, is not likely to cause significant adverse environmental effects.

SOURCE: Record of Proceedings, Including Reasons for Decision. December 10, 2008.
Subject: Screening Environmental Assessment of the Pickering Nuclear Generating Station B Refurbishment and Continued Operations Project, Pickering, Ontario

17. The Commission reviewed the EA Screening Report and concluded that it is complete and in accordance with the requirements of the CEAA.

57. Based on its review of the Screening Report and the above-noted information provided on the record, the Commission concludes that the proposed project, taking into account the mitigation measures, described in section 8 of the EA Screening Report, is not likely to cause significant adverse effects to the environment.

107. To address concerns raised by several intervenors on waste management, the Commission requested that OPG elaborates on the design of the dry-storage container used for used fuel storage and on the fuel cycle after the removal of fuel from the reactor. OPG responded that the dry-storage container was a very robust container consisting of a 13mm-thick steel inner liner and a 13mm-thick steel outer liner with approximately half a metre of high-density reinforced concrete between those two liners. OPG added that the containers, without fuel, weigh approximately 70 tonnes and that they were extremely robust and very similar to those used elsewhere in North America and around the world. OPG noted that they had proven to be adequate for storing spent nuclear fuel for extended periods of time as long as fifty years. To answer the fuel cycle portion of the question, OPG added that the fuel removed from the reactor is stored in water pools at the stations for a minimum of 10 years to allow the fuel to cool to about 0.1 % of the radioactivity levels present at the time of its removal from the reactor. The fuel is then transferred to dry-storage containers for storage until a disposal facility is available.

SOURCE: Record of Decision. April 13, 2017.
Subject: Application to Renew the Waste Facility Operating Licence for the Pickering Waste Management Facility

110. Based on the information considered for this hearing, the Commission is satisfied that the ALARA concept is adequately applied to all PWMF activities.

113. CNSC staff informed the Commission that, in keeping with the ALARA principle, OPG had planned improvements to its radiation protection program during the proposed renewed licence period and CNSC staff would be closely monitoring these initiatives.
115. Based on the information provided for this hearing, the Commission is satisfied that doses to workers at the PWMF are adequately controlled.

121. Based on the information provided on the record for this hearing, the Commission concludes that, given the mitigation measures and safety programs that are in place and will be in place to control radiation hazards, OPG provides, and will continue to provide, adequate protection to the health and safety of persons and the environment throughout the proposed renewed licence period.

122. The Commission is satisfied that OPG's radiation protection program at the PWMF meets the requirements of the Radiation Protection Regulations.

131. The Commission concludes that the health and safety of workers and the public was adequately protected during the operation of the facility for the current licence period and that the health and safety of persons would also be adequately protected during the continued operation of the facility in the proposed renewed licence period.

157. Based on the information submitted by CNSC staff in the EA Report, the Commission is satisfied that the EA adequately shows that OPG made and will continue to make adequate provision for the protection of the environment and persons at the PWMF site.

158. The Commission is satisfied that OPG's and the CNSC's environmental monitoring show that the public and the environment around the PWMF site remain protected.

166. Based on the information presented on the record for this hearing, the Commission is satisfied that the ERAs were carried out satisfactorily and showed that OPG was adequately protecting the environment in the vicinity of the Pickering NGS, and therefore, the PWMF site.

168. Based on the assessment of the application and the information provided on the record at the hearing, the Commission is satisfied that, given the mitigation measures and safety programs that are in place to control hazards, OPG will provide adequate protection to the health and safety of persons and the environment throughout the proposed licence period.

218. Based on the information presented on the record for this hearing, the Commission is satisfied that OPG is meeting, and will continue to meet, regulatory requirements regarding packaging and transport.

ENCLOSURE #2

OPG report
“Safety Assessment Storing Lower Aged Fuel in PWMF SB3”
92896-REP-01320-00012

ONTARIO POWER GENERATION

Report

OPG Proprietary	
Document Number: 92896-REP-01320-00012	Usage Classification: N/A
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Title:
SAFETY ASSESSMENT STORING LOWER AGED FUEL IN PWMF SB3

Safety Assessment Storing Lower Aged Fuel in PWMF SB3

92896-REP-01320-00012 R000
2020-06-30

Order Number: N/A
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LICENSED PROFESSIONAL ENGINEER
K. T. TSANG
PROVINCE OF ONTARIO

Prepared By:
Eva Bartos
Digitally signed by Eva Bartos
DN: cn=Eva Bartos, o=Candu Energy Inc., ou=Nuclear, email=eva.bartos@snclavalin.com, c=CA
Date: 2020.06.30 11:56:03 -04'00'
Eva Bartos Date
Senior PSA Analyst
Candu Energy Inc.

Prepared By:
Digitally signed by Stephen Smith
Date: 2020.06.30 11:51:57 -04'00'
Stephen Smith Date
Health and Radiation Physicist
Candu Energy Inc.

Reviewed By:
Digitally signed by Ralph Bettig
Date: 2020.06.30 11:59:55 -04'00'
Ralph Bettig Date
Senior PSA Analyst
Candu Energy Inc.

Reviewed By:
Digitally signed by Anas Khaial
Date: 2020.06.30 12:06:05 -04'00'
Anas Khaial Date
Senior Reactor Physicist
Candu Energy Inc.

Reviewed By:
Ricky Khaloo
Digitally signed by Ricky Khaloo
Date: 2020.06.30 12:21:54 -04'00'
Ricky Khaloo Date
Specialist Health Physicist
Candu Energy Inc.

Reviewed and Verified By:
Digitally signed by Kwok Tsang
Date: 2020.06.30 12:37:01 -04'00'
Kwok Tsang Date
Specialist Radiation Physicist
Candu Energy Inc.

Reviewed and Verified By:
Silvia Aprodu
Digitally signed by Silvia Aprodu
Date: 2020.06.30 13:20:54 -04'00'
Sylvia Aprodu
Senior PSA Specialist
Candu Energy Inc.

Approved By:
Paul Santamaura
Digitally signed by Paul Santamaura
Date: 2020.06.30 13:15:40 -04'00'
Paul Santamaura Date
Manager, PSA
Candu Energy Inc.

Approved By:
Yahui Zhuang
Digitally signed by Yahui Zhuang
Date: 2020.06.30 14:13:31 -04'00'
Yahui Zhuang Date
Manager, Radiation Physics & RadWaste
Candu Energy Inc.

Accepted By:
Digitally signed by Paul Crowley
Date: 2020.06.30 15:25:48 -04'00'
Paul Crowley Date
Senior Technical Officer
Fuel and Nuclear Waste Safety Assessment
Ontario Power Generation

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Revision Summary

Revision Number	Date	Comments
R000	2020-06-30	Initial issue.

1.0 INTRODUCTION

The Pickering Waste Management Facility (PWMF) Used Fuel Dry Storage (UFDS) area consists of a processing workshop for preparing the Dry Storage Containers (DSCs) loaded with used fuel bundles for storage and three storage buildings (SB1 and SB2 at the Phase I site location and SB3 at the Phase II site location) for storing the DSCs. SB4 is currently being constructed at the Phase II site.

The existing operation of the PWMF involves the processing and storage of DSCs containing used fuel with a minimum of ten (10) years of decay. In order to support PWMF operations, an analysis to determine the impacts of loading used fuel cooled for less than ten (10) years is being completed. These lower-fuel-age DSCs are planned to be stored in SB3.

The existing PWMF Safety Report [1] has considered the storage of DSCs containing used fuel with a minimum of ten (10) years decay for normal operations and malfunctions/accidents. As such, a safety assessment is required to determine if storing fuel that has been out of the core for a period of less than ten (10) years is acceptable from a nuclear safety point of view.

The objective of the current safety assessment is to incorporate the transfer, handling, and storage of DSCs containing used fuel that has only cooled for six (6) years[1] during normal operations and for malfunctions/accident conditions based on the SB4 safety assessment [2]. In addition to the existing hazards identified in Reference [2], potential hazards associated with the transfer of the lower-fuel-age DSCs from the station for processing, re-arrangement and/or removal of a number of the existing DSCs in order to place the lower-fuel-age containers to their storage location in SB3 are included in the assessment.

2.0 SCOPE

This report documents the safety assessment for the processing, transfer, handling, and storage in SB3 of up to 100 DSCs[2] containing 6 year decayed used fuel as well as the re-arrangement and removal of a number of the existing DSCs[3] to accommodate the incoming lower-fuel-age DSCs. The safety assessment for the lower-fuel-age DSCs as well as any impact on the whole-site events, such as earthquakes, floods and tornadoes are included herein.

In addition, a qualitative discussion is provided in Section 5.5 of this report on the fuel sheath temperature for 6 year decayed used fuel.

3.0 QUALITY ASSURANCE

The Project activities will be performed by Candu Energy Inc. in accordance with the Quality Assurance (QA) program described in 147-912020-QAP-001 “CANDU Services Projects (CSA Z299 Series)”, CE-912020-QAM-002 “Candu Energy Inc. – Quality Assurance Manual” and

1 A cooling period of six (6) years represents the conservative limit for the fuel age to be stored in the DSCs.
2 100 DSCs represents the conservative limit for the number of DSCs to be replaced in SB3.
3 The loading pattern of DSCs in SB3 is proposed to ensure the DSCs containing 6 year decayed used fuel are surrounded by DSCs containing used fuel decayed for longer periods with the intention of minimizing dose rates external to the building. In addition, because DSCs are being transferred out of SB3 to SB4 (e.g. to make room in SB3 to allow younger fuel to be stored), older DSCs will be selected for the transfer into SB4 due to their lower dose rates.

CE-912020-QAM-003 “Quality Assurance Manual – Analytical, Scientific and Design Computer Programs” to satisfy the QA requirements of the following standards applicable to the Project scope of work:

- CSA CAN3-Z299.1-85 “Quality Assurance Program Category 1”;
- CSA N286-12 “Management System Requirements for Nuclear Facilities”; and
- CSA N286.7-16 “Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants”.

4.0 DSC STORAGE DESCRIPTION

The PWMF site has undergone an orderly development in phases to facilitate the growing number of DSCs over the years. These phases are:

- Phase I: The PWMF Phase I site is located within the Pickering Nuclear Generating Station (PNGS) protected area, southeast of PNGS Unit 8, adjacent to the east side of the station security fence. The Phase I site consists of a DSC Processing Building (PB), SB1 and SB2, and the Retube Components Storage (RCS) area.
- Phase II: The PWMF Phase II site is located approximately 500 m northeast of the PWMF Phase I site, east of the PNGS powerhouse, within its own protected area in the Pickering Nuclear site. The Phase II site consists of SB3 with provision for future DSC SB 4.

The existing facilities within the PWMF, including the RCS area, a PB, and the three DSC storage buildings (SB1, SB2, and SB3), are shown in Figure 4-1 and Figure 4-2. The future SB4 is being constructed to the south of SB3 and is shown in Figure 4-3.

The DSC preparation process is shown in Figure 4-4 and the DSC storage arrangement is shown in Figure 4-5, similar to the storage arrangement at the existing storage buildings.

A general description of the used fuel stored in DSCs is provided in Table 4-1.

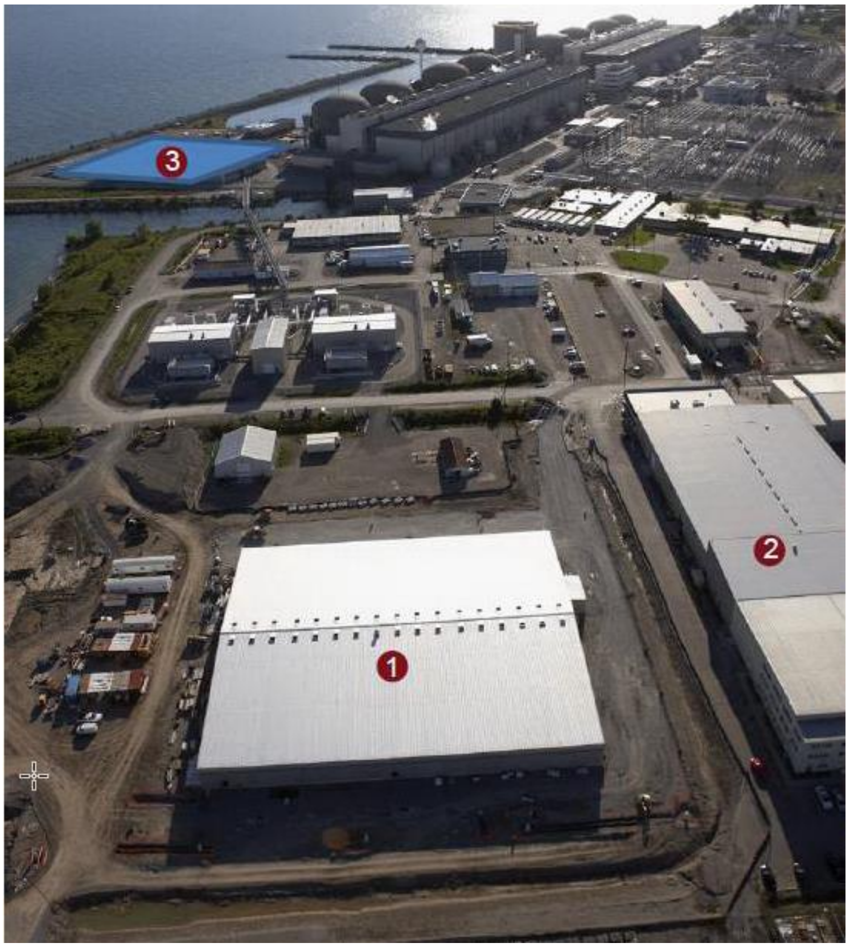
Table 4-1: Summary of Used Fuel Storage in DSCs at PWMF

Parameter	Unit	Value
Fuel bundles per DSC	-	384
Number fuel elements per bundle	-	28
Bundle length	mm	495
Mass of UO_2	kg	22.87
Mass of Zircaloy	kg	1.67
Mass of U	kg	20.16
Mass of bundle	kg	24.54
Bundle fission power	kW	373
Burnup	MWh/kgU	230
Fuel decay age	year	≥6

❶ Pickering A Re-tube Components ❷ DSC Processing ❸ DSC Storage Building 1
❹ DSC Storage Building 2 ❺ Pickering B Station

Figure 4-1: Aerial View of PWMF Phase I

From Figure 2.1 of Reference [1]



❶ DSC Storage Building 3
❷ East Warehouse Complex
❸ PWMF – Phase 1

Figure 4-2: Aerial View of PWMF Phase II

From Figure 2.2 of Reference [1]

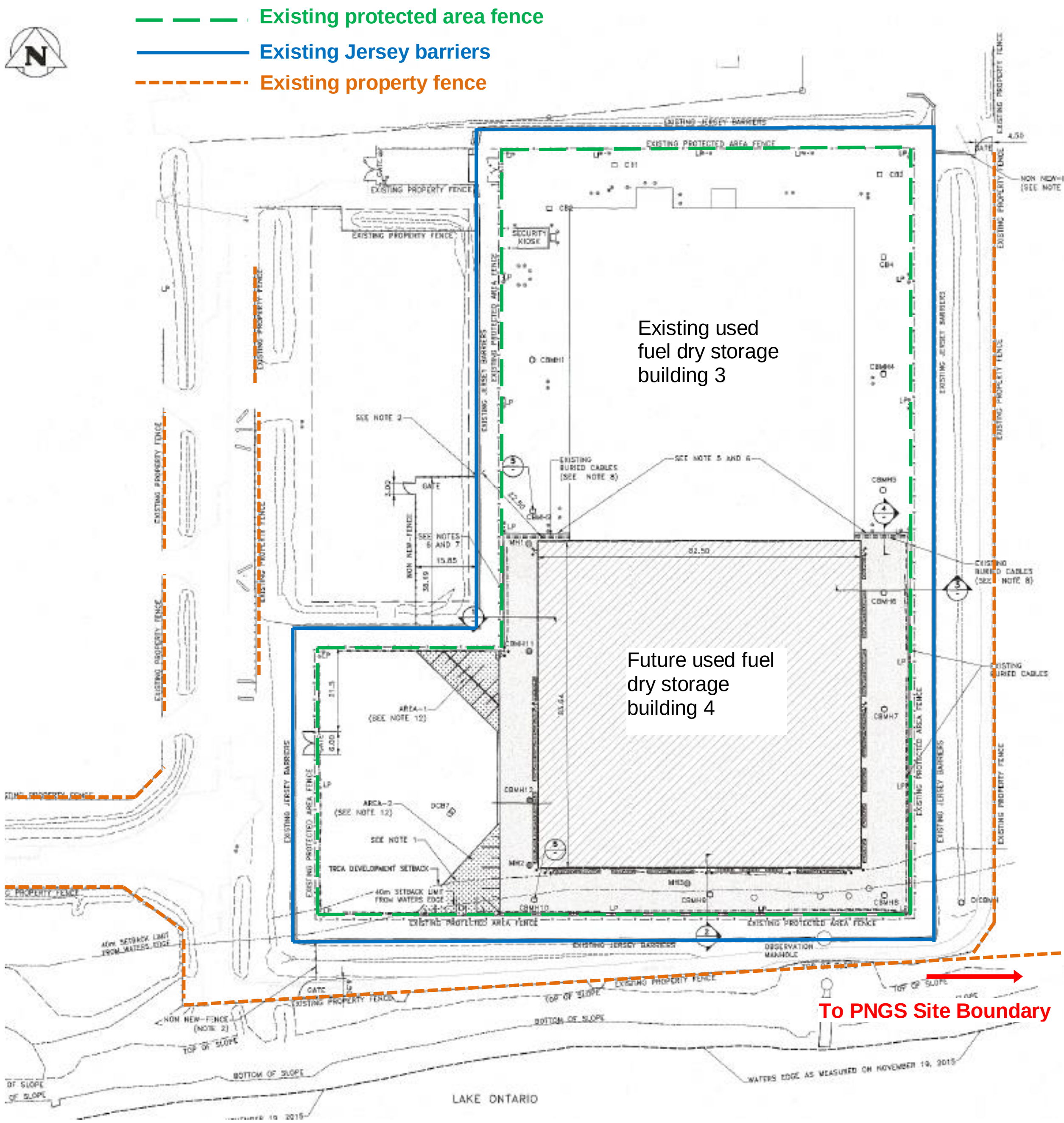


Figure 4-3: General Layout of SB3 and SB4

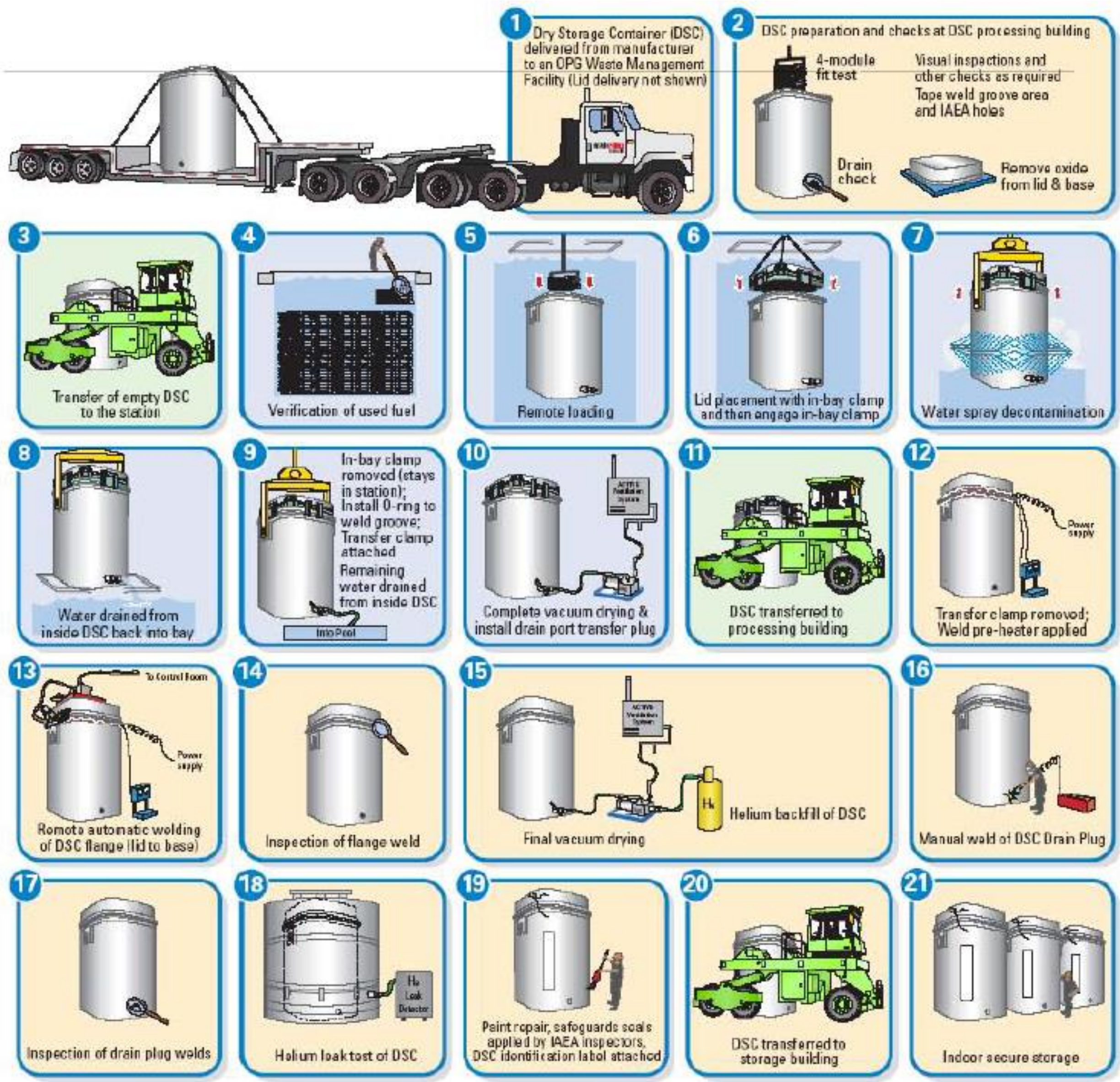


Figure 4-4: Used Fuel Dry Storage Process

From Figure 3-1 of Reference [3]

Figure 4-5: Typical DSC Storage at PWMF

From Figure 2.3 of Reference [1]

5.0 SAFETY ASSESSMENT OF DSC STORAGE IN SB3

5.1 Safety Assessment Approach

Under normal operating conditions, storage containers, buildings, and structures at the PWMF are expected to provide reasonable assurance that the radioactive waste can be stored and retrieved without undue radiological risk to workers, members of the general public, or the environment. 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.

Guidance documented in the CSA N292.0-14 [4], CSA N288.1-14 [5], and CSA N288.2-14 [6] standards was used in performing the safety assessment.

5.2 Acceptance Criteria

The radiation safety requirements for the PWMF under normal operation[4] are the following:

- <10 µSv per year for a member of the general public at or beyond the PNGS site boundary. This dose rate target is one percent (1%) of the CNSC regulatory dose limit [7] of 1 mSv per year for a member of the public [1].
- <0.5 µSv/h at the fence (boundary of the PWMF licensed facility), based on the 1 mSv/a effective dose limit for non-Nuclear Energy Workers (non-NEWs) and a maximum occupancy of 2000 hours per year [1].
- The effective dose limit for NEWs is 50 mSv/a in a one-year dosimetry period and 100 mSv in a five-year dosimetry period [1].

The radiation safety requirements following malfunctions or accident conditions are given below[5]:

- The dose acceptance criterion for members of the public at or beyond the site boundary for a period of 30 days[6] [6] after the analyzed event shall be less than 1 mSv [1].
- The dose acceptance criterion for NEWs following malfunctions or credible accident conditions shall be less than 50 mSv [1].

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4 As per Section 4.2 of Reference [1], these requirements are for the operation of the PWMF only and are exclusive of the dose from the PNGS. Additional discussion on the radiation safety requirements and dose rate targets is provided in Reference [1].

5 The 1 mSv public dose acceptance criterion follows the prescribed limit to the general public given in the Radiation Protection Regulations [7].
The worker dose acceptance criterion follows the maximum annual dose to NEW given the Radiation Protection Regulations [7].

6 The 30-day period of exposure follows the recommendation given in the CSA N288.2-14 Section 7.8.2.2 “For demonstration of compliance with regulatory limits, a period of residence of 30 days over contaminated ground shall be used”. This is a departure from the approach used in the existing PWMF Safety Report [1] in which considers the duration of the plume release as the exposure duration.

5.2.1 ALARA – Worker Dose Management Target

The ALARA dose target for an individual PWMF worker is 3 mSv per year [1].

5.3 Normal Operating Conditions

5.3.1 Radioactive Emissions

Chronic releases of radionuclides from the normal operation of the PWMF are measured and reported in PWMF quarterly reports[7]. Reported data include survey results from the retube component storage facility, used fuel dry storage facility stack samples, and active liquid waste tank samples. Almost all of the weekly measured values from the used fuel dry storage facility stack sampler particulate sample were below the minimum detectable activity of 3.3×10^3 Bq[8]. Following the approach outlined in the PWMF Safety Assessment Update for the purpose of estimating the normal operation releases, weekly measurement values that were below the minimum detectable activity are set to 3.3×10^3 Bq and the particulate release is represented by cobalt-60 release [3]. The annual particulate releases are listed in Table 5-1.

Detailed discussion on sources of radioactive emissions is provided in the Reference [3]. The storage of 100 DSCs containing 6 year decayed used fuel in SB3 is not expected to introduce detectable radioactive emissions during normal operating conditions of the PWMF because DSCs being transferred and stored will not contain fuel known to be damaged. While no significant releases are expected from DSCs under normal operating conditions, small quantities of fixed surface contamination may become airborne during welding operations [1].

Table 5-1: Annual Particulate Releases from PWMF Stack Sampler

Year	Bq/a release
2007	1.72×10^5
2008	1.92×10^5
2009	2.00×10^5
2010	1.85×10^5
2011	1.72×10^5
2012	1.72×10^5
2013	1.72×10^5
2014	1.79×10^5
2015	1.85×10^5
2016	1.72×10^5
2017	1.72×10^5
2018	1.72×10^5
2019	1.75×10^5

7 Quarterly reports are available from the 92896-REP-00531-* series of reports.
8 The minimum detectable activity is provided in the PWMF quarterly reports.

5.3.1.1 Postulated Chronic Release from DSC Processing

The 100 DSCs containing 6 year decayed used fuel to be stored in SB3 follow the same preparation process as existing DSCs. However, the postulated chronic release from DSC processing in the latest safety assessment update [3] is based on DSCs containing 10 year decayed used fuel. The radioactive inventory available for release from the used fuel within the DSC is dependent on both the burnup and the decay age of the used fuel. As no emissions for noble gases, carbon-14, or tritium were available, following the methodology presented in Reference [3] the postulated chronic releases from processing of DSCs containing 6 year decayed used fuel (of 230 MWh/kgU burnup) are as follows:

Radionuclide	Annual Release from DSCs Containing 6 Year Decayed Used Fuel (Bq/a)	Annual Release from DSCs Containing 10 Year Decayed Used Fuel [3] (Bq/a)
Kr-85	3.09×10^{11}	2.38×10^{11}
Tritium (HTO)	5.00×10^{10}	4.32×10^{10}
C-14	4.10×10^{6}	4.91×10^{6}

Notes:
- The Bq/a release values for 6 year decayed used fuel were derived from the radionuclide inventory of a 6 year decayed used fuel bundle (bundle radionuclide inventory presented in Appendix A).
- The Bq/a release values for 10 year decayed used fuel were calculated as part of work document in Reference [3] based on a bundle-wise calculation. The fuel bundle radionuclide inventory has since been revised using a ring-wise calculation, resulting in a decrease of C-14 (see the discussion provided in Appendix A).
- The Bq/a release values were calculated based on releases from 280 failed fuel elements (i.e. 4 failed fuel elements per DSC and 70 DSCs being processed each year).
- For H-3 and Kr-85, the release fraction for the failed fuel element is 0.0218, which is f_{gap} + 10% f_{gb}:
  - f_{gap} = fraction of gap inventory = 0.0095
  - f_{gb} = fraction of grain boundary inventory = 0.123
- For C-14, the amount released to the gap and grain boundary is set to 0.1% [1].
- The Bq/a release values were calculated based on releases from 280 failed fuel elements (i.e. 4 failed fuel elements per DSC and 70 DSCs being processed each year).

The postulated chronic releases from processing of DSCs containing 10 year decayed used fuel (of 230 Mwh/kgU burnup) [3] are provided for comparison.

5.3.1.2 Postulated Chronic Release from DSMs

Dry Storage Modules (DSMs) at the PWMF are used to store legacy retube waste, including pressure tubes, end fittings, and shield plugs. Used fuel bundles are not stored in DSMs. Therefore, the releases from DSMs are expected to be the same as those documented in Reference [3]. For the purpose of evaluating the potential emissions from DSMs, the chronic release of carbon-14 is taken to be $1.6\text{x}10^{10}$ Bq per year [3].

5.3.2 External Gamma Dose Rates

5.3.2.1 Dose Rates from Single DSC

Calculated dose rates using Monte Carlo transport methodology documented in References [8], [9], and [10] have been demonstrated to be conservative compared with actual DSC dose rates measured during UFDS storage operations. For DSCs loaded with reference used fuel bundles (decayed by 10 years or older), the measured contact dose rates to date are in the range of 9 to 13 μSv/h [1]. This is about a factor of 4 conservative compared with the calculated estimate of the near contact (at the DSC long side) dose rate of 37.9±0.2 μSv/h for reference used fuel (decayed by 10 years). At 1 m distance, measured dose rates are about 5 to 7 μSv/h [1], compared with calculated dose rate estimates of 20.0±0.1 μSv/h.

The estimated dose rates from a DSC containing 6 years decayed used fuel are 97.4±2.6 μSv/h at near contact and 51.2±0.9 μSv/h at 1 m [9]. These dose rates are a factor of approximately 2.6 times larger than the dose rates calculated for used fuel decayed 10 years. Therefore, it is expected that the measured dose rates for DSCs containing 6 year decayed used fuel will increase by a similar factor (2.6x)[9] compared to the dose rates measured from the DSCs containing the reference used fuel[10]. The impact of the expected increase in measured dose rates on the dose to workers is discussed in Section 5.3.4.

The calculated dose rates from a DSC, fully loaded with Pickering 6 year decayed used fuel bundles (230 MWh/kgU burnup) are listed in Table 5-2 and Figure 5-1. The dose rates as a function of distance from the DSC and cooling time are tabulated in Table 5-3.

9 The majority of this increase is driven by the greater amount of Rh-106 and Pr-144 in the 6 year decayed used fuel [9].

10 This is generally true if the energy spectra are the same. A check was performed which indicates that there is no significant difference in the average energy per energy group in the binning used in the dose rate calculation [9].

Table 5-2: Calculated Dose Rate (µSv/h) vs. Distance from a Single DSC Containing 6 Year Decayed Used Fuel [9]

Distance From Surface (cm)	Dose Rate (µSv/h)		
	Long Side	Short Side	Top
Near Contact[a]	97.4	86.2	70.5
50	77.4	62.7	49.8
100	51.2	42.9	n/a[b]
150	37.7	29.1	17.0
200	27.7	20.7	10.8
250	23.7	15.5	10.5
300	17.4	12.5	6.4
350	12.8	10.5	5.4
400	10.4	9.0	4.1
450	8.9	6.7	3.2
500	7.2	6.0	2.1

Notes:

a) Contact dose rates were calculated at a distance of 5 cm from the DSC surface.

b) The dose rate at 100 cm from the top of the DSC surface is excluded as the associated statistical uncertainty is larger than the 10% target presented in Section 3.10 of Reference [9].

Table 5-3: Calculated Dose Rate (μSv/h) vs. Distance from a Single DSC (Long Side) at Various Decay Times [9]

	Used Fuel Bundle Decay Time (year)								
Distance (cm)	**6 y**	**10 y**	**15 y**	**20 y**	**25 y**	**30 y**	**35 y**	**40 y**	**Maximum 1σ Uncertainty[b]**
Near Contact[a]	97.4	37.9	26.6	21.0	17.1	14.2	11.9	10.1	2.7%
50	77.4	29.0	20.3	16.0	13.0	10.8	9.1	7.7	2.7%
100	51.2	20.0	14.1	11.1	9.0	7.5	6.3	5.4	1.7%
150	37.7	14.4	10.1	7.9	6.4	5.3	4.5	3.8	2.6%
200	27.7	10.6	7.4	5.8	4.7	3.9	3.3	2.8	3.4%
250	23.7	8.3	5.7	4.5	3.7	3.0	2.6	2.2	9.7%
300	17.4	6.4	4.5	3.5	2.8	2.4	2.0	1.7	5.5%
350	12.8	5.1	3.6	2.8	2.3	1.9	1.6	1.4	2.8%
400	10.4	4.2	2.9	2.3	1.9	1.6	1.3	1.1	2.6%
450	8.9	3.5	2.4	1.9	1.6	1.3	1.1	0.9	4.8%
500	7.2	2.9	2.0	1.6	1.3	1.1	0.9	0.8	3.4%

Notes:

a) Contact dose rates were calculated at a distance of 5 cm from the DSC surface.

b) The uncertainty listed is the maximum uncertainty in the dose rate across all decay times for a given distance.

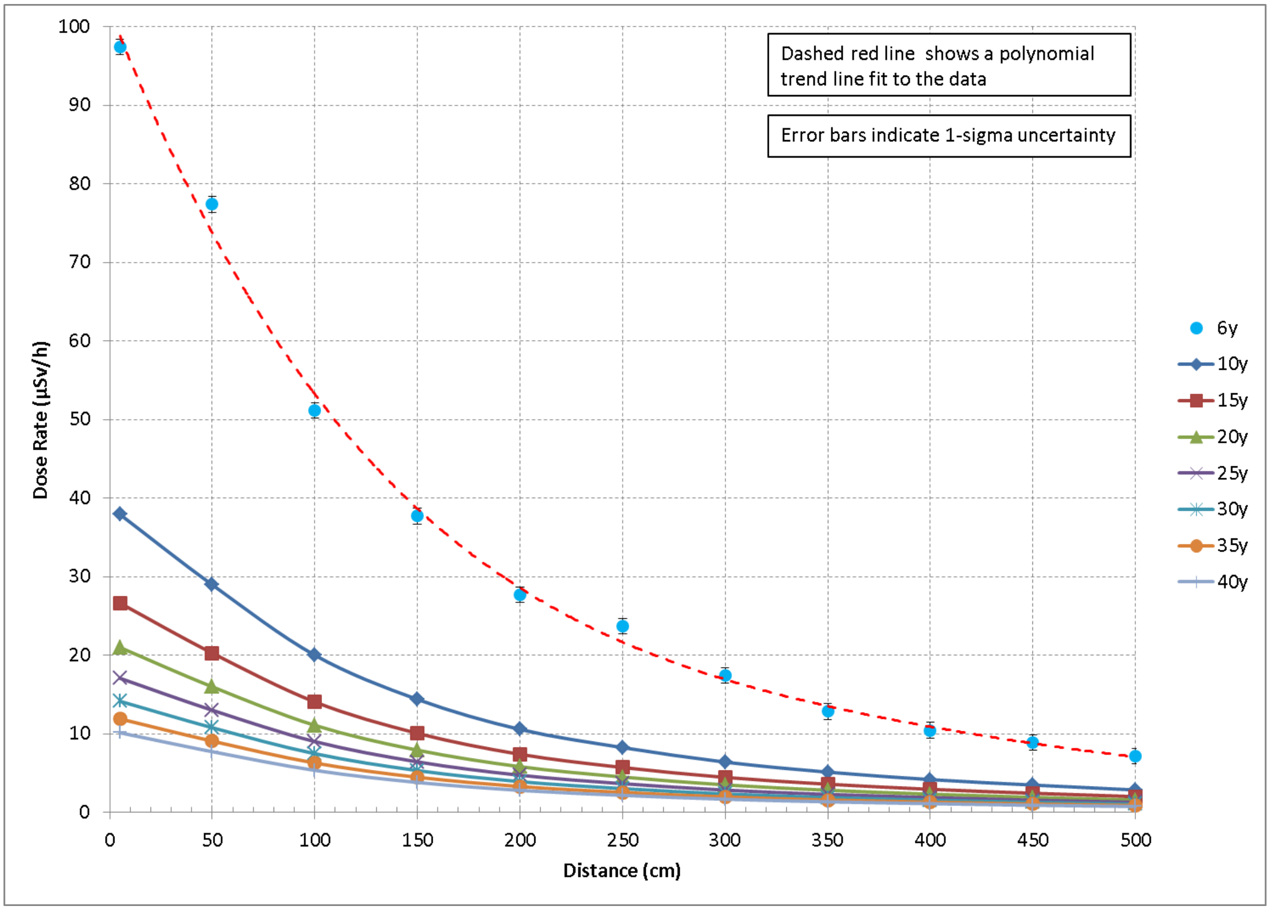


Figure 5-1: Calculated Dose Rate vs. Distance from the Long Side of a Single DSC at Various Fuel Decay Times [9][11]

[11] Note, the dose rate is estimated to drop below 0.5 µSv/h at approximately 7.5 m from the surface of the DSC containing 6 year decayed used fuel.

5.3.2.2 Dose Rates inside Storage Building 3

The occupational dose rate inside SB3 was estimated using the DSC layout shown in Figure 5-2 as part of the shielding assessment for DSCs containing 6 year decayed used fuel stored in SB3 [9]. The dose rate profile in the main aisle way of SB3 is shown in Figure 5-3. The maximum dose rate is estimated to be 13.6±0.5 µSv/h while the average dose rate in the main aisle way is estimated to be approximately 4.7±0.1 µSv/h.

These estimated dose rates across the SB3 main aisle way are significantly less than the maximum (33.9±1.6 µSv/h) and average (9.4±0.2 µSv/h) dose rates calculated for the SB4 main aisle way [10]. The difference in dose rates across the SB3 and SB4 aisles results from the different decay age used fuel stored in the two buildings. Compared to SB3, SB4 contains a larger portion of lower decay age used fuel (10 – 20 years) which contributes to a higher dose rate. Further, the loading pattern for SB4 has a larger amount of DSCs with low decay age fuel bordering the main aisle way which, as seen in Table 5-3, have larger dose rates than the 25 year decayed used fuel stored along the main aisle way in SB3

The proposed loading pattern of SB3 (see Figure 5-2) has the aisle way lined with DSCs containing 25 year decayed used fuel. When compared to the dose rates from a single DSC containing 6 year decayed used fuel, the calculated dose rates across the main aisle way are significantly lower. The DSCs containing the 25 year decayed used fuel lining the aisle way provide shielding from the DSCs with the lower fuel decay age.

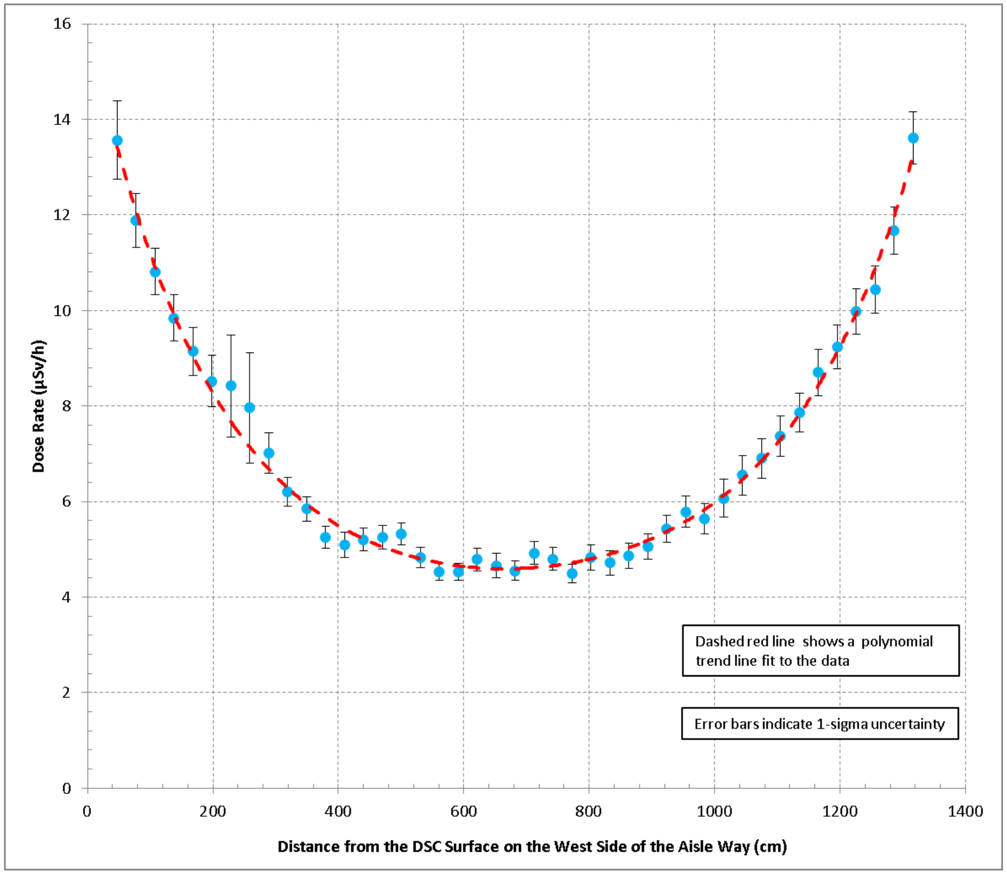
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	6	6	6	6	**25**	**25**	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25

25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25

Values shown in the figure correspond to the DSC decay time in years

Figure 5-2: DSC Layout in SB3[12]

[12] 19 DSCs to be moved back to SB3, after loading the DSCs containing 6 year decayed used fuel, into locations indicated in bold.



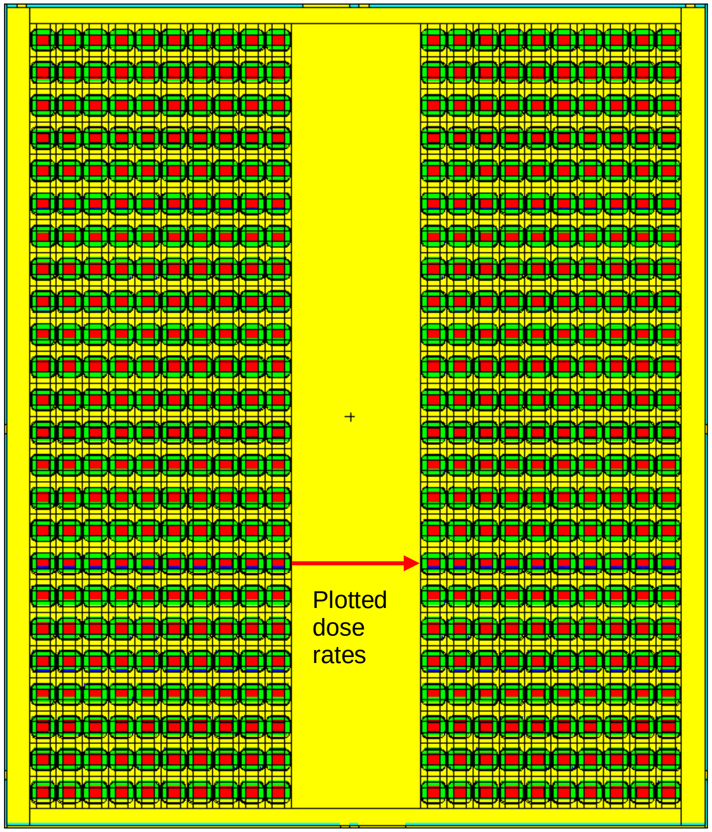


Figure 5-3: Dose Rates inside SB3

5.3.2.3 Dose Rates around PWMF

The limiting dose rates at the Pickering site boundary are calculated to be $(1.7\pm0.1)\times10^{-3}$ μSv/h (at the Montgomery Park Rd turnaround location) and $(1.6\pm0.1)\times10^{-3}$ μSv/h (lakeside exclusion zone boundary) [9]. The limiting dose rates are primarily from radiation sources from the existing SB3 and the DSCs to be stored in SB4. The contribution of radiation sources from DSCs and DSMs stored in the Phase I site to the direct external radiation field at the limiting locations around the Phase II site are negligible [11].

Dose rates at the existing protected area fence around SB3 and SB4 are expected to meet the 0.5 μSv/h dose rate criterion. However, calculations documented in Reference [9] suggest that dose rates at some sections of the fence could potentially be higher than the dose rate criterion (see Figure 5-4). It should be noted that the majority of the contribution to the dose rates at the fence locations that exceed 0.5 μSv/h are from DSCs stored in SB4, and not from the 6 year decayed used fuel DSCs stored in SB3. During the SB3 and SB4 operation, dose rates at the Phase II protected area fence will be measured and monitored and mitigating actions taken if required.

5.3.3 Public Dose

Contributors to the doses to members of the public during normal operation of the PWMF include the airborne radioactive emission and the external gamma dose rate from the radionuclides inside the DSCs and DSMs.

Dose to members of the public from emissions from normal operation of the PWMF have been determined based on the latest information on radionuclide emissions, representative group locations, and meteorological data [3]. The normalized annual doses resulting from the emission of radionuclides of interest are listed in Table 5-4 for releases from the Phase I site and Table 5-5 for releases from Phase II site. The dose receptor and critical group locations are shown in Figure 5-5. Locations included in the model are those identified by the Pickering site survey [12] and hypothetical locations along Pickering site exclusion zone boundary (see Figure 5-6). For landside hypothetical locations, receptors were assumed to be at their locations 100% of the time, i.e., 8760 hours per year. Receptors at the lakeside hypothetical locations were assumed to be present at that location 1000 hours per year.

Potential releases from normal operation of PWMF include chronic releases from DSCs during processing and chronic releases from existing DSMs. Both of these potential contributors to the normal operation releases are located in Phase I site. DSC storage in Phase II site is not expected to generate chronic releases during normal operation of the PWMF.

Hypothetical limiting public doses due to normal operations[13] of the PWMF calculated based on the radioactive emissions outlined in Section 5.3.1.1 are listed in Table 5-6. It is shown that hypothetical limiting public doses are significantly below the 1 mSv acceptance criterion outlined in Section 5.2. At the limiting landside and lakeside locations, the public doses are dominated by the external gamma radiation from DSCs described in Section 5.3.2.

[13] Considering 70 DSCs containing 6 year decay used fuel (of 230 MWh/kgU burnup) are processed in a year of operation of the PWMF and the releases outlined in Section 5.3.1.

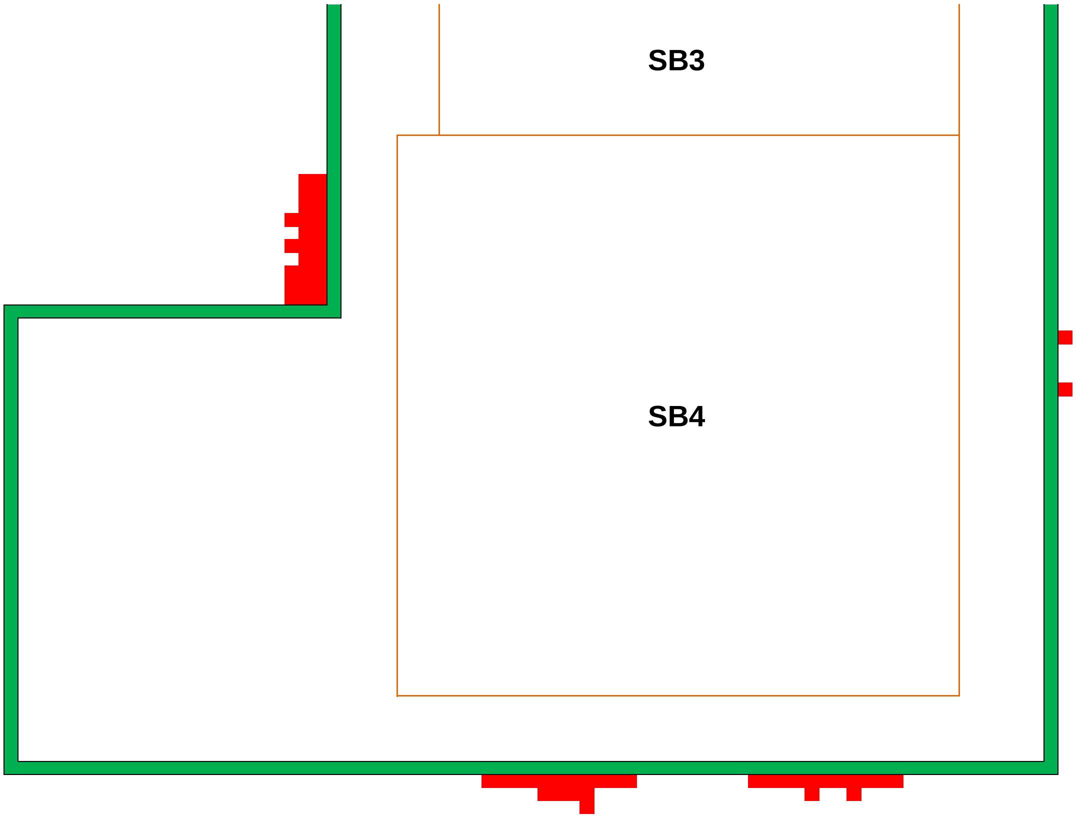


Figure 5-4: Representation of Dose Rates Exceeding Acceptance Criterion (red) Outside PWMF Phase II Protected Area Fence (green) [9][14]

14 Dose rates (best estimate + 2σ uncertainty) are compared against the 0.5 μSv/h acceptance criterion [9]. Each cell corresponds to a 2 m x 2 m x 2 m volume.

Table 5-4: Normalized Annual Doses from Chronic Releases at Phase I Site [3]

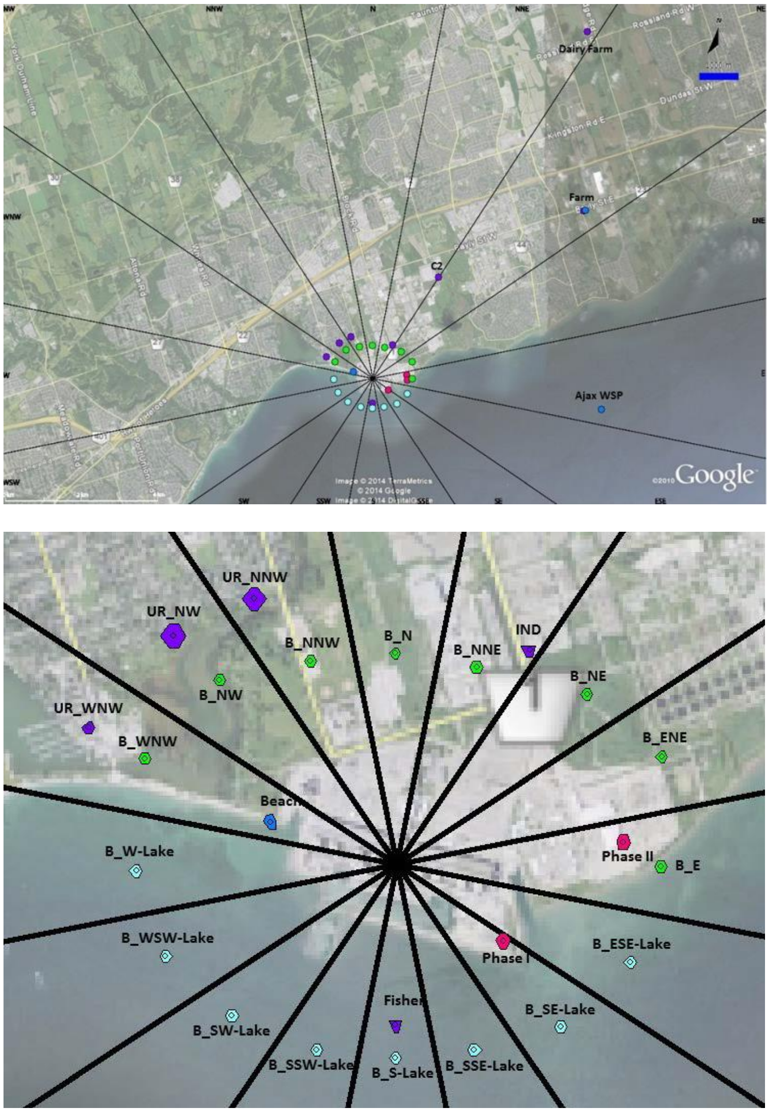
Age group	Location (see Figure 5-5)	Dose (µSv/a) per 1 Bq/s release rate			
		Kr-85	HTO	C-14	Co-60
Adult	B_E	1.50E-08	5.50E-07	2.20E-07	7.30E-02
	B_ENE	6.70E-09	2.50E-07	9.90E-08	4.20E-02
	B_NE	7.30E-09	2.70E-07	1.10E-07	5.10E-02
	B_NNE	1.90E-09	7.10E-08	2.80E-08	1.20E-02
	B_N	1.70E-09	6.50E-08	2.60E-08	1.20E-02
	B_NNW	1.90E-09	7.20E-08	2.90E-08	1.80E-02
	B_NW	2.10E-09	7.60E-08	3.00E-08	2.30E-02
	B_WNW	2.70E-09	1.00E-07	4.00E-08	3.20E-02
	B_W-Lake	3.90E-10	1.40E-08	5.70E-09	5.00E-06
	B_WSW-Lake	4.70E-10	1.70E-08	6.90E-09	6.00E-06
	B_SW-Lake	5.40E-10	2.00E-08	8.00E-09	6.90E-06
	B_SSW-Lake	8.10E-10	3.00E-08	1.20E-08	1.00E-05
	B_S-Lake	1.60E-09	6.10E-08	2.40E-08	2.10E-05
	B_SSE-Lake	6.80E-09	2.50E-07	1.00E-07	8.70E-05
	B_SE-Lake	7.80E-09	2.90E-07	1.10E-07	9.90E-05
	B_ESE-Lake	3.80E-09	1.40E-07	5.60E-08	4.90E-05
	Fisher	1.50E-10	5.70E-09	2.30E-09	2.00E-06
	C2	1.30E-09	4.80E-08	1.90E-08	9.40E-03
	IND	6.20E-10	2.30E-08	9.20E-09	7.90E-06
	UR_NNW	1.40E-09	5.90E-08	6.30E-07	1.30E-02
	UR_NW	1.60E-09	6.70E-08	9.10E-07	1.70E-02
	UR_NNW	2.10E-09	8.70E-08	1.10E-06	2.50E-02
	Dairy Farm NNE	2.10E-10	2.60E-08	2.80E-06	1.60E-03
	Farm NE	3.50E-10	4.00E-08	1.40E-06	2.30E-03
Child	B_E	1.50E-08	6.50E-07	3.10E-07	7.30E-02
	B_ENE	6.70E-09	3.00E-07	1.40E-07	4.20E-02
	B_NE	7.30E-09	3.20E-07	1.50E-07	5.10E-02
	B_NNE	1.90E-09	8.40E-08	4.10E-08	1.20E-02
	B_N	1.70E-09	7.70E-08	3.70E-08	1.20E-02
	B_NNW	1.90E-09	8.50E-08	4.10E-08	1.80E-02
	B_NW	2.10E-09	9.00E-08	4.30E-08	2.30E-02
	B_WNW	2.70E-09	1.20E-07	5.70E-08	3.20E-02
	B_W-Lake	3.90E-10	1.70E-08	8.20E-09	7.00E-06
	B_WSW-Lake	4.70E-10	2.10E-08	9.90E-09	8.40E-06
	B_SW-Lake	5.40E-10	2.40E-08	1.10E-08	9.80E-06
	B_SSW-Lake	8.10E-10	3.60E-08	1.70E-08	1.50E-05
	B_S-Lake	1.60E-09	7.20E-08	3.50E-08	3.00E-05
	B_SSE-Lake	6.80E-09	3.00E-07	1.40E-07	1.20E-04
	B_SE-Lake	7.80E-09	3.40E-07	1.60E-07	1.40E-04
	B_ESE-Lake	3.80E-09	1.70E-07	8.10E-08	6.90E-05
	Fisher	1.50E-10	6.80E-09	3.30E-09	2.80E-06
	C2	1.30E-09	5.80E-08	2.80E-08	9.40E-03
	UR_NNW	1.40E-09	7.70E-08	7.40E-07	1.30E-02
	UR_NW	1.60E-09	7.60E-08	8.00E-07	1.70E-02
	UR_NNW	2.10E-09	9.90E-08	9.90E-07	2.50E-02
	Dairy Farm NNE	1.90E-10	2.00E-08	2.80E-06	1.30E-03
	Farm NE	3.20E-10	2.30E-08	8.10E-07	1.90E-03

Age group	Location (see Figure 5-5)	Dose (μSv/a) per 1 Bq/s release rate			
		Kr-85	HTO	C-14	Co-60
Infant	B_E	1.90E-08	4.50E-07	2.10E-07	9.50E-02
	B_ENE	8.80E-09	2.00E-07	9.70E-08	5.50E-02
	B_NE	9.50E-09	2.20E-07	1.00E-07	6.70E-02
	B_NNE	2.50E-09	5.80E-08	2.80E-08	1.60E-02
	B_N	2.30E-09	5.30E-08	2.50E-08	1.60E-02
	B_NNW	2.50E-09	5.90E-08	2.80E-08	2.30E-02
	B_NW	2.70E-09	6.20E-08	3.00E-08	2.90E-02
	B_WNW	3.50E-09	8.20E-08	3.90E-08	4.10E-02
	B_W-Lake	5.10E-10	1.20E-08	5.60E-09	5.20E-06
	B_WSW-Lake	6.10E-10	1.40E-08	6.70E-09	6.30E-06
	B_SW-Lake	7.10E-10	1.60E-08	7.80E-09	7.30E-06
	B_SSW-Lake	1.10E-09	2.50E-08	1.20E-08	1.10E-05
	B_S-Lake	2.10E-09	5.00E-08	2.40E-08	2.20E-05
	B_SSE-Lake	8.80E-09	2.10E-07	9.80E-08	9.10E-05
	B_SE-Lake	1.00E-08	2.40E-07	1.10E-07	1.00E-04
	B_ESE-Lake	5.00E-09	1.20E-07	5.50E-08	5.10E-05
	Fisher	2.00E-10	4.70E-09	2.20E-09	2.10E-06
	UR_NNW	1.80E-09	4.70E-08	5.80E-07	1.70E-02
	UR_NW	2.10E-09	5.30E-08	6.40E-07	2.30E-02
	UR_NNW	2.70E-09	6.90E-08	7.80E-07	3.20E-02
	Dairy Farm NNE	2.50E-10	2.50E-08	5.00E-06	1.70E-03
	Farm NE	4.20E-10	1.60E-08	6.80E-07	2.40E-03

Table 5-5: Normalized Annual Doses from Chronic Releases at Phase II Site [3]

Age group	Location (see Figure 5-5)	Dose (μSv/a) per 1 Bq/s release rate			
		Kr-85	HTO	C-14	Co-60
Adult	B_E	2.63E-07	9.72E-06	3.89E-06	3.08E-01
	B_ENE	4.05E-08	1.50E-06	5.99E-07	2.87E-01
	B_NE	6.00E-09	2.22E-07	8.88E-08	4.05E-02
	B_NNE	4.16E-09	1.54E-07	6.16E-08	4.15E-02
	B_N	3.34E-09	1.23E-07	4.94E-08	3.72E-02
	B_NNW	3.06E-09	1.13E-07	4.52E-08	3.56E-02
	B_NW	2.51E-09	9.28E-08	3.72E-08	3.03E-02
	B_WNW	2.22E-09	8.20E-08	3.29E-08	2.54E-02
	B_W-Lake	2.54E-10	9.38E-09	3.76E-09	3.24E-06
	B_WSW-Lake	2.40E-10	8.86E-09	3.55E-09	3.07E-06
	B_SW-Lake	2.57E-10	9.50E-09	3.81E-09	3.29E-06
	B_SSW-Lake	3.47E-10	1.28E-08	5.14E-09	4.44E-06
	B_S-Lake	4.76E-10	1.76E-08	7.04E-09	6.08E-06
	B_SSE-Lake	9.35E-10	3.45E-08	1.38E-08	1.19E-05
	B_SE-Lake	2.33E-09	8.63E-08	3.45E-08	2.98E-05
	B_ESE-Lake	8.17E-09	3.02E-07	1.21E-07	1.04E-04
	Fisher	4.52E-11	1.67E-09	6.68E-10	5.77E-07
	C2	1.58E-09	5.83E-08	2.33E-08	1.11E-02
	IND	7.69E-10	2.84E-08	1.14E-08	9.84E-06
	UR_NNW	2.10E-09	8.74E-08	1.16E-06	2.41E-02
	UR_NW	1.96E-09	8.19E-08	1.10E-06	2.35E-02
	UR_NNW	1.80E-09	7.53E-08	1.03E-06	2.08E-02
	Dairy Farm NNE	2.30E-10	2.83E-08	3.06E-06	1.80E-03
	Farm NE	3.99E-10	4.59E-08	1.63E-06	2.65E-03
Child	B_E	2.63E-07	1.16E-05	5.55E-06	3.10E-01
	B_ENE	4.05E-08	1.78E-06	8.55E-07	2.87E-01
	B_NE	6.00E-09	2.64E-07	1.27E-07	4.05E-02
	B_NNE	4.16E-09	1.83E-07	8.79E-08	4.15E-02
	B_N	3.34E-09	1.47E-07	7.04E-08	3.72E-02
	B_NNW	3.06E-09	1.34E-07	6.45E-08	3.56E-02
	B_NW	2.51E-09	1.10E-07	5.30E-08	3.04E-02
	B_WNW	2.22E-09	9.75E-08	4.69E-08	2.54E-02
	B_W-Lake	2.54E-10	1.12E-08	5.36E-09	4.58E-06
	B_WSW-Lake	2.40E-10	1.05E-08	5.06E-09	4.33E-06
	B_SW-Lake	2.57E-10	1.13E-08	5.43E-09	4.64E-06
	B_SSW-Lake	3.47E-10	1.53E-08	7.33E-09	6.27E-06
	B_S-Lake	4.76E-10	2.09E-08	1.00E-08	8.58E-06
	B_SSE-Lake	9.35E-10	4.11E-08	1.97E-08	1.69E-05
	B_SE-Lake	2.33E-09	1.03E-07	4.93E-08	4.21E-05
	B_ESE-Lake	8.17E-09	3.59E-07	1.73E-07	1.47E-04
	Fisher	4.52E-11	1.98E-09	9.53E-10	8.15E-07
	C2	1.58E-09	6.93E-08	3.33E-08	1.11E-02
	UR_NNW	2.10E-09	9.99E-08	1.02E-06	2.41E-02
	UR_NW	1.96E-09	9.35E-08	9.72E-07	2.35E-02
	UR_NNW	1.80E-09	8.59E-08	9.12E-07	2.08E-02
	Dairy Farm NNE	2.12E-10	2.23E-08	3.09E-06	1.48E-03
	Farm NE	3.68E-10	2.60E-08	9.30E-07	2.17E-03

Age group	Location (see Figure 5-5)	Dose (μSv/a) per 1 Bq/s release rate			
		Kr-85	HTO	C-14	Co-60
Infant	B_E	3.43E-07	7.97E-06	3.79E-06	4.00E-01
	B_ENE	5.28E-08	1.23E-06	5.84E-07	3.73E-01
	B_NE	7.82E-09	1.82E-07	8.65E-08	5.27E-02
	B_NNE	5.43E-09	1.26E-07	6.00E-08	5.39E-02
	B_N	4.35E-09	1.01E-07	4.81E-08	4.83E-02
	B_NNW	3.99E-09	9.27E-08	4.41E-08	4.63E-02
	B_NW	3.28E-09	7.61E-08	3.62E-08	3.94E-02
	B_WNW	2.90E-09	6.73E-08	3.20E-08	3.30E-02
	B_W-Lake	3.31E-10	7.69E-09	3.66E-09	3.42E-06
	B_WSW-Lake	3.13E-10	7.27E-09	3.46E-09	3.23E-06
	B_SW-Lake	3.35E-10	7.80E-09	3.71E-09	3.47E-06
	B_SSW-Lake	4.53E-10	1.05E-08	5.01E-09	4.69E-06
	B_S-Lake	6.20E-10	1.44E-08	6.85E-09	6.41E-06
	B_SSE-Lake	1.22E-09	2.83E-08	1.35E-08	1.26E-05
	B_SE-Lake	3.04E-09	7.08E-08	3.36E-08	3.15E-05
	B_ESE-Lake	1.07E-08	2.48E-07	1.18E-07	1.10E-04
	Fisher	5.89E-11	1.37E-09	6.51E-10	6.09E-07
	UR_NNW	2.74E-09	7.00E-08	8.09E-07	3.14E-02
	UR_NW	2.56E-09	6.56E-08	7.70E-07	3.05E-02
	UR_NNW	2.35E-09	6.03E-08	7.23E-07	2.70E-02
	Dairy Farm NNE	2.77E-10	2.77E-08	5.48E-06	1.92E-03
	Farm NE	4.81E-10	1.79E-08	7.86E-07	2.82E-03



	Distance from release location to receptor	
Receptor	Phase 1 (km)	Phase 2 (km)
B_N	1.2	1.1
B_NNE	1.1	0.8
IND	1.2	0.8
B_NE	1.0	0.5
B_ENE	1.0	0.4
B_E	0.7	0.3
B_ESE-Lake	0.4	0.5
B_SE-Lake	0.4	0.7
B_SSE-Lake	0.4	1.0
B_S-Lake	0.5	1.1
Fisher	0.5	1.1
B_SSW-Lake	0.8	1.4
B_SW-Lake	1.1	1.7
B_WSW-Lake	1.2	1.7
B_W-Lake	1.4	1.9
B_WNW	1.6	1.9
Beach	1.0	1.3
UR_WNW	1.8	2.2
B_NW	1.4	1.6
UR_NW	1.7	1.9
B_NNW	1.4	1.3
UR_NNW	1.6	1.7
C2	3.2	2.6
Dairy Farm	10.5	9.9
Farm	6.9	6.3
Ajax WSP	6.2	5.8

Figure 5-5: Near- and Far-Site Critical Group Receptor Locations for PWMF Safety Assessment

Notes: Locations included in the model are those identified by the Pickering site survey [12] and hypothetical locations along Pickering site exclusion zone boundary. For landside hypothetical locations, receptors were assumed to be at their locations 100% of the time. Receptors at the lakeside hypothetical locations were assumed to be present at that location 1000h per year.

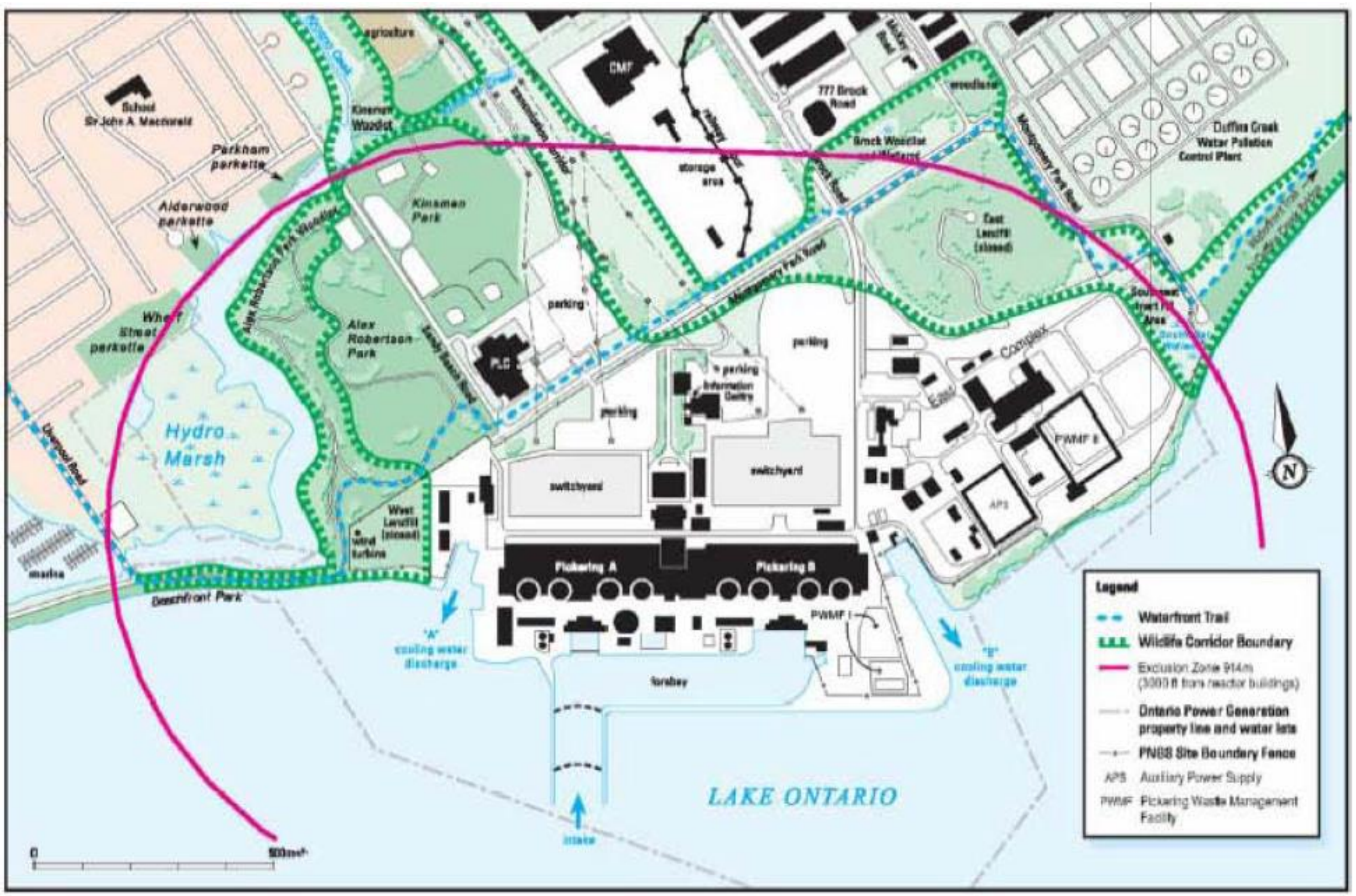


Figure 5-6: Pickering Site

Table 5-6: Annual Individual Dose from PWMF Normal Operation Considering DSCs Containing 6 Year Decayed Used Fuel[15]

Radiation Source	Maximum Annual Individual Dose (μSv/a)	Dose Receptor Location	Notes
External gamma radiation from DSCs and DSMs. Including storage of DSCs containing 6 year decayed used fuel in SB3.	3.56E+00	PNGS site boundary[a]	Based on the Best Estimate + 2σ dose rate at Montgomery Park Rd turnaround (1.78×10^{-3} μSv/h [9]) and 2000 hours annual occupancy. The listed dose and dose rate are due to the DSCs in the Phase II site. The contribution from DSCs and DSMs in the Phase I site is negligible [11].
	1.64E+00	Lakeside exclusion zone boundary	Based on the Best Estimate + 2σ dose rate at the lake where the shoreline intersects with the land site boundary (1.64×10^{-3} μSv/h [9]) and 1000 hours[b] annual occupancy for boaters and fishermen.
Chronic particulate emission from PWMF measurements reported in quarterly reports.	6.02E-04	PNGS site boundary	Based on 2.00×10^{5} Bq/a release from Phase I site (Table 5-1).
	1.01E-07	Lakeside exclusion zone boundary	
Postulated volatile releases from DSC processing.	1.18E-03	PNGS site boundary	Based on 280 failed elements per year from Phase I site.
	7.02E-05	Lakeside exclusion zone boundary	
Postulated release from DSMs.	1.57E-04	PNGS site boundary	Based on $1.60 \text{X} 10^{10}$ Bq/a carbon-14 release from Phase I.
	9.26E-06	Lakeside exclusion zone boundary	
Total annual individual dose	**3.56E+00**	**PNGS site boundary**	**The limiting annual individual dose for a member of the public is 0.4% of the 1 mSv regulatory limit.**
	1.64E+00	**Lakeside exclusion zone boundary**	

Notes:
a) The PNGS boundary is selected because of it is the location with the potential highest dose. There is no temporary or permanent population at the PNGS boundary. A conservative occupancy of 2000 hours per year is assigned for receptors at the hypothetical landside boundary locations. The partial occupancy is conservative since the location is not identified as one of the potential critical group representative locations around the Pickering site [12]. External gamma dose rates at representative locations identified in Reference [12] will be significantly lower than the dose rate at Montgomery Park Rd and are expected to be indistinguishable from the natural background radiation level.
b) The 1000 hours per year occupancy assumption is significantly more conservative than the 1% (~88 hours per year) occupancy factor assumed in the PNGS assessment and site-specific survey [12].

15 Considering 70 DSCs containing 6 year decay used fuel (of 230 MWh/kgU burnup) are processed in a year of operation of the PWMF and the releases outlined in Section 5.3.1.

5.3.4 Worker Dose

The worker doses during normal operation of the PWMF were estimated based on the recorded doses from the PWMF quarterly reports (92896-REP-00531-* series of reports). The maximum recorded individual whole body dose for PWMF worker during the period of 2007-2019 is 1.6 mSv[16], which is 3% of the 50 mSv dose limit and 53% of the 3 mSv ALARA dose target. The highest annual dose is typically for an individual in the Operations group.

At most, if every DSC processed in a year contained 6 year decayed used fuel, collective dose could be expected to increase by a factor of 2.6 (see Footnote 17). However, since not all DSCs will contain 6 year decayed used fuel, and the individual dose will be managed by the existing OPG Radiation Protection Program [13], worker doses are expected to remain well within the 3 mSv ALARA dose target documented in the PWMF Safety Report [1].

5.4 Malfunctions and Accidents

5.4.1 Screening of Potential Accident Scenarios and Identification of Bounding Accident Scenarios

5.4.1.1 Hazard Identification

The basis of the PWMF hazard identification is the list of internal hazards [14] and external hazards [15] developed in support of OPG's Probabilistic Safety Assessment (PSA) guide. This list is supplemented, as applicable, by hazards derived from other PWMF documentation.

5.4.1.2 Pre-Screening of Existing Hazards

A pre-screening of the identified hazards was undertaken to screen out those events which are known to have no impact on the Pickering site or PWMF. Hazards may be eliminated at this stage when it can easily be determined without any additional analysis, that the hazard has a negligible impact on the safety of the PWMF. Hazards that are screened in are then part of the detailed screening analysis. Appendix B lists the results of the hazard pre-screening assessment.

5.4.1.3 Events Screening

Potential hazards associated with the on-site transfer, processing and storage in SB3 of the lower-fuel-age DSC, containing 6 year decayed used fuel, and transfer of a number of existing DSCs between SB3 and SB4, were identified and the events were screened in or out. Events that were screened out were deemed to be incredible (the frequency is less than 10^{-6} events per year) or to have a negligible contribution to risk. This process followed the OPG screening criteria (References [14] and [15], as applicable) against which the events were assessed and summarily dismissed.

First, a qualitative screening was conducted to identify hazards that were judged to have negligible impact on risk without the need to perform any detailed quantitative assessments. Part of the

[16] The maximum recorded individual whole body dose of 1.6 mSv occurred in 2008 for an individual of the Civil Maintenance group.

[17] As identified in Section 5.3.2.1, compared to a DSC containing 10 year decayed used fuel the dose rate at 1 m from a single DSC containing 6 year decayed used fuel may increase by a factor of 2.6.

qualitative screening was performed during hazard pre-screening documented in Appendix B. Some of the identified hazards may not have an impact on the DSC due to its location and orientation with respect to the rest of the structures, or certain hazards may not be applicable due to consideration of those hazards during design. Hazards that have been determined to not have any impact on radiological consequence 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 [4], 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} events per year. Using this definition of a credible event, an event screening frequency of 10^{-6} events per year was applied to quantitative screening.

Assumptions listed in Table 5-7 were applied for event screening.

Table 5-7: Assumptions for Events Screening

Assumption	Justification
The probability of a human error is assumed to be 10^{-3} per movement or activity.	NUREG/CR-1278 [16] provides an estimated human error probability of 0.001 for an operator placing a manual control in an incorrect setting.
100 DSCs containing 6 year decayed used fuel will be transported within one year from the Station IFB to the Phase I Processing Building and from the Processing Building to SB3.	The transfer of 100 lower-fuel-age DSCs in one year is a conservative, bounding value.
A total of 119 existing DSCs stored in SB3 will be moved to SB4 to accommodate the arrival of 100 DSCs with lower-fuel-age into SB3. 19 of the 119 existing DSCs will be moved back from SB4 to the SB3 and placed between the DSCs containing 6 year decayed used fuel and the aisle. These DSC moves will occur within one year.	Refer to Figure 5-2 for DSCs layout in SB3. The move of the existing 119 DSCs between SB3 and SB4 within one year is a conservative, bounding value.
The maximum speed of the DSC transporter is 12 km/h (GEN IV) [17]. The total distance the Transporter needs to travel between the Station IFB to the Phase I Processing Building and from the Processing Building to SB3 is approximately 2 km. However, the travel time of the Transporter is conservatively assumed to take a longer time and be on the road for 1 hour.	Conservatively slow speed maximizes time-at-risk for hazards while the DSC is in transit.

Assumption	Justification
Buildings and structures designed based on NBCC 1995 or prior are assumed to collapse at a frequency of $2x10^{-3}$/year. The PWMF Phase I Processing Building was designed based on NBCC 1990 [3]	Although it can be damaged, the NBCC requires the structure to not collapse at the specified seismic risk level. However, with no supporting analysis to determine when the structure actually fails, the frequency of building failure is assumed to be equivalent to the earthquake frequency. The seismic risk for NBCC 1995 is based on an earthquake with a 500 year return period. 1/500 years = $2x10^{-3}$ events/year
Buildings and structures designed based on NBCC 2005 or later are assumed to collapse at a frequency of $4x10^{-4}$/year [3].	The seismic risk for NBCC 2005 and later is based on an earthquake with a 2500 year return period. 1/2500 years = $4x10^{-4}$ events/year
The functional and performance requirements for DSCs containing 10 year decayed used fuel [18] apply to DSCs loaded with 6 year decayed used fuel.	Because the Design Requirements for a DSC loaded with 6 year decayed used fuel are not currently available, it is assumed that the DSC functional and performance requirements in a DSC containing 10 year decayed used fuel [18], will also be requirements in the Design Requirements for a DSC loaded with 6 year decayed used fuel. This assumption is part of the work scope and may be re-visited when the Design Requirements for a DSC containing 6 year decayed used fuel are issued.

5.4.2 Malfunctions/Accidents during DSC On-Site Transfer

It is conservatively assumed that 100 lower-fuel-age DSCs will be transferred from the Station IFBs to the Phase I Processing Building and after processing the seal-welded DSCs will be transferred to the SB3, within one year.

During normal operation of the existing UFDS facility, up to 70 DSCs are transferred from the PNGS IFB for processing each year that will be stored in either the Phase I or Phase II storage buildings. Hazard frequencies documented in Reference [3] have been calculated based on this number.

In summary, the present hazard screening assessment focuses on the 100 lower-fuel-age containers to be transferred from the Stations' Irradiated Fuel Bay (IFB) for processing and then to storage in SB3 within one year, and hazard frequencies were calculated accordingly.

5.4.2.1 Transporter Failure

5.4.2.1.1 Between the Station IFB and Processing Building

The transporter, carrying a loaded DSC, travels on the south side of the powerhouse between the PBIFB and the DSC Processing Building. From the Pickering A AIFB the transporter travels on the north side of the powerhouse to the DSC Processing Building [19].

In the event of Transporter failure, the containment barrier provided by the transfer clamp and elastomeric seal is assumed to fail [1] as a result of the longer than expected time taken to transfer the DSC from the Pickering B Primary Irradiated Fuel Bay (PBIFB) or from the Pickering A Auxiliary Irradiated Fuel Bay (AIFB) to the DSC Processing Building.

Conservatively it is assumed that the free inventory of tritium, carbon-14, and krypton-85 in four damaged fuel elements is released into the DSC cavity (if 1 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 Western Waste Management Facility (WWMF) operating experience, the frequency for this event is 3 events per year [1]. This event is screened in.

5.4.2.1.2 Between the Processing Building and SB3

The scenario is a Transporter failure while transferring a seal-welded DSC from the PWMF Phase I Processing Building to the Phase II SB3. 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 on-site transfer of the DSC. The lid closure weld is a groove weld between the base plate of the lid and the perimeter flange of the base. After the weld has been completed and cooled, a Phased Array Ultrasonic Testing (PAUT) system is used for the inspection of the DSC lid-to-base seal-weld. The DSC is subsequently filled with inert helium and leak tested prior to storage.

As there is no external force acting upon the DSC, it is considered that a longer than expected transfer time from the Processing Building to the SB3 associated with transporter failure will not have any impact on the integrity of the seal-welded DSC. This event is screened out.

5.4.2.2 Transporter Operator Health-Related Emergency

The Transporter operator could have a health-related emergency resulting in loss of consciousness during the DSC transfer. The Transporter operates at low speed and is escorted by at least one person in addition to the driver. The second person could intervene to stop the Transporter. Even if operator illness 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 the transfer clamp with elastomeric seal or the seal-weld. The dose consequences from this postulated scenario would be within the envelope of those of the container drop presented in Section 5.4.2.3. This event is screened out.

5.4.2.3 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 possible scenario involving collision of the DSC Transporter with another vehicle, resulting in the drop of the DSC, could take place during the on-site transfer of the DSC:

- Between the station IFB and the Phase I Processing Building; along this transfer route the transporter carries a DSC with transfer clamp. 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 [3]. Therefore, only the airborne release of tritium, carbon-14, and krypton-85 from the DSC cavity is considered for this assessment.
- Between the PWMF Phase I Processing Building and Phase II SB3: along this transfer route a transporter carries a seal-welded DSC. Although the seal-weld is extremely robust, the collision is postulated to compromise the seal-weld.

Even though high level of control and security is maintained for the transfer route, a considerable impact may occur with another vehicle during the DSC transfer. Therefore, failure of 100 percent of a clamped or seal-welded 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 (384x28) failed fuel elements [3]. The free inventory of tritium, carbon-14 and krypton-85 in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barrier provided by the transfer clamp or seal-weld, it is assumed that the radionuclides are released at once into the environment.

This event is screened in.

5.4.2.4 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 include acetylene cylinders and propane gas tanks [20], 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 PNGS emergency response team as the primary responder inside the PNGS protected area.

The effect of a fire could potentially be to increase the temperature of the DSC and the used fuel bundles inside the DSC. A thermal analysis was conducted to investigate the heating and cooling process of a DSC during and after a fire of the DSC transporter carrying a container: the DSC in transfer is on the Transporter, located in the fire, refer to Section D4.5.3 of [21]. It was concluded, that given the large thermal inertia of the DSC and the limited duration of the event, the fire scenario involving a DSC would not breach the containment and will not result in radiological release.

This event is screened out.

5.4.2.5 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 sanding or salting of the transfer route. Even if the transporter 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. In the worst case scenario, where the transporter topples over as a result of adverse road conditions, the radiological consequences would be within the envelope of those described in Section 5.4.2.3.

Therefore, this event can be screened out.

5.4.2.6 Earthquake

The Pickering B Design Basis Earthquake (DBE) is defined as an earthquake with peak ground acceleration (PGA) of 0.05g and a frequency of reoccurrence of once in 1000 years [22]. The integrity of the DSC has been evaluated by determining its stability against overturning and sliding under the postulated DBE seismic acceleration [23]. The seismic ground response spectrum of the PNGS B site was used for the evaluation.

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. As mentioned in Section 5.4.2, frequencies for DSC transfer were calculated based on the 100 lower-fuel-age containers to be transferred between the stations' IFB to the SB3 via the Processing Building.

The following assumptions were made:

- 100 lower-fuel-age DSCs are transferred within one year from the station IFBs to the Processing Building and from the Processing Building to SB3;
- The transporter is assumed to travel up to a maximum speed of 12 km/h, refer to Table 5-7;
- The total distance the Transporter needs to travel is approximately 2 km: between the AIFB and the DSC Processing Building is approximately 1 km [24] and from the Processing Building to SB3 is approximately 1 km [25], [1]; and
- The Transporter is conservatively assumed to take a longer transfer time and be on the road for 1 hour to increase the time at risk.

With these assumptions, the probability of finding a loaded DSC in transit during a 1-year period would be:

$$100 \times 1 \times (1/24) \times (1/365) = 1.14 \times 10^{-2}$$

The frequency of a DBE occurring at a time when a lower-fuel-age DSC is being transferred is:

$$(1 \times 10^{-3}) \times (1.14 \times 10^{-2}) = 1.14 \times 10^{-5} \text{ events/year}$$

This event cannot be screened out based on frequency. However, in the worst case scenario, where the Transporter topples over and drops the DSC as a result of a seismic event, the radiological consequences, as per the Safety report [1], would be within the envelope of those in Section 5.4.2.3. This event is screened out.

5.4.2.7 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 [26] 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 [26]. Based on the PNGS site wind speed frequencies listed in Table 1 of Reference [27], the DBT-definition rotational wind speeds correspond to a mean frequency of 3.13×10^{-6} events/year.

During tornado winds, objects can be picked up by the wind forces and accelerated to high velocities. Reference [26] has an established spectrum of tornado- generated missiles considered in the Darlington Nuclear Generating Station (DNGS) design as part of the DBT. The following tornado-generated missiles have been assessed for their impact on the clamped DSC during on-site transfer:

a) Woodplank, 102 mm × 305 mm × 3.7 m, weight 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, weight 35.4 kg, velocity 168 km/h (40 percent of total tornado velocity).

c) Steel rod, 25 mm diameter × 914 mm long, weight 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, weight 129 kg, velocity 168 km/h (40 percent of total tornado velocity).

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

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

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

Safety of the DSC against overturning was investigated for a severe wind load simulating a tornado wind speed of 425 km/h [23]. 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.

As calculated in Section 5.4.2.6, the frequency of having a loaded DSC transferred from the station's IFB to the SB3 via the Phase I Processing Building is 1.14×10^{-2} events/year. The frequency of a tornado occurring at the time when a DSC is being transferred is

$1.14 \times 10^{-2} \times 3.13 \times 10^{-6} = 3.57 \times 10^{-8}$ events/year,

which is below the cut-off frequency of 10^{-6} per year. Therefore, this event can be screened out.

5.4.2.8 Thunderstorms

Thunderstorms can potentially involve lightning striking a loaded DSC on the Transporter during on-site transfer.

According to the DSC design requirements [18], the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during on-site transfer. As severe atmospheric conditions are within the design basis of the DSC, this hazard is screened out.

5.4.2.9 Flooding

The only possibility for flooding at the Pickering site would be as a result of extreme local meteorological events. A review level Probable Maximum Precipitation (PMP) has been developed to be used at the OPG sites [28], which represents a rainfall of 420 mm in a 12-hour period, of which 51% (214 mm) fall within a one hour period.

Transfer procedures require that loaded DSC not to 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 transfer of a lower-fuel-age DSC during an extreme rainfall were to occur, extensive flooding would likely affect the operation of the Transporter, however it is not expected to have any detrimental effect on the DSC. The DSCs are designed to tolerate water immersion at 2MPa [18], so the temporary flooding waters would not be of a concern to radiological safety. This event is screened out.

5.4.2.10 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 AIFB/IFB to the Phase I Processing Building [24], 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 cylinders have been assessed in Section 5.6 of Reference [24], and the combined hazard frequency has been calculated to be 9.74×10^{-8} events/year, lower than the 10^{-6} cut-off frequency. Thus, this hazard is screened out based on frequency.

Explosion hazards along the onsite transfer route of the DSC from the PWMF Phase I Processing Building to the Phase II SB3 have been assessed [20]. The following fire and non-fire initiated hazard scenarios have been taken into consideration to be capable of toppling a passing DSC:

- Acetylene cylinder detonation
- Propane storage tank BLEVE (Boiling Liquid Expanding Vapour Explosion)
- Vapour cloud explosion (VCE) due to a propane storage tank rupture.

The combined explosion frequency of the above-mentioned hazards on the DSC in transfer has been determined to be 5.2×10^{-8} per year, refer to Section 3.4 of Reference [20]. The calculations were done assuming 20 DSC shipments a week (about 1000 shipments per year), which is considerably a higher number than the assumed annual shipments of 100 lower-fuel-age DSCs to SB3. The explosion hazard frequencies calculated above are lower than the 10^{-6} cut-off frequency, therefore this hazard is screened out.

5.4.2.11 Turbine Missile Strike

The frequency of turbine missiles impacting structures, systems and components (SSCs) has been determined in Section 6.2.1 of Reference [28] to be 6×10^{-6} events/year.

The probability of having a loaded DSC in transit between the Phase I Processing Building and SB3 is 1.14×10^{-2} over a year, refer to Section 5.4.2.6.

The frequency of turbine missiles impacting a loaded DSC while the DSC is being transferred from the IFB to the Processing Building or from the Processing Building to Storage Building 3 is:

$$6 \times 10^{-6} \times 1.14 \times 10^{-2} = 6.84 \times 10^{-8} \text{ events/year,}$$

which is below the event cut-off frequency of 10^{-6}/year. This hazard is screened out.

5.4.2.12 Aircraft Crash

The probability of an aircraft strike is proportional to the target area.

To assess this hazard, the size of the Liftking transporter was used as it is slightly larger than the GEN IV transporter. The transporter has an overall length of 27.833 ft (8.5 m), an overall width of 10.875 ft (3.3 m) and an overall height of 15.521 ft (4.7 m) [17].

Using total crash rates determined for the PNGS site [28], the aircraft impact frequency on the transporter has been calculated, considering the limited time that a loaded 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 transporter carrying the DSC during on-site transfer is 4.12×10^{-10} events/year (see Appendix C), which is below the 10^{-6} cut-off frequency and therefore this hazard can be screened out.

5.4.2.13 Toxic Gas Release - Chlorine originated 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 site. The facility is in the route of the DSC onsite transfer from the Processing Building to SB3. The Screening Distance Value (SDV) for chlorine is 4.4 km [15], hence this hazard cannot be screened out based on distance.

Chlorine leak from the Ajax Water Treatment Plant can have an impact on the Transporter operator ability to keep the Transporter safely on road. However, even if the operator illness were to result in the Transporter leaving the road, the impact of this hazard would be bounded by the DSC drop as a result of the Transporter collision with another vehicle, refer to Section 5.4.2.3. This event is screened out.

5.4.2.14 Soil Failures/Slope Instability

It has been identified that the DSC transport route between the AIFB/ IFB and the DSC Processing Building has some anomalous portions and a detailed geotechnical assessment has been recommended [24].

In the worst case scenario, where the Transporter topples over and drops the clamped or seal-welded DSC as a result of soil failure, the radiological consequences would be within the envelope of those in Section 5.4.2.3.

This event is screened out.

5.4.3 Malfunctions/Accidents during Processing

5.4.3.1 Drop of a Dry Storage Container during Handling

Failure of the crane, the lifting beam, lift plates, or the DSC trunnions could potentially result in dropping a loaded DSC while it is being lifted during operations at the DSC Processing Building.

The failure probability of a crane lifting very heavy loads, based on US nuclear plant operating experience, (Section 3.5 of NUREG-1774 [29]), is estimated to be 5.6 x 10^{-5} per demand. Based on the combined operating experience of the PWMF and WWMF, the number of lifts to be carried out in the DSC Processing Building using the crane is approximately 600 per year [1]. Therefore, the total postulated frequency of crane failure would be

$5.6 \times 10^{-5} \times 600 = 3.36 \times 10^{-2}$ events per year, which is greater than the cut-off frequency of 10^{-6}.

A handling accident involving the dropping or tip over of multiple DSCs is not considered to be a credible event. The normal lift height of a DSC in the Processing Building with transfer clamp installed is 200 mm. 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 reduce the likelihood of the container from tipping over and striking a second DSC.

Realistically, fuel sheath failure is not expected to result from an accidental DSC drop from the low lift height of the crane in the DSC Processing Building. In the worst-case scenario, dropping a clamped 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 (0.3x384x28) [1]. The free inventory of tritium, carbon-14 and krypton-85 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. This event is screened in.

5.4.3.2 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 will be performed.

However, to calculate the event frequency of equipment dropping onto a DSC, the following assumptions were made:

- A maximum of 100 lower-fuel-age DSCs will be processed within an year;
- A total of four pieces of equipment have the potential to drop onto a DSC lid [1];
- With the exception of the transfer clamp, each structure is lifted twice over the DSC (installation and removal), totaling seven lifts per DSC [1].

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:

$$5.6 \times 10^{-5} \times 100 \times 7 = 3.92 \times 10^{-2} \text{ events per year.}$$

Given that the lift height of equipment over a loaded DSC is limited by procedural controls, the dose consequences from this scenario would be bounded by the drop of a dry storage container described in Section 5.4.3.1. Therefore, this event is screened out.

5.4.3.3 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 with other process building equipment or structure.

Assuming that 100 lower-fuel-age DSCs are processed within a one year period, the total number of times a loaded and unwelded DSC is lifted would be approximately 110 (one lift per DSC plus 10 percent of them are assumed to have weld failure and require weld repairs). The assumed operator error probability is 10^{-3} per movement [16].

The postulated frequency of a loaded and unwelded DSC craning collision accident is

$$10^{-3} \times 110 = 1.1 \times 10^{-1} \text{ events per year}$$

Given that the overhead crane bridge and trolley maximum speeds are limited by design, the dose consequences from this scenario would be bounded by the drop of a dry storage container described in Section 5.4.3.1. This event is screened out.

5.4.3.4 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 Processing Building floor or with another Transporter in the Processing Building. The Transporter collision could occur while it is carrying a loaded or empty DSC.

It is assumed that a maximum 100 lower-fuel-age DSCs are loaded within one year and that the Transporter is used three times to move each DSC within the Processing Building: 1) transfer of a loaded DSC to the DSC Processing Building, 2) transfer of a seal-welded DSC to the paint station within the Phase I site, and 3) transfer the DSC from processing 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

10^{-3} x 100 x 3 = 3 × 10^{-1} events per year.

Given that the Transporter's 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 the drop of a dry storage container described in Section 5.4.3.1. This event is screened out.

5.4.3.5 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:

- Maximum 100 lower-fuel-age DSCs will be processed within one year;
- A total of four 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) [1]; and
- With the exception of the transfer clamp, each structure is lifted twice over the DSC (installation and removal), totalling seven lifts per DSC [1].

The total frequency of this event due to operator error is 10^{-3} x 100 x 7 = 7 x 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 would be bounded by the drop of a dry storage container described in Section 5.4.3.1. This event is screened out.

5.4.3.6 Dry Storage Container Processing Building Fire

The DSC Processing Building is a non-combustible building with a reinforced slab-on-grade, steel frame structure, concrete on metal deck floors, concrete block interior walls and, insulated precast concrete panel and insulated metal cladding panel exterior walls. The roof consists of a built-up insulated roof on metal deck on the steel structure [21].

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).

Based on the PWMF fire hazard assessment (FHA) [21], the bounding fire scenario for the Processing Building was a fire involving one GEN IV DSC Transporter in the workshop. The safe separation distances between the transporter and combustible materials and transient work equipment (forklift and floor polishers) was assessed and it was determined that the combustible materials would be within the 5.3 m horizontal safe separation distance. As a result, a DSC Transporter fire would likely ignite the adjacent combustibles located in the workshop.

To evaluate the design basis fire event, the quantitative analysis was performed as a worst-case scenario without crediting any fire mitigation systems. After that, the fire scenario was assessed crediting the available fire detection and suppression systems and fire-fighting. The FHA determined that the Emergency Response Team (ERT) would provide suppression within 23 minutes after ignition and the fire would be ended within 30 minutes from ignition. The fire scenarios were used to calculate the impact of the design basis fire on the DSC, structural steel trusses and adjacent DSC transporters.

It was concluded, that due to the fire detection and alarm systems in place in the Processing Building, and the expected prompt arrival of the ERT, the fire would be of short duration and localized.

It is to be noted that no thermal analysis was conducted to investigate the effects of an external fire on the DSC containing 6 year decayed used fuel; however a thermal analysis was conducted to investigate the heating and cooling process of a DSC storing 10 year decayed used fuel during and after a fire under an accident transportation scenario; the DSC is on the Transporter, located in the fire [21]. The thermal assessment concluded that given the large thermal inertia of the DSC and the limited duration of the event, a fire scenario in the Processing Building involving a DSC would not breach the containment and will not result in radiological release. This event is screened out.

5.4.3.7 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 smaller 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 for the WWMF Processing Building [30]. It was concluded that the flange of the DSC base interfacing with the lid may experience some permanent damage allowing some airborne release. However, there is no likelihood of the fuel to be exposed based on the magnitude of lid slippage or DSC tipping. The impact of the PWMF Processing Building collapse on an unclamped and un-welded DSC lid is bounded by the collapse of the WWMF Processing Building scenario [31].

The DSC has a safety factor of 7 against overturning and 4 against sliding under the loads described for the earthquake scenario using the Pickering B design basis earthquake (DBE) of 10^{-3} event per year and ground motion parameters of 0.05g PGA [23]. The structure of the container is adequately strong to ensure the integrity of the DSC in case of an earthquake with the above parameters.

Calculations were performed [32] assessing the DSC seismic stability for the lower probability (10^{-4} per year) 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 revealed that the safety factor against overturning of the DSC is 3 and the safety factor against sliding is 1.54 [32]. While the safety factors are lower than the ones for the Pickering B DBE, in both cases they are greater than 1, meaning that the DSC will not overturn or slide during an earthquake scenario using the more stringent Pickering A DBE. The lower safety factors for Pickering A DBE are acceptable as the probability of the ground motion is lower as well.

However, 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.

An earthquake causing the Processing Building to collapse on an unclamped and un-welded DSC is screened in.

5.4.3.8 Tornado

The effect of tornado-generated missiles on a clamped DSC has been considered in Section 5.4.2.7 and it was concluded that the DSC's transfer clamp will keep the lid in place, the containment will not be breached, and the DSC will not overturn under the impact of the postulated tornado missiles.

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.

A DSC can resist overturning in tornado winds of up to 425 km/h [23].This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the interceding building structures. Therefore, a missile striking the DSC is expected to have negligible consequences, even if the DSC is un-welded.

This event is screened out.

5.4.3.9 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC Processing Building. Based on the design requirements [18], the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions, during storage. As severe atmospheric conditions are within the design basis of the DSC, this event is screened out.

5.4.3.10 Flood

Water entry originating from a PMP into the Processing Building is possible, however, the consequences are assumed to be negligible. The expected flood water depth is too shallow to reach near the level of the DSC lid at approximately 2.5 m height [33]. In addition, no loose contamination is permitted on the exterior surface of the DSC or on accessible surfaces within the Processing Building. This event is screened out.

5.4.3.11 Turbine Missile Strike

Phase I of the PWMF is located southeast of PNGS Unit 8. SB2 is situated the closest, at approximately 30 m, to Unit 8. SB2 is attached to the north wall of the Processing Building and SB1.

The frequency of turbine missiles impacting SSCs has been determined in Section 6.2.1 of [28] to be $6x10^{-6}$ events/year. Given 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. Therefore, this event is screened out.

5.4.3.12 Aircraft Crash

The probability of an aircraft strike is proportional to the target area. The aircraft crash frequency calculated for the Phase I site, which comprises of the Processing Building, SB1 and SB2, is 2.42×10^{-7} events/year, which is below the cut-off frequency of 10^{-6}, therefore this event is screened out.

In addition, a total aircraft crash frequency was calculated for the PWMF site where DSCs and DSMs are stored and it is presented in Appendix C.

5.4.3.13 Release of Oxidizing, Toxic, Corrosive 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, as stated in Reference [3]:

- Acute Toxicity
- Corrosive
- Oxidizing/Reactive
- Asphyxiant

The first step to screening the hazard from chemicals held within the PWMF Processing Building is to compile a list of chemicals held in each area. A preliminary screening was made based on whether the chemical falls into any of the above categories. Chemicals not included in the above four categories were screened out.

A secondary screening, based on quantities held on-site could not be performed as quantities are required to be shown on the HazMat Inventory sheets only for flammable liquids. Therefore, toxic materials stored in the Processing Building and SB3 are listed in Table 5-8 without quantities.

5.4.3.13.1 Toxic Materials

Table 5-8 describes the inhalation consequences of the toxic materials. Direct exposure to highly toxic chemicals may cause an operator to become incapacitated, leading to container mishandling errors.

The Material Safety Data Sheet (MSDS) of these chemicals revealed that most of them are minimally toxic when inhaled based on component assessment or based on test data for structurally similar materials. The only exception is the HYVOLT II transformer oil, which, based on its MSDS can be fatal if it enters the airways.

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.

This event is bounded by the drop of a dry storage container described in Section 5.4.3.1 and it is screened out.

5.4.3.13.2 Asphyxiants

Some hazardous chemicals can be used within the PWMF Processing Building as part of normal operations. Table 5-8 shows the chemicals that are stored in the Processing Building's gas bottle storage room and in the workshop [3]. These materials, such as helium and welding cover gas are considered asphyxiants and may lead to container mishandling caused by human error.

However, 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.

This event is bounded by the drop of a dry storage container described in Section 5.4.3.1 and it is screened out.

Table 5-8 Hazardous Chemicals Stored in Processing Building and SB3

Hazardous Material	CAT ID	Toxic	Oxidizing/ Reactive	Asphyxiant	Quantity/Location	Remarks
Antifreeze, Coolant Ethylene Glycol	328179	X			Location: PWMF, Processing Building, Room 110, Cabinet 33.	Inhalation of ethylene glycol will cause irritation of the eye and respiratory tract, but it is unlikely to result in toxicity. Extremely dangerous in case of ingestion, refer to Material Safety Data Sheet (MSDS).
Silica Gel	473059	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	No acute toxicity information is available for this product, refer to the MSDS. If inhaled, it may cause allergy or asthma symptoms or breathing difficulties.
SIKADUR 32 PART A-B	31222	X			Location: PWMF, Processing Building, Room 110, Cabinet 33	No acute toxicity information is available for this product, Skin and eye irritation
SIKATOP 123 PLUS	634398				Location: PWMF, Processing Building, Room 110, Cabinet 33	May cause skin irritation.

Hazardous Material	CAT ID	Toxic	Oxidizing/ Reactive	Asphyxiant	Quantity/Location	Remarks
Hydraulic Fluid, NUTO H-C 68	323268	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	Inhalation: Acute Toxicity: Based on assessment of the components it is minimally toxic. Irritation: Negligible hazard at ambient/normal handling temperatures, refer to MSDS.
Transformer Oil, Insulating, HYVOLT II	685854	X			Location: PWMF, Storage Building 3, Cabinet 10100	Inhalation: May be fatal if swallowed and enters airways, refer to the MSDS.
Oil, Synthetic Hydro Carbon, Mobil SHC-632	462010	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	Inhalation: Acute Toxicity: Minimally Toxic based on component assessment. Irritation: Negligible hazard at ambient/normal handling temperatures, refer to MSDS.
OIL, VACUUM PUMP P-150	658772				Location: PWMF, Processing Building, Room 110, Cabinet 36	Acute Toxicity: Respiratory tract irritation refer to MSDS.

Hazardous Material	CAT ID	Toxic	Oxidizing/ Reactive	Asphyxiant	Quantity/Location	Remarks
OIL, INDUSTRIAL STEAM TURBINE, TERESTIC-C100	323231	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	Acute Toxicity: Minimally Toxic based on component assessment, refer to the MSDS Irritation: Negligible hazard at ambient/normal handling temperatures.
OIL, GEAR, SPARTAN EP220	323285	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	Acute Toxicity: Minimally Toxic based on component assessment, refer to the MSDS Irritation: Negligible hazard at ambient/normal handling temperatures.
Mobile Delvac 1300 Super 15W30 Engine Oil	670659	X			Location: PWMF, Storage Building 3, Cabinet 10100	MSDS for Mobile Delvac 1300 Super 15W30 has not been located. *MSDS for Engine Oil, DELVAC, 15W40*: Inhalation: Acute Toxicity: Minimally Toxic based on test data for structurally similar materials. *MSDS for Engine Oil, DELVAC, 10W30*: Inhalation: Acute Toxicity: Minimally Toxic based on assessment of components.

Hazardous Material	CAT ID	Toxic	Oxidizing/ Reactive	Asphyxiant	Quantity/Location	Remarks
ENGINE OIL, MOBIL DELVAC, 1300 SUPER 15W40 SZE, CJ-4 API	670659	X			Location: PWMF, Storage Building 3, Cabinet 10100	Inhalation: Acute Toxicity: Minimally Toxic based on test data for structurally similar materials.
HYDRAULIC FLUID, NUTO H-C 32	323228	X			Location: PWMF, Processing Building, Room 110, Cabinet 36	Inhalation: Acute Toxicity: Based on assessment of the components it is minimally toxic.
THREADLOCK ADHESIVE, LOCTITE 242	329117	X			Location: PWMF, Storage Building 3, Cabinet 10100	Inhalation of vapors or mists may be irritating to the respiratory system. Skin and eye irritation

Hazardous Material	CAT ID	Toxic	Oxidizing/ Reactive	Asphyxiant	Quantity/Location	Remarks
Welding cover gas (92% Argon + 8% Carbon Dioxide)				X	Stored in the gas bottle storage room, Processing Building. This room is located in the northwest corner of the workshop and is only accessible from the outside.	Acute exposure to welding fume and gases can result in eye, nose and throat irritation, dizziness and nausea. However, PWMF technicians operate the weld machine remotely from the welding control room. Both the Workshop (Room 117) and the Welding Platform (Room 211) have active ventilation to avoid accumulation of fume and gas levels.
Helium				X	Stored in the gas bottle storage room, Processing Building. This room is located in the northwest corner of the workshop and is only accessible from the outside.	Helium bottles are stored in a separate room. Strict safety procedures and processes are in place for storage and handling of the hazardous chemicals.

5.4.4 Malfunctions/Accidents during DSC Storage

5.4.4.1 Dry Storage Container 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 lid closure weld is a groove weld between the base plate of the lid and the perimeter flange of the base. After the weld has been completed and cooled, a PAUT system is used for the inspection of the DSC lid-to-base seal-weld. 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 the DSCs during storage, it is concluded that random weld failures are not a credible event. This event is screened out.

5.4.4.2 Dry Storage Container Drop during Transfer to Storage

It is assumed that 119 existing DSCs stored in SB3 will be moved to the adjacent SB4 in order to accommodate the newly transferred 100 lower-fuel-age DSCs in SB3. Then 19 DSCs will be moved back from SB4 to SB3 and placed between the lower-fuel-age DSCs and the aisle, see Figure 5-2. It is further assumed, that the 238 (119+100+19) DSC movements in SB3 and between SB3 and SB4 will be performed within one year.

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 within the Phase II storage buildings, SB3 and SB4, 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. Failure of one independent mechanical locking mechanism is 1.0×10^{-4} events/year [1].

The postulated frequency of both mechanical locking mechanisms failing simultaneously, while carrying a loaded DSC within the Phase II storage buildings would be:

$$(1.0 \times 10^{-4}) \times (1.0 \times 10^{-4}) \times 238 = 2.38 \times 10^{-6} \text{ events per year}$$

This value is greater than the cut-off frequency of 10^{-6} events per year.

Given that the Transporter is equipped with front and rear bumper emergency stops or sensors, and taking into account the low-lift height of the DSC while in the Transporter and that the loaded DSC has been seal-welded at this stage of the process, no releases would result from this scenario [1]. This event is screened out.

5.4.4.3 Transporter Collision with a Dry Storage Container or another Transporter

Operator error during the Transporter operations could result in a collision with a loaded DSC on the floor of the SB3 or with another Transporter within SB3.

It is assumed that a maximum of 100 lower-fuel-age DSCs are transferred to SB3 and 119 existing DSCs are re-arranged/moved to accommodate the newly arrived lower-fuel-age DSCs resulting in 238 DSC movements as discussed in Section 5.4.4.2. The 238 movements are assumed to be performed within a single year. The postulated frequency of a DSC drop or collision event within the DSC storage building due to operator error (10^{-3} per movement) would be:

$$10^{-3} \times 238 = 2.38 \times 10^{-1} \text{ events/year.}$$

This event cannot be screened out based on frequency. However, 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 [1]. There would be no public or occupational dose consequences as a result of this event. This event is screened out.

5.4.4.4 DSC Storage Building 3 Fire

Storage Building 3 is of non-combustible construction with a reinforced slab-on-grade, steel structure, and concrete exterior walls. 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 [21]. The design basis fire scenario postulated an oil spill and a rubber tire fire without any fire mitigation measures.

The safe separation distances between the transporter involved in the fire and other targets/transient materials was assessed and it was determined that the combustible materials in Storage Building 3 were located outside the 5.3 m horizontal safe separation distance from the DSC Transporter parking area. As a result, transient materials located along the SB3 walls were not expected to be affected by a Transporter fire.

For the design basis fire, the quantitative analysis was performed without crediting the fire detection and suppression systems. Then the fire scenario was assessed crediting the available fire detection, suppression and fire-fighting. The FHA determined that the Emergency Response Team would provide suppression within 23 minutes after ignition and the fire would be ended within 30 minutes from ignition. The fire scenarios were used to calculate the impact of the design basis fire on the DSC, structural steel trusses and adjacent DSC transporters.

Due to the fire detection and alarm systems in place in all three storage buildings, and the expected prompt arrival of the emergency response personnel, the fire would be of short duration and localized.

The effects of a fire could potentially increase the temperature of the DSC and the used fuel bundles inside the container. It is to be noted that no thermal analysis was conducted to investigate the effects of an external fire on the DSC containing 6 year decayed used fuel; however a thermal analysis was conducted to investigate the heating and cooling process of a DSC storing 10 year decayed used fuel during and after a fire with the DSC on the Transporter, refer to Section E4.5.3 of Reference [21]. It was assumed that the DSC was subjected to a steady state of 800°C fire for 30 minutes and then cooled naturally for 48 hours. A very slow and gradual rise of the fluid temperature in the DSC cavity was experienced after the cessation of the fire. It was found that the maximum fuel sheath temperature will rise to 143°C, and the maximum internal pressure would be 99.47 kPa(a), with the maximum temperature rise of 17.13°C. As previously noted, no specific fire scenario for the 6 year decayed used fuel has been performed, however it can be concluded that a similar temperature rise could be expected considering a similar fire scenario and DSC design characteristics.

Section 5.5 details the fuel sheath temperature of the 6 year decayed used fuel was predicted to be between 175°C and 265°C[18], depending on the operating pressure in the DSC cavity. Based on the maximum 6 year decayed used fuel sheath temperature predictions and the fire-induced temperature increase it is expected that the maximum temperature rise in the DSC cavity to be less than the maximum allowed sheath temperature of 300°C.

Given the large thermal inertia of the DSC and the limited duration of the event, the fire scenario involving a DSC within SB3 would not breach the containment and will not result in radioactive emissions or radiological release. This hazard is therefore screened out.

5.4.4.5 Tornado

A DSC can resist overturning in tornadoes with winds of up to 425 km/h [23]. This scenario considers the DSC to be subject to the full force of the horizontal wind and ignores the storage building structures.

The effect of tornado missiles has been evaluated on a clamped DSC and it was concluded 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 tornado missiles [3]. It is also expected that the wind associated with a DBT will not overturn a DSC loaded with used fuel, which has a safety factor greater than 5 against overturning due to tornado winds [3]. Based on these findings, it can be concluded that a seal-welded DSC will not overturn under the impact of postulated tornado missiles/ severe wind load and the containment will not be breached as a result of a tornado-generated missiles.

Based on the design requirements [18], the DSC, while in storage, will withstand tornado generated missile impacts, strong winds and storage building structural failure or collapse without loss of shielding or containment. Therefore, it is expected that a tornado would result in no releases from DSC in storage and there would be no public or occupational dose consequences. This event is screened out.

5.4.4.6 Thunderstorms

Thunderstorms can potentially involve lightning striking the DSC storage building.

Based on the design requirements [18], the DSC was designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions, during storage. As severe atmospheric conditions are within the design basis of the DSC, this event is screened out.

5.4.4.7 Flooding due to Runoff

Water entry originating from a PMP into SB3 is possible, however, the consequences are assumed to be negligible. The DSCs are seal-welded and they are designed to tolerate water immersion at 2 MPa [18], so the temporary waters of the PMP flooding does not represent radiological safety concern. This event is screened out.

[18] The 265°C corresponds to 0.01 kPa(a) (vacuum pressure). The DSC is not stored under total vacuum, therefore it is unlikely that the fuel sheath's temperature will be 265°C during storage.

5.4.4.8 Earthquake

The DSC has a safety factor of 7 against overturning and 4 against sliding under the loads described for the earthquake scenario using the Pickering 'B' seismic ground response spectra and accelerations of 0.0625g horizontal and 0.05g vertical PGA [23]. The structure of the container is adequately strong to ensure the integrity of the DSC in case of an earthquake with the above parameters.

The DSC, by design is required to withstand a storage building collapse without loss of shielding or containment [18].

It can be concluded that an earthquake would result in no releases from the seal-welded DSCs stored in SB3 and there would be no public or occupational dose consequences. This event is screened out.

5.4.4.9 Toxic Materials stored in Storage Building 3

Table 5-8 shows that the only toxic chemical stored in Storage Building 3 that can be fatal if enters the airways based on its MSDS is the HYVOLT II transformer oil. 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 Transporter operates at a low speed within the SB3. Even if operator illness were to result in the Transporter collision with a container in storage, a release of radioactivity from a DSC is not expected given the design of the DSC and that the container has already been seal-welded at this stage of the process. This event is screened out.

5.4.4.10 Aircraft Crash

The probability of an aircraft strike is proportional to the target area. The aircraft crash frequency calculated for the Phase II site, which includes Storage Buildings SB3 and SB4, is 2.92×10^{-7} events/year, which is below the cut-off frequency of 10^{-6}, therefore this event is screened out.

In addition, a total aircraft crash frequency was calculated for the PWMF where DSCs and DSMs are stored/processed and it is presented in Appendix C.

5.4.5 Bounding Scenarios

5.4.5.1 DSC during Transport

There are two events related to DSC on-site transfer that have been screened in:

- Transporter failure during on-site transfer of the DSC between the station IFB to the Processing Building
- Drop of the DSC

For the transporter failure, release from four damaged fuel elements is assumed as a result of the event, whereas for the DSC drop, 100% of the fuel is assumed failed in both scenarios. Therefore, the DSC drop is deemed to be the bounding credible scenario.

5.4.5.2 DSC Bounding Event during Processing

There are two events related to DSC processing that have been screened in:

- DSC drop
- Earthquake

For the DSC drop scenario it is assumed that 30% of the fuel fails and the containment is impaired.

For the earthquake scenario, it is postulated that the processing building collapses onto an unclamped and un-welded DSC, without the fuel being exposed, but lid damage can result in some airborne release.

The impact force to the fuel is considered larger when the loaded DSC is dropped compared to the case where an object falls onto the DSC. Based on the radiological consequences, the DSC drop is assessed to be the bounding credible scenario.

5.4.5.3 DSC during Storage

All of the events related to DSC storage were screened out and therefore there is no bounding scenario.

5.4.6 Inventory and Releases

The PWMF UFDS facility stores used fuel bundles from the PNGS. The PNGS use 28-element CANDU fuel bundles. Approximately 3000 fuel bundles are discharged each year from each of the reactors at PNGS [1]. After a minimum of 6 years of cooling in the Pickering irradiated fuel bays, fuel bundles may be transferred to DSCs for interim dry storage.

For the purpose of future operation of the PWMF, the current safety assessment considers the storage of up to 100 DSCs containing 6 year decayed used fuel. The description of a used fuel bundle is given in Table 4-1.

Under normal operating conditions, no significant airborne emissions are expected from the DSCs because the uranium dioxide matrix, the used fuel sheath, and the seal-weld provide multiple barriers towards preventing the release of radioactive materials. While no significant releases are expected from DSCs under normal operating conditions, small quantities of fixed surface contamination may become airborne during welding operations [1].

In the event that a used fuel bundle should become damaged during PWMF operations, the only significant radionuclide species that are volatile and available for release are tritium, carbon-14, and krypton-85. For a fuel element damaged under abnormal operating conditions, it is postulated that the free inventory of tritium, carbon-14, 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 free inventory is released following a postulated malfunction and accident scenario.

For a fuel bundle which has cooled for a period of 6 years, the releases per failed fuel element are listed in Table 5-9.

The bounding scenario (see Section 5.4.5) for the DSC storage in SB3 is the drop of a DSC during transfer, which conservatively assumes to result in all (10,752) fuel elements in the DSC to fail.

Table 5-9: Activity Released per Failed Fuel Element

Nuclide	Bq released per failed fuel element
Krypton-85	1.10E+09
Tritium (HTO)	1.79E+08
Carbon-14	1.46E+04
Note: The Bq released was calculated based on the following values: + Fraction of inventory in gap = 0.0095 + Fraction of inventory in grain boundary = 0.123 + Percent of gap inventory being released = 100% + Percent of grain boundary inventory being released = 10%	

5.4.7 Public Dose

The bounding scenario identified in Section 5.4.5 for dose consequence to the public is the DSC drop which conservatively assumes to result in all (10,752) fuel elements in the DSC to fail. Using the ADDAM code (see Appendix D) the 95th percentile doses were calculated for the scenario where the DSC drop occurs near Phase I or Phase II and are summarized in Table 5-10.[19] The 95th percentile individual dose following the limiting malfunction / accident scenario is:

Release from the Phase I site:

- 6.02E-03 mSv (adult), which is 0.60% of the 1 mSv dose limit.
- 7.28E-03 mSv (infant), which is 0.73% of the 1 mSv dose limit.

The limiting 95th percentile doses occur at the lakeside boundary where the fishermen are assumed to be located.

Release from the Phase II site:

- 7.38E-03 mSv (adult), which is 0.74% of the 1 mSv dose limit.
- 9.00E-03 mSv (infant), which is 0.90% of the 1 mSv dose limit.

The limiting 95th percentile doses occur at the East landside boundary (receptor B_E in Figure 5-5).

19 The ADDAM input files (Phase I and Phase II sites, without buildings) used in the analysis documented in Reference [3] were used as input for the current calculations. All meteorological data, radionuclide data, and site descriptions in the input files remain unchanged from the analysis in Reference [3]; the release source term was updated as per Section 5.4.6.

Table 5-11 shows the public and occupational dose consequences due to those malfunctions and accidents deemed credible (i.e., events with frequency of occurrence $\geq 10^{-6}$ events per year) during DSC on-site transfer.

The bounding dose consequences during this stage of the dry storage process are associated with the drop of a DSC during on-site transfer. Although fuel sheath failure is not expected to result from a DSC drop from the low lift height of the 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. The free inventory of tritium, krypton-85, and carbon-14 in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring that DSCs being transferred from the PWMF Phase I site to the PWMF Phase II site are already seal-welded, it is assumed that these radionuclides are released at once into the environment.

Assuming that this event occurs at or near the PWMF Phase II site, the limiting individual dose to the public (95th percentile) was calculated to be 7.38 µSv for an adult and 9.00 µSv for an infant at the Pickering site boundary [3]. The dose to a NEW would be 5.92 mSv (see Section 5.4.8).
Table 5-12 shows the public and occupational dose consequences for postulated malfunctions / accidents which occur during DSC processing. The dose consequence for DSCs during storage in SB3 is shown in Table 5-13.

Table 5-10: 95th Percentile Public Dose following the Limiting Malfunction Accident Scenario

	Phase I		Phase II	
Location	**Adult (µSv)**	**Infant (µSv)**	**Adult (µSv)**	**Infant (µSv)**
B_N	0.00	0.00	0.00	0.00
B_NNE	0.33	0.40	0.47	0.57
B_NE	0.42	0.54	1.22	1.47
B_ENE	0.27	0.32	1.37	1.66
B_E	1.18	1.41	7.38	9.00
B_ESE-Lake	1.35	1.56	1.32	1.43
B_SE-Lake	2.57	3.58	1.11	1.34
B_SSE-Lake	1.78	2.13	0.46	0.57
B_S-Lake	5.72	6.93	1.52	1.83
B_SSW-Lake	0.68	0.94	0.34	0.36
B_SW-Lake	0.00	0.00	0.00	0.00
B_WSW-Lake	0.00	0.00	0.00	0.00
B_W-Lake	0.10	0.12	0.07	0.08
B_WNW	0.00	0.00	0.00	0.00
B_NW	0.00	0.00	0.00	0.00
B_NNW	0.00	0.00	0.00	0.00
IND	0.30	0.36	0.53	0.63
Fisher	6.02	7.28	1.48	1.79
Beach	0.00	0.00	0.00	0.00
UR_WNW	0.00	0.00	0.00	0.00
UR_NW	0.00	0.00	0.00	0.00
UR_NNW	0.00	0.00	0.00	0.00
C2	0.07	0.08	0.09	0.11
Dairy Farm	0.02	0.02	0.02	0.02
Farm	0.03	0.03	0.03	0.04
Ajax WSP	0.05	0.06	0.05	0.06
Max. Dose on Landside	1.18	1.41	7.38	9.00
Location	B_E	B_E	B_E	B_E
Max. Dose on Lake	6.02	7.28	1.52	1.83
Location	Fisher	Fisher	B_S-Lake	B_S-Lake

Notes:
Doses shown correspond to the most limiting accident scenario at PWMF, which is a DSC event during transfer resulting in 100% failed fuel elements inside the DSC.

Table 5-11: Postulated Malfunctions / Accidents during DSC On-Site Transfer

Malfunction or accident	Potential for occurrence	Potential maximum dose consequence to the public (mSv)		Potential maximum individual occupational dose consequence (mSv)
		Adult	Infant	
Transporter failure between the station IFB and Processing Building	Credible	<6.02E-03	<7.28E-03	<5.9
Transporter failure between Processing Building and SB3	Credible	<6.02E-03	<7.28E-03	<5.9
Transporter operator health-related emergency	Credible	<6.02E-03	<7.28E-03	<5.9
DSC drop during on-site transfer from IFB to DSC Processing Building	Credible	6.02E-03	7.28E-03	5.9
DSC drop during on-site transfer from Phase I to Phase II site	Credible	7.38E-03	9.00E-03	5.9
Fire	Credible	0	0	0
Adverse road conditions	Credible	<7.38E-03	<9.00E-03	<5.9
Earthquake[a]	Credible	<7.38E-03	<9.00E-03	<5.9
Tornado[a]	Incredible[b]	-	-	-
Thunderstorms[a]	Credible	0	0	0
Flooding[a]	Credible	0	0	0
Explosions along transfer route	Incredible	-	-	-
Turbine missile strike	Incredible	-	-	-
Aircraft crash	Incredible	-	-	-
Toxic gas releases – chlorine from Ajax water treatment plant	Credible	<7.38E-03	<9.00E-03	<5.9
Soil failure/slope instability	Credible	<7.38E-03	<9.00E-03	<5.9

Notes:
a) For common cause events, the potential maximum dose consequence as reported is related to DSC on-site transfer only and does not include contribution from other sources on site.
b) The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.

Table 5-12: Postulated Malfunctions / Accidents during DSC Processing

Malfunction or accident	Potential for occurrence	Potential maximum dose consequence to the public (mSv)		Potential maximum individual occupational dose consequence (mSv)
		Adult	Infant	
Drop of a DSC during handling	Credible	<6.02E-03	<7.28E-03	<5.9
Equipment drop onto a DSC	Credible	<6.02E-03	<7.28E-03	<5.9
DSC collision during craning	Credible	<6.02E-03	<7.28E-03	<5.9
Transporter collision with a loaded DSC or other transporter	Credible	<6.02E-03	<7.28E-03	<5.9
Equipment collision with loaded DSC during craning	Credible	<6.02E-03	<7.28E-03	<5.9
Processing Building Fire	Credible	0	0	0
Earthquake[a]	Credible	0	0	0
Tornado[a]	Incredible[b]	-	-	-
Thunderstorms[a]	Credible	0	0	0
Flooding[a]	Credible	0	0	0
Turbine missile strike	Incredible	-	-	-
Aircraft crash	Incredible	-	-	-
Release of oxidizing, toxic, corrosive liquids stored in the PB	Credible	<6.02E-03	<7.28E-03	<5.9

Notes:
a) For common cause events, the potential maximum dose consequence as reported is related to DSC processing only and does not include contribution from other sources on site.
b) The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.

Table 5-13: Postulated Malfunctions/Accidents during DSC Storage in SB3

Malfunction or accident	Potential for occurrence	Potential maximum dose consequence to the public (mSv)		Potential maximum individual occupational dose consequence (mSv)
		Adult	Infant	
Seal weld failure during storage	Incredible[a]	-	-	-
DSC drop during transfer to storage	Credible	<7.38E-03	<9.00E-03	<5.9
Transporter collision with a DSC or another transporter	Credible	<7.38E-03	<9.00E-03	<5.9
Fire	Credible	0	0	0
Tornado[b]	Credible	0	0	0
Thunderstorms[b]	Credible	0	0	0
Flooding[b]	Credible	0	0	0
Earthquake[b]	Credible	<7.38E-03	<9.00E-03	<5.9
Toxic Materials in SB3	Credible	<7.38E-03	<9.00E-03	<5.9
Aircraft Crash	Incredible	-	-	-

Notes:
a) The term incredible is used for those events with frequency of occurrence below 10^{-6} events per year.
b) For common cause events, the potential maximum dose consequence as reported is related to DSC storage in SB3 only and does not include contribution from other sources on site.

5.4.8 Worker Dose

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 leave the accident location under emergency back-out conditions is assumed to be 120 seconds [3]. Normal radiation protection procedures would have the worker leave the building or general area at the time of the accident to allow any airborne particulate levels to subside before any other activities, such as clean-up if needed, can be initiated.

The dose to workers following a postulated accident scenario was calculated by including the contributions from the inhalation and cloudshine exposure pathways. The release and mixing of radionuclides are assumed to be instantaneous.

Dose from inhalation:

$$D_{inhalation} = \sum_{n=1} \left(R_n \times BR \times sk_{a,n} \times DCF_{inhalation,n} \right) \times T/V$$

where:

- $D_{inhalation}$ = worker dose from inhalation (Sv);

- R_n = release amount (Bq) during the exposure time (krypton-85 = 1.10x10^9 Bq, tritium = 1.79x10^8 Bq, and carbon-14 = 1.47x10^4 Bq);
- BR = worker breathing rate (4.17X10^{-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 (tritium = 1.8x10^{-11}Sv/Bq, carbon-14 = 6.5x10^{-12} Sv/Bq);
- T = exposure time (120 s); and
- V = contaminated cloud volume (500 m^3).

Dose from cloudshine:

$$D_{cloudshine} = \sum_{n=1} \left(R_n \times DCF_{cloudshine,n}\right) \times T/V$$

where:

- $D_{cloudshine}$ = worker dose from cloudshine (krypton-85 = 6.75x10^{-8} Sv, carbon-14 = 9.13x10^{-15} Sv); and
- $DCF_{cloudshine,n}$ = cloudshine dose coefficient krypton-85 = 2.55x10^{-16} Sv.m^3.Bq^{-1}.s^{-1}, carbon-14 = 2.60x10^{-18} Sv.m^3.Bq^{-1}.s^{-1}) .

The dose to the worker following a DSC event during transport scenario where 100% of the fuel elements in the DSC are assumed failed is estimated to be 5.92 mSv, which came from the following contributors[20]:

- 5.19 mSv from inhalation (including skin absorption) of tritium;
- 0.73 mSv from cloudshine of krypton-85.

Carbon-14 releases from the DSC makes an insignificant contribution to the dose (<0.1%).

The calculated worker dose is approximately 12% of the 50 mSv worker dose limit.

5.5 Fuel Sheath Temperature

As identified in Section 2.0, a qualitative assessment of the fuel sheath temperature for 6 year decayed used fuel is provided in this Section. One of the key requirements for the DSC is to maintain a fuel sheath temperature below the allowable maximum fuel sheath temperature under dry storage conditions in a DSC. A detailed discussion of the potential for oxidation of fuel stored in the Pickering DSCs is given in Appendix H of the 1998 issue of the PWMF Safety Report, concluding that when the used fuel is stored in a helium atmosphere, temperatures of up to 300°C can be considered safe for the planned storage period. That is, CANDU fuel can be stored in the Pickering DSCs without rick of fuel sheath temperatures exceeding 300°C.

20 Groundshine contributions are not applicable for the current scenario since tritium and carbon-14 are beta emitters and krypton-85 does not deposit on the ground.

Two different thermal assessments, estimating the dry storage sheath temperature for the 6 years old used fuel, were previously performed and documented in Appendix A of the 1998 issue of the PWMF Safety Report (92896-SR-01320-10002 R000) and Reference [34]. However, there was a difference of about 100°C between the maximum sheath temperatures predicted in the above two analyses as a result of the different conditions assumed in these assessments.

An independent review of the two previous assessments, including independent Computational Fluid Dynamics simulations using commercial code ANSYS CFX 17.2, was performed and documented in Reference [35].

As discussed in Reference [35], the first assessment documented in Appendix A of the 1998 issue of the PWMF Safety Report was conducted using helium at 145 kPa(a) operating pressure for the DSC cavity and resulted in a maximum sheath temperature prediction of 175°C. On the other hand, the second assessment documented in Reference [34] was conducted using near vacuum (0.01 kPa(a)) which almost eliminated natural convection in the DSC cavity and resulted in a maximum sheath temperature prediction of 265°C. The difference between the temperatures obtained by the two previous thermal assessments was mainly due to the different internal pressures used for their simulations as confirmed by the independent simulations performed in Reference [35].

The calculated fuel sheath temperatures for the storage of 6 year decayed used fuel are found to be less than 300°C even at near vacuum conditions. Therefore, no issues with respect to fuel sheath temperature are expected for the storage of DSCs containing 6 year decayed used fuel.

6.0 CONCLUSIONS

The current report summarizes the safety assessment of the storage of DSCs containing 6 year decayed used fuel in SB3 at the PWMF and any impact on common-mode accidents that impact the entire PWMF site. Dose consequences under normal operation and malfunction/accident conditions were assessed.

The assessment demonstrates compliance with the radiation safety requirements during normal operation of the PWMF when SB3 is in service. With the addition of the 100 DSCs containing 6 year decayed used fuel, the annual public dose estimates have increased compared to that of the existing PWMF configuration. The maximum annual dose to individual member of the public with the addition of these 100 DSCs is still a small percentage of the 1 mSv limit. With respect to malfunction / accident scenarios, the estimated bounding doses to members of the public (see Table 5-10) are less than the 1 mSv acceptance criterion. The dose to workers following a postulated accident scenario is found to be much less than the 50 mSv limit (see Section 5.4.8). It is concluded that the dose consequences to workers and members of the public as a result of credible postulated malfunction / accident scenarios meet the acceptance criteria outlined in Section 5.2.

The review of the fuel sheath temperature did not indicate any concerns with the previously conducted analyses and the fuel sheath temperature is not expected to exceed 300°C. Therefore, there are no issues with respect to fuel sheath temperature expected for the storage of DSCs containing 6 year decayed used fuel.

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8.0 ACRONYMS

AIFB	Auxiliary Irradiated Fuel Bay
ALARA	As Low As Reasonably Achievable
BLEVE	Boiling Liquid Expanding Vapour Explosion
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DNGS	Darlington Nuclear Generating Station
DSC	Dry Storage Container
DSM	Dry Storage Module
ERT	Emergency Response Team
FHA	Fire Hazard Analysis
IFB	Irradiated Fuel Bay
MSDS	Material Safety Datasheet
NBCC	National Building Code of Canada
NEW	Nuclear Energy Worker
NFCC	National Fire Code Canada
NGS	Nuclear Generating Station
OHSA	Occupational Health and Safety Act
OPG	Ontario Power Generation
PAUT	Phased Array Ultrasonic Testing
PB	Processing Building
PBIFB	Pickering B Primary Irradiated Fuel Bay
PGA	Peak Ground Acceleration
PMP	Probable Maximum Precipitation
PNGS	Pickering Nuclear Generating Station
PSA	Probabilistic Safety Assessment
PWMF	Pickering Waste Management Facility
RCS	Retube Components Storage
SB	Storage Building
SDV	Screening Distance Value
SSC	Structure, System, and Component
UFDS	Used Fuel Dry Storage
VCE	Vapour Cloud Explosion
WWMF	Western Waste Management Facility

Appendix A PWMF 6 YEAR DECAYED USED FUEL BUNDLE SOURCE TERM

The radionuclide inventory for the PWMF reference used fuel bundle was generated as part of the work for the Safety Assessment Update (see discussion in Appendix B of Reference [3). The generation of the radionuclide inventory in Reference [3] involved the use of a 28-element Pickering-specific cross-section library and a bundle-wise calculation methodology. The cross-section library has since been revised in Reference [36]. Further, Reference [36] recommends using a ring-wise methodology to determining the radionuclide inventory, where calculations are performed for each ring of the fuel bundle and then summed to obtain the bundle-wise quantities.

There are no significant differences in the radionuclide inventory of tritium, carbon-14, and krypton-85 (the radionuclides of interest with respect to airborne releases following postulated credible accident scenarios involving PWMF DSCs) when calculated using the revised cross-section library compared to the inventory listed in Reference [3]. However, following the recommendation identified in Reference [36] for using the ring-wise calculation methodology, the summed inventory of carbon-14 is decreased. Further discussion is provided in Reference [36].

The revised cross-section library generated in Reference [36] was used to generate the radionuclide inventory for a 28-element Pickering-specific fuel bundle following the ring-wise calculation methodology. The inventory was generated for various decay times using the reference fuel bundle properties [9], these include:

- Pickering-type 28-element fuel bundle;
- Mass of uranium per bundle is 20.2 kg;
- Exit burnup of 230 MWh/kgU; and
- Fuel bundle power of 373 kW (fission) representing the core average of 100% full power operations.

The radionuclide inventories for a PWMF used fuel bundle at reference burnup (230 MWh/kgU) are presented in Table A-1 for a decay time of 6 year. For comparison, the updated 10 year decayed used fuel bundle inventory is included.

Table A-1 PWMF Used Fuel Bundle Radionuclide Inventory

Bq/bundle					
Nuclide	6 Year Decayed	10 Year Decayed	Nuclide	6 Year Decayed	10 Year Decayed
Ac-225	1.61E+00	2.30E+00	Os-185	4.24E+00	8.47E-05
Ac-226	0.00E+00	0.00E+00	Os-186	1.10E-04	1.10E-04
Ac-227	1.39E+02	2.47E+02	Os-189m	0.00E+00	0.00E+00
Ac-228	4.29E-02	5.90E-02	Os-191	0.00E+00	0.00E+00
Ag-105	6.11E-14	0.00E+00	Os-194	8.67E+04	5.46E+04
Ag-106m	0.00E+00	0.00E+00	P-32	1.94E+03	1.90E+03
Ag-108	4.00E+05	3.98E+05	P-33	2.54E-18	0.00E+00
Ag-108m	4.60E+06	4.57E+06	Pa-230	0.00E+00	0.00E+00
Ag-109m	1.07E+07	1.20E+06	Pa-231	9.28E+02	1.15E+03
Ag-110	1.66E+07	2.87E+05	Pa-232	9.09E-02	9.09E-02
Ag-110m	1.22E+09	2.11E+07	Pa-233	2.16E+07	2.20E+07
Ag-111	0.00E+00	0.00E+00	Pa-234	3.95E+05	3.95E+05
Al-26	5.79E+00	5.79E+00	Pa-234m	2.47E+08	2.47E+08
Am-241	2.05E+11	3.04E+11	Pb-202	4.19E-02	4.19E-02
Am-242	3.00E+08	2.94E+08	Pb-204	1.43E-05	1.44E-05
Am-242m	3.02E+08	2.96E+08	Pb-205	2.36E+02	2.36E+02
Am-243	1.04E+09	1.04E+09	Pb-209	1.61E+00	2.30E+00
Am-244	1.95E-06	1.43E-06	Pb-210	5.69E+00	9.08E+00
Am-245	2.44E-05	1.03E-06	Pb-211	1.39E+02	2.47E+02
Am-246m	1.08E-08	1.08E-08	Pb-212	4.00E+05	5.38E+05
Ar-37	5.46E-09	0.00E+00	Pb-214	2.11E+01	4.96E+01
Ar-39	5.24E+07	5.18E+07	Pd-103	0.00E+00	0.00E+00
Ar-42	2.59E+01	2.38E+01	Pd-107	3.00E+07	3.00E+07
As-73	8.80E-04	2.93E-09	Pm-143	7.58E-01	1.66E-02
As-74	0.00E+00	0.00E+00	Pm-144	4.20E+01	2.58E+00
At-217	1.61E+00	2.30E+00	Pm-145	2.56E+06	2.21E+06
At-218	4.21E-03	9.92E-03	Pm-146	4.39E+06	2.66E+06
Au-194	4.60E-04	4.57E-04	Pm-147	1.20E+13	4.17E+12
Au-195	1.54E+00	6.69E-03	Pm-148	2.49E-05	0.00E+00
Au-196	0.00E+00	0.00E+00	Pm-148m	5.15E-04	1.15E-14
Ba-131	0.00E+00	0.00E+00	Po-208	3.86E-01	1.48E-01
Ba-133	2.81E+08	2.15E+08	Po-209	6.83E+01	6.65E+01
Ba-136m	0.00E+00	0.00E+00	Po-210	3.67E+04	3.34E+01
Ba-137m	1.91E+13	1.74E+13	Po-211	3.84E-01	6.82E-01
Ba-140	0.00E+00	0.00E+00	Po-212	2.56E+05	3.45E+05

Bq/bundle					
Nuclide	6 Year Decayed	10 Year Decayed	Nuclide	6 Year Decayed	10 Year Decayed
Be-10	3.71E+03	3.71E+03	Po-213	1.57E+00	2.25E+00
Bi-205	0.00E+00	0.00E+00	Po-214	2.11E+01	4.96E+01
Bi-207	3.97E+03	3.63E+03	Po-215	1.39E+02	2.47E+02
Bi-208	6.38E+02	6.38E+02	Po-216	4.00E+05	5.38E+05
Bi-209	1.35E-06	1.35E-06	Po-218	2.11E+01	4.96E+01
Bi-210	5.69E+00	9.08E+00	Pr-143	0.00E+00	0.00E+00
Bi-210m	3.37E+02	3.37E+02	Pr-144	1.77E+12	5.06E+10
Bi-211	1.39E+02	2.47E+02	Pr-144m	1.69E+10	4.83E+08
Bi-212	4.00E+05	5.38E+05	Pt-188	0.00E+00	0.00E+00
Bi-213	1.61E+00	2.30E+00	Pt-190	2.21E-04	2.21E-04
Bi-214	2.11E+01	4.96E+01	Pt-193	1.31E+08	1.24E+08
Bk-247	4.26E-10	4.26E-10	Pu-236	3.16E+06	1.20E+06
Bk-248	1.95E-06	1.43E-06	Pu-237	2.03E-07	4.67E-17
Bk-249	1.68E+00	7.11E-02	Pu-238	9.92E+10	9.62E+10
Bk-250	1.26E-07	7.88E-09	Pu-239	1.25E+11	1.25E+11
Bk-251	0.00E+00	0.00E+00	Pu-240	2.24E+11	2.24E+11
C-14	4.10E+08	4.09E+08	Pu-241	1.73E+13	1.42E+13
Ca-41	6.07E+06	6.07E+06	Pu-242	3.30E+08	3.30E+08
Ca-45	2.73E+06	5.39E+03	Pu-243	1.69E-01	1.69E-01
Ca-48	5.40E-08	5.40E-08	Pu-244	1.80E+01	1.80E+01
Cd-109	1.07E+07	1.20E+06	Pu-246	1.08E-08	1.08E-08
Cd-113	1.09E-05	1.09E-05	Ra-222	0.00E+00	0.00E+00
Cd-113m	5.82E+07	4.78E+07	Ra-223	1.39E+02	2.47E+02
Cd-115m	3.95E-04	5.31E-14	Ra-224	4.00E+05	5.38E+05
Cd-116	1.29E-07	1.29E-07	Ra-225	1.61E+00	2.30E+00
Ce-139	1.95E+04	1.24E+01	Ra-226	2.11E+01	4.96E+01
Ce-141	3.03E-06	8.94E-20	Ra-228	4.29E-02	5.90E-02
Ce-144	1.77E+12	5.06E+10	Rb-82	0.00E+00	0.00E+00
Cf-248	2.37E-07	1.14E-08	Rb-83	6.85E-02	5.41E-07
Cf-249	5.09E-01	5.09E-01	Rb-84	4.18E-13	0.00E+00
Cf-250	3.60E+00	2.91E+00	Rb-86	0.00E+00	0.00E+00
Cf-251	1.37E-02	1.37E-02	Rb-87	4.16E+03	4.16E+03
Cf-252	5.89E-01	2.06E-01	Re-183	2.07E-06	1.08E-12
Cf-253	0.00E+00	0.00E+00	Re-184	9.83E+00	2.45E-02
Cf-254	4.85E-14	2.50E-21	Re-184m	1.04E+01	2.60E-02
Cl-36	9.96E+06	9.96E+06	Re-186	1.55E+03	1.55E+03
Cm-240	0.00E+00	0.00E+00	Re-186m	1.55E+03	1.55E+03

Bq/bundle					
Nuclide	**6 Year Decayed**	**10 Year Decayed**	**Nuclide**	**6 Year Decayed**	**10 Year Decayed**
Cm-241	2.33E-15	0.00E+00	Re-187	3.95E+01	3.95E+01
Cm-242	5.62E+08	2.45E+08	Re-188	3.88E-01	1.93E-07
Cm-243	4.47E+08	4.06E+08	Rh-99	0.00E+00	0.00E+00
Cm-244	3.24E+10	2.78E+10	Rh-101	2.47E+03	1.07E+03
Cm-245	1.06E+06	1.06E+06	Rh-102	2.19E+05	1.70E+04
Cm-246	2.16E+05	2.15E+05	Rh-102m	1.19E+07	5.68E+06
Cm-247	1.69E-01	1.69E-01	Rh-103m	1.02E-02	5.60E-14
Cm-248	1.94E-01	1.94E-01	Rh-106	3.11E+12	2.04E+11
Cm-249	0.00E+00	0.00E+00	Rn-217	1.13E-04	1.61E-04
Cm-250	6.02E-08	6.02E-08	Rn-218	4.21E-06	9.92E-06
Co-56	7.60E-06	1.53E-11	Rn-219	1.39E+02	2.47E+02
Co-57	3.35E+05	8.07E+03	Rn-220	4.00E+05	5.38E+05
Co-58	8.67E+00	5.38E-06	Rn-222	2.11E+01	4.96E+01
Co-60	2.01E+09	1.19E+09	Ru-103	1.03E-02	6.37E-14
Co-60m	3.96E+01	3.96E+01	Ru-106	3.11E+12	2.04E+11
Cr-51	2.02E-12	0.00E+00	S-35	1.33E+03	1.25E-02
Cs-131	0.00E+00	0.00E+00	Sb-120m	0.00E+00	0.00E+00
Cs-132	0.00E+00	0.00E+00	Sb-124	1.08E+00	5.34E-08
Cs-134	1.99E+12	5.19E+11	Sb-125	5.94E+11	2.18E+11
Cs-135	4.01E+07	4.01E+07	Sb-126	8.54E+06	8.54E+06
Cs-136	0.00E+00	0.00E+00	Sb-126m	6.10E+07	6.10E+07
Cs-137	2.02E+13	1.84E+13	Sc-44	7.17E-08	6.85E-08
Dy-154	3.28E-08	3.28E-08	Sc-45m	5.19E+01	1.02E-01
Dy-159	1.45E+02	1.30E-01	Sc-46	1.01E+05	5.71E-01
Er-169	0.00E+00	0.00E+00	Se-75	2.66E+05	5.66E+01
Es-252	4.88E-11	5.71E-12	Se-79	1.54E+07	1.54E+07
Es-253	0.00E+00	0.00E+00	Si-32	1.94E+03	1.90E+03
Es-254	1.21E-07	3.06E-09	Sm-145	4.06E+05	2.07E+04
Es-255	2.11E-21	0.00E+00	Sm-146	1.44E+00	1.48E+00
Eu-147	0.00E+00	0.00E+00	Sm-147	1.43E+03	1.62E+03
Eu-148	1.53E-16	0.00E+00	Sm-148	4.28E-03	4.28E-03
Eu-149	2.33E-06	4.39E-11	Sm-151	4.49E+10	4.36E+10
Eu-150	5.06E+02	4.69E+02	Sn-113	3.85E+05	5.81E+01
Eu-152	6.65E+07	5.41E+07	Sn-117m	0.00E+00	0.00E+00
Eu-154	4.44E+11	3.22E+11	Sn-119m	2.50E+09	7.88E+07
Eu-155	3.14E+11	1.75E+11	Sn-121	4.31E+09	4.04E+09
Eu-156	0.00E+00	0.00E+00	Sn-121m	5.55E+09	5.21E+09

Bq/bundle					
Nuclide	6 Year Decayed	10 Year Decayed	Nuclide	6 Year Decayed	10 Year Decayed
Fe-55	4.44E+10	1.62E+10	Sn-123	5.44E+06	2.14E+03
Fe-59	2.83E-05	3.68E-15	Sn-125	0.00E+00	0.00E+00
Fe-60	3.96E+01	3.96E+01	Sn-126	6.10E+07	6.10E+07
Fr-221	1.61E+00	2.30E+00	Sr-82	0.00E+00	0.00E+00
Fr-223	1.92E+00	3.41E+00	Sr-85	4.26E-04	7.01E-11
Ga-68	6.52E-06	1.55E-07	Sr-89	2.51E+01	4.96E-08
Gd-148	5.94E-07	5.72E-07	Sr-90	1.28E+13	1.16E+13
Gd-149	0.00E+00	0.00E+00	Ta-178	0.00E+00	0.00E+00
Gd-150	1.19E-02	1.19E-02	Ta-179	4.76E+04	1.04E+04
Gd-151	3.35E-01	9.49E-05	Ta-182	5.27E+05	8.00E+02
Gd-152	1.47E-04	1.48E-04	Ta-183	0.00E+00	0.00E+00
Gd-153	2.42E+06	3.58E+04	Tb-157	3.17E+05	3.05E+05
Ge-68	6.52E-06	1.55E-07	Tb-158	8.99E+04	8.85E+04
Ge-71	0.00E+00	0.00E+00	Tb-160	1.08E+02	8.94E-05
Ge-73m	8.80E-04	2.93E-09	Tb-161	0.00E+00	0.00E+00
H-3	2.29E+11	1.83E+11	Tc-95	2.00E-09	0.00E+00
Hf-172	1.23E-01	2.79E-02	Tc-95m	5.08E-08	3.13E-15
Hf-174	6.42E-06	6.42E-06	Tc-97	1.13E+02	1.13E+02
Hf-175	7.48E+00	3.89E-06	Tc-97m	4.05E-02	5.94E-07
Hf-177m	1.35E+03	2.45E+00	Tc-98	2.31E+02	2.31E+02
Hf-181	9.91E-05	4.17E-15	Tc-99	3.20E+09	3.20E+09
Hf-182	7.23E+02	7.23E+02	Te-121	1.96E+02	4.11E-01
Hg-194	4.60E-04	4.57E-04	Te-121m	1.96E+02	4.10E-01
Hg-203	3.54E-05	1.29E-14	Te-123m	2.27E+03	4.65E-01
Hg-206	1.08E-07	1.73E-07	Te-125m	1.46E+11	5.33E+10
Ho-163	1.21E+04	1.21E+04	Te-127	2.00E+06	1.84E+02
Ho-166m	3.90E+05	3.89E+05	Te-127m	2.04E+06	1.88E+02
I-125	8.76E-08	3.45E-15	Te-128	7.16E-06	7.16E-06
I-126	0.00E+00	0.00E+00	Te-129	2.16E-07	0.00E+00
I-129	6.26E+06	6.26E+06	Te-129m	3.42E-07	2.78E-20
I-131	0.00E+00	0.00E+00	Th-226	0.00E+00	0.00E+00
In-113m	3.86E+05	5.81E+01	Th-227	1.37E+02	2.44E+02
In-114	4.61E-04	6.02E-13	Th-228	4.00E+05	5.38E+05
In-114m	4.77E-04	6.23E-13	Th-229	1.61E+00	2.30E+00
In-115	7.38E-03	7.38E-03	Th-230	1.30E+04	2.00E+04
In-115m	4.19E-08	0.00E+00	Th-231	2.60E+06	2.60E+06
Ir-188	0.00E+00	0.00E+00	Th-232	8.07E-02	8.86E-02

Bq/bundle					
Nuclide	6 Year Decayed	10 Year Decayed	Nuclide	6 Year Decayed	10 Year Decayed
Ir-189	0.00E+00	0.00E+00	Th-234	2.47E+08	2.47E+08
Ir-190	0.00E+00	0.00E+00	Ti-44	7.17E-08	6.85E-08
Ir-191m	0.00E+00	0.00E+00	Tl-202	4.19E-02	4.19E-02
Ir-192	7.68E+01	8.46E-05	Tl-204	9.66E+08	4.64E+08
Ir-193m	0.00E+00	0.00E+00	Tl-206	3.37E+02	3.37E+02
Ir-194	8.68E+04	5.46E+04	Tl-207	1.39E+02	2.47E+02
K-40	5.79E+02	5.79E+02	Tl-208	1.44E+05	1.93E+05
K-42	2.59E+01	2.38E+01	Tl-209	3.54E-02	5.06E-02
Kr-81	2.34E+04	2.34E+04	Tl-210	4.42E-03	1.04E-02
Kr-83m	5.10E-02	4.02E-07	Tm-167	0.00E+00	0.00E+00
Kr-85	1.42E+12	1.09E+12	Tm-168	1.76E-01	3.31E-06
La-137	1.73E+04	1.73E+04	Tm-170	1.09E+06	4.14E+02
La-138	1.93E-01	1.93E-01	Tm-171	6.00E+08	1.42E+08
La-140	0.00E+00	0.00E+00	U-230	0.00E+00	0.00E+00
Lu-172	1.23E-01	2.79E-02	U-232	5.38E+05	5.96E+05
Lu-172m	1.23E-01	2.79E-02	U-233	1.63E+03	2.01E+03
Lu-173	4.22E+03	5.57E+02	U-234	1.90E+08	1.91E+08
Lu-174	3.06E+05	1.33E+05	U-235	2.60E+06	2.60E+06
Lu-174m	2.80E+01	2.24E-02	U-236	3.96E+07	3.97E+07
Lu-176	4.06E-02	4.06E-02	U-237	4.24E+08	3.49E+08
Lu-177	3.83E+02	6.95E-01	U-238	2.47E+08	2.47E+08
Lu-177m	1.72E+03	3.11E+00	U-240	1.80E+01	1.80E+01
Mn-53	1.30E+01	1.30E+01	V-48	0.00E+00	0.00E+00
Mn-54	9.63E+07	3.75E+06	V-49	6.36E+04	2.96E+03
Mo-93	4.15E+05	4.15E+05	V-50	2.06E-06	2.06E-06
Mo-100	9.99E-05	9.99E-05	W-178	0.00E+00	0.00E+00
Na-22	3.25E+03	1.12E+03	W-180	1.60E-08	1.60E-08
Nb-91	6.48E+03	6.45E+03	W-181	1.22E+04	2.87E+00
Nb-91m	1.27E-05	7.55E-13	W-183m	0.00E+00	0.00E+00
Nb-92	3.42E+00	3.42E+00	W-185	1.02E+02	1.42E-04
Nb-92m	0.00E+00	0.00E+00	W-186	3.84E-08	3.84E-08
Nb-93m	1.84E+08	2.15E+08	W-188	3.84E-01	1.91E-07
Nb-94	1.17E+07	1.17E+07	Xe-127	7.70E-13	0.00E+00
Nb-95	6.53E+04	8.83E-03	Xe-129m	0.00E+00	0.00E+00
Nb-95m	3.39E+02	4.58E-05	Xe-131m	0.00E+00	0.00E+00
Nd-144	2.99E-01	2.99E-01	Xe-133	0.00E+00	0.00E+00
Nd-147	0.00E+00	0.00E+00	Y-88	7.85E+01	5.89E-03

Bq/bundle					
Nuclide	**6 Year Decayed**	**10 Year Decayed**	**Nuclide**	**6 Year Decayed**	**10 Year Decayed**
Nd-150	1.19E-05	1.19E-05	Y-89m	2.42E-03	4.78E-12
Ni-59	2.56E+07	2.56E+07	Y-90	1.28E+13	1.17E+13
Ni-63	3.32E+09	3.23E+09	Y-91	2.05E+03	6.24E-05
Np-235	2.19E+03	1.70E+02	Yb-169	1.68E-15	0.00E+00
Np-236	4.54E+01	4.54E+01	Zn-65	1.20E+08	1.88E+06
Np-237	2.16E+07	2.20E+07	Zr-88	1.99E-05	1.06E-10
Np-238	1.38E+06	1.36E+06	Zr-93	3.86E+08	3.86E+08
Np-239	1.04E+09	1.04E+09	Zr-95	2.96E+04	4.00E-03
Np-240	2.16E-02	2.16E-02	Zr-96	3.70E-04	3.70E-04
Np-240m	1.80E+01	1.80E+01	-	-	-

Appendix B HAZARD PRE-SCREENING

Category	Hazard	Screening Status	Rationale	Reference
H-EXT	**External Hazards – Human Induced**			
Mobile Sources				
H-EXT-1	**Aircraft Impact**			
H-EXT-1.1	Aircraft Strike	IN	This hazard is expected to cause damage to the PWMF and may lead to a radiological release. This hazard is further assessed in Sections 5.4.2.12, 5.4.3.12 and 5.4.4.10.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-2 Rail Transportation Hazards				
H-EXT-2.1	Train Crash	OUT	The CN Rail main line runs north of the PNGS, at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Based on [15], the screening distance for train derailment is estimated to be 80 m (3-rail-car length) from the crash. Therefore, this hazard can be screened out based on the distance from the PWMF.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
H-EXT-2.2	Cold Toxic Gas Release	OUT	The CN Rail mainline runs North of the PNGS, at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Table 3-1 of [15] shows the SDV for Cold Toxic Gases. a) SDV for Ammonia, Hydrochloric Acid and Hydrogen Fluoride releases is 0.9 km and 1.4 km, respectively. This means	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]

Category	Hazard	Screening Status	Rationale	Reference
			that these toxic materials can be screened out based on distance. b) SDV for Chlorine, Sulphuric Acid and Sulphur Dioxide is 4.4 km. This hazard can be screened out based on frequency (8.81E-07), refer to Table 3-9 of [28].	
H-EXT-2.3	Hot Toxic Gas Release	OUT	The CN Rail mainline runs North of the PNGS, at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Table 3-2 of [15] shows that the maximum SDV is 2.3 km (sulphur dioxide) for hot toxic gases. Therefore, this hazard can be screened out based on the distance from the PWMF.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
H-EXT-2.4	BLEVE – Missile Damage	OUT	The CN Rail mainline runs North of the PNGS, at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Based on [15], the BLEVE SDV is estimated to be 1600 m. Therefore, the BLEVE hazard from rail derailment can be screened out based on the distance from the PWMF.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
H-EXT-2.5	BLEVE – Blast Wave	OUT	The blast waves associated with a BLEVE are localized and not as strong as a Vapour Cloud Explosion (VCE). Since this hazard is bounded by the VCE hazard, it is not included in the screening analysis.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-2.6	Vapour Cloud	OUT	The CN Rail mainline runs North of the PNGS,	N-GUID-03611-10001

Category	Hazard	Screening Status	Rationale	Reference
	Explosion (VCE)		at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Based on [15], the Vapour Cloud Explosion SDV is estimated to be 460 m. Therefore, this hazard can be screened out based on the distance from the PWMF.	Vol.8 [15] NK30-REP-03611-00008 [28]
H-EXT-2.7	Explosions	OUT	The CN Rail mainline runs North of the PNGS, at approximately 3 km to the PWMF and the CP Rail mainline is located approximately 6 km north of the site [28]. Based on [15], the SDV is estimated to be 700 m. Therefore, this hazard can be screened out based on the distance from the PWMF.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
H-EXT-3 Road Transportation Hazards				
H-EXT-3.1	Cold Toxic Gas Release, such as: Ammonia, Hydrochloric Acid and Hydrogen Fluoride; Hot Toxic Gases, BLEVEs, VCEs, and Explosions	OUT	As major roads/highways are slightly further away from the plant than the railway, these offsite road transportation accidents can be screened out based on distance.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-3.2	Cold Toxic Gas Release e.g. Chlorine; Sulphuric Acid, Sulphur Dioxide	OUT	As per N-GUID-03611-10001, only 10% of these chemicals are transported on Highway 401 compared to the CN rail line traffic. This hazard can be screened out based on frequency (8.81E-08), refer to Table 3-11 of [28].	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]

Category	Hazard	Screening Status	Rationale	Reference
H-EXT-4 Ship Accidents				
H-EXT-4.1	Small Vessels	OUT	Boats/small vessels are not permitted to dock on the shore near the PWMF. Small vessels will not have an impact on the PWMF site; therefore this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-4.2	Large Vessels	OUT	As per [15], the normal shipping lanes in Lake Ontario are 10 kilometres away from the shoreline in the vicinity of the plant. In addition, there are no commercial wharfs around the Pickering area, see [28]. Therefore, this hazard can be screened out based on the distance from the PWMF.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
Stationary Sources				
H-EXT-5	Nearby Nuclear Event	OUT	An accident at the Pickering A, Pickering B or DNGS, resulting in significant releases would progress slowly enough to ensure notification to PWMF personnel such that the required actions could be taken. Any anticipated dose from the PWMF as a result of a significant event at either Pickering A or Pickering B, would be bounded by the dose received from the station itself. Therefore, this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-6 Toxic Gas Release				
H-EXT-6.1	Toxic Gas Release – Chlorine originated from Ajax Water Treatment Plant	IN	The Ajax Water Treatment Plant is situated near the Ajax Waterfront Park, and uses chlorine cylinders for water treatment. As per Table 3-1 of [15], the SDV for Chlorine is 4.4 km.	N-GUID-03611-10001 Vol.8 [15]

Category	Hazard	Screening Status	Rationale	Reference
			The Ajax Water Treatment facility is located at approximately 4.1 km from Phase II site. Since the SDV for chlorine is 4.4 km, this hazard cannot be screened out based on distance. This hazard is further assessed in Section 5.4.2.13	
H-EXT-6.2	Toxic Gas Release - Chlorine originated from the Duffin Creek Water Pollution Control Plant	OUT	The Duffin Creek Water Pollution Control Plant, using chlorine is located in Pickering at 1.2 km from the PWMF Phase II site. Due to its proximity, it may be able to release sufficient chlorine to impair the DSC transporter operator. Since the SDV for chlorine is 4.4 km, this hazard cannot be screened out based on distance. However, based on the annual frequency of chlorine leak from a fixed storage tank, 2.86E-07, refer to Table 3-12 of [28], this hazard can be screened out.	NK30-REP-03611-00008 [28] N-GUID-03611-10001 Vol.8 [15]
H-EXT-7	BLEVE	OUT	External fixed sources of BLEVEs have been identified within a radius of 5 km of PNGS [28] and it was concluded that none of the sites were within the SDV, which is 1600 m for BLEVE, refer to [15]. Therefore, this hazard can be screened out based on distance.	NK30-REP-03611-00008 [28] N-GUID-03611-10001 Vol.8 [15]
Other Sources				
H-EXT-8	Missiles from Military Activity	OUT	As per [15], this is considered a malevolent act. Therefore, it is out of scope.	N-GUID-03611-10001 Vol.8 [15]
H-EXT-9	Orbital Debris Crashes	OUT	According to [15], there is no SDV for this hazard type. Orbital debris can cause serious damage to the DSCs. However, based on the annual frequencies of	N-GUID-03611-10001 Vol.8 [15] P-REP-03611-00009

Category	Hazard	Screening Status	Rationale	Reference
			this hazard of: • 1.6E-08 occurrence/ year for the UFDS facilities, refer to Section 3.9 of [37] (including the total PWMF Phase I and Phase II storage area, with planned SB4). Taking into account that the Phase II storage area is smaller, the frequency of occurrence is lower; • The annual frequency of orbital debris impacting the DSC transporter was conservatively determined to be 7.2×10^{-7} events/year, refer to Section 5.5 of [24], which is already below the cutoff frequency of 10^{-6}. This value will considerably decrease when the likelihood of having a DSC in transit is taken into account. This hazard can be screened out.	[37] 92896-REP-00120-00005 [24]
N-EXT External Hazards – Natural				
N-EXT-1	Earthquake	IN	The ground motion associated with this event may exceed the design capacity of the PWMF SSCs. This hazard has the potential to lead to a radiological release; therefore, it cannot be screened out. This hazard is further assessed in Sections 5.4.2.6, 5.4.3.7 and 5.4.4.8	N-GUID-03611-10001 Vol.8 [15]
N-EXT-2	**Soil Failures**			
N-EXT-2.1	Slope Instability	IN	1) During DSC On-site Transfer As per [15] and [28], the PNGS site complies with the specific clauses of the Canadian	N-GUID-03611-10001 Vol.8 [15]

Category	Hazard	Screening Status	Rationale	Reference
			Foundation Engineering Manual and the National Building Code of Canada (NBCC). However, it has been identified that the DSC transport route between the PNGS B Irradiated Fuel Bay (IFB) or PNGS A Auxiliary Irradiated Fuel Bay (AIFB) and the DSC Processing Building at the PWMF – Phase I, has some anomalous portions, [92896-REP-00120-00005 R000]. Detailed geotechnical assessment has been recommended. In addition, reference [38] identified a lower priority anomaly along the transfer route from PWMF Phase I to Phase II, near the north-west corner of SB3. This hazard cannot be screened out for the DSC on-site transfer from Phase I to Phase II. This hazard requires further assessment and it is assessed in Section 5.4.2.14	NK30-REP-03611-00008 [28] 92896-REP-00120-00005 [24] P-CORR-76310-0395173 [38]
N-EXT-2.2	Subsidence	OUT	As per [15] and [28], the PNGS site is not situated in a geographical area where subsidence can occur. Therefore, this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
N-EXT-2.3	Swelling Clay	OUT	Based on [15] the foundations of PNGS are not on clay layers. Therefore, this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-2.4	Soil Frost	OUT	Based on Reference [15] this hazard may affect	N-GUID-03611-10001

Category	Hazard	Screening Status	Rationale	Reference
			the integrity of buried piping. This hazard is not applicable to the present PWMF safety assessment.	Vol.8 [15]
N-EXT-3 Flooding				
N-EXT-3.1	Flooding Due to Runoff	IN	A probable maximum precipitation (PMP) event can impact the PWMF SSCs ; this hazard will require, further assessment and it is assessed in Sections 5.4.2.9, 5.4.3.10 and 5.4.4.7.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-3.2	Flooding Due to River	OUT	Main river courses are located at a distance greater than 2 km from the western (Rouge River and the Petticoat Creek) and eastern (Duffin’s Creek) boundary of the PNGS site [28]. Based on distance, the potential for these rivers to represent a potential flood hazard to the PWMF is screened out. Krosno Creek is located immediately to the west of the PNGS and is prone to flooding. Based on Reference [28], an assessment has been conducted in 2011 as part of the Fukushima follow-up and it was determined that Krosno Creek would maintain at minimum approximately 2.7 m of freeboard from a potential spill during flooding due to a PMP event. Based on this, the potential for flooding from this river can be screened out.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
N-EXT-3.3	Flooding Due to	OUT	The Processing Building is at elevation 77.4m	92896-DRAW-29651-

Category	Hazard	Screening Status	Rationale	Reference
	Waves		(254') [39] and at a minimum distance of 75 m (the RCS area width) north of the lakeshore. The SB3 elevation is 83.53 m [40]. For an assumed lake level of 74.5 m (mean winter condition), wave uprushes have been estimated at 0.2 m and 1.8m [28]. The maximum wave run-up heights (76.3 m) are below the PWMF site. For an assumed (100-year) lake level of 75.6 m (Review Level Conditions – Lake Level) and with wave uprushes of 2.20 m, refer to Section 4.1 and Table 4-1 of [28], the maximum wave run-up heights are 77.8 m. Taken into account the elevation of the PWMF Phase I and Phase II sites and the distance of the Processing Building with respect to the lake, this hazard is screened out.	10075 [40]
N-EXT-3.4	Flooding Due to Seiche	OUT	Section 4.4.4 of Reference [28] notes that the site requires protection for water surge of up to 0.75 m, as the highest modeled water level at Darlington resulting from surge or seiche is about 0.75 m. The 100-year maximum lake level is 75.6 m, so the possible maximum level is 76.35 m. Phase II SB3 is situated at 83.53 m elevation; this hazard is screened out.	NK30-REP-03611-00008 [28] 92896-DRAW-29651-10075 [40]
N-EXT-3.5	Flooding Due to	OUT	Based on Section 4.4.5 of Reference [28], a	N-GUID-03611-10001

Category	Hazard	Screening Status	Rationale	Reference
	Tsunami		tsunami in Lake Ontario is an improbable event, with no associated flood hazard potential. Furthermore, the Great Lakes are in a geologically stable, mid-continental region, where the probability of occurrence of earthquakes large enough to generate tsunamis is negligible. Therefore, this hazard can be screened out.	Vol.8 [15] NK30-REP-03611-00008 [28]
N-EXT-3.6	Flooding Due to Sudden Releases of Water from Natural or Artificial Storage	OUT	As per Section 4.4.6 of [28], no large lakes and no man-made water retaining structures creating reservoirs are located within the drainage areas in the vicinity of the PNGS that could influence flooding. For this reason, this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
N-EXT-3.7	Flooding Due to Ice-jamming	OUT	Rapid melting of snow and large blocks of ice accumulated on the buildings' rooftop and at site as the temperature rises above the freezing point (late winter/early spring) can cause flooding. Section 7 of [18] states that the *DSC shall be designed that water from melting snow cannot enter the DSC.* Therefore this hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28] 00104-DR-79171-10000 [18]
N-EXT-3.8	Flooding Due to Other Causes	OUT	Other causes of flooding may include underwater landslides and lake ice. Lake Ontario shorelines as a whole are not susceptible to shore slope failure or landslide [28]. Lake ice can be also screened out as a flood hazard as ice structures are not expected	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]

Category	Hazard	Screening Status	Rationale	Reference
			to create or worsen any coastal flood hazard at Pickering [28].	92896-REP-00120-00005 [24]
N-EXT-4 Meteorological – Extremes				
N-EXT-4.1	Temperature (extreme high/ extreme low)	OUT	DSC on-site transfer: Procedures are in place to prohibit DSC transfer under poor or slippery road conditions. Even if the on-site transfer of a DSC takes longer than expected as result of adverse road conditions, the radiological consequences would be bounded by a transporter failure incident, refer to Section 5.4.2.1. Furthermore, as per Section 4.1, Transportation, of the DSC design requirements [18]: "*the materials and effectiveness of the components of the DSC shall not be degraded within the temperature range -40C to +70C*." This hazard can be screened out.	N-GUID-03611-10001 Vol.8 [15] 00104-DR-79171-10000 [18]
N-EXT-4.2	Snowpack	OUT	Waste transfer activities should not be performed during snow-covered conditions, and are bounded by a transporter failure incident. Section 7 of the DSC design requirements [18] states that the "*DSC shall not be degraded by exposure to snow* and the *DSC shall be designed that water from melting snow cannot enter the DSC".* In addition, the impact of the snowpack load on the DSC has to be taken into consideration.	N-GUID-03611-10001 Vol.8 [15] 00104-DR-79171-10000 [18]

Category	Hazard	Screening Status	Rationale	Reference
			As per Table 2 of [18], the DSC shall withstand *“compressive load for 24 hours of either five times the actual DSC package mass or 13 kPa multiplied by the vertically projected area of the DSC package.”* Based on the above requirements, the snowpack hazard can be screened out.	
N-EXT-4.3	Freezing Rain	OUT	The impact of freezing rain is bounded by the impact of external flood, ice-storms and snowpack. Procedures are in place to prohibit DSC transfer under poor or slippery road conditions. Even if the on-site transfer of a DSC takes longer than expected as result of adverse road conditions, the radiological consequences would be bounded by a transporter failure incident. This hazard is screened out.	N-GUID-03611-10001 Vol.8 [15] 92896-REP-00120-00005 [24]
N-EXT-4.4	Extreme Water Temperature	OUT	Operation of the PWMF does not depend on the use of lake water.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.5	Avalanches	OUT	The PNGS is not situated in a mountainous region with large slopes which would lead for a large avalanche.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.6	Lightning	OUT	Section 4.2 of 00104-DR-79171-10000 [18] states that “*the DSC shall be designed to maintain its structural integrity, appropriate shielding and containment function for severe atmospheric conditions during on-site transfer*	N-GUID-03611-10001 Vol.8 [15] 00104-DR-79171-10000 [18]

Category	Hazard	Screening Status	Rationale	Reference
			and storage." This hazard is bounded by a thunderstorm hazard, refer to Sections 5.4.2.8, 5.4.3.9, 5.4.4.6 and is screened out.	
N-EXT-4.7	Hurricanes	OUT	Tornadoes are more frequent in the region of concern and the impact of a tornado is considered bounding for high-winds category of hazard. Therefore, the wind speeds from tornadoes will be considered a bounding hazard.	
N-EXT-4.8	Tornadoes	IN	This hazard is expected to cause significant damage to the PWMF SSCs; therefore it will require further assessment and it is assessed in Sections 5.4.2.7, 5.4.3.8, and 5.4.4.5	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.9	Sand Storms	OUT	Sandstorms are typically associated with deserts. In the vicinity of the PWMF there are no large sand-bodies, therefore sandstorms are not a credible potential external hazard for Ontario.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.10	Ice Storms	OUT	Waste transfer activities should not be performed during slippery conditions, and are bounded by transporter failure incident and adverse road conditions, refer to Sections 5.4.2.1 and 5.4.2.5, respectively.	92896-REP-00120-00005 [24] 00104-DR-79171-10000 [18]

Category	Hazard	Screening Status	Rationale	Reference
			During storage in SB3: Based on the design requirements [18], the DSC, while in storage, will withstand storage building structural failure or collapse without loss of shielding or containment. This hazard does not require further assessment.	
N-EXT-4.11	Frazil Ice	OUT	Operation of the PWMF does not depend on the use of the lake water.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.12	Low Lake Level/Drought	OUT	Operation of the PWMF does not depend on the use of the lake water.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-4.12	Meteorites	OUT	Similar to the orbital debris hazard, this hazard cannot be screened out based on qualitative screening. However, the annual frequencies of this hazard are[6]: • 5.91E-08 occurrence/ year for the UFDS facilities (including SB4), refer to Table 4-1 of [37]. • The annual frequency of meteorites impacting the DSC transporter was conservatively determined to be 1.96x10-7 events/year, refer to Section 5.4 of [24], which is already below the cutoff frequency of 10-6. This value will considerably decrease when the likelihood of having a DSC in transit is taken into account. These values are lower than the cut-off	N-GUID-03611-10001 Vol.8 [15] P-REP-03611-00009 [37] 92896-REP-00120-00005 [24]

Category	Hazard	Screening Status	Rationale	Reference
			frequency of 10^{-6}, therefore this hazard is screened out.	
N-EXT-4.13	Geomagnetic storm	OUT	Geomagnetic storm events will impact the power distribution system equipment and may cause loss of off-site power. This hazard does not impact the PWMF site; therefore, it is screened out.	N-GUID-03611-10001 Vol.8 [15] NK30-REP-03611-00008 [28]
N-EXT-5 Other Hazards				
N-EXT-5.1	Forest Fire	OUT	There is no heavily forested area around 3 km of the site [24]. SDV for this hazard is 1 km [15]. Therefore, this hazard is screened out.	92896-REP-00120-00005 [24] N-GUID-03611-10001 Vol.8 [15]
N-EXT-5.2	Corrosion from Salt Water	OUT	This hazard is not applicable in the Great Lakes area; therefore, this hazard is screened out.	N-GUID-03611-10001 Vol.8 [15]
N-EXT-5.3	Animals	OUT	As per reference [15], it would require large numbers of animals to challenge the operation of the station. However, large numbers of animals will be restricted from entering the PWMF as the facility is within the protected area fence. Therefore, this hazard does not have any impact on the PWMF site or the DSC on-site transfer.	N-GUID-03611-10001 Vol.8 [15]
H-INT	**Internal Hazards**			
H-INT-1	Turbine Generated Missiles	IN	**During DSC on-site transfer** Pickering B Unit 8 Turbine is in close proximity to the DSC transfer routes. A missile may have	N-GUID-03611-10001 Vol.9 [14]

Category	Hazard	Screening Status	Rationale	Reference
			an impact on the transfer and processing in the Phase I Processing Building. This hazard requires further assessment and it is assessed in Section 5.4.2.11. **During DSC processing** Phase I of the PWMF is located southeast of PNGS Unit 8 with SB2 is situated the closest, at approximately 30 m, to Unit 8. A missile may have an impact on the Phase I structures, therefore this hazard requires further assessment and it is assessed in Section 5.4.3.11. **During DSC storage in SB3** The Phase II SB3 is located approximately 500 m northeast of Pickering B Unit 8, the closest unit to the PWMF site. The building is separated by distance from the Unit 8 turbine, and it is also shielded by various buildings located between the two facilities. The frequency of turbine missiles impacting SSCs has been determined to be $6x10^{-6}$ events/year [3]. Based on the low frequency of a turbine missile impacting an SSC and taking into account the location of the SB3 with reference to the Unit 8 turbine, this hazard is not further assessed.	
H-INT-2	Other Mechanically Generated Missiles	OUT	The effect of missiles from other components, such as pumps and valves is assumed bounded by the turbine missiles hazard. This hazard can be screened out.	N-GUID-03611-10001 Vol.9 [14]

Category	Hazard	Screening Status	Rationale	Reference
H-INT-3	Acetylene Decomposition Explosion Missile	OUT	Based on Section 3.1.2 of [20], the explosion frequency caused by acetylene cylinder explosion, adversely affecting the transported DSC, is 2.7E-08 events/year. Based on frequency, this event is screened out.	N-GUID-03611-10001 Vol.9 [14] 92896-REP-00120-00003 [20]
H-INT-4	Missiles Generated by a Hydrogen Explosion at the Tritium Removal Facility	OUT	This hazard is associated with the tritium removal facility at DNGS and is not applicable for the Pickering site.	N-GUID-03611-10001 Vol.9 [14]
H-INT-5	Control Rod Ejection Missiles	OUT	This hazard is not applicable due to the design of a CANDU reactor.	N-GUID-03611-10001 Vol.9 [14]
H-INT-6	**Release of Toxic, Radioactive or Corrosive Gases and Liquids from On-Site Storage**			
H-INT-6.1	Acute Inhalation Toxicity	IN	Toxic materials are not considered to be stored along the transfer route. There are toxic materials stored in the Processing Building and in SB3, therefore this hazard requires further assessment. This hazard is discussed in Sections 5.4.3.13 and 5.4.4.9.	N-GUID-03611-10001 Vol.9 [14]
H-INT-6.2	Corrosion	OUT	Quantities for corrosive materials are not shown on the latest Hazardous Material Inventory sheets. Based on previous Hazardous Material Inventory sheets (with quantities included) and the nature of the corrosive materials stored in Room 110, cabinets 32 and 6666 in the	N-GUID-03611-10001 Vol.9 [14]

Category	Hazard	Screening Status	Rationale	Reference
			Processing Building, it was concluded that there are less than 50 gallons of corrosive materials stored in the Processing Building. Therefore this hazard can be screened out.	
H-INT-6.3	Oxidizing/Reactive Chemicals	OUT	It was not confirmed by OPG that there is oxygen gas stored in the PWMF Processing Building. Therefore this hazard is screened out.	N-GUID-03611-10001 Vol.9 [14]
H-INT-6.4	Asphyxiants	IN	There are asphyxiating gases (argon, helium and nitrogen) stored in the workshop and gas bottle storage room in the Processing Building. Therefore this hazard requires further assessment and it is assessed in Section 5.4.3.13.	N-GUID-03611-10001 Vol.9 [14]
H-INT-7	Release of Stored Energy	OUT	Catastrophic failure of pressure vessels are excluded from consideration. There are no other sources of significant stored energy, such as high pressure piping associated with the PWMF.	N-GUID-03611-10001 Vol.9 [14]
H-INT-8	**On-Site Transfer**			
H-INT-8.1	Vehicle Impacts - Onsite Vehicle Movements	IN	Accident of vehicles with the DSC transporter during on-site transfer of the DSC has the potential to lead to radiological release. Therefore, this hazard is screened in. Further assessment is provided in Sections 5.4.2.1, 5.4.2.2 and 5.4.2.3.	N-GUID-03611-10001 Vol.9 [14]

Category	Hazard	Screening Status	Rationale	Reference
H-INT-8.2	Vehicle Impacts Within the PWMF Buildings	IN	The movement (craning, lifting, re-arrangement) of the DSCs within the PWMF buildings has the potential to lead to radiological release. Therefore, this hazard is screened in. This hazard is assessed in Sections 5.4.3.1, 5.4.3.2, 5.4.3.3, 5.4.3.4, 5.4.3.5, 5.4.4.2, 5.4.4.3.	N-GUID-03611-10001 Vol.9 [14]
H-INT-8.3	Toxic and/or Dangerous Goods - Onsite Vehicle Movements	OUT	This hazard is bounded by vehicle accidents involving radiological waste.	
H-INT-8.4	BLEVE – Blast Wave	OUT	The frequency of missile sources originating from propane tank explosion (BLEVE) along the transporter on-site transfer route is 3.4E-09 events/yr, refer to Section 3.2 of [20]. This hazard is screened out.	N-GUID-03611-10001 Vol.9 [14] 92896-REP-00120-00003 [20]
H-INT-8.5	Vapour Cloud Explosion (VCE)	OUT	The frequency of missile sources originating from propane tank explosion (VCE) along the transporter on-site transfer route is 2.1E-08 events/yr, refer to Section 3.3 of [20]). This hazard is screened out.	N-GUID-03611-10001 Vol.9 [14] 92896-REP-00120-00003 [20]
H-INT-9	Collapsed Structures	OUT	This hazard is bounded by earthquakes.	
H-INT-10	Fire – Toxic Effects Only	OUT	The effects of this hazard are bounded by fire.	
H-INT-11	Dropped or impacting loads	IN	The dropping of DSCs during handling can lead to radioactive release. This hazard needs further	N-GUID-03611-10001 Vol.9 [14]

Category	Hazard	Screening Status	Rationale	Reference
			assessment and it is assessed in Sections 5.4.3.1 and 5.4.4.2.	
H-INT-12	Electromagnetic Interference (EMI) and Radio-Frequency Interference (RFI)	OUT	EMI and RFI can affect the proper operation of digital instrumentation, I&C systems or advanced analog systems [14]. DSCs will not be affected by either the EMI or RFI.	N-GUID-03611-10001 Vol.9 [14]
H-INT-13	Static Electricity	OUT	The discharge of static electricity may impact the performance of control systems and control centers [14]. DSCs will not be affected by static electricity.	N-GUID-03611-10001 Vol.9 [14]
H-INT-14	Criticality related events	OUT	Based on the criticality assessment documented in the 1998 issue of the PWMF Safety Assessment, Appendix G, the Pickering used fuel stored in DSCs cannot achieve criticality under normal conditions or credible abnormal scenarios.	N-GUID-03611-10001 Vol.9 [14]
H-INT-15	High Temperature Surfaces	OUT	This hazard is bounded by fire.	N-GUID-03611-10001 Vol.9 [14]
H-INT-16	Fire	IN	Fires may lead to damage of the PWMF SSCs, therefore this hazard will require further assessment and it is assessed in Sections 5.4.2.4, 5.4.3.6 and 5.4.4.4	N-GUID-03611-10001, Vol9

Appendix C AIRCRAFT CRASH FREQUENCY CALCULATIONS

This Appendix presents the PWMF aircraft crash frequency calculations. The calculation is based on Appendix B of Reference [41] and consists of calculating the effective area of the target and multiplying that by the aircraft crash rate.

The effective target area is calculated as:

Aeff = Af + As where

where:

Af = (WS + R) x H x cotΦ + (2 x L x W x WS)/R + L x W, and

As = (WS+R) x S

where:

- Af = effective fly-in area
- As = effective skid area
- WS = aircraft wingspan
- R = length of the diagonal of the facility
- H = facility height
- cotΦ = mean of the cotangent of the aircraft impact angle
- L = length of facility
- W = width of facility
- S = aircraft skid distance

Table C-1 shows the total crash rates calculated for the PNGS site for the five aircraft categories, taken from Table 3-2 of Reference [28]. Airports located in a radius of about 35 kilometres from the PNGS were considered in the airfield crash rate calculation.

The aircraft crash frequency for the PWMF was calculated by the summation of the crash frequencies in those areas where DSCs are stored or processed.

A qualitative aircraft impact assessment (AIA) was performed for the DSC against light general aviation aircraft crashes [42] and concluded that the Category 1 aircraft, which is a light aircraft, will not cause damage to a DSC except for slight concrete cracking or scabbing, therefore this aircraft type was not included in the DSC aircraft crash frequency calculations.

The aircraft crash frequency calculations were performed for Aircraft categories 2 to 5 for the areas occupied by the PWMF Phase I and Phase II structures.

In addition, the total aircraft crash frequency for the PWMF structures holding safety related waste containers, such as DSCs and Dry Storage Modules (DSM) [43] was determined by summation of the frequency of an aircraft crash impacting the DSC processing and storage buildings and Retube Component Storage (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 PWMF Phase I SB1, SB2 and Processing Building occupy a total area of approximately 312 ft x 342 ft [44]. The height of the Phase I facility is 45 ft [39]. The aircraft crash frequency calculated for the PWMF Phase I area is 2.42×10^{-7} events/year.

The RCS facility occupies a total fenced area of approximately 75 m x 85 m [3] and it is located south of the Phase I area. The aircraft crash frequency for the RCS area was calculated to be 2.54×10^{-7} event/year.

SB3 has a total area of 279.06 ft x 238 ft [45]. Storage Building 4 occupies a total area of 275.33 ft x 270.67 ft [46]. The total area of SB3 and SB4 was calculated using the following dimensions: total length of SB3+SB4 (275.33+279.06) ft and the greater width of the two buildings: 270.67 ft. The height of both SB3 and SB4 is 9.6 m (31.5 ft) [45]. The aircraft crash frequency calculated for the PWMF Phase II area is 2.92×10^{-7} events/year.

The summation of the above aircraft crash frequencies calculated for the PWMF site where safety-related containers are stored or processed is:

$$2.42\times10^{-7}+2.54\times10^{-7}+2.92\times10^{-7} = 7.88\times10^{-7}\text{events/year}$$

This value is below the cut-off frequency of 10^{-6} events/year and is therefore screened out.

Table C-1 PNGS Airfield Crash Rates

Aircraft Category	Total Crash Rate (km^{-2} yr^{-1})
Light Aircraft (Category 1)	5.1E-06
Helicopters (Category 2)	3.6E-07
Small Transport (Category 3)	9.3E-07
Large Transport (Category 4)	1.2E-06
Military Combat (Category 5)	6.6E-08
Total for Large Aircraft (Categories 4 and 5)	-
Total for all Categories	-

Taken from Table 3-2 of Reference [28]

Table C-2 PWMF Aircraft Crash Frequency Calculations

		Category 1	Category 2	Category 3	Category 4	Category 5	Total Crash Frequency/Facility
		Light Aircraft	Helicopters	Small Transport	Large Transport	Military Combat	
Wingspan	WS, ft	73	50	59	98	110	
Skid distance	S, ft	60	0	1440	1440	447	
Cot impact angle	Cot q	8.2	0.58	10.2	10.2	10.4	
Crash rate	$km^{-2}\ yr^{-1}$	5.10E-06	3.60E-07	9.30E-07	1.20E-06	6.60E-08	
DSC On-Site Transfer with Liftking Transporter							
Transporter Length	L, ft	N/A	27.83	27.83	27.83	27.83	
Transporter Width	W, ft	N/A	10.88	10.88	10.88	10.88	
Diagonal of Transporter	R, ft	N/A	29.88	29.88	29.88	29.88	
Transporter Height	H, ft	N/A	15.52	15.52	15.52	15.52	
Effective fly area	Af. ft^2	N/A	2,034.68	15,568.33	22,532.27	25,109.22	
Effective skid area	As, ft^2	N/A	0.00	127,990.26	184,150.26	62,527.31	
Total Area	Aeff, ft^2	N/A	2,034.68	143,558.60	206,682.53	87,636.53	
	Aeff, km^2	N/A	0.00019	0.013	0.019	0.008	
Probability of a loaded transporter on-site	yr^{-1}	N/A	0.011	0.011	0.011	0.011	
Crash Frequency	yr^{-1}	N/A	7.77E-13	1.42E-10	2.63E-10	6.13E-12	**4.12E-10**
Phase I (Processing Building, SB1, SB2)							
Facility Length	L, ft	N/A	342.00	342.00	342.00	342.00	
Facility Width	W, ft	N/A	312.00	312.00	312.00	312.00	
Diagonal of Facility	R, ft	N/A	462.93	462.93	462.93	462.93	

		Category 1	Category 2	Category 3	Category 4	Category 5	Total Crash Frequency/Facility
		Light Aircraft	Helicopters	Small Transport	Large Transport	Military Combat	
Facility Height	H, ft	N/A	45.00	45.00	45.00	45.00	
Effective fly area	Af. ft^2	N/A	143,141.08	373,470.17	409,349.78	425,546.07	
Effective skid area	As, ft^2	N/A	0.00	751,585.13	807,745.13	256,101.55	
Total Area	Aeff, ft^2	N/A	143,141.08	1,125,055.30	1,217,094.91	681,647.62	
	Aeff, km^2	N/A	0.013	0.105	0.113	0.063	
Crash Frequency	yr^{-1}	N/A	4.79E-09	9.72E-08	1.36E-07	4.18E-09	**2.42E-07**
DSM/ RCS Area							
Facility Length	L, ft	278.8	278.8	278.8	278.8	278.8	
Facility Width	W, ft	246.00	246.00	246.00	246.00	246.00	
Diagonal of Facility	R, ft	371.81	371.81	371.81	371.81	371.81	
Facility Height	H, ft	16.07	16.07	16.07	16.07	16.07	
Effective fly area	Af. ft^2	154,130.86	90,962.37	160,967.49	181,748.03	189,690.59	
Effective skid area	As, ft^2	26,688.82	0.00	620,371.75	676,531.75	215,370.73	
Total Area	Aeff, ft^2	180,819.68	90,962.37	781,339.25	858,279.78	405,061.32	
	Aeff, km^2	0.017	0.008	0.073	0.080	0.038	
Crash Frequency	yr^{-1}	8.57E-08	3.04E-09	6.75E-08	9.57E-08	2.48E-09	**2.54E-07**
Phase II - DSC Storage Buildings SB3 and SB4							
Facility Length	L, ft	N/A	553.75	553.75	553.75	553.75	
Facility Width	W, ft	N/A	270.68	270.68	270.68	270.68	
Diagonal of Facility	R, ft	N/A	616.37	616.37	616.37	616.37	
Facility Height	H, ft	N/A	31.50	31.50	31.50	31.50	

		Category 1	Category 2	Category 3	Category 4	Category 5	Total Crash Frequency/Facility
		Light Aircraft	**Helicopters**	**Small Transport**	**Large Transport**	**Military Combat**	
Effective fly area	Af. ft^2	N/A	186,379.17	395,561.65	427,059.38	441,326.75	
Effective skid area	As, ft^2	N/A	0.00	972,528.32	1,028,688.32	324,686.00	
Total Area	Aeff, ft^2	N/A	186,379.17	1,368,089.97	1,455,747.70	766,012.75	
	Aeff, km^2	N/A	0.017	0.127	0.135	0.071	
Crash Frequency	yr^{-1}	N/A	6.23E-09	1.18E-07	1.62E-07	4.70E-09	**2.91E-07**
Total Crash Frequency for Phase I, RCS/DSM Area and Phase II:							**7.88E-07**

Notes:

i Light Aircraft dimensions were taken from Tables B-16/B-17 and B-18 of Reference [41], corresponding to General Aviation, TurboProp.

ii Small Transport dimensions were taken from Tables B-16/B-17 and B-18 of Reference [41], corresponding to Commercial Aviation, Air Taxi.

iii Large Transport dimensions were taken from Tables B-16/B-17 and B-18 of Reference [41], corresponding to Commercial Aviation, Air Carrier.

iv Military Combat dimensions were taken from Tables B-16/B-17 and B-18 of Reference [41], corresponding to Military Aviation, Small Aircraft Low Performance.

Appendix D ADDAM COMPUTER CODE

Computer Program Name	ADDAM [47]
Code Version	1.4.2
Operating System	Windows 7

Code description:
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 accident 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 release (timing, composition and quantity), decay, build-up, and deposition. Doses are calculated for various organs, 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 the CSA N288.2-M91 standard [48]. A recent code assessment documented in Reference [49] has confirmed that ADDAM is also in compliance with the CSA N288.2-14.

Use of code:
For the current analysis, the ADDAM code was used to predict dose to members of the public following postulated malfunciton / accident scenarios. Consistent with the CSA N288.2-14 [6], ADDAM has a limitation on the treatment of releases from fire scanario; however, the current safety assessment, there is no releases associated with the fire scenario.

Validation and code applicability:
The use of code for the current analysis is within the current ADDAM code range of applicability. The methodologies implemented in the ADDAM code has undergone a series of validation exercises. The phenomena that govern atmospheric dispersion and dose estimation in the context of safety analysis were identified and documented in the validation matrix for dispersion [50]. Phenomena that were validated were summarized in the ADDAM validation manual [51]:

- Plume rise;
- Downwash;
- Modification of effective height release due to building entrainment;
- Plume broadening due to building entrainment;
- Fumigation;
- Reflection at an elevated inversion;

- Plume transport;
- Plume diffusion;
- Wet deposition;
- Dry deposition;
- Plume depletion;
- Exposure to cloudshine;
- Exposure to groundshine; and
- Internal exposure due to inhalation.

The validation exercises conform with the requirements given in the CSA N286.7 standard. Note that since the ADDAM validation exercises were phenomena-based and the current version (1.4.2) still has the same underlying dispersion methodology, all validation results are still valid for the current version being used for the PWMF safety assessment.

Although there are still uncertainties associated with the ADDAM code in modeling some phenomena such as wet deposition, dry deposition, and plume depletion, the code has been developed with many conservative assumptions to ensure that the calculated doses are not underestimated.

Sofware Quality Assurance Documents:

Problem definition	COG document SQAD-15-5074, ADDAM Version 1.4.2 Problem Definition
Development plan	Section 4 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Theory manual	COG document SQAD-07-5008, ADDAM version 1.4 Theory Manual and Section 6 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Requirements specification	Section 5 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Design description	Section 7 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Verification report	Section 9 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Programmer's manual	Section 12 of COG document SQAD-10-5087, ADDAM version 1.4.2 Model Development and Verification.
Validation report	AECL RC-2674 Validation Reports volumes 1-10.

ENCLOSURE #3

OPG report
“Dose Rate Assessment Considering Lower Aged Fuel in PWMF SB3”
92896-REP-03200-00009

ONTARIOPOWER GENERATION

Report

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Document Number: 92896-REP-03200-00009	Usage Classification: N/A
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Title:
DOSE RATE ASSESSMENT CONSIDERING LOWER AGED FUEL IN PWMF SB3

Dose Rate Assessment Considering Lower Aged Fuel in PWMF SB3

92896-REP-03200-00009 R000
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Prepared By:
Digitally signed by Stephen Smith
Date: 2020.06.12 08:33:56 -04'00'
S. Smith Date
Health and Radiation Physicist
Candu Energy Inc.

Reviewed By:
Digitally signed by Anas Khaial
Date: 2020.06.12 09:13:30 -04'00'
A. Khaial Date
Senior Reactor Physicist
Candu Energy Inc.

Reviewed By:
Ricky Khaloo
Digitally signed by Ricky Khaloo
Date: 2020.06.12 09:27:47 -04'00'
R. Khaloo Date
Specialist Health Physicist
Candu Energy Inc.

Reviewed and Verified By:
Digitally signed by Kwok Tsang
Date: 2020.06.12 09:20:43 -04'00'
K. Tsang Date
Specialist Radiation Physicist
Candu Energy Inc.

LICENSED PROFESSIONAL ENGINEER
K. T. TSANG
PROVINCE OF ONTARIO

Approved By:
Yahui Zhuang
Digitally signed by Yahui Zhuang
Date: 2020.06.12 09:51:59 -04'00'
Y. Zhuang Date
Manager, Radiation Physics & RadWaste
Candu Energy Inc.

Accepted By:
Digitally signed by Paul Crowley
Date: 2020.06.15 10:34:51 -04'00'
P. Crowley Date
Senior Technical Officer
Fuel and Nuclear Waste Safety Assessment
Ontario Power Generation

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Revision Summary

Revision Number	Date	Comments
R000	2020-06-12	Initial issue.

1. INTRODUCTION

1.1 Background

Ontario Power Generation's (OPG's) Pickering Nuclear Generating Station (NGS) is located on the north shore of Lake Ontario in the Regional Municipality of Durham. The Pickering Waste Management Facility (PWMF) handles the transfer, processing and storage of Dry Storage Containers (DSCs) containing used fuel discharged from the Pickering NGS units.

The PWMF is part of the larger OPG Pickering site; the location of the PWMF within the Pickering NGS site is shown in Figure 1.

The PWMF site has undergone an orderly development in phases to facilitate the growing number of DSCs over the years. These phases are:

- Phase I: 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 Phase I site consists of a DSC Processing Building (PB), DSC Storage Buildings (SBs) 1 and 2, and the Retube Components Storage area.
- Phase II: The PWMF Phase II site is located approximately 500 m northeast of the PWMF Phase I site, east of the Pickering NGS powerhouse, within its own protected area in the Pickering Nuclear site. The Phase II site consists of DSC SB3 (see Figure 2 and Figure 3) with provision for future SB4.

The existing operation of the PWMF involves the processing and storage of DSCs containing used fuel with a minimum of ten (10) years of decay. In order to support PWMF operations, an analysis to determine the impacts of loading used fuel cooled for less than ten (10) years is being completed. These lower fuel age DSCs are planned to be stored in SB3.

The existing PWMF Safety Report [1] and the latest safety assessment update [2] have considered the storage of DSCs with a minimum of ten (10) years of decay in SB3. The purpose of this document is to determine the impact, from a dose rate perspective, of storing fuel that has only cooled for six (6) years.

The current report documents the dose rate analysis part of the assessment associated with the storage of the lower fuel age DSCs in SB3. The calculation model was based on the previous shielding assessment documented in Reference [3] with appropriate changes in the DSC loading pattern applied to SB3.

1.2 Scope

This report documents the dose rate assessment for the storage of DSCs with lower fuel age in SB3 at PWMF. The current assessment considers the storage of 100 DSCs containing six (6) year decayed used fuel to replace the equivalent number of existing DSCs stored in SB3. Dose rates at the following locations are calculated and presented:

- Dose rates at and beyond the existing protected area fence surrounding SB3 and SB4;
- At the Pickering NGS property boundary (Montgomery Park Rd);
- At the lakeside exclusion boundary;
- At the Training and Mock-up Building (TMB); and
- At the main aisle way of SB3 in the vicinity of the lower fuel age DSCs.

Note, the limiting dose rates at the above locations are primarily from radiation sources from SB3 and SB4. The contribution of radiation sources from DSCs and Dry Storage Modules (DSMs) stored in the PWMF Phase I site to the direct external radiation field at the limiting locations around the Phase II site are negligible [4]. Therefore, contribution from the Phase I site is not included in the current analysis.

In addition, estimated dose rates near contact[1] and at 1 m from the long side of a single DSC loaded with 230 MWh/kgU fuel bundles with average decay times of 6, 10, 15, 20, 25, 30, 35 and 40 years decay time are reported.

1.3 Quality Assurance

The work was performed by Candu Energy Inc. in accordance with the Quality Assurance (QA) program described in 147-912020-QAP-001 "CANDU Services Projects (CSA Z299 Series)", CE-912020-QAM-002 "Candu Energy Inc. – Quality Assurance Manual" and CE-912020-QAM-003 "Quality Assurance Manual – Analytical, Scientific and Design Computer Programs" to satisfy the QA requirements of the following standards applicable to the scope of work:

- CSA CAN3-Z299.1-85 "Quality Assurance Program Category 1";
- CSA N286-12 "Management System Requirements for Nuclear Facilities"; and
- CSA N286.7-16 "Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants".

1.4 Terms and Abbreviations

AP	Antero-Posterior
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
DSC	Dry Storage Container
DSM	Dry Storage Module
ENDF	Evaluated Nuclear Data File
EPB	Enhanced Processing Building
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IST	Industry Standard Toolset
MCNP™	Monte Carlo N-Particle Transport
NEW	Nuclear Energy Worker
NGS	Nuclear Generating Station
OPG	Ontario Power Generation
PB	Processing Building
PWMF	Pickering Waste Management Facility
QA	Quality Assurance
RCS	Retube Components Storage
ROT	Rotational
SB	Storage Building
SCALE	Comprehensive modeling and sumulation suite for nuclear safety analysis and design
SSR	Surface Source Read – MCNP option
SSW	Surface Source Write – MCNP option
TMB	Training and Mock-up Building
UFDS	Used Fuel Dry Storage

[1] Contact dose rates were calculated at a distance of 5 cm from the DSC surface.

2. ACCEPTANCE CRITERIA

The radiation safety requirements for the PWMF under normal operation[2] are the following:

- ≤ 10 µSv per year for the general public at, or beyond the Pickering NGS site boundary. This dose rate target is one percent (1%) of the CNSC regulatory dose limit of 1 mSv per year for a member of the public [1].
- ≤ 0.5 µSv/h outside of the protected area fence (boundary of the PWMF licensed facility), based on the 1 mSv/a effective dose limit for non-Nuclear Energy Workers (NEWs) and an occupancy rate of 2000 hours per year [1].

3. METHODOLOGY

Dose rates are calculated for normal operation of the PWMF following the reference methodology for heavily shielded containers [5]. Computer codes used for the dose rate calculations are listed in Section 3.6. Additional discussions on the various aspects of the dose rate calculations are provided in the following subsections.

3.1 General Approach

Dose rate calculations during normal operations follow the OPG reference methodology for heavily shielded containers [5]. The general calculation approach is using the two-stage approach, for both site shielding and single DSC assessments. The main steps in applying the two-stage analysis to DSC problems are:

1. Perform a Stage-1 analysis of a single DSC to capture the photon source escaping from the DSC by using the Surface Source Write (SSW) and Surface Source Read (SSR) options in MCNP. Photons escaping the DSC are recorded as they pass through user-selected planes outside the DSC. Recorded photon information is then used as a boundary source in the Stage-2 calculations. The single-stage analysis is performed for representative source energy groups for an irradiated fuel bundle of a specific decay time and burnup. DSCs containing irradiated fuel of various decay times and burnup are assessed as recommended in the reference methodology [5]. This step has been performed for fuel with ten (10) years decay as part of the work documented in Reference [6].
2. Perform Stage-2 analyses to calculate dose rates near DSCs, within buildings, and around the area outside of buildings. For the Stage-2 modified Phase II site model, MCNP input files from Reference [3] were used as the starting model. The following changes are implemented in the starting model:
    a. The layout of the DSCs in SB3 is modified to replace one hundred (100) DSCs with DSCs containing six (6) year decayed fuel; and
    b. The MCNP tally definitions are modified to include the existing protected area fence, the Pickering NGS property fence, and the SB3 aisle way.

Note that, as indicated in in the shielding assessment for SB4 [3], neutrons are generated in irradiated fuel in addition to gamma radiation. However, due to use of heavy concrete in the design of the DSC, the neutron dose rates outside DSCs are negligible compared to gamma dose rates and therefore neutrons are not included in the current shielding assessment.

[2] As per the wording in Section 4.2 of Reference [1], these requirements are for the operation of the PWMF only and are exclusive of the dose from the Pickering NGS. Additional discussion on the radiation safety requirements and dose rate targets is provided in Reference [1].

3.2 Representation of Geometry in MCNP Models

3.2.1 Stage-1 DSC

The representation of the DSC geometry (internal structures, rebar) and fuel bundles contained therein is described in Reference [4]. For the Stage-1 MCNP simulations, the DSC and fuel bundles were modeled in detail (see Figure 4, Figure 5, and Figure 6). The SSW files generated from the existing Stage-1 calculations [6] for reference used fuel, with a burnup of 230 MWh/kgU and 10 years decay time, are utilized.

3.2.2 Stage-2 DSC

For the Stage-2 MCNP simulations, a simplified DSC model is used, consistent with the models documented in Reference [3]. The density of the homogenized mixture of heavy concrete and rebar is 3.57 g/cm[3]. The cavity of the DSC is represented as a single region of homogenized UO_2, Zircaloy cladding, air, and steel with a mixture density of 2.96918 g/cm^3 (see Footnote 3).

This simplification is justified because in the Stage-2 simulations, photons are 'born' from the captured Stage-1 virtual source planes in the air outside of each DSC. Photons entering a DSC in the Stage-2 calculations have a low probability of escaping. As such, there is little benefit in modelling the fine details within a DSC in this stage.

3.2.2.1 Single DSC Model

For the single DSC dose rate calculations, the single DSC model and analysis results from the previous shielding assessment for SB4 [3] are extracted and used in the current work. In the analysis presented in Reference [3], the dose rates are obtained from Stage-2 calculations by applying the existing SSW files as the SSR. Additionally, the DSC is placed on the concrete floor to account for scattering off the floor.

3.2.2.2 PWMF Site Model

The MCNP modelling of the PWMF site and buildings in this analysis is exactly the same as in the previous shielding assessment [3]. A detailed description of the PWMF site MCNP model is given in Reference [6], with the recent changes to the modelling of SB3 and SB4 outlined in Reference [3]. There are no changes to the modelled geometry of the PWMF site for the current analysis.

As part of the PWMF site MCNP model, SB3 currently stores 480 DSCs. One hundred (100) of the stored DSCs are replaced with DSCs containing six (6) year decayed fuel. The physical positions of the DSC arranged in SB3 remains the same as in Reference [3]. Previously, the DSCs in SB3 are assumed to contain used fuel bundles that have decayed for 25 or 30 years. As identified in Section 1.1, 100 of these DSCs are replaced with DSCs containing used fuel bundles that have decayed for 6 years.[4] The loading of the DSCs in SB3 is shown in Figure 7.[5]

The loading of the DSCs in SB4, shown in Figure 8, remains consistent with the shielding assessment for SB4 [3].

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[3] The MCNP input files are carried-over from the analysis presented in Reference [3]. As such, the material definitions are kept constant with Reference [3] and do not reflect the accuracy to which these values are known or estimated.

[4] The 100 replaced DSCs comprise 97 DSCs containing 25 year decayed used fuel and 3 DSCs containing 30 year decayed used fuel.

[5] The loading pattern of DSCs in SB3 is proposed to ensure the DSCs containing 6 year decayed used fuel are surrounded by DSCs containing used fuel decayed for longer periods with the intention of minimizing dose rates external to the building. In addition, because DSCs are being transferred out of SB3 to SB4 (e.g. to make room in SB3 to allow younger fuel to be stored), older DSCs will be selected for the transfer into SB4 due to their lower dose rates.

MCNP representations of the DSC layout in SB3 and SB4 are shown in Figure 9 and Figure 10, respectively.

3.3 Wall Openings

The manway doors in SB3 are represented as low-density (0.07536 g/cm^3) steel and the roll-up doors are represented as being open (i.e. as if no door is present), as per Reference [3]. Similarly, a roll-up door on the south wall of SB3 was added as part of the analysis documented in Reference [3] and is included in the current analysis.

3.3.1 Building Walls and Roofs

For SB3, the bulk shielding material is ordinary concrete, except for the building roof, which is made of steel. SB4 is modelled with 22 gauge (0.0759 cm) steel walls and roofs with an ordinary concrete floor. The shielding material, composition, and density are the same as the existing analysis [3] and follow the reference methodology [5].

3.3.2 Ventilation

The existing ventilation designed for the SB3 (Reference [6]) remains as modelled in Reference [3]. SB4 is modelled without ventilation.

3.4 Representation of Materials

The composition and density of materials used in the previous shielding assessment [3] is utilized in the current analysis. A detailed list of material compositions and densities is available in Table 1.

3.5 Variance Reduction

The current work involves deep penetration photon transport and transport of photons over a long distance. As such, a direct simulation (analog MCNP) would be lengthy in computation and impractical. The Stage-1 SSW file generation involves deep penetration shield transport. A "geometry splitting with Russian roulette"[6] variance reduction technique was employed by splitting the shield materials inside a DSC. Stage-1 simulations are not repeated in the current work. For the Stage-2 simulations, the SSW files from the Stage-1 simulation are used as source terms.[7] The Stage-2 simulations involve transport over a long distance. A similar "geometry splitting with Russian roulette" variance reduction technique is employed by splitting the shield material (walls, roofs) or splitting the air between the source and the receptors.

3.6 Representation of Radiation Source Terms

3.6.1 Photon Source Spectra

The source spectra is supplied as input to the MCNP simulations as a distribution of photons as a function of energy. This energy distribution can be provided either as a set of discrete 'lines' corresponding to the actual decay emission energies of photons produced from individual radionuclides or as a set of 'energy' bins into which the individual photon energies are binned and sampled via a histogram distribution. When the number of radionuclides contributing to the decay photon source term is

[6] In MCNP, particles transported from a region of higher importance to a region of lower importance undergo Russian roulette; that is, some of those particles are stochastically terminated, but the weight of surviving particles is increased.

[7] The SSW files are not modified in the Stage-2 MCNP simulations. The surfaces comprising the SSW source are translated (with the TRn card) to each desired location corresponding to each DSC position. The relative intensity / weighting of each instance of the SSW source is then scaled using the SPn card.

small, the discrete photon energy lines are supplied. When the number of radionuclides is large, it is more practical to define the photon source energy in the form of energy bins.

The resolution of the photon decay spectra is generally determined by the limitations of the calculation method, rather than MCNP, as MCNP is able to handle a detailed spectra. In the current analysis, the photon source spectrum is represented using 500 energy groups. Following the approach in Reference [5], this source spectrum was grouped into six energy bands. A separate SSW file was generated for each of the energy bands (Reference [6]).

Decay photons with energy <0.3 MeV do not escape the thick concrete shield of the DSC and their contribution to the dose rate outside of the DSC is negligible. Therefore, it is judged to be acceptable not to sample decay photons with energy <0.3 MeV. Similarly, the population of decay photons with energy >3 MeV is small (< 0.01%) and their percent contribution to the total dose rate outside of the DSC is judged negligible. Thus, these decay photons are not sampled either.

The six energy bands for the SSW files are listed in Table 2. The surface sources for the DSCs at the PWMF were generated using the reference fuel bundle properties [6], these include:

- Pickering-type 28-element fuel bundle;
- Mass of Uranium per bundle is 20.2 kg;
- Exit burnup of 230 MWh/kgU;
- Fuel bundle decay time of 10 years; and
- Fuel bundle power of 373 kW (fission) representing the core average of 100% full power operations.

3.6.2 Spent Fuel Decay Times

The fuel stored in SBs across the PWMF includes seven binned decay times: 10, 15, 20, 25, 30, 35 and 40 years [3]. In addition, a decay time of 6 years will be applied to the 100 replaced DSCs to be stored in SB3. All DSCs in the SBs have different decay times. The DSC loading pattern is determined based on OPG's input. The loading patterns for SB3 and SB4 are provided in Figure 7 and Figure 8 , respectively. The burnup for all decay times is assumed to be the same as the burnup of the reference fuel (i.e., 230 MWh/kgU).

3.6.3 Two-Stage Decay Source Treatment

The SSW files were generated for the reference fuel burnup and decay time. DSCs with different fuel burnup and decay times use the same SSW files. The total source term for SB3, including 480 DSCs of various ages, is weighted using the SPn/TRn cards in the SSR option of the MCNP file. The total weight distribution considers the relative source terms of each DSC based on the characteristics of the stored used fuel compared to those of the reference fuel burnup and decay. The use of the same SSW files assumes that the fine-group photon spectrum in each energy band is identical. The impact of applying the reference fuel spectra to all fuel burnup/decay age combinations has been investigated and documented in Reference [7].[8] The investigation shows that the difference in the group average energies is negligible and has no impact on the calculated dose rates. Therefore, the use of the reference SSW files for other decay times is justified.

[8] The work in Reference [7] was performed for the Western Waste Management Facility, however, the conclusion regarding the application of the reference fuel spectra to all fuel burnup / decay age combinations remains appropriate for the current analysis.

3.6.4 ORIGEN-S Photon Spectrum

The fine-group photon spectrum used in the previous PWMF shielding assessment [3] was generated using the bundle-wise 28-element ORIGEN-S cross-section library provided with the distribution of SCALE. The 28-element Pickering-specific cross-section library was updated in Reference [2] including ring-wise spectra. The cross-section library was further revised in Reference [8].

The revised cross-section library generated in Reference [8] is used in the current analysis to calculate the source terms used in the MCNP dose rate calculation.

3.7 Dose Rate Tallies

MCNP provides several methods of estimating photon fluxes required to calculate dose rates. The track-length estimate (F4) cell tally is simple and reliable for problems where many photon histories pass through the volume of interest. The FMESH tallies are simple and reliable and have similar applications as F4 tallies. The FMESH tallies can be placed at general locations and are useful to show the spatial distribution of dose rates around components or areas. The current scope of work calculations utilizes F4 and FMESH tallies following the two-stage methodology.

The dose rates from a single DSC at near contact and at 1 m from the wide side of a single DSC with 6, 10, 15, 20, 25, 30, 35 and 40 years decay time are reported. Results are discussed in Section 4.1.

Dose rates surrounding SB3 / SB4 resulting from the inclusion of 100 DSCs containing 6 year decayed used fuel are tabulated at the following locations:

- Across the main aisle way of SB3 - results are discussed in Section 4.2

  An FMESH tally (approximately 30 cm x 30 cm x 50 cm) is used to calculate the dose rates across the SB3 aisle way. The mesh tally location across the aisle way of SB3 is shown in Figure 11. Receptor locations are shown in Figure 12 and also listed in Table 3. A sketch of mesh tally locations around SB3/SB4 to calculate the dose rate at the PWMF existing protected area fence is shown in Figure 13.

- At selected dose receptor locations - results are discussed in Section 4.3.1

  FMESH tallies (10 m x 10 m x 2 m rectangular cuboids) are used to calculate the dose rates at the Pickering NGS site boundary (Montgomery Park Rd), the lakeside exclusion zone boundary, and at the TMB. The FMESH tallies are located above the ground at various locations. The lakeside exclusion zone receptors are located 8 m below ground level, the level at which lake water is modelled in MCNP.

- Protected area fence - results are discussed in Section 4.3.2

  An FMESH tally encompassing the area around SB3 / SB4 is used to calculate the dose rates at the protected area fence. The layout of the protected area fence surrounding SB3 and SB4 is given in Reference [9]. The following distances, to the centre of the protected area fence, are adopted in the analysis of the dose rates:

  - North Fence: 15.00 m from the north wall of SB3;
  - East Fence: 15.33 m from the east wall of SB3;
  - South Fence: 96.68 m from the south wall of SB3;
  - West Fence: 18.00 m from the west wall of SB3; and
  - West Fence, extended[9]: 66.00 m from the west wall of SB3.

[9] Distances derived from the dimensions provided in Reference [9].

All cardinal directions are given based on the site construction north (i.e. the south fence is the fence on the construction south side of SB3).

3.8 Nuclear Data

The photoatomic library mcplib04 [10] from the MCNP6 distribution package is utilized. The library is the most up-to-date photoatomic data set in MCNP. It is based on the ENDF/B-VI release 8 nuclear evaluation data.

3.9 Dose Rate Conversion Coefficients

Dose rate conversion values given in ICRP publication 116 [11] are used.

Dose rates are computed using an “Antero-Posterior” (AP) dose conversion curve, which conservatively assumes that personnel are facing the DSCs. Over most of the range of photon energies normally encountered in DSC applications (less than about 10 MeV), the AP dose conversion curves predicts higher dose rates than if other directional conversion curves are assumed, due to the actual configuration of organs and tissues in human bodies. The calculation of dose rates using the AP conversion curves provides a level of inherent conservatism since in practice, there would be no preferred orientation of personnel at most dose locations outside of processing and storage buildings.

In addition, the dose rates are also computed using a “Rotational” (ROT) dose conversion curve, in which a geometry is defined by rotating the body at a uniform rate about its long axis, while irradiating the body by a broad beam of ionising radiation from a stationary source, in this case DSCs, located on an axis at right angles to the long axis of the body.

Dose rates calculated using the AP and ROT dose conversion factors are reported. Compliance to the dose rate acceptance criteria is demonstrated using the ROT dose conversion factor.

3.10 Target Relative Error

MCNP tally results include the relative error corresponding to one standard deviation. These errors cannot be believed reliable (hence neither can the tally itself) unless the error is fairly low. Results with relative errors less than 10% are generally (but not always) reliable for the F4 and FMESH tally types used in this analysis.

The target relative error for the current dose rate assessment is 5%. If there are dose rate values with relative error >5% but <10%, the values may be conditionally accepted if there is an adequate margin to the acceptance criteria. In order to achieve the target relative error, averaging across neighbouring FMESH tally cells is applied at applicable locations of interest.

Results are presented by listing the calculated dose rates from MCNP as the best estimate dose rates and their associated one-sigma uncertainties. Comparison to the acceptance criterion is done using the best estimate + two-sigma uncertainty dose rate.

3.11 Other Options in MCNP

The MCNP photon treatment options such as the upper energy limit for the detailed photon physics model, generation of bremsstrahlung photons with thick-target bremsstrahlung model, coherent Thompson scattering treatment (on), photonuclear particle production (off), photon Doppler energy broadening (on), photo-fission model (no photo-fission prompt gammas) is set to the MCNP default setting.

3.12 Computer Codes

3.12.1 MCNP

Computer Program Name	MCNP [12]
Code Version	6.1
Operating System	Linux 64 bit

Code description:
MCNP is a general-purpose Monte Carlo code that can be used for neutron, photon, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by 1st and 2nd degree surfaces and 4th degree elliptical tori. Point-wise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (e.g. the ENDF Evaluated Nuclear Data File) are accounted for. Thermal neutrons are described by both the free gas and scattering kernel S(α,β) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, and absorption in electron positron pair production. The mcplib04 photon cross-section library from the MCNP distribution package [10] is used for gamma photon transport calculations.

Use of code:
For the current analysis, the MCNP code is used to calculate the gamma dose rates at various locations in the PWMF site.

Validation and code applicability:
The use of code described above is within the validation range of applicability described below.
The MCNP code has been used worldwide for neutron, photon, and electron transport calculations. MCNP validation results are available from the following documents provided by the code developer:
- LA-UR-03-9032, Bibliography of MCNP Verification & Validation: 1990-2003
- LA-UR-04-8965: Bibliography of MCNP Verification & Validation: 2004
- LA-UR-02-0878: Validation Suites for MCNP, Proc. of the American Nuclear Society, Radiation Protection and Shielding Division.
- LA-UR-12-26307: V&V of MCNP and Data Libraries at Los Alamos.

MCNP has been used to perform the dose rate calculations for the Western, Darlington, and Pickering waste management facilities. The reference methodology described in Reference [5] is applied. Reference [5] also discusses the validation and benchmarking of the MCNP with respect to dose rate analysis involving DSCs.

3.12.3 SCALE

Computer Program Name	SCALE [13]
Code Version	6.1 (COG IST)
Operating System	Linux 64 bit

Code description:

The ORIGEN-S module of SCALE is employed in the current analysis.

ORIGEN-S is the depletion module to calculate neutron activation, actinide transmutation, fission product generation, and radiation source terms. It applies a matrix exponential expansion model to calculate time-dependent concentrations, activities, and radiation source terms for a large number of isotopes simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay. The ORIGEN-S libraries include nuclear data for 2226 nuclides produced by neutron activation, fission, and decay. All decay data are based on the ENDF/B-VII.0.

Use of code:

The ORIGEN-S module is used to provide the gamma source terms for use in MCNP dose rate calculations. The cross-section library recently generated for the bundle-wise cross-section library in Reference [8] is used for this analysis.

Validation and code applicability:

For the use of the code described above, the ORIGEN-S code has also been validated for various nuclear reactor types, including CANDU reactors. A list of publicly available documents on the ORIGEN-S validation is available from the SCALE code developer website: https://www.ornl.gov/scale/scale/spent-fuel-isotopic-characterization

3.13 Analysis Assumptions

The assumptions in the current analysis are the same are those adopted in the SB4 shielding assessment [3]. The list of assumptions and the justification of each assumption are listed below.

Assumption	Justification of Assumption
100 DSCs containing 6 year fuel are incorporated in the loading pattern of SB3.	The total number of DSCs containing 6 year decayed used fuel to be stored in SB3 is expected to be equal to or less than 100. Therefore, consideration of 100 DSCs is bounding in the dose rate estimation.
All personnel doors and rollup doors are assumed open (air material composition is applied at the location of the door)	Provides conservatism in the dose rate estimation.
The DSC placement at the east and west of the SB4 follows a uniform gap between the DSCs. However, the north-south gap for the centre row is about 55 cm more than the rest of the rows in SB4 (see Reference [3]).	The assumption has no impact on the final results and is consistent with previous analyses (see References [3] and [6]).

Assumption	Justification of Assumption
DSCs with different fuel burnup and decay times use the same SSW files in the MCNP calculations. The use of the same SSW files assumes that the fine-group photon spectrum in each energy band is identical.	The impact of applying the reference fuel spectra to all fuel burnup/decay age combinations has been investigated and documented in Reference [7]. The investigation shows that the difference in the group average energies is negligible and has no impact on the calculated dose rates.

4. RESULTS

Best estimate dose rate values are presented along with the associated 1σ uncertainty (the statistical uncertainty quoted by MCNP). Compliance to the acceptance criteria is evaluated using the best estimate + 2σ uncertainty values. The reported dose rates do not include the natural background contribution, which is approximately 0.2 µSv/h based on Reference [1].[10]

4.1 Dose Rates from Single DSC

The calculated dose rates from a single DSC loaded with 6 year decayed used fuel (230 MWh/kgU burnup) are listed in Table 4.

The calculated dose rates on the long side of a DSC as a function of used fuel decay time are given in Table 5 and shown in Figure 14. The dose rates are presented for used fuel decay times between 6 years and 40 years.

For DSCs loaded with reference used fuel bundles (decayed by 10 years or older), the measured contact dose rates to date are about 9 to 13 µSv/h [1]. This compares with the calculated estimate of the near contact (at the DSC long side) dose rate of 37.9±0.2 µSv/h for reference used fuel (decayed by 10 years). At 1 m distance, measured dose rates are about 5 to 7 µSv/h [1], compared with calculated dose rate estimates of 20.0±0.1 µSv/h.

In the current analysis, the estimated dose rates from a DSC containing 6 years decayed used fuel are 97.4±2.6 µSv/h at near contact and 51.2±0.9 µSv/h at 1 m. These dose rates are a factor of approximately 2.6 times larger than the dose rates calculated for used fuel decayed 10 years. Therefore, it is expected that the measured dose rates for DSCs containing 6 year decayed used fuel will increase by a similar factor (2.6x) compared to the dose rates measured from the DSCs containing the reference used fuel.

While the calculated dose rate in all individual energy groups (G00 – G05) increases for the 6 year decayed used fuel as a result of the shorter decay time, the largest relative increase is seen in energy group G05. With respect to 10 year decayed used fuel, the intensity of the photon spectrum for energy group G05 increases approximately 25 times for the 6 year decay used fuel. The increase is driven by the greater amount of Rh-106 (half-life of approximately 30 seconds) and Pr-144 (half-life of approximately 17 minutes) in the 6 year decayed used fuel.

10 The natural background contribution of 0.2 µSv/h includes all sources of background radiation, such as internal doses from K-40, cosmic radiation, and radon doses in the home. The TLDs used for on-site measurements will not include all these sources, so it is expected that the background contribution to TLD results is much lower than 0.2 µSv/h.

4.2 Dose Rates across the SB3 Main Aisle Way

The dose rate profile across the SB3 main aisle way was evaluated for the DSC loading pattern[11] including the 100 DSCs containing 6 year decayed used fuel. The calculated dose rate profile across the SB3 aisle way is shown in Figure 15. The profile was calculated for a location near the centre of where the DSCs containing 6 year decayed used fuel will be stored. The maximum estimated dose rate across the aisle way is 13.6±0.5 µSv/h. The average dose rate in the main aisle way at distances 5 m or more away from the DSCs is estimated to be 4.7±0.1 µSv/h.

These estimated dose rates across the SB3 main aisle way are significantly less than the maximum (33.9±1.6 µSv/h) and average (9.4±0.2 µSv/h) dose rates calculated for the SB4 main aisle way [3]. The difference in dose rates across the SB3 and SB4 aisles results from the different decay age used fuel stored in the two buildings. Compared to SB3, SB4 contains a larger portion of lower decay age used fuel (10 – 20 years) which contributes to a higher dose rate. Further, the loading pattern for SB4 has a larger amount of DSCs with low decay age fuel bordering the main aisle way which, as seen in Table 5, have larger dose rates than the 25 year decayed used fuel stored along the main aisle way in SB3.

The proposed loading pattern of SB3 (see Figure 7) has the aisle way lined with DSCs containing 25 year decayed used fuel. When compared to the dose rates from a single DSC containing 6 year decayed used fuel, the calculated dose rates across the main aisle way are significantly lower. The DSCs containing 25 year decayed used fuel lining the aisle way provide shielding from the DSCs with the lower fuel decay age.

4.3 Dose Rates around PWMF

4.3.1 Dose Rate at Selected Receptors

The dose rates at selected receptor locations were calculated based on the DSCs stored in SB3. Table 6 shows the calculated dose rates for the existing SB3 DSC loading pattern (analyzed in Reference [3]) compared to the proposed loading pattern for 100 DSCs containing 6 year decayed used fuel. The calculated dose rates at all receptor locations increase with the storage of lower fuel age in SB3. This increase in dose rate is expected as the dose rates from DSCs containing 6 year decayed used fuel are larger than the dose rates of the replaced DSCs from SB3 which contain used fuel decayed for longer periods of time (see Section 4.1 and Table 5).

The total dose rates at the selected receptor locations resulting from the DSCs stored in SB3 and SB4 are given in Table 7. The contribution from SB3 includes the 100 DSCs containing 6 year decayed used fuel. The dose rate at the Montgomery Park Rd Pickering NGS site boundary (PW24) represents the Pickering NGS site boundary fence.[12] The estimated dose rate at the Montgomery Park Rd site boundary (PW24) receptor location from SB3 and SB4 is $(1.7\pm0.1)\times10^{-3}$ µSv/h. Based on a yearly occupancy of 2000 hours, the annual dose at the Montgomery Park Rd site boundary is 3.6 µSv/a (best estimate + 2σ uncertainty).

Comparison against the acceptance criteria is given in Table 8. The AP dose conversion factor is applied in calculations for dose receptor locations within the PWMF protected area fence. The ROT[13] dose conversion factor is applied for calculation for the remaining receptor locations. It is shown that, with the inclusion of 100 DSCs containing 6 year decayed used fuel in SB3, the estimated dose rates and annual doses are within the acceptance criteria.

11 The DSC loading pattern for SB3 is provided in Figure 7.

12 As part of the analysis document in Reference [3], receptor location PW24 was shifted to represent the Pickering NGS site boundary fence instead of then the walking path directly east of the fence.

13 In practice, there would be no preferred orientation of personnel at most dose locations outside of the processing and storage buildings. As such, the ROT dose conversion factor would be appropriate at such locations.

4.3.2 Dose Rate at the Existing Protected Area Fence

The PWMF protected area fence is shown in Figure 3. The distances between the protected area fence and SB3 are given in Section 3.7. An FMESH tally in MCNP with large mesh voxel volumes[14] was used to calculate the dose rate across and beyond the protected fence area (mesh tally area shown in Figure 13).

The dose rates are calculated using the ROT dose conversion factor and provide the dose rate profile at and beyond the protected area fence. The contribution to the dose rates from the DSCs stored in SB3 is calculated as part of the current analysis. The analysis presented in Reference [3] provides the contribution to the dose rate from the DSCs stored in SB4. The maximum estimated dose rates (best estimate + 2σ uncertainty) along the protected area fence, considering the contribution form DSCs stored in both SB3 and SB4, are shown in Table 9.

Along sections of the east, south, and west protected area fence the calculated dose rate exceeds the 0.5 µSv/h acceptance criterion. A representation of these locations is provided in Figure 16. The majority of the dose rate at the areas exceeding the acceptance criterion is due to the DSCs stored in SB4. The distance from the existing protected area at which the dose rate would fall below the acceptance criterion is specified in Table 9.

5. CONCLUSIONS

With respect to the normal operation of SB3 at the PWMF, the inclusion of 100 DSCs containing 6 year decayed used fuel does not pose an unacceptable risk to workers or members of the public. The external radiation exposure at the TMB and the public dose at the site boundary remain below the acceptance criteria outlined in Section 2. The dose rates within SB3 are not adversely affected by the storage of 100 DSCs containing 6 year decayed used fuel and remain comparable to those calculated for other DSC storage buildings ([1]). While dose rates at some locations along the existing protected area fence are over 0.5 µSv/h, the distances beyond the fence to where the dose rates fall below the 0.5 µSv/h acceptance criterion have been provided (see Table 9). It should be noted that the majority of the dose rates at the fence locations that exceed 0.5 µSv/h are from PWMF SB4, and not from the 6 year old fuel in SB3. The risk to workers is low, and, if necessary, other site fences can be used as boundaries at which the target of 0.5 µSv/h can be applied.

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14 FMESH tally voxels are 2 m x 2 m x 2 m in size.

6. REFERENCES

[1] Ontario Power Generation, "Pickering Waste Management Facility Safety Report", 92896-SR-01320-10002 R005, November 2013.

[2] Ontario Power Generation. "Pickering Waste Management Facility Safety Assessment Update", 92896-REP-01320-00005 R002, August 2018.

[3] Ontario Power Generation, "Shielding Assessment for PWMF Storage Building SB4", 92896-REP-03200-00008 R000, June 2019.

[4] Ontario Power Generation, "PMWF Holistic Shielding Analysis and Optimization Study", 92896-REP-03200-00003 R000, August 2016.

[5] Ontario Power Generation, "Reference Nuclear Waste Shielding Analysis Methodology for Heavily Shielding Containers", W-REP-03200-00002 R002, December 2018.

[6] Ontario Power Generation, "Assessment of Outdoor Storage of DSCs at the PWMF", 92896-REP-03200-00007 R000, June 2018.

[7] Ontario Power Generation, "Unshielded DSC Dose Rate Parametric Study", W-REP-03200-00003 R000, June 2018.

[8] Candu Energy Inc., "Impact of Bug Reported in CANDU Burnup-Dependent Neutron Cross-Section Libraries Generated with TRITON version SCALE 6.1.3 on PWMF Safety Assessment", 19-CEOPG-19-0014L, October 2018.

[9] Ontario Power Generation, "Storage Building 3 Architectural Site Plan", 92896-DRAW-29651-10075 R000, March 2009.

[10] LA-UR-03-1019 "Photoatomic Data Library MCPLIB04: A New Photoatomic Library Based On Data from ENDF/B-VI Release 8", 2003.

[11] International Commission on Radiological Protection, "Conversion Coefficients for Radiological protection Quantities for External Radiation Exposures", ICRP Publication 116, 2010.

[12] MCNP: Monte Carlo N-Particle Transport Code version 6.1. Available from Radiation Safety Information Computational Center as code package CCC-810.

[13] SCALE 6.1: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design: includes ORIGEN, version 6.1, available from Radiation Safety Information Computational Center as code package CCC-785.

Table 1 Stage-2 MCNP Material Compositions [3]

Material	Density (g/cm^3)	Element	Un-normalized Weight Fractions
DSC Homogenized Source	2.96918	Oxygen	0.23636
		Iron	2.24677
		Zirconium	0.18575
		Uranium	0.30031
ASTM A516 Grade 70 Steel	7.85	Carbon	0.27
		Silicon	0.4
		Phosphorus	0.025
		Sulfur	0.025
		Manganese	1.2
		Iron	98.08
Dry Air at 35ºC	1.1214E-03	Hydrogen	8.80E-07
		Helium	5.20E-06
		Carbon	1.10E-04
		Nitrogen	0.780851
		Oxygen	0.209682
		Neon	1.82E-05
		Argon	9.33E-03
		Xenon	1.00E-07
High Density Concrete with Homogenized Rebar	3.57	Hydrogen	0.34
		Carbon	0.39
		Oxygen	50.18
		Magnesium	1.1
		Aluminium	1.57
		Silicon	2.56
		Calcium	6.86
		Chromium	5.84
		Iron	31.15
Concrete	2.35 (Normal) 1.175 (Hollow Concrete Block) 2.08 (Grout Concrete)	Hydrogen	0.56
		Oxygen	49.83
		Sodium	1.71
		Magnesium	0.24
		Aluminium	4.56
		Silicon	31.58
		Sulfur	0.12
		Potassium	1.92
		Calcium	8.26
		Iron	1.22

Material	Density (g/cm^3)	Element	Un-normalized Weight Fractions
Ground - Sand	1.6	Oxygen	53.26
		Silicon	46.74
Fiberglass Insulation	0.03	Oxygen	46.14
		Fluorine	0.7
		Sodium	10.53
		Magnesium	1.51
		Aluminium	0.318
		Silicon	33.66
		Calcium	7.15
Rockwool Insulation	0.1	Oxygen	41.72
		Sodium	1.699
		Aluminium	3.45
		Silicon	24.74
		Phosphorus	0.0655
		Potassium	1.303
		Calcium	21.64
		Titanium	0.306
		Manganese	0.0465
		Iron	1.82
Aluminium	0.124876 (Homogenized Louvres)	Aluminium	1
High Density Concrete without Rebar	3.5	Hydrogen	0.35
		Carbon	0.4
		Oxygen	50.19
		Magnesium	1.13
		Aluminium	1.6
		Silicon	2.61
		Calcium	7
		Chromium	5.95
		Iron	29.77
Water	1.0	Hydrogen	0.11191
		Oxygen	0.88809

Table 2 Energy Bands for SSW / SSR Surface Sources in MCNP

Energy Group	Photon Energy Band (MeV)
00	0.30 ≤ E < 0.65
01	0.65 ≤ E < 1.00
02	1.00 ≤ E < 1.25
03	1.25 ≤ E < 1.50
04	1.50 ≤ E < 2.00
05	2.00 ≤ E < 3.00

Table 3 Dose Receptor Locations[15]

Dose Point #	X (cm)	Y (cm)	Z (cm)	Description
PW10	2726	17246	1319	1ft below TMB roof peak (height 41 ft above TMB floor)
PW24	38800	12380	100	Montgomery Park Rd turnaround
PW26	28700	33330	400	Bend in bike path northeast of PWMF Phase II
LS03	-20009	-81613	-700	Off shoreline
LS04	16230	-57143	-700	Off shoreline
LS05	32391	-32700	-700	Lake 282 m off shoreline
LS06	37411	-18888	-700	Lake 144 m off shoreline
LS07	40208	-4460	100	Lake, where shoreline intersects with land site boundary
Figure 11	-	-	100	Mesh tally location across the aisle way in SB3
Figure 13	-	-	100	Mesh tally locations around SB3/4 (green overlay)

15 The origin location (x,y,z) = (0,0,0) corresponds to a location near the centre of the array of DSCs stored in SB4.

Table 4 Calculated Dose Rate (µSv/h) vs. Distance from a Single DSC Containing 6 Year Decayed Used Fuel

Distance From Surface (cm)	Dose Rate (µSv/h)		
	Long Side	Short Side	Top
Near Contact[a]	97.4	86.2	70.5
50	77.4	62.7	49.8
100	51.2	42.9	n/a[b]
150	37.7	29.1	17.0
200	27.7	20.7	10.8
250	23.7	15.5	10.5
300	17.4	12.5	6.4
350	12.8	10.5	5.4
400	10.4	9.0	4.1
450	8.9	6.7	3.2
500	7.2	6.0	2.1

Notes:

a) Contact dose rates were calculated at a distance of 5 cm from the DSC surface.

b) The dose rate at 100 cm from the top of the DSC surface is excluded as the associated statistical uncertainty is significantly larger than the 10% target presented in Section 3.10.

Table 5 Calculated Dose Rate (μSv/h) vs. Distance from a Single DSC (Long Side) at Various Decay Times

	Used Fuel Bundle Decay Time (year)								
Distance (cm)	**6 y**	**10 y**	**15 y**	**20 y**	**25 y**	**30 y**	**35 y**	**40 y**	**Maximum 1σ Uncertainty[b]**
Near Contact[a]	97.4	37.9	26.6	21.0	17.1	14.2	11.9	10.1	2.7%
50	77.4	29.0	20.3	16.0	13.0	10.8	9.1	7.7	2.7%
100	51.2	20.0	14.1	11.1	9.0	7.5	6.3	5.4	1.7%
150	37.7	14.4	10.1	7.9	6.4	5.3	4.5	3.8	2.6%
200	27.7	10.6	7.4	5.8	4.7	3.9	3.3	2.8	3.4%
250	23.7	8.3	5.7	4.5	3.7	3.0	2.6	2.2	9.7%
300	17.4	6.4	4.5	3.5	2.8	2.4	2.0	1.7	5.5%
350	12.8	5.1	3.6	2.8	2.3	1.9	1.6	1.4	2.8%
400	10.4	4.2	2.9	2.3	1.9	1.6	1.3	1.1	2.6%
450	8.9	3.5	2.4	1.9	1.6	1.3	1.1	0.9	4.8%
500	7.2	2.9	2.0	1.6	1.3	1.1	0.9	0.8	3.4%

Notes:

a) Contact dose rates were calculated at a distance of 5 cm from the DSC surface.

b) The uncertainty listed is the maximum uncertainty in the dose rate across all decay times for a given distance.

Table 6 Dose Rate Contribution at Receptor Locations from DSCs Stored in SB3

	Existing DSC Loading [3]		DSC Loading with 6 Year Decayed Used Fuel		Ratio of Best Estimate Dose Rates
Dose Receptor	Best Estimate (μSv/h)	1σ Uncertainty (relative)	Best Estimate (μSv/h)	1σ Uncertainty (relative)	
PW10	5.27E-02	3%	7.27E-02	2%	1.38
PW24	4.23E-04	2%	7.36E-04	2%	1.74
PW26	4.62E-04	2%	7.90E-04	2%	1.71
LS03	8.33E-07	4%	2.59E-06	5%	3.11
LS04	1.16E-05	3%	3.09E-05	4%	2.66
LS05	7.13E-05	3%	1.66E-04	3%	2.33
LS06	1.54E-04	2%	3.01E-04	2%	1.96
LS07	2.72E-04	2%	5.05E-04	2%	1.86

Table 7 Dose Rates (µSv/h) from SB3 and SB4

		Contribution from DSCs in SB3		Contribution from DSCs in SB4 [3]		Contribution from SB3 and SB4		
Dose Receptor	Dose Conversion Factor	Best Estimate (µSv/h)	1σ Uncertainty (relative)	Best Estimate (µSv/h)	1σ Uncertainty (relative)	Best Estimate (µSv/h)	1σ Uncertainty (relative)	Best Estimate + 2σ Uncertainty (µSv/h)
PW10	AP	7.27E-02	2%	1.96E-02	4%	9.23E-02	2%	9.61E-02
PW24	ROT	7.36E-04	2%	9.64E-04	4%	1.70E-03	2%	1.78E-03
PW26	ROT	7.90E-04	2%	4.76E-04	3%	1.27E-03	1%	1.30E-03
LS03	ROT	2.59E-06	5%	7.20E-06	6%	9.79E-06	5%	1.07E-05
LS04	ROT	3.09E-05	4%	9.39E-05	4%	1.25E-04	3%	1.33E-04
LS05	ROT	1.66E-04	3%	4.22E-04	5%	5.88E-04	4%	6.32E-04
LS06	ROT	3.01E-04	2%	6.77E-04	2%	9.78E-04	2%	1.01E-03
LS07	ROT	5.05E-04	2%	1.05E-03	4%	1.55E-03	3%	1.64E-03

Table 8 Annual Dose Rates from SB3 and SB4

Dose Point	Best Estimate + 2σ Uncertainty (μSv/h)	Dose Conversion Factor	Occupancy (hours)	Annual Dose based on Occupancy[a] (μSv/a)	Acceptance Criterion
PW10	9.61E-02	AP	2000	192.29	0.5 μSv/h (1000 μSv/a)[b]
PW24	1.78E-03	ROT	2000	3.56	10 μSv/a (1% of public dose limit)
PW26	1.30E-03	ROT	2000	2.61	
LS03	1.07E-05	ROT	1000	0.01	
LS04	1.33E-04	ROT	1000	0.13	
LS05	6.32E-04	ROT	1000	0.63	
LS06	1.01E-03	ROT	1000	1.01	
LS07	1.64E-03	ROT	1000	1.64	

Notes:

a) The presented annual doses include only the contribution from SB3 and SB4. Contribution from other PWMF radiation sources is not included.

b) 1000 μSv/a is the prorated annual dose based on the acceptance criterion of 0.5 μSv/h and 2000 hours occupancy.

Table 9 Dose Rates at Phase II Protected Area Fence from DSCs in SB3 and SB4

Protected Area Fence[a]	Distance From the Edge of SB3 to the Protected Area Fence[b] (m)	Maximum Dose Rate Along Protected Area Fence from DSCs Stored in SB3[c,d] (μSv/h)	Maximum Dose Rate Along Protected Area Fence from DSCs Stored in SB3 and SB4[c,d,e] (μSv/h)	Distance[f] Beyond Existing Protected Area Fence where Dose Rate ≤ Acceptance Criterion (m)	Acceptance Criterion
North	15.00	0.21	0.21[g]	-	0.50 μSv/h
South	96.68	0.02	0.71	7	
East	15.33	0.14	0.56	3	
West	18.00	0.13	0.85	7	
West (extended)	66.00	0.03	0.16	-	

Notes:

a) Directions correspond to the construction site cardinal directions.
b) Distances derived from the dimensions provided in Reference [9].
c) Presented dose rates are the Best Estimate + 2σ uncertainty.
d) The presented dose rates include only the contribution from DSCs stored in SB3 and SB4. Contribution from other PWMF radiation sources is not included.
e) The maximum dose rate from SB3 along the protected area fence occurs at different locations than for the maximum dose rate from SB3 and SB4.
f) Distances are estimated based on the FMESH tally voxel size of 2 m x 2 m x 2 m. A representation of the locations the dose rate exceeds the acceptance criterion is shown in Figure 16.
g) The contribution to the dose rate at the north fence from DSCs stored in SB4 was not calculated as part of the analysis in Reference [3].

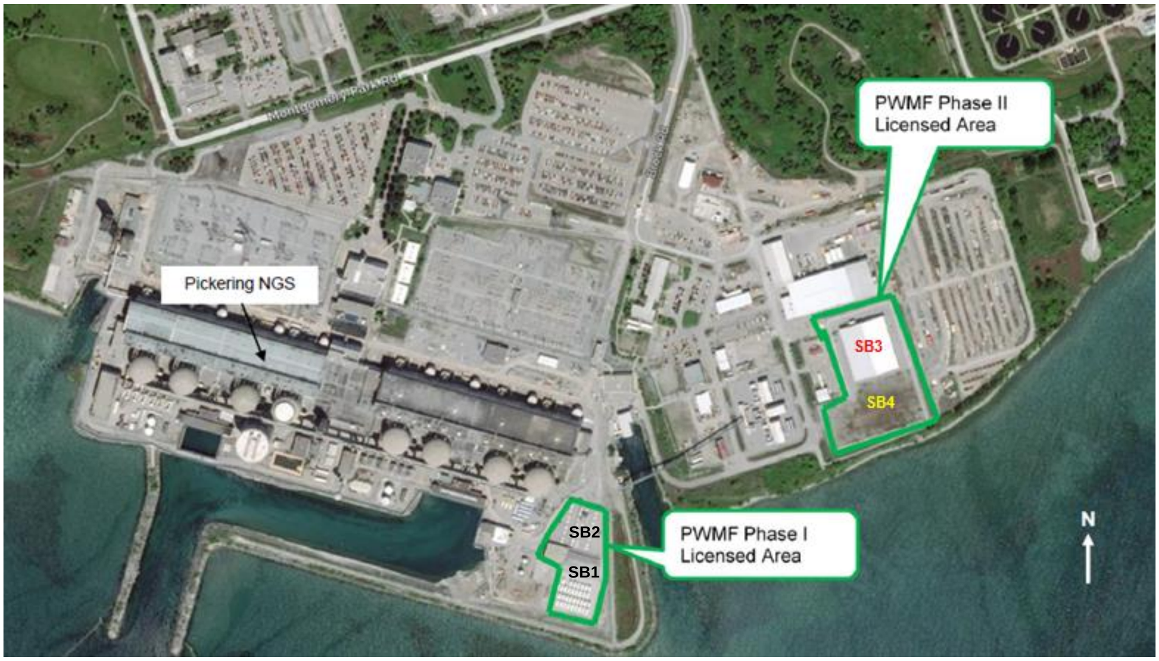

Figure 1 Aerial View of the PWMF Site

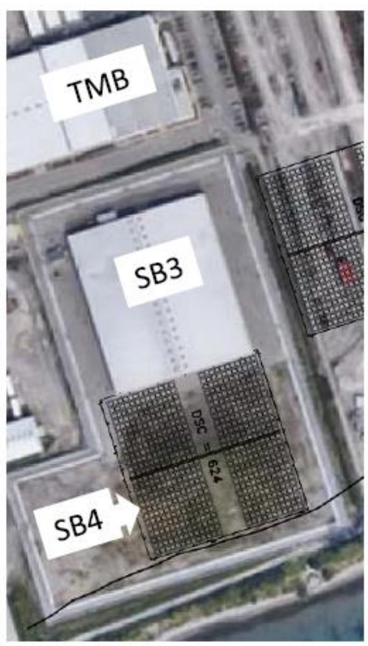

Figure 2 PWMF Layout for Current SB3 and Future SB4

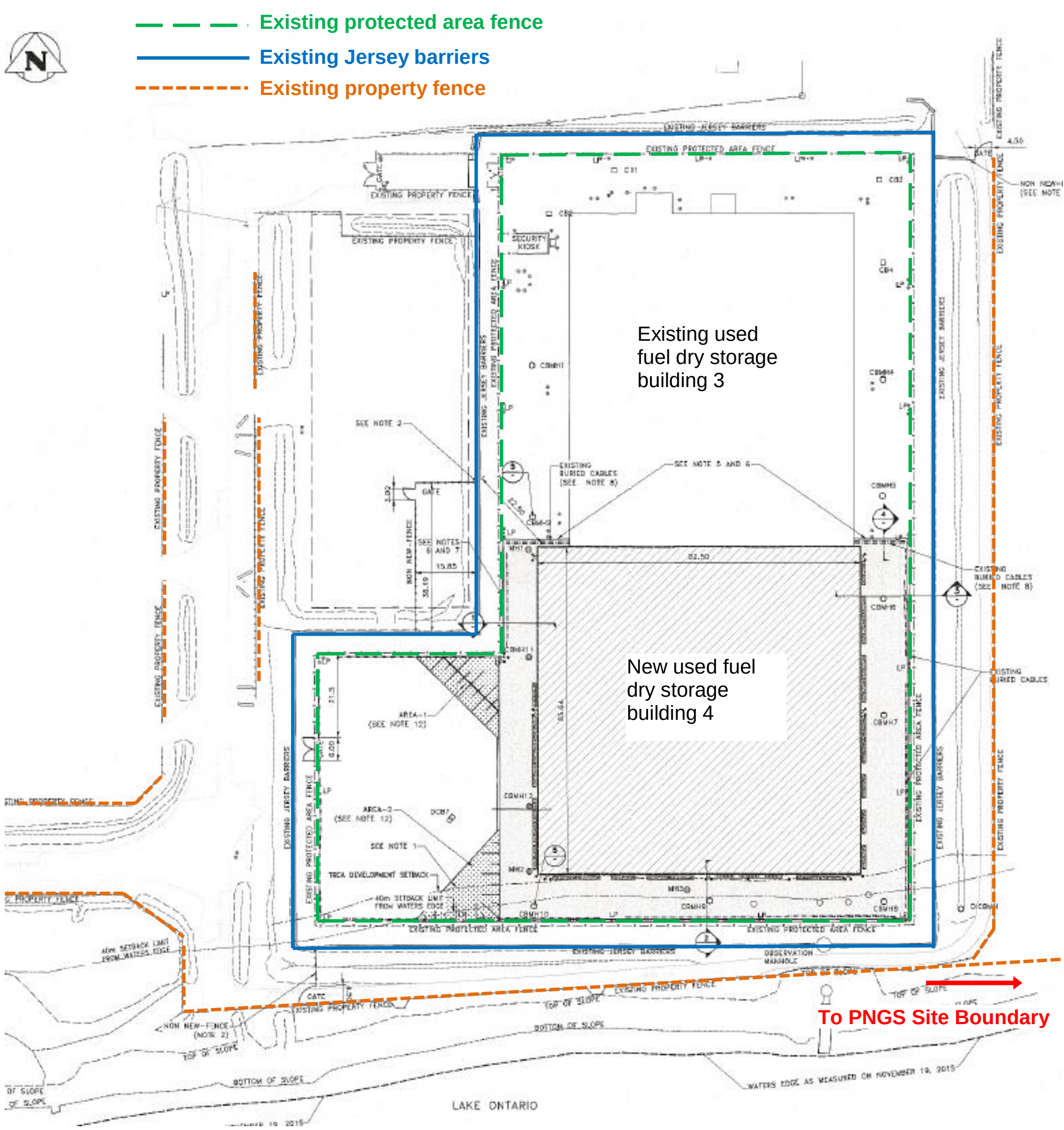

Figure 3 PWMF Phase II Protected Area Fence [3]

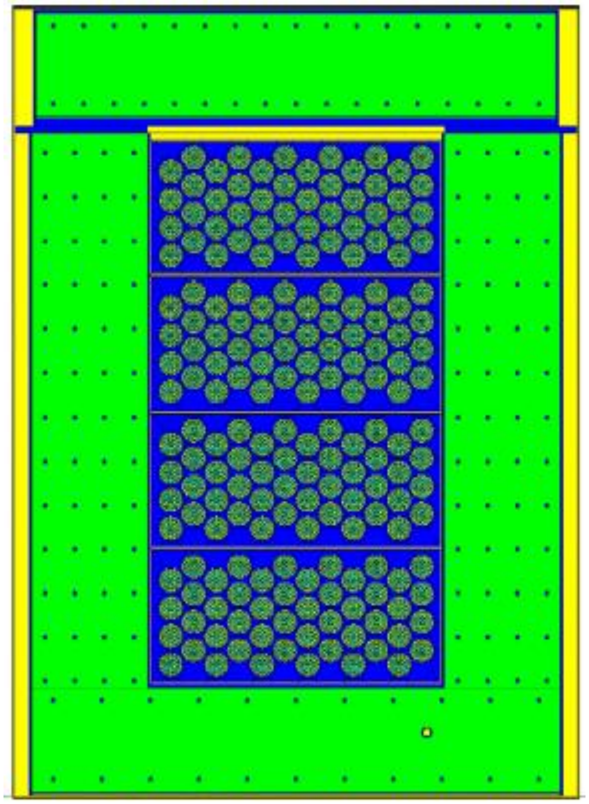

Figure 4 MCNP Representation of a Vertical Slice through a DSC Showing Used Fuel

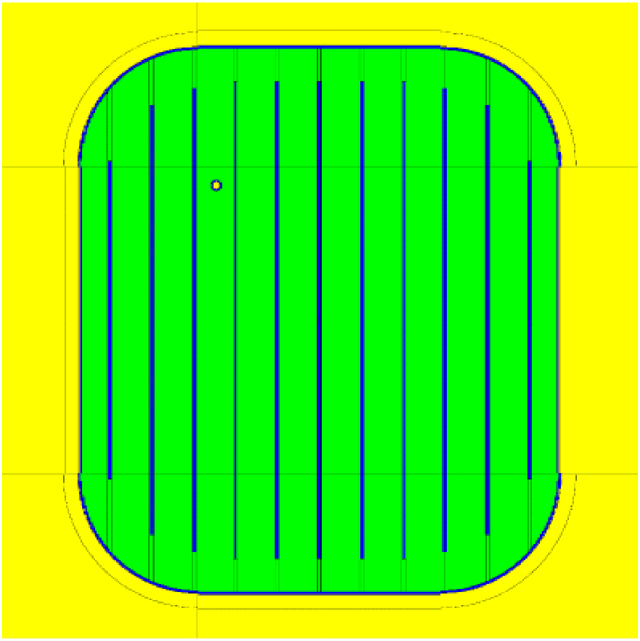

Figure 5 MCNP Representation of a Horizontal Slice through a DSC Showing Rebar Present in the DSC Base

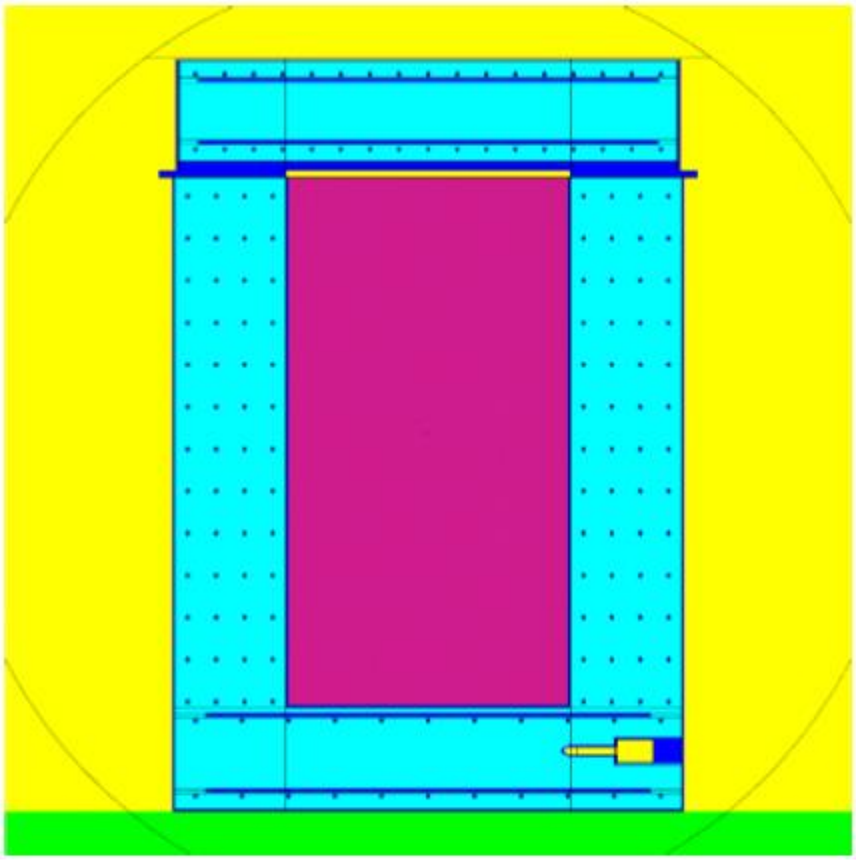

Figure 6 MCNP Representation of a Vertical Slice through a DSC Showing the Homogenized Used Fuel Region

30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	25	25	25	25	25	25	25
30	30	30	6	6	6	6	**25**	**25**	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	6	6	6	6	6	6	**25**
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25
30	30	30	30	25	25	25	25	25	25

25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	6	6	6	6	6	6	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25
25	25	25	25	25	25	25	25	25	25

Values shown in the figure correspond to the DSC decay time in years

Figure 7 SB3 Loading Pattern[16]

16 19 DSCs to be moved back to SB3, after loading the DSCs containing 6 year decayed used fuel, into locations indicated in bold.

35	20	20	20	20	20	20	20	20	20	20	20	20
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
35	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
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30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
30	20	15	15	10	10	10	10	10	10	10	10	10
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30	20	15	15	15	15	15	15	15	15	15	15	15
40	20	15	15	15	15	15	15	15	15	15	15	15
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40	20	20	20	20	20	20	20	20	20	20	20	20
40	35	35	35	35	35	35	35	35	35	35	35	40
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20	20	20	20	20	20	20	20	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
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10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
10	10	10	10	10	15	15	15	20	25	40
15	15	15	15	15	15	15	15	20	25	40
15	15	15	15	15	15	15	15	20	25	40
15	15	15	15	15	15	15	15	20	25	40
20	20	20	20	20	20	20	20	20	25	40
20	25	25	25	25	25	25	25	25	25	40
40	35	35	40	40	40	40	40	40	40	40
40	40	40	40	40	40	40	40	40	40	40

Values shown in the figure correspond to the DSC decay time in years

Figure 8 SB4 Loading Pattern [3]

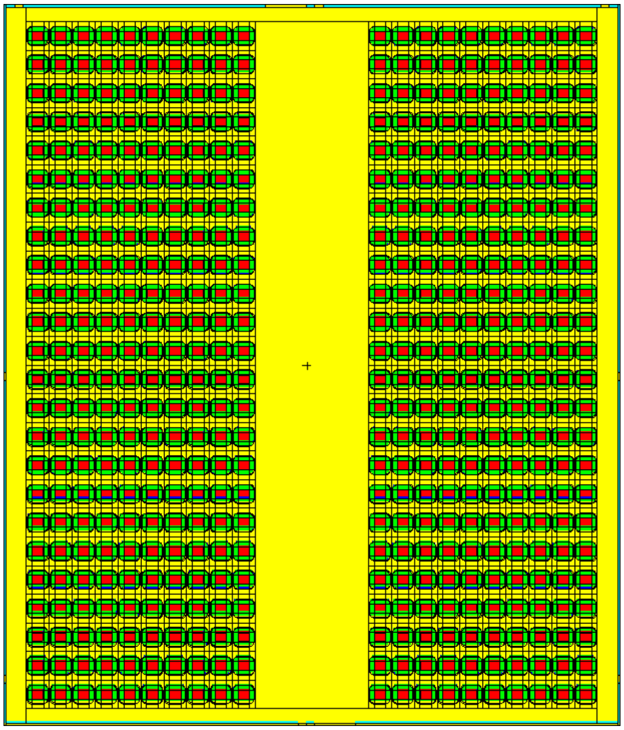

Figure 9 MCNP Representation of the DSC Layout in SB3

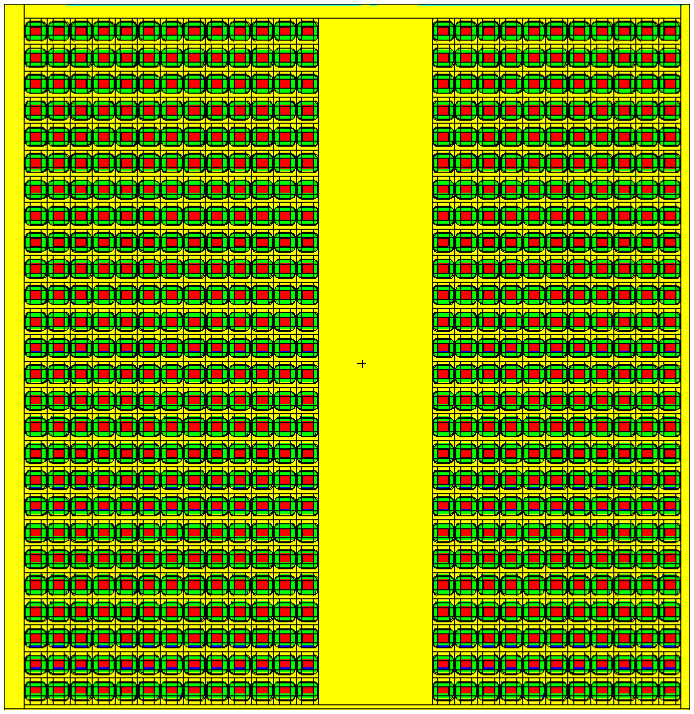

Figure 10 MCNP Representation of the DSC Layout in SB4

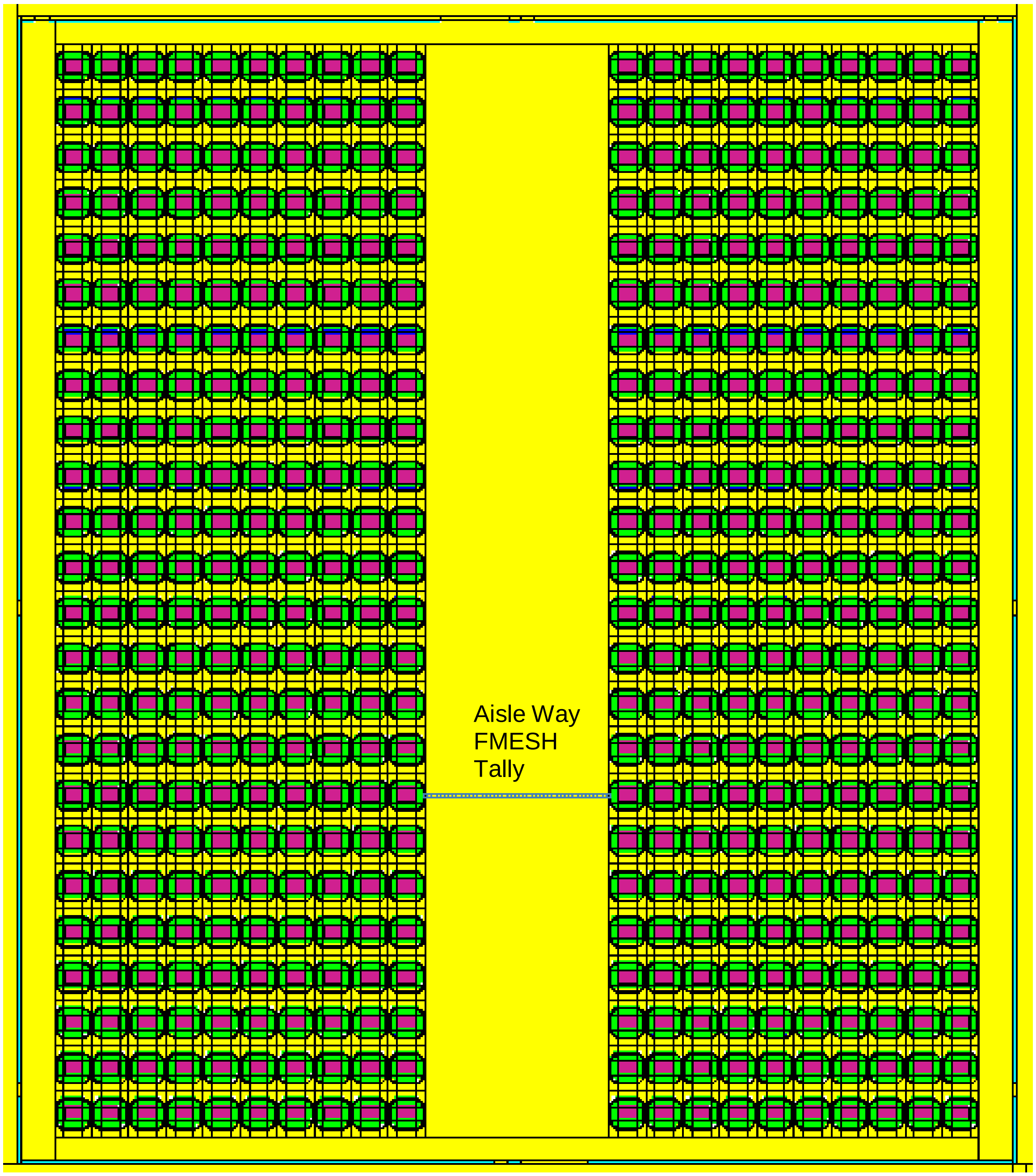

Figure 11 MCNP Representation of FMESH Tally across Aisle Way of SB3[17]

17 The dose rate profile across the aisle way is shown in Figure 15.

Figure 12 Dose Receptor Locations

Figure 13 Sketch of Mesh Tally Locations (green) around SB3 and SB4

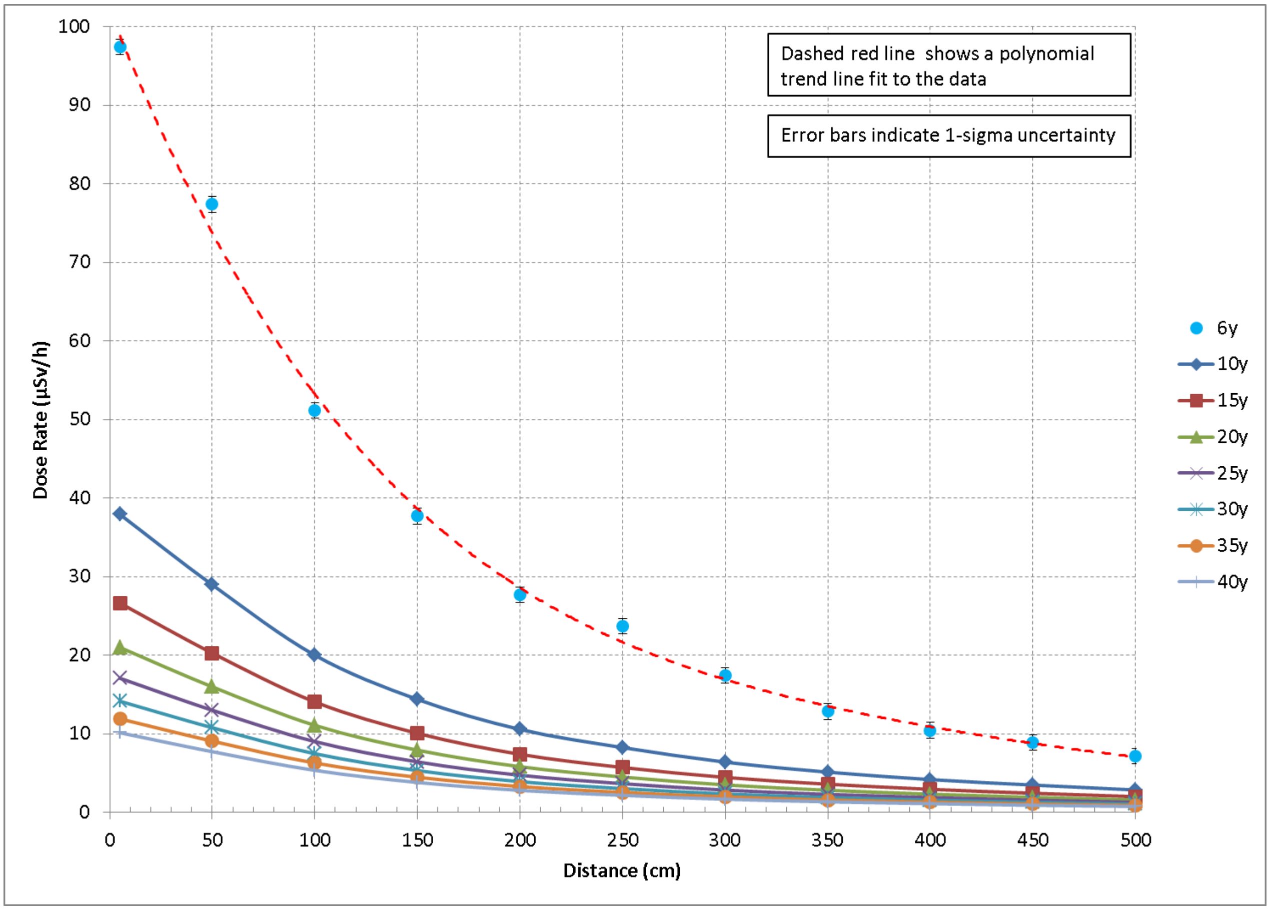

Figure 14 Calculated Dose Rate vs. Distance from the Long Side of a Single DSC at Various Fuel Decay Times

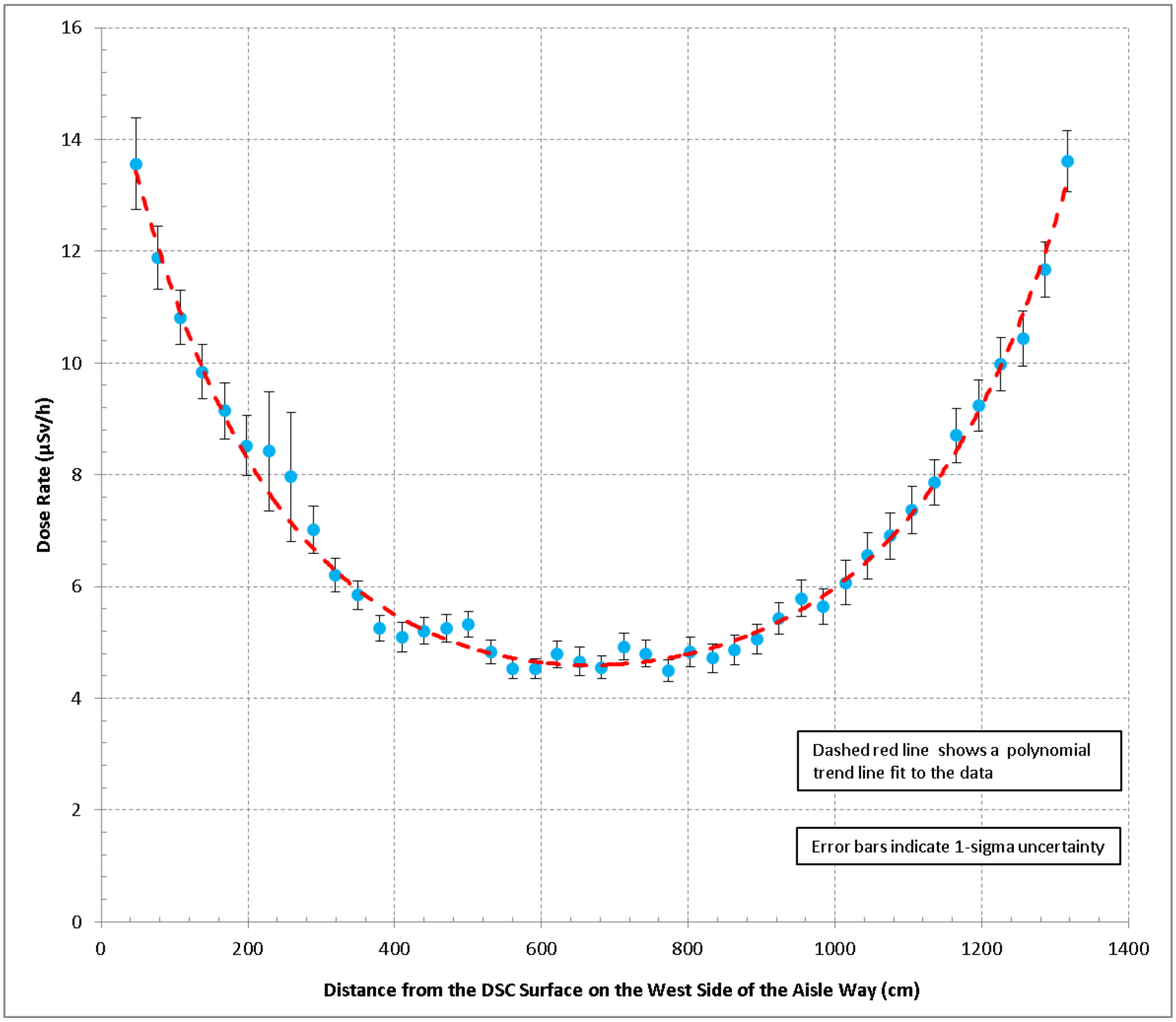

Figure 15 Dose Rate (µSv/h) Across the Main Aisle Way of SB3[18]

18 The data points at ~230 cm and ~260 cm have associated uncertainties outside the target specified in Section 3.10. These 2 points are included in the figure for presentation purposes only and are excluded from the generation of the trend line fit.

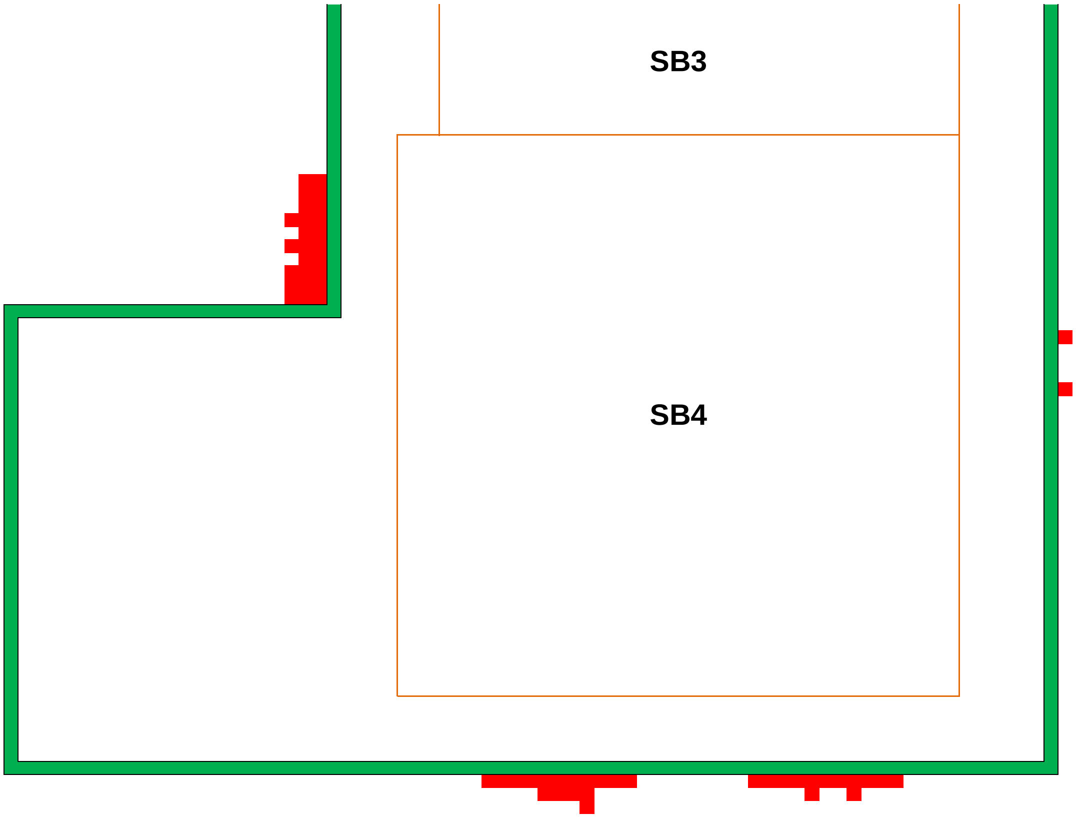

Figure 16 Representation of Dose Rates Exceeding Acceptance Criterion (red) Outside PWMF Phase II Protected Area Fence (green)[19]

19 Dose rates (best estimate + 2σ uncertainty) are compared against the 0.5 µSv/h acceptance criterion (see Section 2). Each cell corresponds to a 2 m x 2 m x 2 m volume.

ENCLOSURE #4

OPG letter, P. Dinner to Y. Mroueh, “Additional Information Concerning: Thermal Gradients Pertaining to Dry Storage Containers (DSCs)” 00104-CORR-79171-0139942

ONTARIO POWER GENERATION

700 University Avenue Toronto, Ontario M5G 1X6

May 4, 2005

File#: 00104-79171 (P)

Dr. Youssef Mroueh
2120 Blue Ridge Crescent
Pickering, Ontario
L1X 2N3

Dear Dr. Mroueh:

Additional Information Concerning Thermal Gradients Pertaining to Dry Storage Containers (DSCs)

Reference: Paul Dinner, letter to Dr Youssef Mroueh, "Questions in Connection with the Dry Storage Container for Used Fuel", September 10, 2004.

This letter is a follow-up to our discussions last August and my letter of September 10, 2004. At that time we discussed, amongst other issues, the subject of heat transfer in the loaded DSC. In section (b) of my letter, I described the process of heat transfer in the DSC, the temperature monitoring program conducted in the field in 1998 on a specially instrumented DSC containing 6-year old fuel (DSC 0024), and the simulation modeling carried out for us by ANSYS using their computational fluid dynamics software. I noted in summary that temperatures, both calculated and measured, are much less than permitted by the DSC design.

00104-79171 - 0139942

Recently, I understand that you spoke with my colleague Dr. Atika Khan concerning the early work done on the Concrete Integrated Container (CIC), an experimental fuel storage container which was built and tested at Pickering during the early 1990's. You requested information from that testing program pertaining to the thermal behaviour of the CIC. I believe that you may have been given the impression that a report containing consolidated information on the dismantling of the CIC's existed. I have not been able to locate such a report (indeed, the CIC's have never been dismantled – they were unloaded, decontaminated, and put in storage at our Western Waste Management Facility. However, Dr. Khan has been able to provide me with results of the thermal measurements conducted between 1990 and 1993 (from a CIC designated as number 2). These data are plotted in Figure 1 (attached). Please note that the maximum inner liner temperature observed was about 50°C.

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p

The modest temperature conditions observed in the CIC containers were as expected and supported Ontario Hydro's decision to move forward with used fuel dry storage.

In order to provide you with the basis for a deeper understanding of the spatial temperature distribution in dry storage containers, I am enclosing a copy of our report on the Thermal Monitoring of Pickering DSC 0024. The data in this report provide the basis for the statements in paragraph 2 (above), discussed with you in our meeting of last August.

You will note that the temperatures from the CIC are consistent with those observed for the Dry Storage Container of the type in use today, although the DSC temperatures are nominally greater owing to the fact that DSC 0024 was situated amongst an array of other loaded DSCs in storage.

Thank you again for your interest in our Dry Storage program. We appreciate the efforts you have made to understand the key safety aspects of the DSC design.

Sincerely,

Paul Dinner
Manager, Nuclear Waste Engineering
Nuclear Waste Management Division

/attach.

cc: Frank King, H17-F20
Atika Khan, H17-B22

bcc: John Vincent
Terry Squire, H17-F26
Janice Hudson, P26 W2

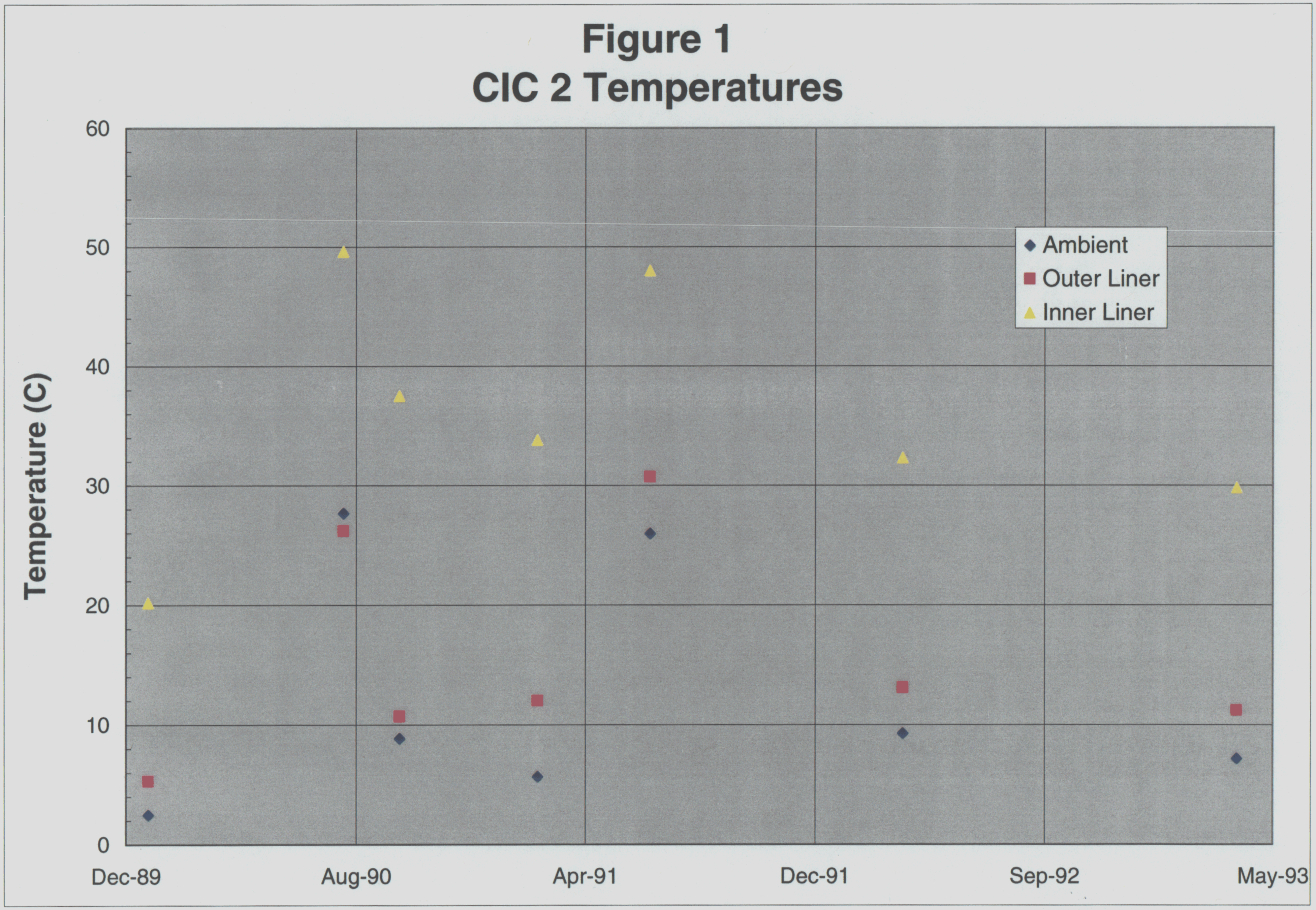

Figure 1
CIC 2 Temperatures
Temperature (C)
60
50
40
30
20
10
0
Dec-89
Aug-90
Apr-91
Dec-91
Sep-92
May-93
Ambient
Outer Liner
Inner Liner

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File:6119-1997	Revision: 0	Date: October 27, 1998

Temperature Measurements Summary Sheet

Ontario Hydro Technologies

Description: Thermal Monitoring of a Pickering Dry Storage Container (DSC) stored inside the Pickering Used Fuel Dry Storage Facility (PUFDSF) during June, July and August 1998.

Method Used: The required temperature data were obtained through a remote controlled Data Logging (DL) system. Measurements were made for a DSC instrumented with a total of 34 Thermocouples (TCs). The data were processed to produce plots of temperature histories and time-averages for all the TCs monitored. In addition, standard deviation, standard error and coefficients of variation were also calculated.

Conclusion: The average measured indoor air temperature is 26°C while the measured average outdoor ambient temperature is 22°C. That is 12°C lower than the design indoor air temperature of 38°C used for the DSC analysis. The recorded maximum temperature at the DSC's inner liner is 61°C. When this maximum inner liner temperature is corrected for the 12°C difference between the design and measured indoor air temperatures it gives a maximum inner liner temperature of 73°C. This is 21°C lower than the predicted maximum inner liner temperature of 94°C. Therefore the temperature results of the monitoring program confirm that the thermal performance of the DSC is much better than predicted.

Prepared by: D. Taralis *D. Taralis*
Title/Department: Sr. Research Engineer/ Mechanical Systems Performance/ OHT
Date: October 27, 1998

Reviewed by: B.E. Mills
Title/Department: Sr. Research Engineer/ Mechanical Systems Performance/ OHT
Date: October 27, 1998

Approved by: W.H.S. Lawson
Title/Department: Manager/ Mechanical Systems Performance/ OHT
Date: October 27, 1998

THERMAL PERFORMANCE VERIFICATION OF PICKERING DRY STORAGE CONTAINER

1. Introduction

The thermal monitoring program of a full-scale Pickering Dry Storage Container (DSC) was developed to address AECB requirements regarding safe temperatures for used fuel during dry storage. The objective of the temperature measurements was to confirm that the thermal performance of the container stored within the Pickering Used Fuel Dry Storage Facility (PUFDSF) is not worse than predicted/1/ under the worst case scenario of 6 year old fuel and summer temperatures.

To accomplish the above objective, Pickering DSC #24 was instrumented with 24 Thermocouples (TCs) at the inner and outer surfaces of the walls. The instrumented container was loaded with 6 year cooled fuel and placed within an array of DSCs containing 10 year old fuel. A remotely controlled Data Logging (DL) System designed for this application was used to collect the required temperature data at regular intervals of time over a three-month period (June, July and August, 1998).

This report describes the instrumented DSC, the Data Logging (DL) System used to collect the data as well as the method for processing and analysing the results. Furthermore, the maximum temperatures registered on the inner liner are identified and compared with the predicted inner liner temperatures. The final conclusions reached from the monitoring program are also presented.

a) Description of Test Equipment & Monitoring System

This Section describes the main components of the equipment used for making the required temperature measurements:

- **Instrumented DSC**
- **DSC Test Array**
- **Data Logging System**

Instrumented DSC: Pickering DSC #24 was instrumented with 24 TCs during its construction phase. The specifications for the TCs used and their locations are described below. All TCs were tested before and after installation to ensure proper operation.

Thermocouple Specifications: The temperature measurements were made using type K TCs which conform to QA standards of ANSI M96.1-1992 and ASTM E230-1993. The TC wire size used was 30 AWG. An inconel outer sheath of 1.5 mm (1/16") diameter contained the TC wires which were embedded in compressed magnesium dioxide powder insulation. The length of the TCs varied, depending on the location of their junction, and were sized to within 0.5 m of the actual routed length. The TCs leads were terminated with integral miniature female connectors to facilitate connections to the DL system.

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Thermocouple Locations: Figures 1 and 2 show the location of the TC junctions where the temperature measurements were made. These TCs were grouped (for installation purposes) as follows:

a. Fourteen TCs on the outer surface of the inner liner

b. One TC at the outer surface of the outer liner (bottom)

c. Nine TCs at the inner surface of the outer liner

d. Six TCs in the air spaces between the instrumented DSC and the surrounding DSCs

   These TCs were installed on two adjacent vertical sides of the instrumented DSC at three elevations and were attached to the DSC outer surfaces using magnetic mounting. The TCs extend out horizontally from the vertical surfaces (0.1 m from one side and 0.3 m from the other).

e. One TC above the DSC top (0.1 m away from the top surface)

f. Three TCs for ambient air (2 TCs for air within the Dry Storage Facility (DSF) and 1 TC for outdoor air)

A total of 34 TCs (24 on DSC and 10 for air) were used to make the temperature measurements. For redundancy purposes there were two locations on the DSC lid (inner and outer liners) which were instrumented with 3 TCs at each location.

Thermocouple Installation: The installation of the TCs was carried out by co-ordinating the fabrication schedule of the DSC with the installation procedure of the instrumentation (actual installation of TCs was carried out by OHT personnel with the assistance of the DSC manufacturer/2/). To interface with the normal production process all the required information and drawings were provided to the manufacturer well in advance. Any changes considered necessary were made to the standard DSC produotion procedure with hold points to facilitate the installation of the instrumentation. Further details are available in Reference/3/.

DSC Test Array: The thermal performance verification test was carried out within the PUFDSF. The instrumented DSC was fully loaded with 384 fuel bundles of approximately 6 year old (estimated decay heat of 7.4 W/Bundle) and stored within the facility in a test array as shown in Figure 3. The DSCs in the immediate vicinity of the instrumented container were loaded with 10 year used fuel as defined in the test plan submitted to the AECB/1/. When the required number of loaded DSCs were in place, the monitoring program began.

Data Logging System: The temperature data were collected using a Campbell Scientific portable Data Logging (DL) System (Model 21X). This system (see Figure 4) is capable of accepting up to 64 TCs or other voltage inputs. It can store up to several days of data (based on half-hour interval scanning) before it becomes necessary to download the data to the controlling PC. The DL System was located in a nearby location within the PUFDSF (see Figure 2) and connected to the TCs by a standard thermocouple extension cable which is normally used for such applications. The DL System was calibrated prior to the collection of temperature data and also at the end of the monitoring period. Based on this calibration the accuracy of all the temperature measurements made was within ± 2.2 °C (standard error of type K TCs is 2.2 °C or ± 0.75% whichever is greater in the temperature range of 0 to 1250 °C).

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To control the monitoring of the instrumented DSC remotely, a telephone/modem connection was made to the data logger at the Remote Site and to a PC at the receiving end. This enabled real-time monitoring and data collection from the OHT site (800 Kipling Ave., Toronto) and minimized the requirements for frequent site visits.

The monitoring program conducted covered a period of three months. Since the main purpose of the study was to collect data that represent the maximum possible temperatures, the program was carried out during the hottest months of the year (i.e. June through August). The results of this study are presented below:

b) Presentation of Temperature Measurements

To facilitate the presentation of the field measurements, the temperature data obtained from all the TCs have been grouped as follows:

Air & Air Spaces

- Outdoor ambient air temperature and its arithmetic and time-averages (Figure 5)
- Indoor air temperature (Figure 6)
- Air temperature between two DSC vertical sides and surrounding DSCs including top of DSC (Figure 7)

Outer Liner:

- Two adjacent vertical sides (Figure 8)
- Top and bottom (Figure 9)

Inner Liner

- Two adjacent vertical sides (Figure 10)
- Top and bottom (Figure 11)

The maximum temperatures obtained over the entire monitoring period at locations of interest are presented in Table 1. The corresponding temperature distribution throughout the container is illustrated in Figure 12.

The maximum temperature measured (61°C) occurred on July 24, 1998 (about 45 days after the monitoring program began). This temperature corresponds to TC location #4 (which is located near the top of the inner liner approximately 1.89m from the bottom of the cavity as shown in Figure 1).

In addition to the above results, the time-average of all the TCs including the ambient outdoor temperature and standard deviation have been calculated. The equations used for obtaining the statistical quantities are presented in Table 2 and the results obtained in Table 3. These results have been processed further to produce the average values shown in Table 4 for the TC groups presented above.

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c) **Analysis and Assessment of Temperature Measurements**

This Section describes the analysis of the temperature measurements including the assessment of conditions under which the measurements were made.

Temperature difference across the walls: An examination of the inner and outer liner temperatures (see Table 3 for TC#4 & TC#20) reveals a maximum temperature difference of the order of 20 °C, occurring across the DSC vertical wall adjacent to the drain and vent side. This is to be expected, since this side and opposite side are closer to the surrounding DSCs. The minimum difference occurs across the bottom wall (approximately 7 °C). This smaller temperature difference indicates that the heat transfer rate to the floor is not as high.

Variation of wall temperatures with DSC height: Further examination of the results obtained (see Figure 12 reveals that the inner liner temperature increases with the vertical distance from the bottom of the DSC cavity until a height of 1.89 m is reached (location of maximum temperature of 61 °C). Beyond this height the vertical wall temperatures begin to fall. This observation is consistent with previous analytical results presented in the Pickering Safety Report (SR)/1/.

Correlation of Outdoor versus Indoor Air Temperatures: Figure 13 shows the outdoor (TC#34) and indoor (TC#33) air temperatures as well as the outer (TC#20) and inner (TC#4) liner temperatures. As this figure reveals, a given fluctuation in outdoor ambient temperature is followed by a much smaller fluctuation in the indoor air temperature with a slight delay. The outer surface temperature fluctuations due to the DSC thermal inertia are even smaller. Furthermore, the maximum inner liner temperature is rather insensitive to these outdoor air temperature fluctuations. Although the highest outdoor temperature recorded occurred much later (August 11/98), the maximum inner liner temperature did not increase any further. In fact this maximum temperature began to decline, reaching 58 °C at the end of the monitoring period (September 9, 1998). It appears that due to the large thermal inertia of the container the DSC temperatures are only a function of the time-average of the outdoor ambient temperature.

Assessment of Conditions During Measurements: The above results are assessed with respect to the following parameters:

a. Test array containing a DSC loaded with 6 year old fuel at the centre of the array
b. Hottest weather conditions with daily temperature variation versus DSC analysis carried out using constant indoor air temperature of 38°C
c. Facility being less than full capacity at time of measurements
d. Covered ventilation louvres versus openings for ventilation in the analysis
e. Time between loading and beginning of measurements

Effects of Fuel Age: The fuel stored within the DSCs will be much older than 6 years (minimum 10 year old; 15 years old on the average). The corresponding decay heat of the fuel will be lower and consequently the DSC temperatures are expected to be lower.

Effects of Lower Ambient Conditions: As Figure 5 and Table 1 (TC#34) indicate, the maximum outdoor air temperature during the entire 3-month monitoring period reached 38.6 °C for a short instant of time (occurred August 11/98 at 3:30 pm). The time-average of the ambient air temperature experienced during the monitoring period was determined to be about 22°C (see Figure 5 and Table 3). The difference between the time-average of the indoor air temperature (26°C) and

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the temperature used in the DSC analysis/1/ (38 °C) is 12°C. A previous study /4/ indicated, a nearly linear relationship between ambient air temperature and DSC temperatures. Therefore, adding the 12°C difference to the actual maximum inner liner temperature of 61°C results in 73°C. This is 21°C below the predicted maximum inner liner temperature of 94°C.

The difference between measurements and predictions can be attributed to the various conservative assumptions made in the analysis, such as DSC being in non-black environment (zero radiant heat loss from the DSC outer surface) and conservative decay heat value.

Effects of Closed Ventilation Louvres: During the entire monitoring period the facility's ventilation louvres remained closed. The effect of this would be to restrict the indoor warm air from exiting the facility. If the louvres were kept open, the indoor air as well as DSC temperatures would have been somewhat lower than the present measured values.

Effects of Partially Filled Facility: Since the facility was not near full capacity (about 33% occupied with loaded DSCs and about 20% with empty DSCs) at the time of measurements, the temperatures for a full facility would be expected to be somewhat higher during the hottest season. This is because the difference between the indoor and outdoor average temperatures is only 4°C for the presently 1/3 filled facility. Assuming the decay heat of the fuel remains constant, when the facility becomes 100% full with loaded DSCs, the maximum additional increase in facility and DSC temperatures would be expected to be 8°C (assuming 4°C for each additional one third of facility being filled). However, even if such an increase occurred, the maximum inner liner temperature would still be lower by 13°C (21°C - 8°C) than the predicted maximum of 94°C.

Based on the present results and the fact that the decay heat of the fuel decreases with age, it seems very unlikely that the DSC temperatures will increase appreciably (as the facility becomes full) in subsequent years.

Effects of DSC Thermal Inertia: Although the DSC has large thermal inertia due to its mass, it takes of the order of two weeks for the container to reach steady-state conditions /4,5/ following a given change in outdoor ambient air temperature. Therefore, since the monitoring period of 3 months is considered sufficiently long for the DSC to reach thermal equilibrium within the facility, it follows that the maximum DSC temperatures were reached during this test period.

Effects of Time Between Loading & Measurements: The elapsed time between the loading of the DSC with 6 year old fuel and the time the DSC was placed in the facility (within the test array) was relatively long (about two weeks). Therefore, at the time of measurements the DSC was close to thermal equilibrium with a maximum inner liner temperature of about 57 °C (see Figure 10). As the monitoring results indicate at the beginning of measurements (about 10 to 15 days after loading the container) the DSC temperatures were decreasing as a result of the lower outdoor temperature (see Figure 5).

d) Conclusions

Based on the analysis of the temperature data obtained from the thermal monitoring of the Pickering DSC loaded with 6 year old fuel the following conclusions can be made:

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a. The time-averaged air temperature within the facility during summer was 26°C. The corresponding time-averaged outdoor ambient temperature was 22°C. The inner wall temperature reached a maximum value of 61 °C. This temperature (when adjusted by 12°C to account for the difference between the design indoor ambient air temperature of 38°C and the measured value of 26°C) is 73°C, which is still 21°C below the predicted value of 94°C.

b. The effect of the facility not being full at the time of DSC temperature measurements was estimated to be of the order of 8°C higher. Accounting for this effect, the DSC inner liner maximum temperature is still expected to remain below the predicted maximum of 94°C (about 13°C lower).

c. Using the measured maximum inner liner temperature (adjusted for the difference in the indoor air temperatures between measurements and analysis) the fuel sheath temperature is expected to be lower by about 21°C than the previously predicted value of 173°C (i.e.152°C).

In the long term, as the fuel ages, the heat load of the containers will gradually fall resulting in lower fuel sheath and DSC wall temperatures, thus further reducing the likelihood of any temperature-related degradation of the fuel and the container.

References

1. Pickering Used Fuel Dry Storage Facility, Safety Report, File 907-92896-00531-P, January 21, 1994.

2. Taylor Forge Ltd., Niagara Falls, Ontario, Canada (DSC Manufacturer).

3. Mills, B.E. and Taralis, D., "Dry Storage Container Thermal Performance Verification Test Plan", OHT Document, File: 825.122, October 24, 1995.

4. Bruce Used Fuel Dry Storage Facility Safety Report, A Report to the Atomic Energy Control Board, 1997.

5. Taralis, D., Mills, B. E. and Shah, N. N., "Thermal Testing of a Concrete Integrated Container (CIC) for the Ukraine Dry Storage Program: Phase I", Ontario Hydro Technologies, Report No. A-NBP-96-84-CON.

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TABLE 1
MEASUREMENTS OF MAXIMUM TEMPERATURES

Description/Location[1]	Temperature, °C
Inner Liner	
•Adjacent to Drain &Vent Side	60.9
•Opposite to Drain & Vent Side	52.5
Outer liner	
•Adjacent to Drain &Vent Side	42.3
•Opposite to Drain & Vent Side	41.9
Air Space	34.6
Indoor Ambient	33.7
Outdoor Ambient	38.6

(1) For Location See Figures 1 & 2

TABLE 2
EQUATIONS FOR CALCULATING STATISTICAL QUANTITIES FROM TEMPERATURE MEASUREMENTS

Statistical Quantity	Equation
Time-Average, °C	$\overline{T_{ji}} = \{ T_{j1}\,\Delta t_1 + T_{j2}\,\Delta t_2 + \bullet\bullet\bullet + T_{jn}\,\Delta t_n \}/t$
Arithmetic Average, °C	$T^{*}_{j} = 1/n \sum_{i=1}^{n} T_{ji}$
Standard Deviation, °C	$\sigma_j = \{ 1/n \sum_{i=1}^{n} (T_{ji} - \overline{T_j})^2 \}^{1/2}$
Standard Deviation of the Mean, °C (Standard Error)	$\varepsilon_j = \sigma_j / n^{1/2}$
Coefficient of Variation	$C_j = \sigma_j / \overline{T_j}$

j=1 to 34 (number of channels)
i = 1 to n (number of time steps/sampling points)
t = time, hours (measured from beginning of monitoring period)

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TABLE 3
SUMMARY OF TEMPERATURE MEASUREMENTS

TC #	Maximum Temperature (°C)	Time[1] of Occurrence of Maximum Temperature (Days)	Minimum Temperature (°C)	Time[1] of Occurrence Of Minimum Temperature (Days)	Mean[2] Temperature (°C)	Standard Deviation	Standard Error	Coefficient of variation
1	44.3	82.38	35.9	4.17	42.30	2.20	0.03	0.05
2	54.3	80.02	45.9	4.27	52.38	2.10	0.03	0.04
3	59.7	44.69	51.2	4.25	57.69	2.02	0.03	0.03
4	60.9	44.69	52.3	4.02	58.88	1.97	0.03	0.03
5	50.5	43.67	41.4	3.14	48.40	1.97	0.03	0.04
6	42.6	80.21	34.0	3.69	40.55	2.19	0.03	0.05
7	48.6	80.21	40.0	3.69	46.69	2.11	0.03	0.05
8	52.2	44.77	43.6	4.23	50.22	2.04	0.03	0.04
9	52.5	44.64	43.7	3.38	50.49	2.00	0.03	0.04
10	48.1	43.69	39.1	3.17	46.04	1.97	0.03	0.04
11	43.3	79.98	34.8	4.69	41.12	2.28	0.03	0.06
12	56.0	44.52	47.3	3.17	53.98	1.89	0.03	0.03
13	55.9	44.56	47.3	3.19	53.94	1.89	0.03	0.03
14	56.0	44.56	47.4	3.19	54.03	1.89	0.03	0.03
15	37.0	81.61	28.6	4.27	34.73	2.33	0.04	0.07
16	39.5	43.14	27.7	2.69	36.25	2.09	0.03	0.06
17	39.4	69.21	27.7	2.67	36.24	2.07	0.03	0.06
18	39.4	43.17	27.6	2.67	36.22	2.09	0.03	0.06
19	38.1	79.29	28.8	2.71	35.89	2.13	0.03	0.06
20	41.9	78.19	31.8	2.67	39.53	2.05	0.03	0.05
21	42.3	43.27	31.9	2.79	39.93	2.07	0.03	0.05
22	37.1	79.23	27.3	2.71	34.84	2.23	0.03	0.06
23	41.7	43.11	31.8	2.71	39.33	2.09	0.03	0.05
24	41.9	43.33	32.1	2.88	39.69	2.10	0.03	0.05
25	30.9	69.17	16.1	2.64	26.44	2.42	0.04	0.09
26	32.6	76.17	16.5	2.64	27.40	2.56	0.04	0.09
27	33.5	69.19	16.9	2.67	28.41	2.67	0.04	0.09
28	31.0	76.19	16.2	2.64	26.42	2.44	0.04	0.09
29	32.7	76.14	16.8	2.67	27.30	2.53	0.04	0.09
30	34.6	76.19	18.0	0.96	29.66	2.52	0.04	0.08
31	33.8	76.17	17.0	2.67	28.45	2.69	0.04	0.09
32	33.7	76.19	17.4	2.64	28.74	2.62	0.04	0.09
33	29.8	76.19	17.0	2.67	26.35	2.20	0.03	0.08
34	38.6	63.02	11.4	1.04	21.90	3.93	0.06	0.18

Notes:

(1) Time in days is measured from the beginning of the monitoring period (June 9, 1998)
(2) Time-Average over the entire monitoring period (see Figure 5 for comparison with arithmetic average)
(3) End of monitoring period = September 9, 1998

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TABLE 4
DSC WALL AND AIR SPACE AVERAGE TEMPERATURES
(BASED ON TIME-AVERAGED VALUES)

Description	Temperatures, °C
Ambient Air:	
Indoor Air (Near DSC)	28.7
Indoor Air (Near DL System)	26.3
Outdoor Air	21.9
Air Spaces:	
Vertical Wall (Adjacent to Drain & Vent)	27.4
Vertical Wall (Opposite to Drain & Vent)	27.8
Top	28.4
Outer Liner:	
Vertical Wall (Adjacent to Drain & Vent)	38.4
Vertical Wall (Opposite to Drain & Vent)	37.9
Top	36.2
Bottom	34.7
Inner Liner:	
Vertical Wall (Adjacent to Drain & Vent)	51.9
Vertical Wall (Opposite to Drain & Vent)	46.8
Top	54.0
Bottom	41.1

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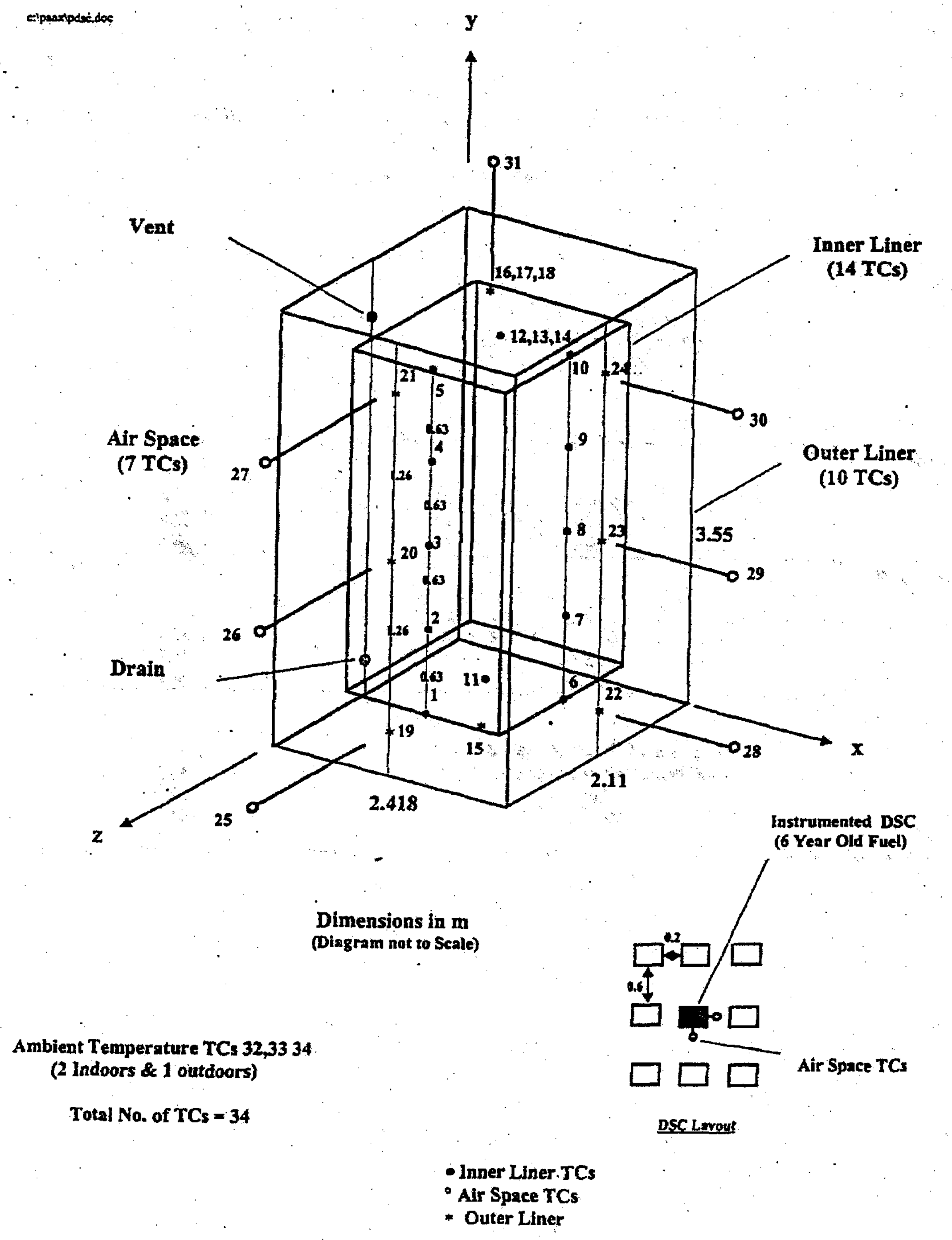

FIGURE 1
INSTRUMENTED PDSC
(THERMOCOUPLE LOCATIONS AND NUMBERING SCHEME)

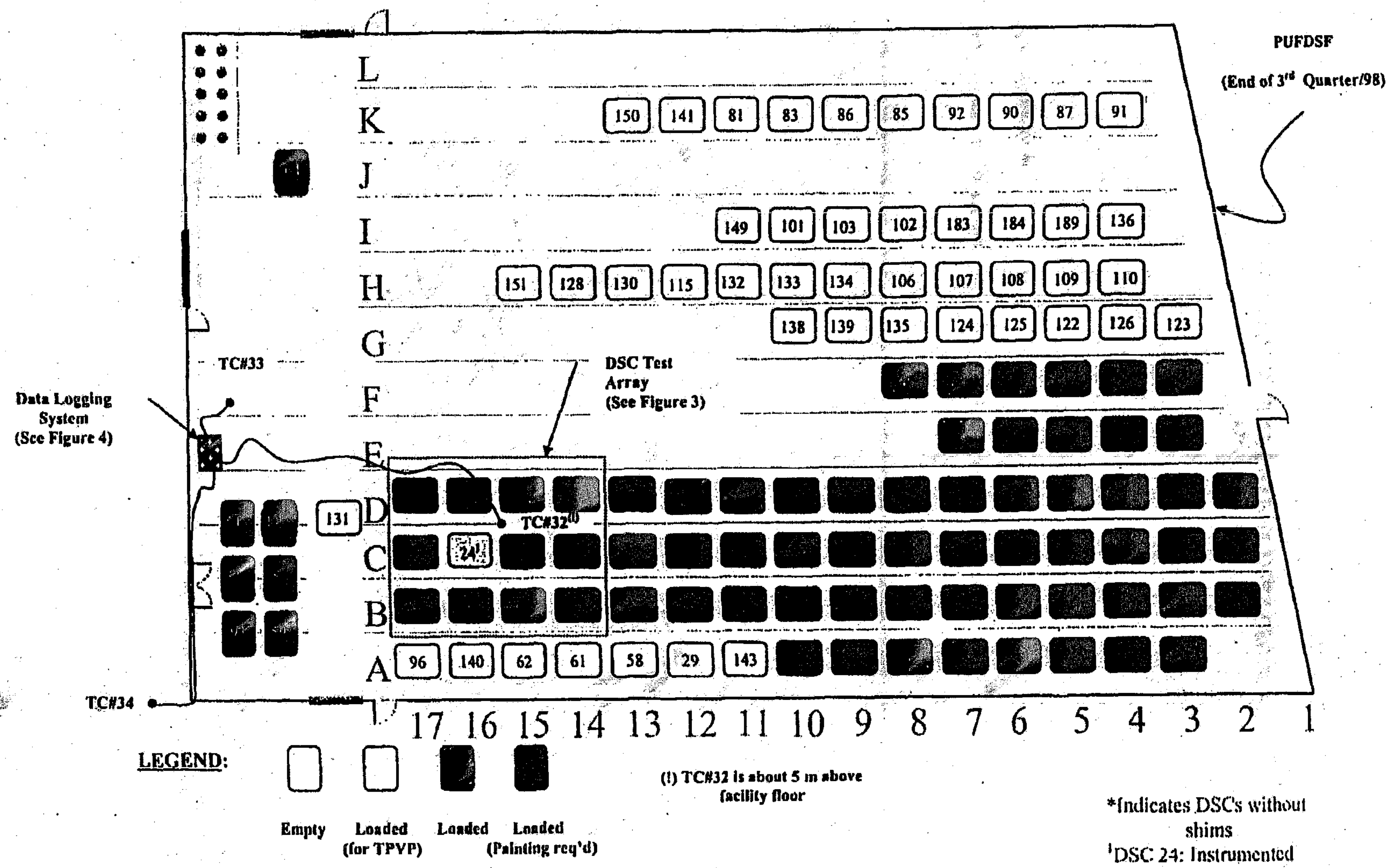

FIGURE 2
LOCATIONS OF INDOOR AND OUTDOOR AMBIENT TEMPERATURE MEASUREMENTS

*Indicates DSC's without shims
[1]DSC 24: Instrumented for TPVP
[2]Indicates DSC's needing painting

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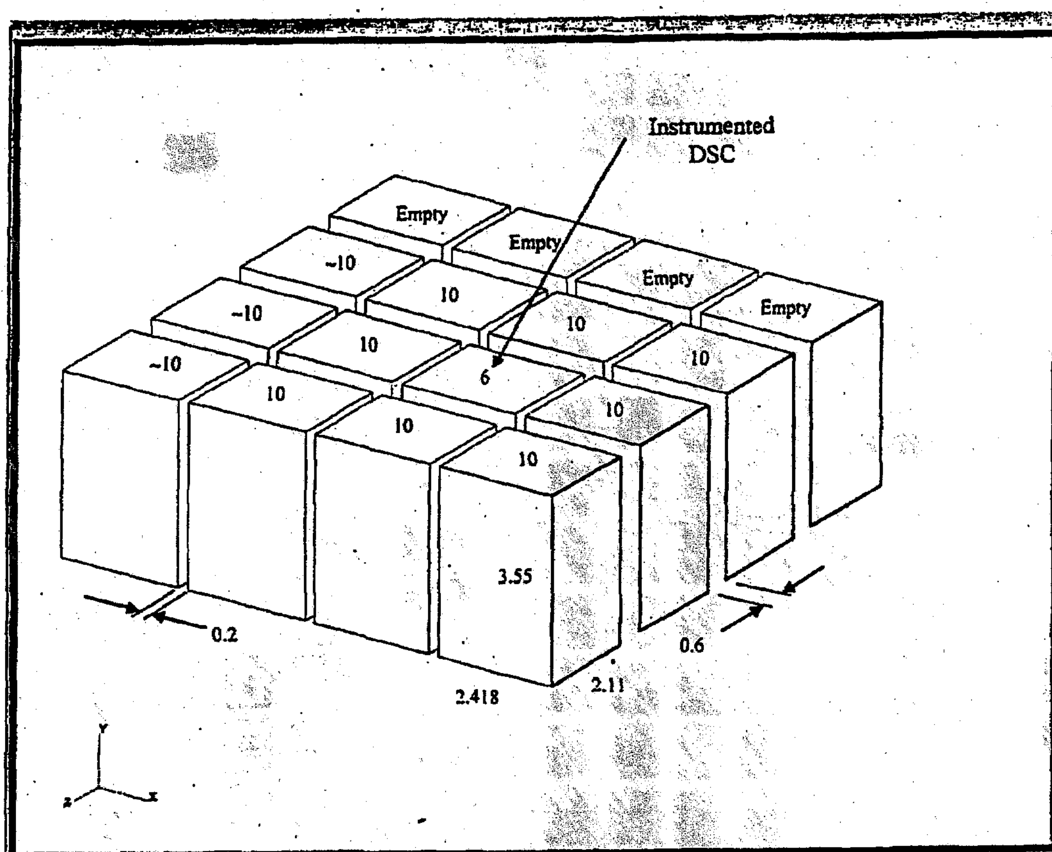

Note: Numbers on the Top Plane Indicate Age of Fuel

Dimensions in m

FIGURE 3
DSC TEST ARRAY WITHIN THE PUFDSF

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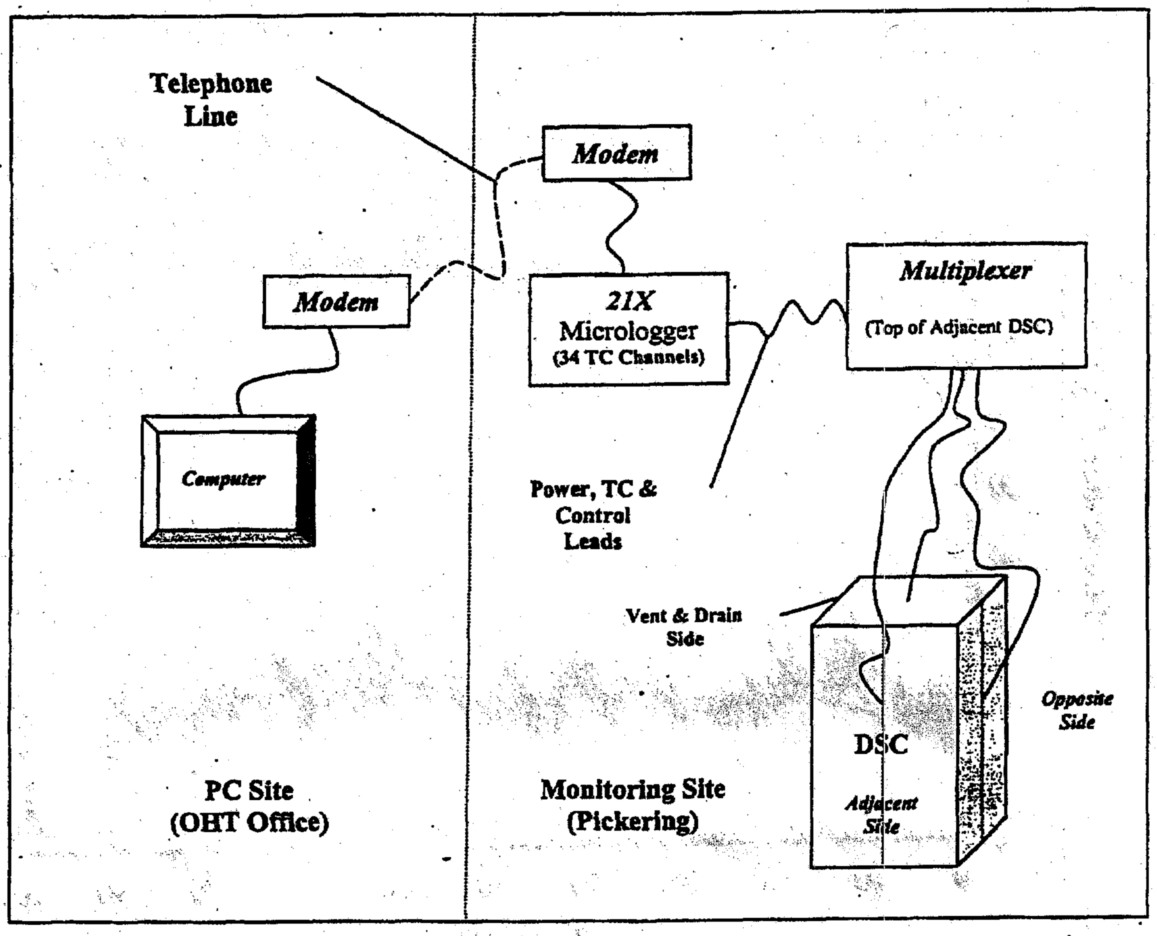

FIGURE 4
REMOTELY CONTROLLED DATA LOGGING SYSTEM

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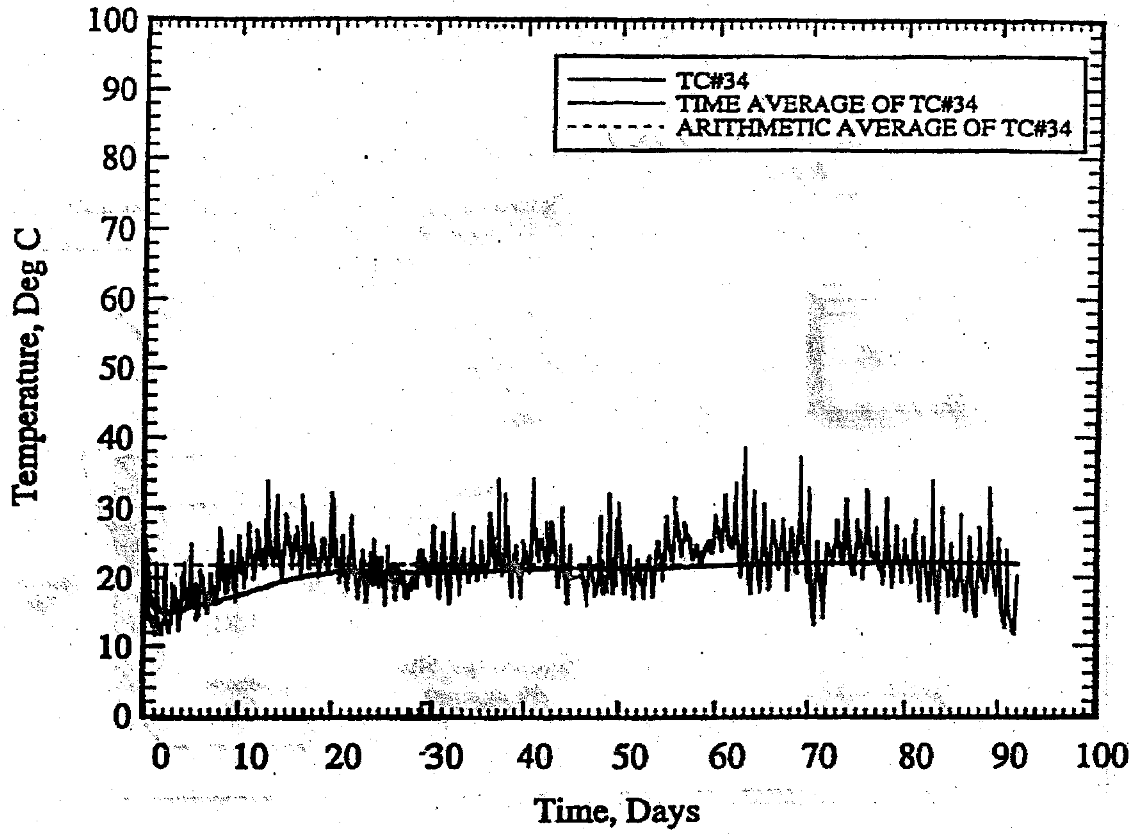
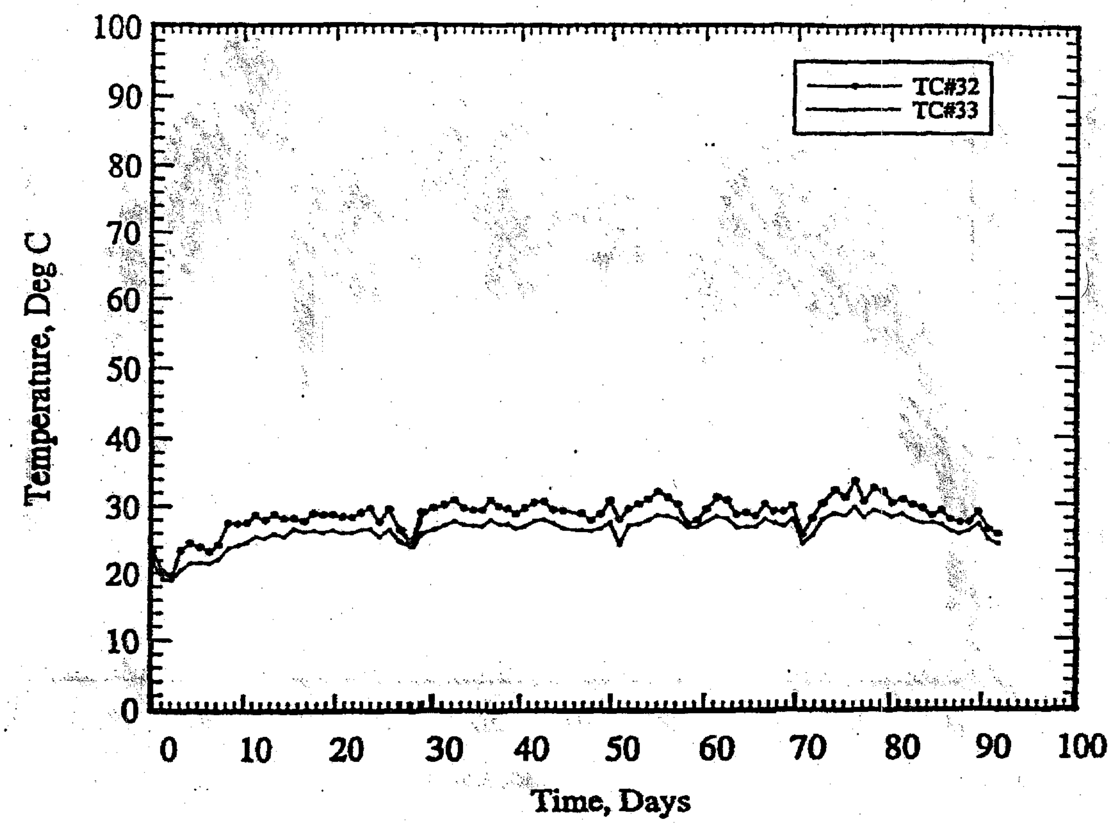


FIGUE 5
OUTDOOR TEMPERATURE HISTORY



FIGUE 6
INDOOR AIR TEMPERATURE HISTORIES

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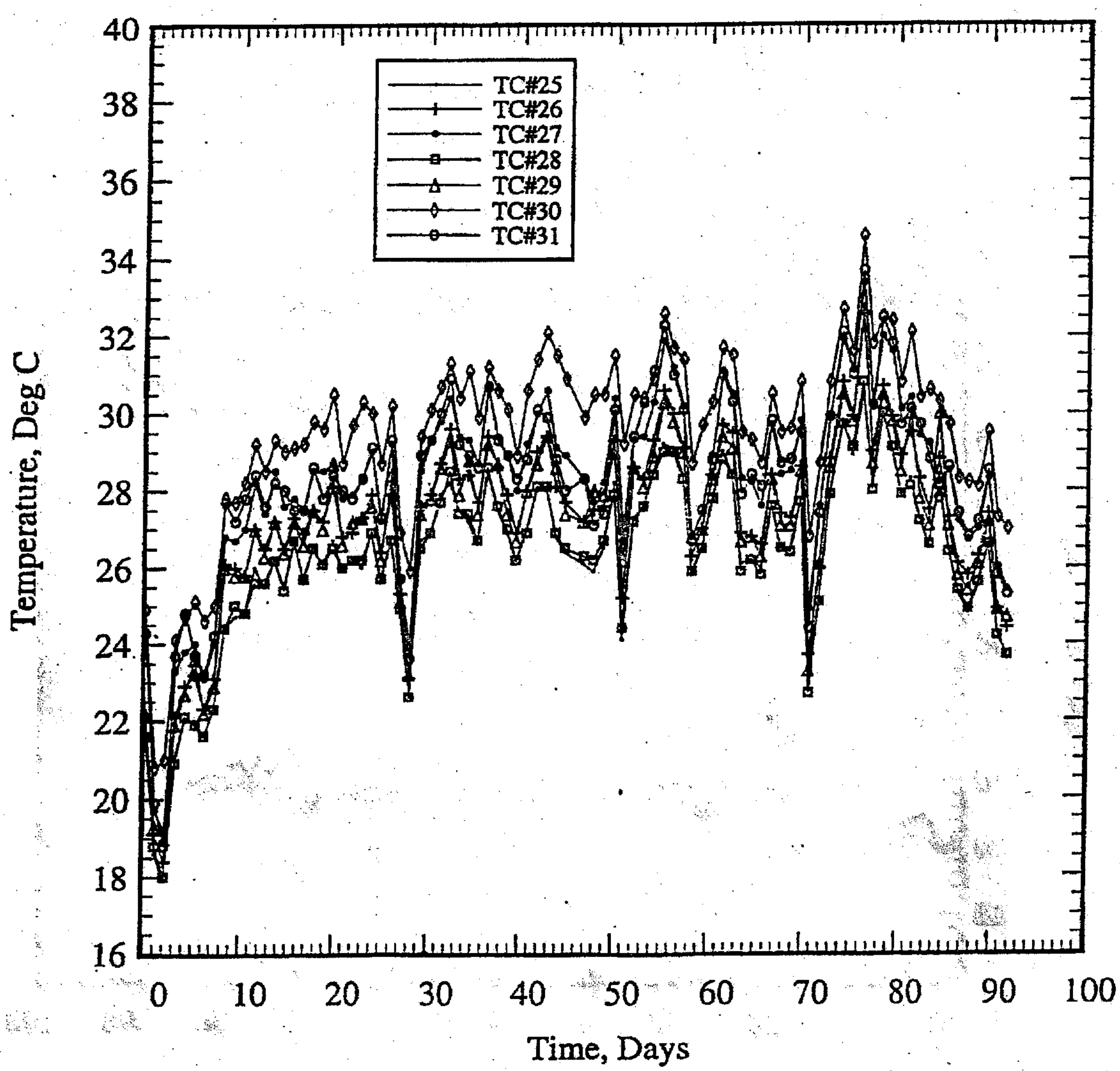
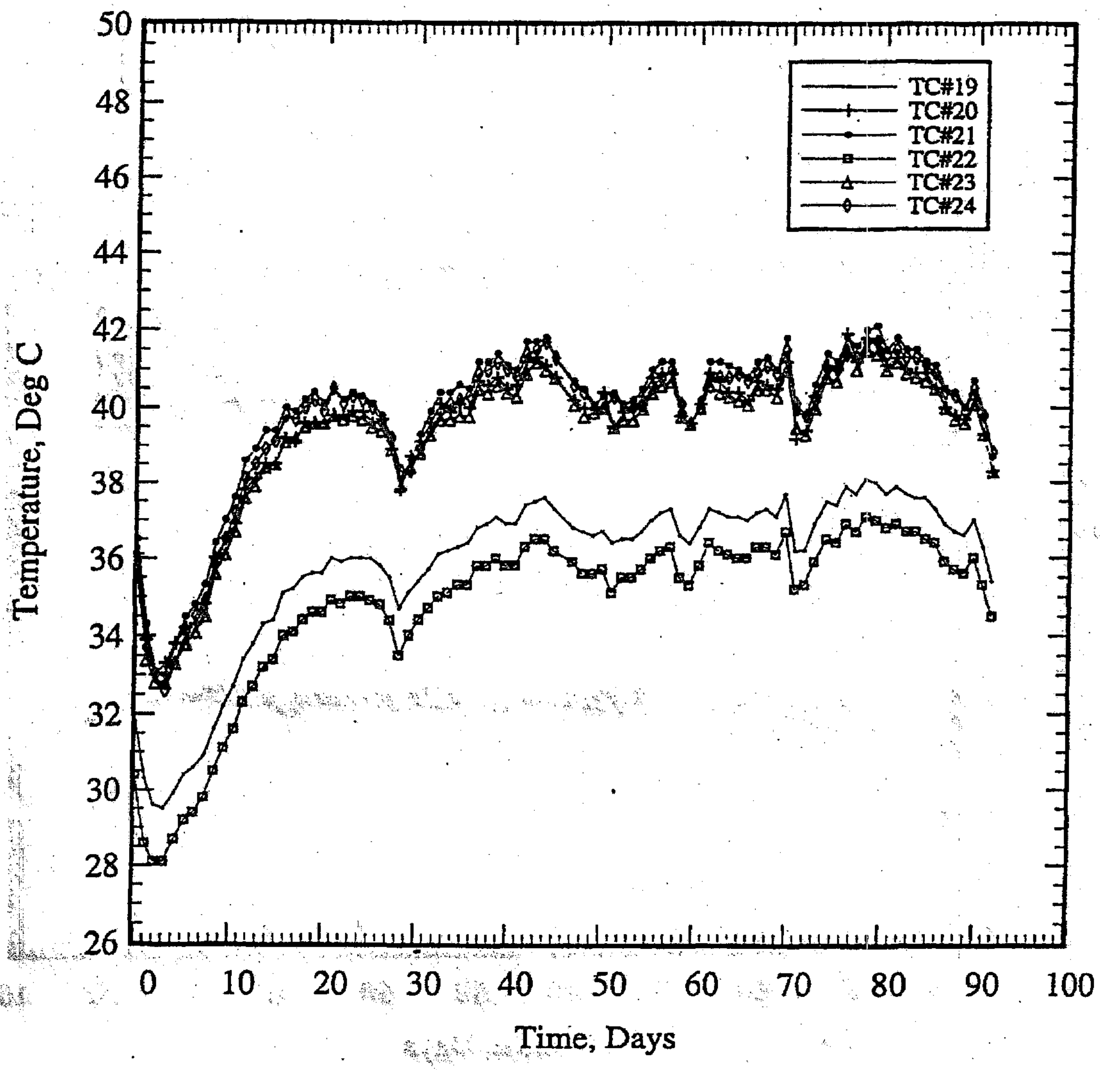


FIGUE 7
AIR SPACE TEMPERATURE HISTORIES
(VERTICAL SIDES & TOP)



FIGUE 8
OUTER LINER TEMPERATURE HISTORIES
(VERTICAL SIDES)

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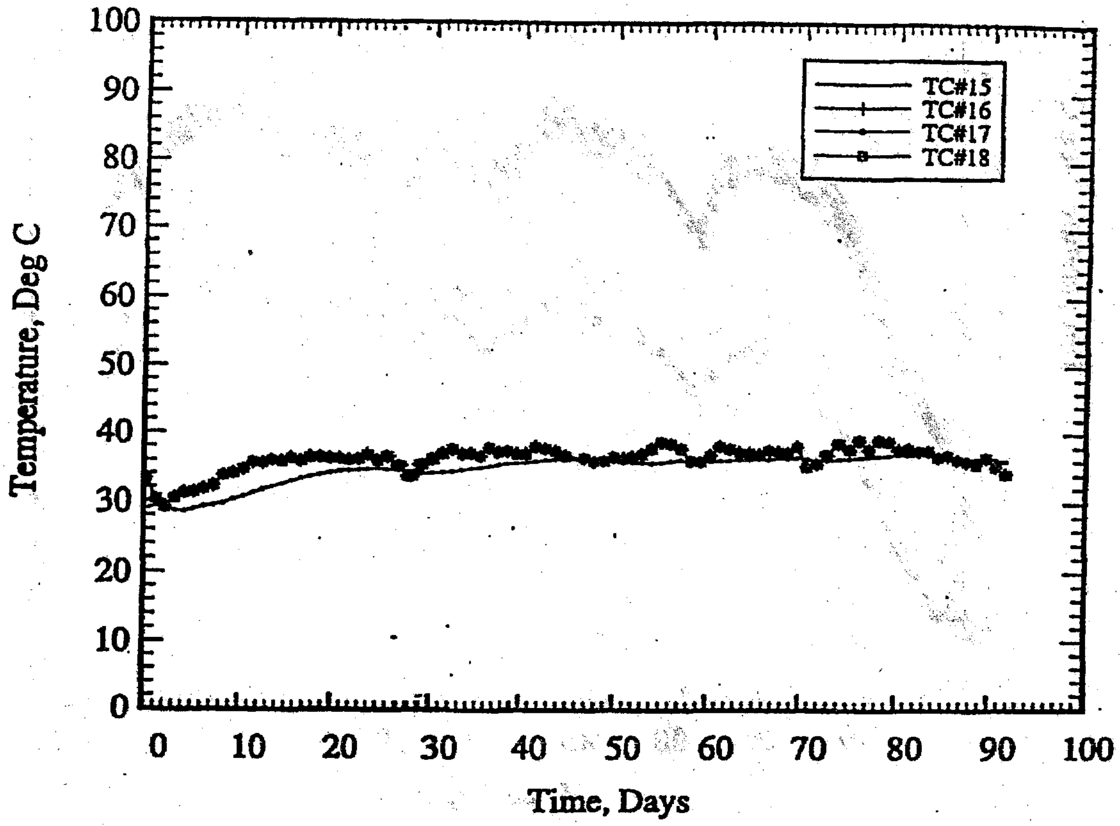

FIGUE 9
OUTER LINER TEMPERATURE HISTORIES
(TOP & BOTTOM)

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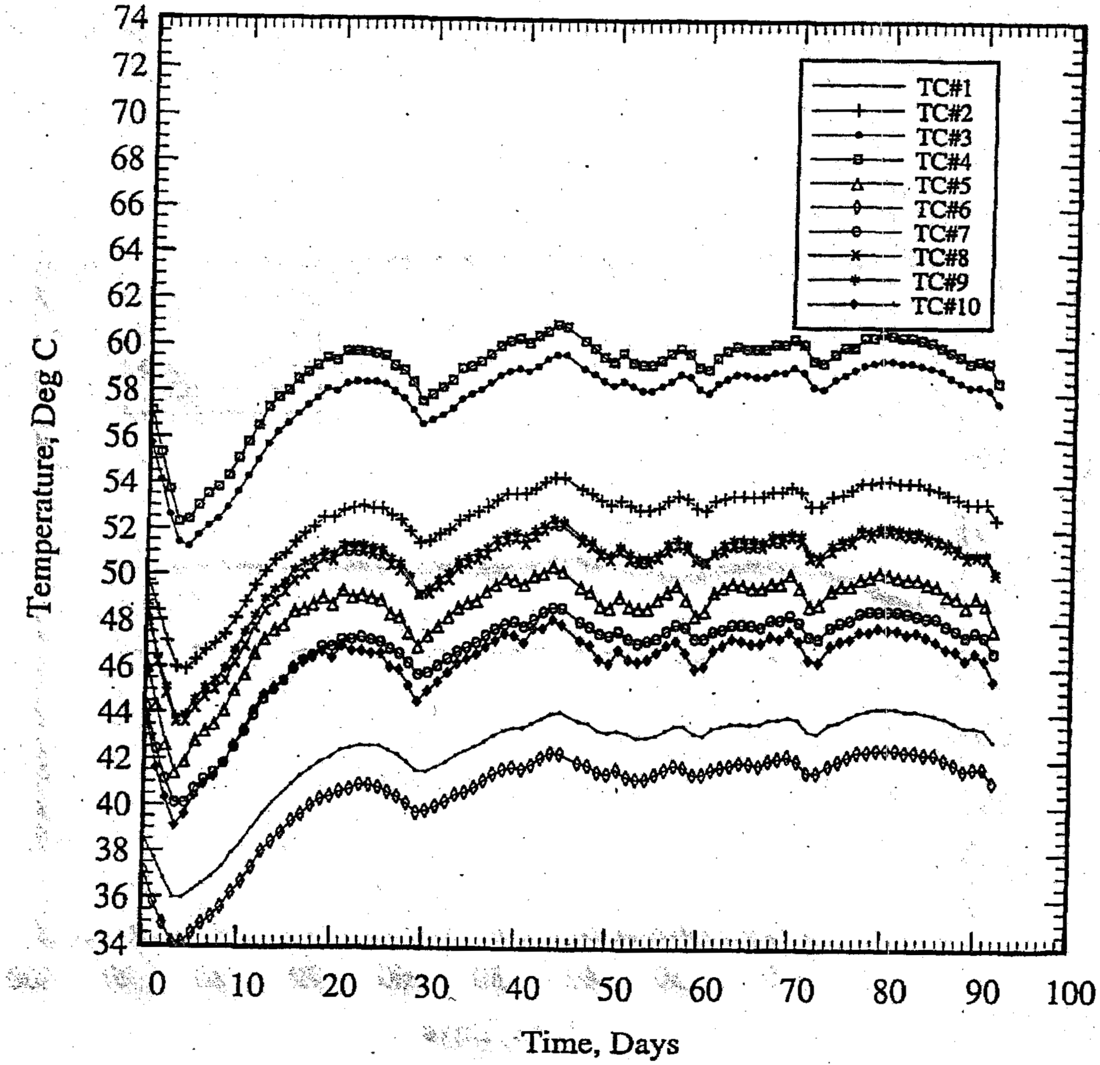

FIGUE 10
INNER LINER TEMPERATURE HISTORIES
(VERTICAL SIDES)

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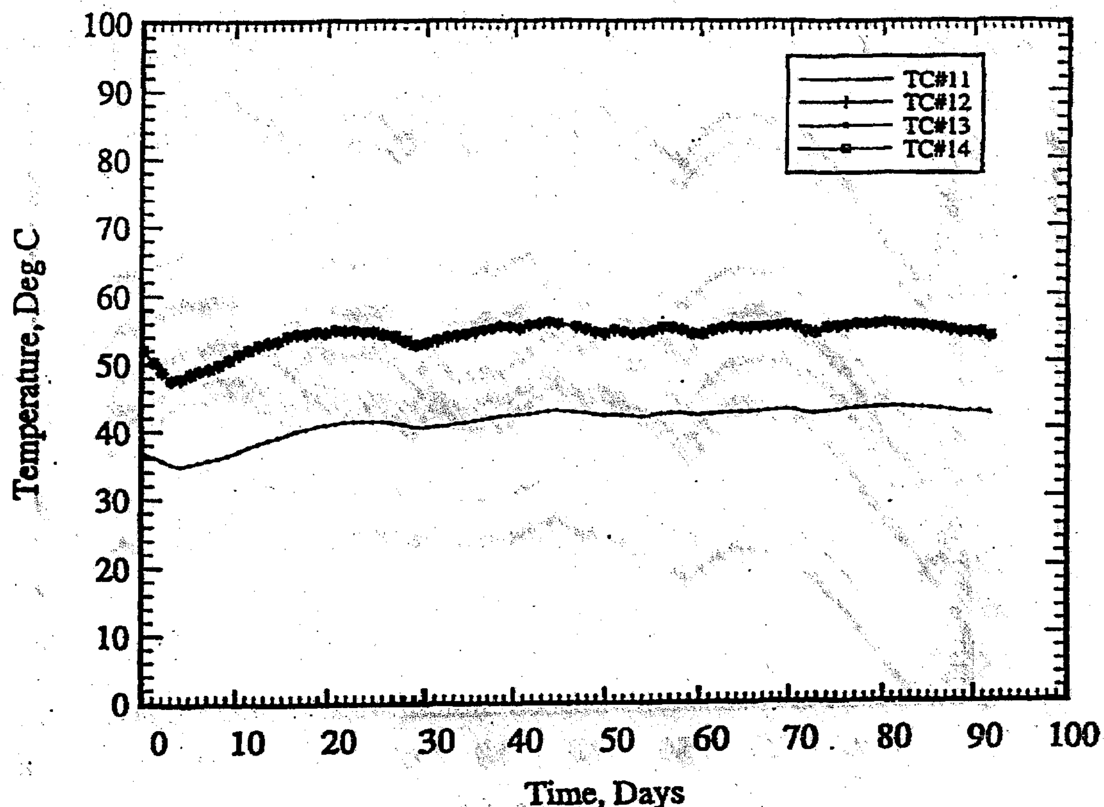

FIGUE 11
INNER LINER TEMPERATURE HISTORIES
(TOP & BOTTOM)

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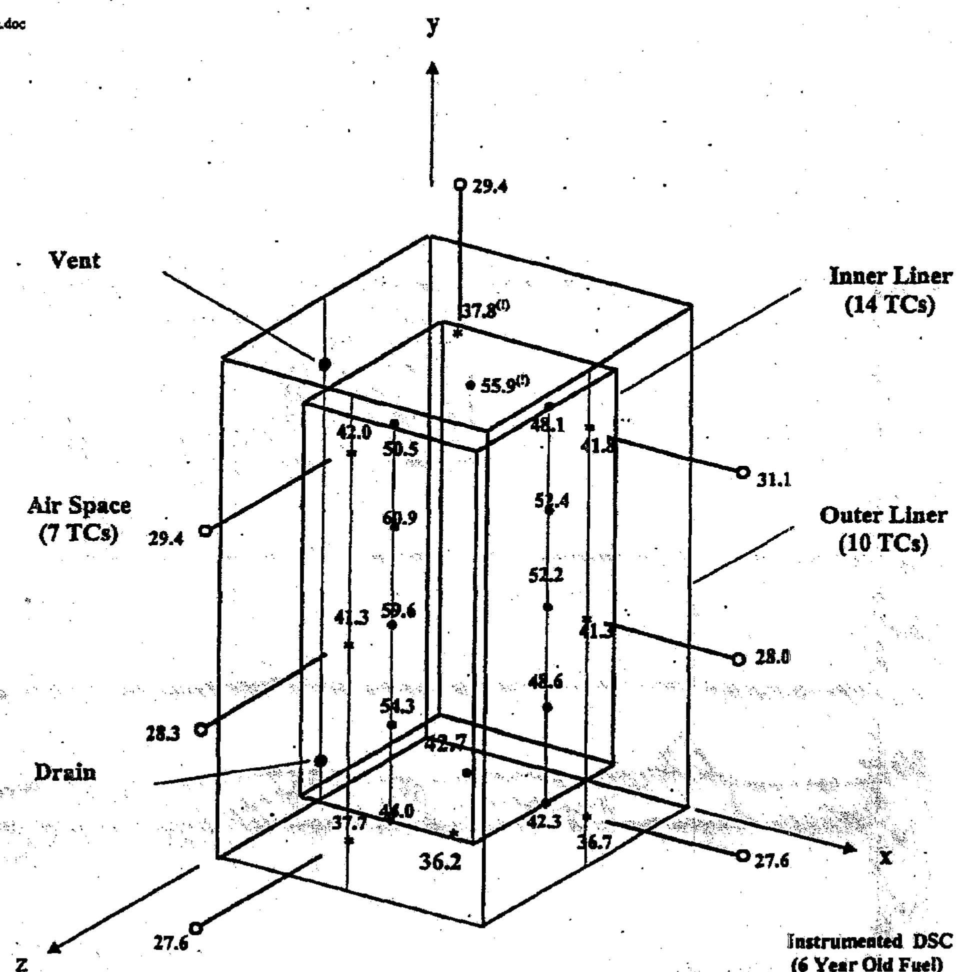

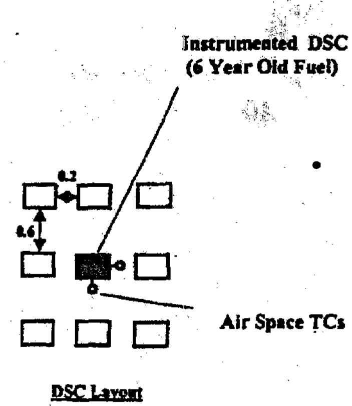

Dimensions in m
(Diagram not to Scale)

Ambient Temperature TCs 30.2,28.0,19.6
(2 Indoors & 1 outdoors)

Total No. of TCs = 34

(!) Average value of 3 TCs

- • Inner Liner TCs
- ° Air Space TCs
- * Outer Liner

FIGURE 12
TEMPERATURE DISTRUBUTION AT TIME OF MAXIMUM INNER LINER TEMPERATURE
(JULY 24, 1998)

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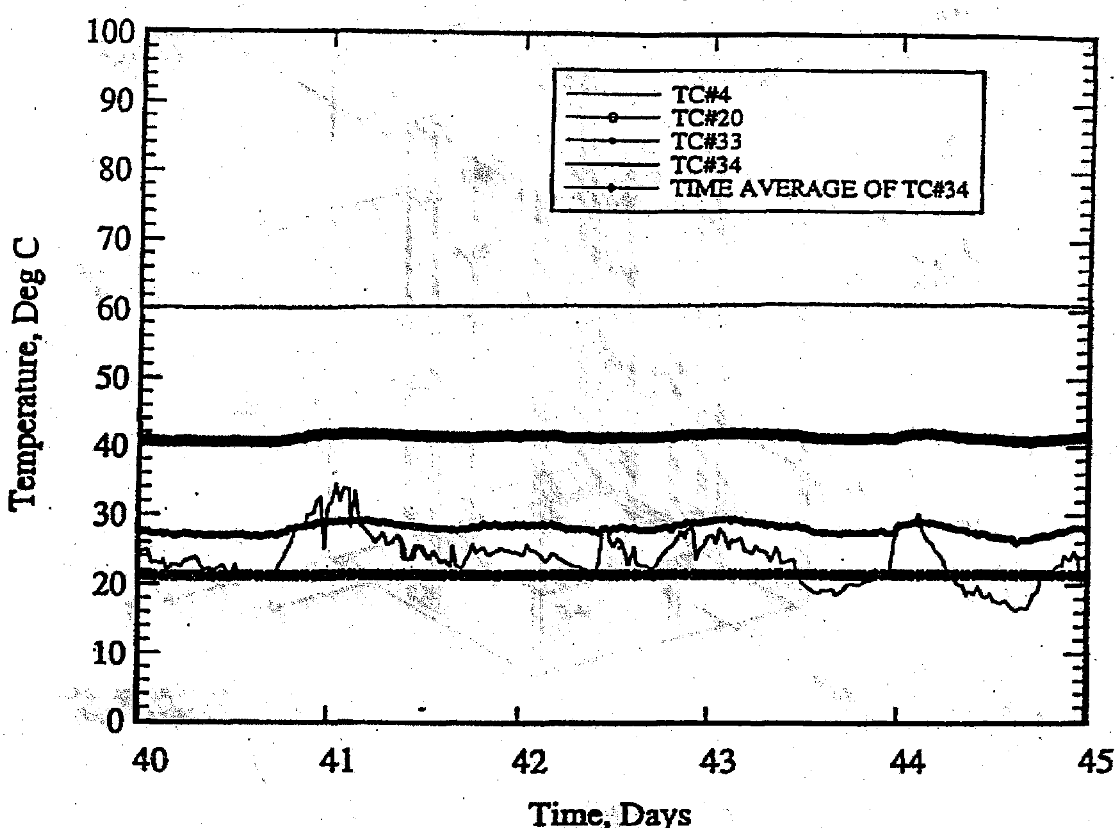

FIGUE 13
CORRELATION OF INDOOR AIR AND DSC TEMPERATURES WITH OUTDOOR AIR TEMPERATURE