Canadian Nuclear Laboratories | Laboratoires Nucléaires Canadiens

Supplier Document

In Situ Decommissioning of Whiteshell Reactor 1 Project – Decommissioning Safety Assessment Report

WLDP-26000-SAR-001

Revision 4

Accepted by: 2021-12-23

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Date

REPORT

Decommissioning Safety Assessment Report Revision 4

In Situ Decommissioning of Whiteshell Reactor 1 Project

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Executive Summary

Canadian Nuclear Laboratories (CNL) was created by Atomic Energy of Canada Limited (AECL) in 2014, as a wholly-owned subsidiary. AECL is a Federal Government Crown Corporation. CNL is a private-sector company that is contractually responsible for the management and operation of the nuclear sites, facilities and assets owned by AECL. CNL is the Site Operating Company and holder of the Canadian Nuclear Safety Commission (CNSC) licences for all AECL sites. In September of 2015, a contract was awarded to Canadian National Energy Alliance (CNEA) and ownership of CNL was transferred to CNEA. CNL retained ownership of all CNSC licences and is the operator of all AECL sites. The Whiteshell Laboratories (WL) is a government-owned and contractor-operated facility in Pinawa, Manitoba. All WL assets and liabilities are owned by AECL. Under an agreement with the Canadian National Energy Alliance, CNL operates the WL facility and performs decommissioning work for the WL site. As the proponent of the decommissioning activities, following completion of the decommissioning contract CNL will be responsible for implementing and managing the monitoring and follow-up program for the WL site, under contract to AECL.

The WL was established by AECL in the 1960s to conduct nuclear research to demonstrate the organic-cooled reactor concept using heavy water (D_2O) as the moderator. The Whiteshell Reactor 1 (WR-1) also provided a facility for engineering tests and scientific studies on alternative fuels, fuel channels and reactor coolants. Whiteshell Laboratories was also home to other significant research programs, including the Nuclear Fuel Waste Management Program, the SLOWPOKE Demonstration Reactor, and CANDU Reactor Safety research projects and accelerator projects.

The WL site is located in southeastern Manitoba (Figure ES-1), approximately 100 kilometres (km) northeast of the City of Winnipeg. The WL site was operated for approximately 40 years under an operating licence issued by the Atomic Energy Control Board. In May 2000, the *Nuclear Safety and Control Act* came into force creating the CNSC. Since then, the WL site has been operated first under an operating licence and then under a decommissioning licence (since March 2002) issued by CNSC.

The WL site has a total area of 4,375 hectares and is within the boundaries of the Local Government District of Pinawa. The Winnipeg River forms the western boundary of the WL site. Nearby communities include the Village of Lac Du Bonnet and the Local Government District of Pinawa. Both communities are located on the shore of the Winnipeg River.

Three Indigenous communities are in proximity to the WL site. The Ojibway community of Sagkeeng First Nation (also known as Fort Alexander, Manitoba) is located on the shore of the Winnipeg River at Lake Winnipeg approximately 50 km northwest of the WL site. The Little Black River First Nation is located approximately 60 km northwest of the WL site, in proximity to Sagkeeng First Nation. The Brokenhead Ojibway Nation (also known as Scanterbury, Manitoba) is located along the shore of the Brokenhead River approximately 50 km west of the WL site. The Project is located on Treaty 3 land, while the overall WL site extends west of the Winnipeg River into Treaty 1 land. Communities that form part of these Treaties and Treaty 5 have historical and current traditional land uses with the area. The Project is also located in the homeland of the Manitoba Métis Nation, as represented by the Manitoba Métis Federation.

The WL site encompasses the WR-1 Complex (i.e., Building 100, which includes the WR-1 Building, the east annex and the service wing), waste management facilities and other research laboratories. The WL site occupies four different land use designations in the Local Government District of Pinawa (LGD of Pinawa 2004) including general agriculture, heavy industrial, recreational/commercial, and natural area.

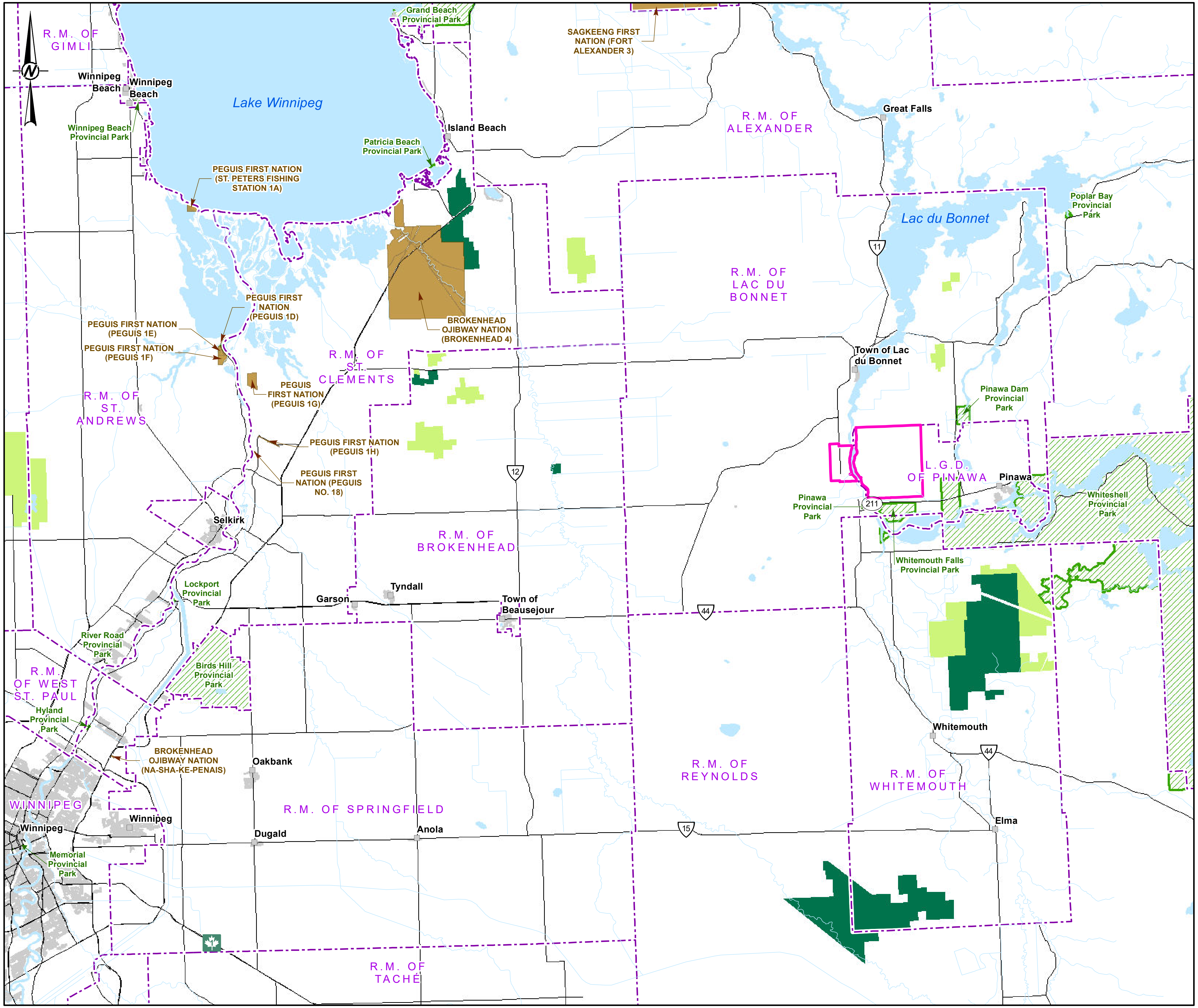
In 1998, the Government of Canada made a decision to decommission the WL site. A Comprehensive Study Report under the *Canadian Environmental Assessment Act, 1992*, was completed for the WL decommissioning project. The Environmental Assessment Decision to approve the decommissioning project was announced by the Minister of Environment in March 2002 and a decommissioning licence was issued (Licence No. NRTEDL-W5-8.00/2024). Enabled by the Comprehensive Study Report and the site decommissioning licence, CNL is authorized to decommission WR-1 by means of dismantling and demolition. Over the past 10 years, multiple buildings and facilities at the WL site have been decommissioned and the space occupied by the facilities has been remediated in an effort to meet this objective.

The decommissioning approach for WR-1 in the Comprehensive Study Report includes a deferment period prior to dismantlement and decommissioning of the reactor. The deferment period was planned to allow time for the reduction of radiation fields within WR-1. The currently approved decommissioning approach includes complete removal of the WR-1 Building with waste generated being classified, segregated, and placed in interim storage on-site or disposed of at appropriate off-site disposal facilities.

AECL's mandate is to enable nuclear science and technology and manage its radioactive waste and decommissioning liabilities in a safe and environmentally responsible manner. AECL has asked CNL to perform the work, and in keeping with international best practices, the decommissioning timeframe has been accelerated with the goal of completing decommissioning of the WL site by 2027. In an effort to continue to safely reduce AECL's nuclear legacy liabilities and reduce the need for interim storage of radioactive waste, CNL began to investigate other options that would allow the WR-1 Building to be decommissioned safely. The new proposed approach for WR-1 is in situ disposal (ISD), which allows CNL to decommission the facility in a safer, compliant manner that reduces interim storage and provides protection of the public and the environment.

The ISD approach represents a permanent, passive decommissioning end-state, increases worker safety, provides protection of the environment and the public, reduces interim storage and multiple handling, enables permanent nuclear liability reduction and utilizes less resources. The ISD approach incorporates proven technologies and best industry practices, including documented experience from the International Atomic Energy Agency and other similar international facilities.

As part of the ISD of WR-1 at the WL site (the Project), the below-grade reactor systems, and associated radiological and non-radiological hazards, will be permanently encased in grout within the WR-1 Building foundation. The above-grade structures will be demolished and removed using traditional demolition methods. During decommissioning, consideration will be given to place some equipment from the heat transport system that is currently located on the ground-level reactor floor to a below-grade position for incorporation in the disposal system. A concrete cap and engineered cover will then be constructed over the Whiteshell Reactor Disposal Facility to resist intrusion and divert precipitation and surficial runoff. All other decommissioning activities will be conducted as described in the Comprehensive Study Report and as currently approved under the existing Decommissioning Licence NRTEDL-W5-8.00/2024.



LEGEND

▭ WHITESHELL LABORATORIES SITE

BASE FEATURES

- ▪ CITY/TOWN
- — MAJOR ROAD
- — WATERCOURSE
- 🟦 WATERBODY
- ▭ MUNICIPAL BOUNDARY
- 🟫 FIRST NATION RESERVE

PROTECTED AREAS

- ▨ PROVINCIAL PARK
- 🟩 ECOLOGICAL RESERVE
- 🟩 WILDLIFE MANAGEMENT AREA

0 10 20

1:350,000 KILOMETRES

NOTE(S)

REFERENCE(S)

1. BASE DATA - CANVEC AND MLI, 2016
2. PROJECT DATA - CNL, 2016
3. PROJECTION: TRANSVERSE MERCATOR DATUM: NAD 83 COORDINATE SYSTEM: UTM ZONE14N

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
GENERAL LOCATION OF THE WHITESHELL LABORATORIES SITE

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	CGE
PREPARED	CGE/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
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The Project considers a closure phase and a post-closure phase. The closure phase includes preparation for ISD, grouting of below-grade structures and systems, removal of above-grade WR-1 structures and systems, installation of the concrete cap and the engineered cover, implementation of environmental controls, and final site restoration. These activities are expected to occur from 2022 to 2026.

The post-closure phase has two discrete periods, institutional control and post-institutional control. Institutional control is estimated to last a minimum of 100 years during which long-term performance monitoring and maintenance activities will continue, to demonstrate compliance with the safety case assumptions. Although institutional control is estimated to last a minimum of 100 years, it is recognized that it will continue until the CNSC agrees institutional controls are no longer needed. Post-institutional control occurs thereafter (expected in year 2126) and continues indefinitely.

Under the *Nuclear Safety and Control Act*, CNL's proposal requires an amendment to the existing Decommissioning Licence NRTEDL-W5-8.00/2024. Before the responsible federal authority (i.e., CNSC) can permit the Project to proceed, a decision must be made based on the results of an environmental assessment prepared pursuant to the *Canadian Environmental Assessment Act, 2012*. The environmental assessment is an iterative Project planning process, intended to affirm that proposed activities will not cause significant adverse environmental effects.

This Decommissioning Safety Assessment Report (DSAR) has been prepared as a supporting document for the Environmental Impact Statement (EIS) being completed for the Project. A formal safety assessment or safety analysis is required under the Class I Nuclear Facilities Regulations. CNL has performed safety analyses for its Class II nuclear facilities and radioisotope laboratories, as well as other locations where nuclear material is used. These assessments are completed to demonstrate that decommissioning activities can be safely completed and prescribed environmental and human health protective limits will not be exceeded, including limits governing radiological doses to workers and members of the public, and releases of radioactive material to the surrounding environment.

The DSAR has been prepared specific to the ISD of WR-1 to support the justification of the selected decommissioning strategy, and identify controls, conditions, and mitigation necessary to accomplish compliance with regulatory requirements and industry best practices. The scope of the assessment considers the closure phase (which includes decommissioning and reclamation) and long-term performance during the post-closure phase (which includes institutional control and post-institutional control).

A detailed safety analysis still needs to be completed for the ISD of the Waste Management Area (WMA) trenches, as only the current state of the WMA is included in the base case of this analysis. Specific cleanup criteria and defined end-states would be defined as part of the detailed safety analysis of the ISD of the WMA trenches. It is expected that this analysis will capture the potential loadings to the Winnipeg River from the WMA trenches in combination with the predictions for the Project. Specific to this assessment, it is recognized that the WMA trenches are encompassed by the Comprehensive Study Report and existing Decommissioning Licence NRTEDL-W5-8.00/2024.

The safety strategy refers to the approach that will be taken to comply with the safety objectives and principles, to comply with regulatory requirements, to confirm that good engineering practice has been adopted, and that safety and protection are optimized. The ISD approach provides a permanent, passive decommissioning end-state, and incorporates proven technologies and best industry practices, including documented experience from the International Atomic Energy Agency and other similar international facilities. The safety of the Project post-closure is provided by means of passive features so that there is no need for active management, which is in alignment with International Atomic Energy Agency requirements.

The Project encompasses closure and post-closure (institutional control, including verification of end-state and post-institutional control) activities. The transition between these phases will be marked by CNL decision hold-points. The DSAR assesses the closure and post-closure phases of the Project separately, as the hazards associated with the two phases of the Project are substantially different. Central to the safety assessment, a wide range of scenarios are considered to develop an understanding of the system and provide a thorough safety case for the Project.

The DSAR provides the necessary information to decision-makers so that the decommissioning activities can be completed safely and that the public and environment will be protected over the long-term. The DSAR evaluation encompasses the period of time when the maximum effect is predicted to occur from Project activities. The maximum dose for each radionuclide was conservatively assessed at a single point in time, corresponding to the peak loading rate from groundwater to the Winnipeg River. The closure and post-closure assessments consider not only normal operating conditions or the normal evolution of the site, but also potential upset conditions (i.e., disruptive events).

The key information that will factor into decisions on how the Project will be executed and that has been taken into consideration in the environmental assessment are:

- Identification of controls and mitigation required to confirm closure activities can be completed safely, meeting regulatory requirements and protecting workers, the public and the environment.
- Identification of institutional controls, including the timeframe required for regulatory requirements to be met and to ensure the protection of the public and the environment over the long-term.

A robust compliance system was established for the WL site to maintain compliance with regulatory requirements and industry best practices during the operational period, and has been updated as required to encompass the evolution of the site, including decommissioning activities.

Closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. During the closure phase, potential effects are primarily related to changes in air quality from demolition activities and grouting of the WR-1. The safety assessment confirms that the total radiation dose to all human receptors during closure activities (demolition prior to grouting and grouting) is well below the public dose limit and dose constraint for the Project. Proven operational programs will be in effect so that radiological doses are also below regulatory criteria and are As-Low-As-Reasonably-Achievable.

Environmental effects will occur as the Whiteshell Reactor Disposal Facility degrades over time, due to mechanical stresses and chemical reactions, resulting in the release of contaminants. Over time these contaminants will migrate, being discharged into shallow groundwater and ultimately realized in the surface environment (i.e., post-closure phase). The ISD approach is designed to control the rate of release of nuclear and hazardous substances from the Whiteshell Reactor Disposal Facility and retain the waste away from people and the environment. The design considers possible events that could affect the integrity of the Whiteshell Reactor Disposal Facility. It is recognized that there are inevitable uncertainties associated with predicting the performance of the Whiteshell Reactor Disposal Facility over a long-time scale (i.e., thousands of years);

therefore, safety is established through multiple barriers of protection and verified through long-term environmental monitoring during institutional control.

Central to the safety assessment, a wide range of scenarios are considered to develop an understanding of the system and provide a thorough safety case for a project. The scenarios selected for detailed assessment are those most likely to occur (i.e., the Normal Evolution Scenario) and various unlikely disruptive events that could result in substantially higher exposure doses to the public and the environment (i.e., bounding scenarios).

The Normal Evolution Scenario is the expected long-term evolution of the WL site after closure has been completed, and is a reasonable extrapolation of present day site features and receptor lifestyles. The doses to the public during the post-closure phase for the Normal Evolution Scenario is predicted to be below the public dose limit and will not exceed the CNL's dose constraint. As such, no discernable health effects from exposure to radiological releases are anticipated due to Project activities. For non-radiological releases, predicted exposure to the public are below the acceptable risk level, with one exception, which is driven by background water concentrations in the Winnipeg River and the Project contribution to the predicted exposure is negligible. The doses predicted for non-human biota during post-closure were also well below benchmarks and protective target values. Therefore, it is unlikely that there would be significant adverse health effects on either aquatic or terrestrial populations or communities as a result of the Project.

The safety assessment assessed multiple disruptive events. These disruptive events included a conservatively defined future hypothetical exposure group, specifically an On-site Farm that was also assessed as part of the Normal Evolution Scenario. Three disruptive events were identified as "worst-case", with consequences greater than the other disruptive events considered, and selected as bounding scenarios. The three bounding scenarios are Human Intrusion, Whiteshell Reactor Disposal Facility Barrier Failure, and Well in Plume.

For all Bounding Scenarios, the total radiological doses were compared to the International Atomic Energy Agency (IAEA) reference level ranging from 1 to 20 mSv/a for Disruptive Events. For the Human Intrusion Bounding Scenario, total doses were below both the upper (20 mSv/a) and lower (1 mSv/a) IAEA reference levels. For non-radiological hazardous material, the assessment demonstrates that human intrusion into the Whiteshell Reactor Disposal Facility could result in exposures of human receptors to HB-40 and lead in waste material brought to the surface above target levels. As such, while this is a very unlikely worst case scenario, reasonable effort is warranted to reduce the probability of these unplanned events from occurring. During the Post-Institutional Control period, passive controls will still be in place including the limited footprint, the Whiteshell Reactor Disposal Facility composition being relatively impervious and made of material of no economic value, and any remaining land use restriction acting to reduce the likelihood of a human intrusion event.

For the Whiteshell Reactor Disposal Facility Barrier Failure Bounding Scenario, all radionuclide doses to human receptors were below the IAEA reference level (lower and upper level) and all radionuclide doses to non-human biota receptors were well below benchmarks. Predicted exposure to the public from non-radiological releases from a Whiteshell Reactor Disposal Facility Barrier Failure are below the acceptable risk level, with the exception of lead, which is driven by background water concentrations in the Winnipeg River and the Project contribution to the predicted exposure is negligible. Exposure of the non-human biota receptors are all below the acceptable risk level; therefore, it is unlikely that there would be significant adverse effects on either aquatic or terrestrial populations or communities as a result of WRDF barrier failure.

For the Well in Plume Bounding Scenario, the total radiation dose does not exceed the upper IAEA reference level for any receptor, but does exceed the lower IAEA reference level for most receptors. For non-radionuclides, the assessment demonstrates that human habitation with groundwater use for drinking water could result in exposures to cadmium and lead at levels above target values. However, the assessment is considered conservative as it assumes that the maximum concentrations of cadmium and lead occur at the same time, where in reality maximum concentrations occur at different times during the Post-closure period. Additionally, this scenario is very unlikely as the capacity for a well to provide sufficient water for domestic use is very low and because of the close proximity to the Winnipeg River.

The safety assessment illustrates that the Whiteshell Reactor Disposal Facility components and the characteristic of the environmental setting, will provide long-term protection for the public and the environment. The design of the Whiteshell Reactor Disposal Facility meets the criteria of providing long-term safety by passive means and minimizing the need for active controls and systems (active management of the site during which monitoring, and surveillance activities are completed).

CNL's Environmental Protection Program is designed to provide protection of the environment and the public with respect to environmental aspects that result from operation of CNL's facilities. CNL operates an extensive Environmental Monitoring Program that will be maintained throughout the Project to monitor the effects of disposal activities and to verify that the requirements and objectives of the Environmental Protection Program are met. During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions, for a minimum of 100 years. CNL has revised the Environmental Assessment Follow Up Program for the WL site to incorporate the proposed monitoring and reporting specific to the Project. Towards the end of the institutional control period, a Licence to Abandon will be sought and, as a prerequisite for this, it will need to be demonstrated that the facility is in a long-term, passive, safe state. If abandonment of the facility is allowed, monitoring and surveillance will no longer be required as, at this time, the facility will have been demonstrated to no longer pose a hazard to humans or the environment.

The DSAR is a "living document" (i.e., continued iterative use as needed) and will be periodically reviewed and updated (as required), approximately every 5 years, over the lifetime of the facility. The safety envelope delineated by the current safety assessment is based on preliminary design information that was conservatively developed based on experience from similar long-term waste management and decommissioning projects. As outlined in IAEA SSR-5, safety assessments are updated as necessary to reflect actual experience and increasing knowledge. Currently, it is anticipated that adequate conservatism has been integrated into the assessment and assumptions to accommodate future detailed design decisions and outcomes.

Table of Contents

TABLES

FIGURES

APPENDICES

APPENDIX A
Detailed Concordance Table

APPENDIX B
CNL WR-1 In Situ Decommissioning Activities Hazard Identification

APPENDIX C
WL Whiteshell Reactor 1 Decommissioning Features, Events and Process Analysis

APPENDIX D
Derived Release Limits for AECL’s Whiteshell Laboratories

List of Abbreviations

Acronym	Definition
AC	alternating current
ACM	Asbestos-Containing Materials
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
ALWTC	Active Liquid Waste Treatment Centre
CANDU	Canadian Deuterium Uranium
CCME	Canadian Council of Ministers of the Environment
CCSF	Concrete Canister Storage Facilities
CNEA	Canadian Nuclear Energy Alliance
CNL	Canadian Nuclear Laboratories
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COPC	constituent of potential concern
CSA	Canadian Standards Association
DC	direct current
DDP	Detailed Decommissioning Plan
DRL	derived release limit
DSAR	Decommissioning Safety Assessment Report
EAFP	Environmental Assessment Follow up Program
EcoRA	Ecological Risk Assessment
EIS	Environmental Impact Statement
ERA	Environmental Risk Assessment
FEPs	Features, Events and Processes
HAZOP	Hazard and Operability Study
HB-40	Organic Coolant (hydrogenated terphenyl)
HCF	Hot Cell Facilities
HEPA	high-efficiency particulate absorber
HHRA	Human Health Risk Assessment
HHW	high temperature, high pressure water
HLW	High Level Waste
HQ	hazard quotient
HVAC	heating, ventilating and air conditioning
IAEA	International Atomic Energy Agency
IFTF	Immobilized Fuel Test Facility
ILW	Intermediate Level Radioactive Waste
ISBN	International Standard Book Number
ISD	in situ disposal

Acronym	Definition
LLW	Low Level Radioactive Waste
LSA	local study area
NBCC	National Building Code of Canada
NEA	Nuclear Energy Agency
NPARB	Nuclear Performance Assurance Review Board
PAC	Protective Action Criteria (Department of Energy)
PCB	polychlorinated biphenyls
PGA	peak ground acceleration
PHT	Primary Heat Transport
PM_{10}	particles nominally smaller than 10 µm in diameter
$PM_{2.5}$	particles nominally smaller than 2.5 µm in diameter
Project	in situ decommissioning of Whiteshell Reactor-1
REGDOC	Regulatory Document
RSA	regional study area
RSZ	Radiological Safety Zones
RTL	registered trapline
sp.	species
SPM	suspended particulate matter
spp.	multiple species
TRV	toxicity reference value
US DOE	United States Department of Energy
US EPA	United States Environmental Protection Agency
VC	Valued Component
WL	Whiteshell Laboratories
WMA	Waste Management Area
WNRE	Whiteshell Nuclear Research Establishment
WR-1	Whiteshell Reactor 1
WRDF	Whiteshell Reactor Disposal Facility

Units of Measure

Unit of Measure	Definition
%	percent
~	approximately
<	less than
>	more than
≤	equal to or less than
≥	equal to or more than
°C	degrees Celsius
µg/m^3	micrograms per cubic metre
µm	micrometres
µSv/h	microsieverts per hour
Bq/cm^2	becquerels per square centimetre
Bq/g	becquerels per gram
Bq/kg	becquerels per kilogram
Bq/L	becquerels per litre
Bq/s	becquerels per second
Bq/wk	becquerels per week
cm	centimetre
cm^2	square centimetre
g/m^3	grams per cubic metre
g/yr	grams per year
GBq	gigabecquerel
h/wk	hours per week
ha	hectare
kg	kilogram
kg/m^3	kilograms per cubic metre
km	kilometre
km/h	kilometres per hour
km^2	square kilometre
L	litre
L/d	litres per day
L/s	litres per second
m	metre
m/s	metres per second
m/yr	metres per year
m^2	square metre
m^3	cubic metre
m^3/d	cubic metres per day

Unit of Measure	Definition
m^3/s	cubic metres per second
masl	metres above sea level
Mg	Megagrams
mg/kg bw/d	milligrams per kilogram of body weight per day
mg/kg/d	milligrams per kilogram per day
mg/L	milligrams per litre
mGy/d	milligray per day
mGy/d	milligray per hour
mm	millimetre
mm/yr	millimetres per year
mm[eq]	millimetres equivalent
MPa	megapascal
mSv/a	millisieverts per year
p-mSv	person-millisieverts
psi	pounds per square inch
s/m^3	seconds per cubic metre
t	metric tonne
TBq	terabecquerel
vol. %	percent volume
yr	year

1.0 INTRODUCTION

The Whiteshell Laboratories (WL) site at Pinawa, Manitoba was established in the 1960s by Atomic Energy of Canada Limited (AECL) to conduct nuclear research. The Whiteshell Reactor 1 (WR-1) also provided a facility for engineering tests and scientific studies on alternative fuels, fuel channels and reactor coolants. Whiteshell Laboratories was also home to other significant research programs, including the Nuclear Fuel Waste Management Program, the demonstration reactor, and reactor safety analysis.

The WL site is in southeastern Manitoba (Figure 1.0-1). The WR-1 operated from 1965 to 1985. A first phase of decommissioning occurred in the early 1990s and included the removal of easily mobilized radioactive material (e.g., fuel and fluids) and decontamination of the main floor and first sublevel space. Equipment was decommissioned and removed from the main reactor hall floor and one floor below-grade and placed in interim storage. The removal of fuel, liquids and equipment substantially reduced radioactivity. Activated materials and residual corrosion and fission products have been undergoing decay for over 30 years. Currently, WR-1 is under a storage with surveillance program.

In 1998, AECL decided to decommission the WL site. In March 2002, a decommissioning licence was issued by the Canadian Nuclear Safety Commission (CNSC), authorizing CNL to decommission WR-1 by means of dismantling and demolition. In 2013, the Minister of Natural Resources announced plans to restructure AECL and move to a Government Owned, Contractor Operated (GoCo) management model. In 2014, AECL created Canadian Nuclear Laboratories (CNL), a wholly-owned subsidiary of AECL, as the Site Operating Company and holder of all CNSC site licences to manage all work performed at AECL sites on behalf of AECL. In September of 2015, a contract was awarded to Canadian National Energy Alliance (CNEA) and ownership of CNL was transferred to CNEA. CNL retained ownership of all CNSC licences and is the operator of all AECL sites.

CNL is performing the decommissioning of the WL site with the planned outcome of complete site closure by 2027. CNL is licensed to perform this work under a CNSC Decommissioning Licence (NRTEDL-W5-8.00/2024). CNL is applying for an amendment to the current Decommissioning Licence, to propose in situ disposal (ISD) in place of dismantling and demolition. The ISD approach includes partial dismantling and demolition, along with passive, permanent disposal of the below-grade portions of WR-1 (the Project).

This Decommissioning Safety Assessment Report (DSAR) presents the assessments and the analyses carried out to demonstrate that the Project and associated activities comply with applicable regulatory requirements and established guidance (Section 1.2 Regulatory Requirements and Guidance Documents). This report provides a description of the environmental setting within which the WR-1 Building[1] is located, an understanding of existing site facilities, previous decommissioning activities completed, and current condition of the WR-1 Building (Section 2.0 Background Information). The WR-1 decommissioning plan is described in Section 3.0 Project Description. The basis for the safety objectives and design criteria are presented (Section 4.0 Safety Strategy), which provide the logic and rationale for the plan. The principles, scenarios, and management of uncertainty in the assessment (Section 5.0 Assessment Approach, and the sensitivity analyses completed to support defence-in-depth (Section 6.0 Defence-in-depth for the In Situ Disposal System) are also included. The Project design is supported by the safety assessments for the closure (Section 7.0 Closure Safety Assessment) and post-closure (Section 8.0 Post-closure Safety Assessment) phases. The report also includes a summary of the results from the closure and post-closure analyses (Section 9.0 Results Summary), presents institutional control

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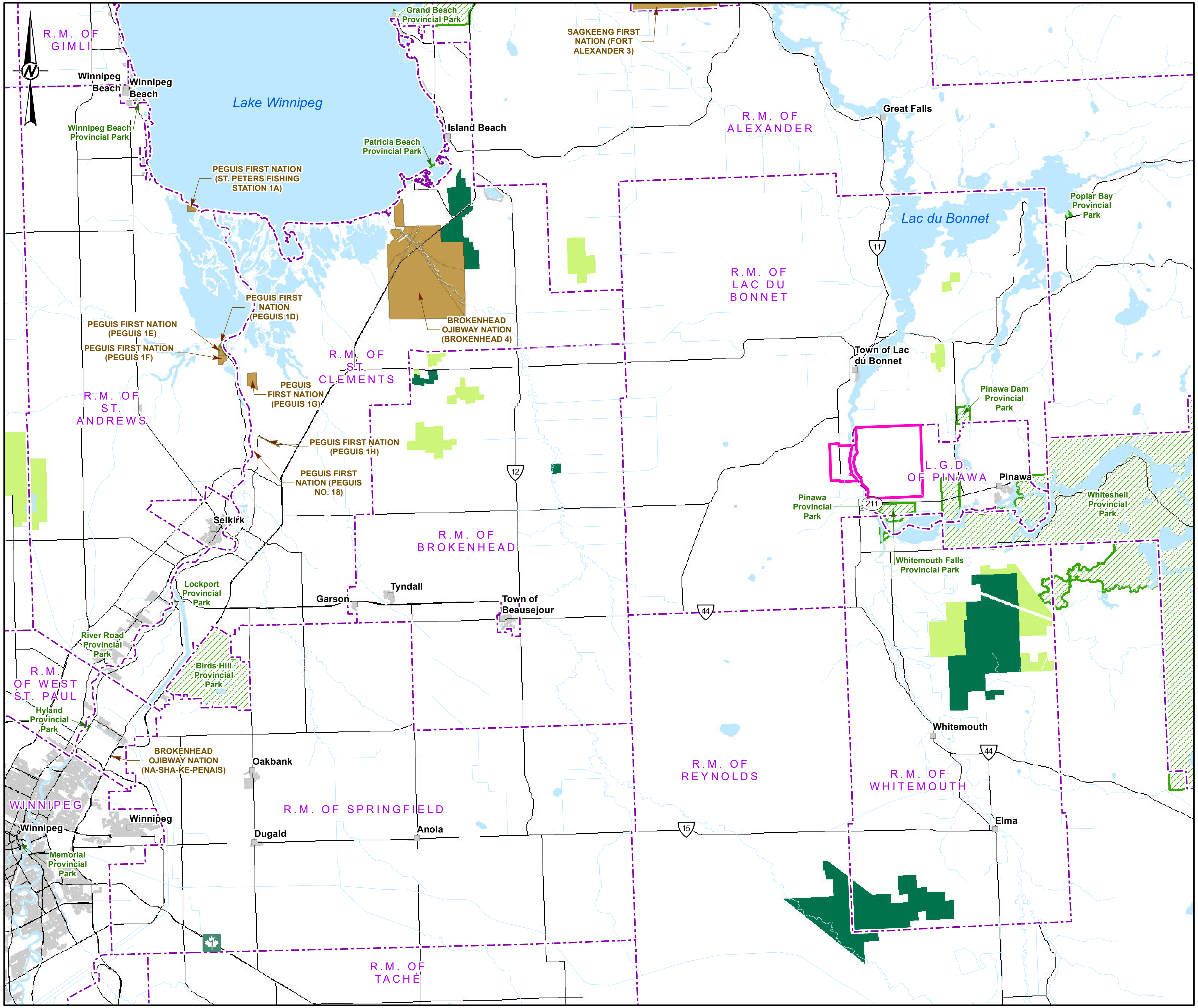
[1] The WR-1 Building is seven storeys tall, with two storeys above-grade and five below-grade and contains the WR-1.

requirements (Section 10.0 Institutional Control), and monitoring and surveillance requirements (Section 10.0 Monitoring and Surveillance) for the Project during closure and post-closure.



LEGEND

▭ WHITESHELL LABORATORIES SITE

BASE FEATURES

- ▪ CITY/TOWN
- — MAJOR ROAD
- — WATERCOURSE
- ■ WATERBODY
- ▭ MUNICIPAL BOUNDARY
- ■ FIRST NATION RESERVE

PROTECTED AREAS

- ▨ PROVINCIAL PARK
- ■ ECOLOGICAL RESERVE
- ■ WILDLIFE MANAGEMENT AREA

0 10 20

1:350,000 KILOMETRES

NOTE(S)

REFERENCE(S)

1. BASE DATA - CANVEC AND MLI, 2016
2. PROJECT DATA - CNL, 2016
3. PROJECTION: TRANSVERSE MERCATOR DATUM: NAD 83 COORDINATE SYSTEM: UTM ZONE14N

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
GENERAL LOCATION OF THE WHITESHELL LABORATORIES SITE

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	CGE
PREPARED	CGE/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**1.0.1**

Path: S:\Clients\Canadian_Nuclear_Laboratories\Manitoba\99_PROJ\20145046\40_PROD\0001_DSAR\20145046-0001-G-0001.mxd

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1.1 Scope and Purpose

The purpose of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public, and the releases of contaminants to the surrounding environment. The scope of the assessment considers the closure phase (which includes decommissioning and reclamation) and long-term performance during the post-closure phase (which includes institutional control and post-institutional control).

The DSAR has been prepared in accordance with CNSC's *REGDOC-2.11.1 Waste Management, Volume III: Assessing the Long-Term Safety of Radioactive Waste Management* (*REGDOC-2.11.1, Volume III* [CNSC 2018a]) and incorporates guidance outlined by the International Atomic Energy Agency (IAEA), specifically *SSG-23 The Safety Case and Safety Assessment for the Disposal of Radioactive Waste* (IAEA 2012) and *SSR-5 Disposal of Radioactive Waste* (IAEA 2011). As per CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), demonstrating long-term safety consists of providing reasonable assurance that waste management will be completed in a manner that protects human health and the environment. This is achieved through the development of a safety case, which includes a safety assessment supported by various arguments based on:

- appropriate selection and application of assessment strategies;
- demonstration of system robustness;
- the use of complimentary indicators of safety; and
- any other evidence that is available to provide confidence in the long-term safety of radioactive waste management.

The DSAR provides a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy. The DSAR demonstrates the level of protection provided to people and the environment by the Project and provides assurance that regulatory safety requirements will be met. This is accomplished through:

- documentation of evidence illustrating how the proposed decommissioning activities can be completed in compliance with regulatory requirements and established guidelines;
- systematic evaluation of safety consequences for both the planned activities and potential disruptions (e.g., accidents or upset scenarios), which is accomplished through the completion of a Hazard and Operability Study (HAZOP) for the closure phase (see Section 5.4.1 Hazard and Operability Study) and a Features, Events, and Processes (FEPs) Analysis for the post-closure phase (see Section 5.4.2 Features, Events, and Processes);
- analysis of selected bounding events; and
- documenting the long-term safety case, and providing the information required by regulatory authorities to support the approval of the preferred decommissioning strategy and the assessment of its long-term performance.

The decommissioning safety assessment:

- provides the rationale for the proposed decommissioning strategy;
- provides evidence that the decommissioning activities can be completed safely;
- presents the institutional controls that will need to be established; and
- presents the timeframe for institutional control.

The DSAR also provides an additional level of detail and framework for the detailed design and systematic work plans that will be completed to confirm compliance and optimization. The DSAR is a “living document” (i.e., continued iterative use as needed) and will be periodically reviewed and updated (as required), approximately every 5 years, over the lifetime of the facility. At subsequent stages in the facility’s lifecycle, as-built information and operational data will be used, when found necessary, to refine the model of the disposal system for assessment purposes. As with the site model, the model of the disposal system will evolve to become more realistic, and less conservative, based on real data.

1.2 Regulatory Requirements and Guidance Documents

The DSAR is limited to ISD of the WR-1 Building and the long-term performance of the Whiteshell Reactor Disposal Facility[2] (WRDF). Therefore, applicable regulatory requirements, guidance documents and safety standards are related to the safety assessment completed for the closure and post-closure phases.

Other decommissioning activities for the WL site are assumed to be unmodified and are covered under CNL’s existing Decommissioning Licence for the WL site (Licence No. NRTEDL-W5-8.00/2024). These activities are documented in the *Whiteshell Laboratories Decommissioning WR-1 Project Comprehensive Study Report* (Comprehensive Study Report; AECL 2001a).

1.2.1 Regulatory Requirements

The Project is required to comply with applicable federal and provincial legislation in accordance with the Decommissioning Licence NRTEDL-W5-8.00/2024, and the Licence Conditions Handbook for Whiteshell Laboratories (CNSC 2020a). Design of the WRDF is governed by CNL’s Engineering Change Control and related procedures, which are accepted by the CNSC as part of the current Decommissioning Licence. Classification of systems and selection of appropriate quality assurance requirements for design and construction have been performed in accordance with these procedures. The ISD approach will require the WL site to apply for a licence as a disposal facility and will need to meet the associated regulatory requirements for this type of facility. This section provides an overview of the federal and provincial requirements applicable to the Project.

1.2.1.1 Federal Acts and Regulations

Proponents wishing to carry out activities related to the construction and operation of facilities for the long-term management or disposal of nuclear waste in Canada must first obtain a licence from the CNSC. The CNSC regulates these activities under the *Nuclear Safety and Control Act*, which establishes the CNSC’s authority to set regulatory requirements for all nuclear-related activities in Canada.

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[2] The Whiteshell Reactor Disposal Facility (WRDF) is the end-state of the Project, after the WR-1 Building is demolished, grouted and covered with a concrete cap and engineered cover.

The relevant regulations under the *Nuclear Safety and Control Act* include:

- *General Nuclear Safety and Control Regulations* (SOR/2000-202)*;*
- *Nuclear Security Regulations* (SOR/2000-209)*;*
- *Radiation Protection Regulations* (SOR/2000-203)*;*
- *Class I Nuclear Facilities Regulations* (SOR/2000-204; although the decommissioning is not operation of a Class IA or IB nuclear facility);
- *Nuclear Substances and Radiation Devices Regulations* (SOR/2000-207); and
- *Packaging and Transport of Nuclear Substances Regulations* (SOR/2015-145).

In accordance with Sections 24 and 26 of the *Nuclear Safety and Control Act*, the ISD approach requires a submission for a Decommissioning Licence, which includes a Detailed Decommissioning Plan (DDP), a Post-closure Safety Assessment and a Safety Case. Although WR-1 decommissioning preparations occur within an existing Decommissioning Licence, changes to future work to allow ISD have not yet been approved by the CNSC. The Project will result in a closed disposal facility; therefore, the licence submission addresses waste disposal facility requirements in addition to the decommissioning requirements.

In addition to the CNSC, the Project is also regulated by the following authorities and legislation:

- Canadian Environmental Assessment Agency. *Canadian Environmental Assessment Act, 2012*;
- Transport Canada. *Transportation of Dangerous Goods Act, 1992;*
- Natural Resources Canada. *Nuclear Liability and Compensation Act*;
- Environment Climate Change Canada. *Canadian Environmental Protection Act*, *1999*;
- Environment Climate Change Canada. *Migratory Birds Convention Act*, *1994*;
- Environment Climate Change Canada. *Species At Risk Act*;
- Environment Climate Change Canada. Polychlorinated Biphenyls (*PCB) Regulations*; and
- Fisheries and Oceans Canada. *Fisheries Act*.

The CNSC requires the environmental effects of all licensed activities to be assessed and considered when licensing decisions are made. An environmental assessment is a review of information used to support the Commission’s determination on whether the licensee will make adequate provisions for the protection of the environment and the health and safety of people while carrying out a licensed activity. Environmental assessments under the *Canadian Environmental Assessment Act, 2012* are required for designated projects, which are defined under the Regulations Designating Physical Activities, and include the construction and operation of facilities for the long-term management or disposal of nuclear waste. An Environmental Impact Statement (EIS; Golder et al. 2022) has been prepared for the Project to meet the requirements of the *Canadian Environmental Assessment Act, 2012.*

1.2.1.2 Provincial Regulations

The Manitoba Conservation and Climate Department (formerly Manitoba Sustainable Development) has been notified of the federal environmental assessment being conducted for the Project. Manitoba Conservation and Climate Department is a member of the WL Public Liaison Committee to maintain awareness of the environmental and socio-economic effects of the Project. Decommissioning of the WR-1 Building is not considered by the Manitoba Conservation and Climate Department as a development under the Government of Manitoba's *The Environment Act*; however, regulations applying to eventual site re-use are listed in the Classes of Development Regulations under the Act. In addition, under the *Canada-Manitoba Agreement for Environmental Assessment Cooperation* (Government of Canada 2007), information on the Project has been provided to Manitoba's Conservation and Climate Department by the Impact Assessment Agency of Canada, and provincial technical staff have been invited to participate in the technical review of the environmental assessment.
The Manitoba Conservation and Climate Department has formed a Technical Advisory Committee to maintain awareness of the environmental and socio-economic effects of the Project and to provide advice to the Director and the Minister of Manitoba Conservation and Climate Department, as required.

1.2.2 Guidance Documents and Safety Standards

Regulating nuclear safety in Canada is the responsibility of the CNSC. Therefore, the Project has been designed to be compliant with the CNSC guidance documents. The IAEA is a valuable resource to provide guidance for decisions concerning safety related to CNL's plans to decommission WR-1. The Project has been designed to be in alignment with the IAEA safety standards.

To demonstrate compliance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), CNSC *REGDOC 2.11.2 Decommissioning* and alignment with IAEA's SSR-5, and to facilitate access to information within the DSAR document, a concordance table has been prepared (Appendix A) that lists the requirements and the location for the corresponding information provided within the DSAR.

1.2.2.1 Canadian Guidance Documents and Safety Standards

In addition to the *Nuclear Safety and Control Act* and associated regulations, the DSAR was developed considering CNSC's *Regulatory Guide G-219 Decommissioning Planning for Licensed Activities* (CNSC 2000a), which describes CNSC's guidance concerning the planning of decommissioning activities. Further, since the ultimate goal of the Project is to place the WL site into a safe long-term condition, this DSAR also considers CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), which provides guidance on the long-term waste management for any radioactive waste that will remain on-site.

The purpose of *REGDOC-2.11.1, Volume III* (CNSC 2018a) is to assist applicants for new licences and for licence renewals in assessing the long-term safety of radioactive waste management, including:

- determining long-term care and maintenance considerations;
- setting post-decommissioning objectives;
- establishing assessment criteria;
- establishing assessment strategies and level of detail;
- selecting timeframes and defining assessment scenarios;
- identifying receptors and critical groups; and
- interpreting assessment results.

The *REGDOC-2.11.1, Volume III* (CNSC 2018a) describes the philosophy that underlies the CNSC's approach to regulating the management of radioactive waste and the principles considered when making regulatory decisions on waste management. The principles in this document that inform CNSC licensee expectations include:

- the generation of radioactive waste is minimized to the extent practicable by the implementation of design measures, operating procedures, and decommissioning practices;
- the management of radioactive waste is commensurate with its radiological, chemical, and biological hazard to the health and safety of persons and the environment, and to national security;
- the assessment of future effects of radioactive waste on the health and safety of persons and the environment encompasses the period when the maximum effect is predicted to occur;
- the predicted effect on health and safety of persons and the environment from the management of radioactive waste is no greater than the effect that is permissible in Canada at the time of the regulatory decision;
- the measures needed to prevent unreasonable risk to present and future generations from the hazards of radioactive waste are developed, funded, and implemented as soon as reasonably practicable; and
- the trans-border effects on the health and safety of persons and the environment that could result from the management of radioactive waste in Canada are not greater than the effects experienced in Canada.

The CNSC's *REGDOC-2.11-2 Decommissioning* (CNSC 2019) provides requirements and guidance regarding the planning, preparation, execution, and completion of decommissioning. This document is complemented by other CNSC regulatory documents and Canadian Standards Association (CSA) guidance documents. The CSA guidance documents also fit into the regulatory framework for the Project. The CSA documents do not form any part of regulation but they do provide meaningful guidance on how to meet regulatory requirements.

1.2.2.2 International Safety Standards

Canada adheres closely to international standards regarding nuclear safety and the Canadian legislation is largely based on the recommendation of international agencies. The most important recommendations to which Canada adheres are those provided by:

- International Commission on Radiological Protection;
- IAEA;
- The American National Council on Radiation Protection and Measurement; and
- Organization for Economic Co-operation and Development Nuclear Energy Agency.

The safety assessment development considers international recommendations relating to the safe management of radioactive waste, including IAEA's SF-1 *Fundamental Safety Principles* (IAEA 2006), SSR-5 *Disposal of Radioactive Waste* (IAEA 2011), SSG-23 *Safety Case and Safety Assessment for Disposal of Radioactive Waste* (IAEA 2012), GSR Part 4, "Safety Assessment for Facilities and Activities (IAEA 2016), as well as safety assessment guidance including IAEA WS-G-5.2 *Safety Assessment for the Decommissioning of Facilities Using Radioactive Material* (IAEA 2008).

The IAEA has also published a specific safety guide SSG-29 *Near Surface Disposal Facilities for Radioactive Waste* (IAEA 2014a). It is noted that as the Whiteshell Reactor Disposal Facility (WRDF) contains intermediate level waste, SSG-29 is not applicable to the Project; however, where relevant, guidance given in SSG-29 has been taken into account. Much of this guide is applicable to the Project because the final end-state for WR-1 will effectively be a near surface disposal facility. The guide gives detailed guidance on the following, relating to design, including:

- containment;
- isolation;
- multiple safety functions;
- passive safety; and
- surveillance and control of passive safety features.

These criteria have been considered in the Project design and are documented in this DSAR.

The IAEA is a valuable resource to provide guidance for decisions concerning safety related to CNL's plans to decommission WR-1. Current IAEA guidance states that in situ decommissioning should not be the preferred decommissioning strategy for nuclear power reactors, except possibly under exceptional circumstances (IAEA 2014a).

The proposed ISD approach aligns with IAEA's safety standard *Decommissioning of Facilities GSR Part 6* (IAEA 2014b), which lists 15 requirements that should be met when selecting a specific decommissioning pathway for a facility. CNL has met each of these requirements when evaluating the ISD for WR-1.

1) **Optimization of protection and safety in decommissioning - Exposure during decommissioning shall be considered to be a planned exposure situation and the relevant requirements of the Basic Safety Standards shall be applied accordingly during decommissioning.**

   National regulations on the protection of the environment shall be complied with during decommissioning, and beyond if a facility is released from regulatory control with restrictions on its future use.

   All decommissioning work at CNL is carried out under the oversight of the CNL Radiation Protection Program, as per the CNSC-issued site decommissioning licence, which provides the framework and constraints for planned exposures during decommissioning work at the WL site.

   Furthermore, the Project is subject to Federal Legislation including the *Nuclear Safety and Control Act*, and *Canadian Environmental Assessment Act, 2012*. The Project is further subject to oversight and approval from the CNSC, which licenses all activities performed by CNL related to decommissioning and waste management. The Decommissioning Licence NRTEDL-W5-8.00/2024 dictates requirements for CNL to comply with and outline what activities CNL is permitted to perform. Regular inspections by the CNSC ensure compliance with the Decommissioning Licence, and all applicable relevant federal legislation. These conditions remain in place until the CNSC deems them no longer necessary for the safety of workers, the public and the environment.

2) **<u>Graded approach in decommissioning</u> - A graded approach shall be applied in all aspects of decommissioning in determining the scope and level of detail for any particular facility, consistent with the magnitude of the possible radiation risks arising from the decommissioning.**

The conduct and regulatory oversight of decommissioning actions shall be applied in a manner that is commensurate with the hazards and risks associated with the decommissioning of the facility.

All decommissioning activities carried out by CNL are subject to CNSC oversight under the Decommissioning Licence NRTEDL-W5-8.00/2024. The Decommissioning Licence outlines the activities which CNL is permitted to perform and conditions that must be met while performing it.
The Decommissioning Licence further identifies all the relevant CNL policies, programs and procedures that decommissioning activities must be performed in accordance with. The CNSC provides compliance oversight to ensure CNL is following the specified policies, program and procedures. One such requirement is the adherence to the CSA N286 quality assurance standard for nuclear power plants, which has specific provisions for application of graded approaches to performing work, commensurate with the risk level involved. As such this graded approach has been implemented in many of the policies, programs and procedures identified in the Decommissioning Licence, which have been deemed as satisfactory by the CNSC. The graded approach is reflected in procedures for environmental review, radiation and contamination monitoring, occupational health and safety measures, quality assurance and waste management and minimization, among others. The ALARA Principle (As Low As Reasonably Achievable) permeates CNL's safety culture, and the 'Reasonably' portion of that is where the graded approach is applied.

The graded approach is also apparent in CNSC oversight of CNL operations. The Decommissioning Licence also outlines which activities WL may perform without notifying the CNSC, where the CNSC must be notified, or where the CNSC must approve prior to execution. These distinctions are based on the commensurate risk of the activities.

3) **<u>Assessment of safety for decommissioning</u> - Safety shall be assessed for all facilities for which decommissioning is planned and for all facilities undergoing decommissioning.**

The final decommissioning plan shall be supported by a safety assessment addressing the planned decommissioning actions and incidents, including accidents that may occur or situations that may arise during decommissioning.

As part of the ongoing environmental assessment, under the *Canadian Environmental Assessment Act, 2012*, CNL has prepared an EIS, which summarizes the assessed effects of the project on the environment. This is supported through this detailed DSAR, compliant with CNSC's *REGDOC 2.11.1, Volume III* (CNSC 2018a). Detailed calculations and modelling that support the assessment in the DSAR are provided in an Environmental Risk Assessment (ERA) report (EcoMetrix 2021), and a Groundwater Flow and Solute Transport Modelling report (Golder 2021). The DSAR also provides an assessment of accident and malfunction scenarios during decommissioning, and their effect on the environment. All of these documents provide supporting information for the Detailed Decommissioning Plan (DDP) that is prepared in compliance with CSA N294 and submitted for CNSC acceptance as a component of the licence application.

4) **<u>Responsibilities of the government for decommissioning</u> - The government shall establish and maintain a governmental, legal and regulatory framework within which all aspects of decommissioning, including management of the resulting radioactive waste, can be planned and carried out safely. This framework shall include a clear allocation of responsibilities, provision of independent regulatory functions, and requirements in respect of financial assurance for decommissioning.**

The responsibilities of the government shall include:

- *Establishing a national policy for the management of radioactive waste, including radioactive waste generated during decommissioning;*
- *Establishing and maintaining the legal, technical and financial responsibilities for organizations involved in decommissioning, including responsibilities for granting the authorization to conduct decommissioning and for the management of the resulting radioactive waste;*
- *Ensuring that the necessary scientific and technical expertise is available both for the licensee and for the support of regulatory review and other independent national review functions;*
- *Establishing a mechanism to ensure that adequate financial resources are available when necessary for safe decommissioning and for the management of the resulting radioactive waste.*

The Government of Canada provides the legislative framework supporting the CNSC, including defining its mandate and authority as Canada's independent nuclear regulator. The CNSC is responsible for the oversite of all civilian nuclear activities in Canada.

Whiteshell Laboratories is the property of AECL, which is a Schedule III, Part 1 Crown Corporation under the *Financial Administration Act* and an agent of Her Majesty in Right of Canada. As owner, AECL retains responsibility for the site, financial obligations for decommissioning, and long-term management of the site post-closure. These liabilities have been officially recognized by the Minister of Natural Resources in a letter dated July 31, 2015 (Rickford 2015) and satisfy CNSC's *REGDOC-3.3.1 Financial Guarantees for Decommissioning of Nuclear Facilities and Termination of Licensed Activities* (CNSC 2021). AECL has chosen a Government-Owned, Contractor-Operated approach to completing the decommissioning of WL. The tendering process reviewed the proposed approaches to the decommissioning of WL against:

- compliance with AECL mandate, policies and procedures as agent of the Federal Government of Canada;
- adherence to CNSC requirements for the vendor to be the site licence holder and be approved by the CNSC to perform decommissioning work;
- expertise of each vendor in safely performing nuclear decommissioning work; and
- financial commitments of AECL to execute work safely.

5) **Responsibilities of the regulatory body for decommissioning - The regulatory body shall regulate all aspects of decommissioning throughout all stages of the facility's lifetime, from initial planning for decommissioning during the siting and design of the facility, to the completion of decommissioning actions and the termination of authorization for decommissioning. The regulatory body shall establish the safety requirements for decommissioning, including requirements for management of the resulting radioactive waste, and shall adopt associated regulations and guides. The regulatory body shall also take actions to ensure that the regulatory requirements are met.**

The CNSC is the authority having jurisdiction for all nuclear decommissioning work in Canada. The CNSC has a rigorous licencing approach that ensures nuclear safety in all licensed nuclear decommissioning work. The Decommissioning Licence (Licence No. NRTEDL-W5-8.00/2024) currently issued to CNL, and all future licences and licence revisions granted to CNL for decommissioning the WL site, do and will include specific requirements, standards and guidance for maintaining safe decommissioning operations.
These requirements are developed with input from regulatory and industry experience and take into consideration international guidance and best practices. As a member state of the IAEA, Canada (and therefore the CNSC) are committed to pursuing the highest standards in nuclear safety through international collaboration and sharing operational experience.

6) **Responsibilities of the licensee for decommissioning - The licensee shall plan for decommissioning and shall conduct the decommissioning actions in compliance with the authorization for decommissioning and with requirements derived from the national legal and regulatory framework. The licensee shall be responsible for all aspects of safety, radiation protection and protection of the environment during decommissioning.**

The responsibilities of the licensee shall include:

- *Selecting a decommissioning strategy as the basis for preparing and maintaining the decommissioning plans throughout the lifetime of the facility.*
- *Preparing and submitting an initial decommissioning plan and its updates for review by the regulatory body.*
- *Establishing and implementing an integrated management system. If the licensee changes during the lifetime of the facility, procedures shall be put in place to ensure the transfer of responsibilities for decommissioning to the new licensee.*
- *Fostering a safety culture in order to encourage a questioning and learning attitude towards safety, and to discourage complacency.*
- *Estimating the cost of decommissioning and providing financial assurances and resources to cover the costs associated with safe decommissioning, including the management of the resulting radioactive waste.*
- *Notifying the regulatory body prior to the permanent shutdown of the facility.*
- *Submitting a final decommissioning plan and supporting documents for review and approval by the regulatory body, in accordance with national regulations, in order to obtain an authorization to conduct decommissioning.*
- *Managing the decommissioning project and conducting decommissioning or ensuring oversight of the actions conducted by contractors.*

- *Managing the remaining operational waste from the facility and all waste from decommissioning.*
- *Ensuring that the facility is maintained in a safe configuration during the period of transition following permanent shutdown and until the approval of the final decommissioning plan.*
- *Performing safety assessments and environmental impact assessments in support of decommissioning actions.*
- *Preparing and implementing appropriate safety procedures, including emergency plans.*
- *Ensuring that properly trained, qualified and competent staff are available for the decommissioning project.*
- *Performing radiological surveys in support of decommissioning.*
- *Verifying that end state criteria have been met by performing a final survey.*
- *Keeping and retaining records and submitting reports as required by the regulatory body.*

CNL performs many of these responsibilities on a daily basis as part of its core business operations. CNL carries out all work at WL the Decommissioning Licence (Licence No. NRTEDL-W5-8.00/2024) from the CNSC. The requirements for obtaining and maintaining a licence align with the objectives of the bullet list above. All aspects of the decommissioning work, are subject to CNSC oversight and acceptance prior to any work being performed, including:

- preliminary and detailed planning;
- integrated management systems;
- development of company safety culture;
- cost estimating and financial guarantees;
- safe work execution and oversight;
- waste management, facility maintenance and safety;
- safety assessments supporting decommissioning planning;
- emergency planning, training and qualification of staff; and
- record retention.

The preparation of a DDP by CNL summarizes the pertinent information noted above for CNSC review and acceptance. The CNSC performs regular compliance inspections to verify CNL complies with the requirements of the Decommissioning Licence and the work summarized in the DDP. CNL also develops work plans with additional detail on how work scope of the DDP will be carried. These work plans are provided to the CNSC for information, as a means to assess the plans' compliance with the goals outlined in the DDP. Upon completion of the work, end-state reports are prepared to summarize the work performed against the planned activities, noting discrepancies or changes, for CNSC acceptance.

7) **Integrated management system for decommissioning – The Licensee shall ensure that its integrated management system covers all aspects of decommissioning.**

The prime responsibility for safety shall remain with the licensee.

CNL is the licensee for the overall WL Closure Project, including the proposed ISD of WR-1. Under the terms of the Decommissioning Licence, CNL has demonstrated its commitment to safety through both policy and through daily work activities including safe work processes such as work permits, Event Free Tools, and fostering a strong safety culture that permeates the organization.

8) **Selecting a decommissioning strategy - The licensee shall select a decommissioning strategy that will form the basis for the planning for decommissioning. The strategy shall be consistent with the national policy on the management of radioactive waste.**

There may be situations in which immediate dismantling is not a practicable strategy when all relevant factors are considered. The selection of a decommissioning strategy shall be justified by the licensee. The licensee shall demonstrate that, under the strategy selected, the facility will be maintained in a safe configuration at all times and will reach the specified decommissioning end state, and that no undue burdens will be imposed on future generations.

CNL has selected ISD as the decommissioning strategy for WR-1. In the absence of a well-defined national waste strategy, CNL continues to pursue a risk-based approach to radioactive waste management that complies with all CNSC regulations, applicable legislation, and where appropriate aligns with international guidance and best practices. The justification for the selection of this strategy is presented in the EIS (Section 2.0 Purpose of the Project and Alternatives to the Project). The EIS, supported by the DSAR, the ERA, and other technical documents, demonstrate that the effects of this decommissioning strategy do not place an undue burden on future generations of people and the environment.

9) **Financing of decommissioning – Responsibilities in respect of financial provisions for decommissioning shall be set out in national legislation. These provisions shall include establishing a mechanism to provide adequate financial resources and to ensure that they are available when necessary, for ensuring safe decommissioning.**

The requirements for financial guarantees are laid out in CSA N294 for Decommissioning of facilities containing nuclear substances. Adherence to this standard by CNL is required from national legislation to CNSC regulatory requirements, and licence conditions.

10) **Planning for decommissioning - The licensee shall prepare a decommissioning plan and shall maintain it throughout the lifetime of the facility, in accordance with the requirements of the regulatory body, in order to show that decommissioning can be accomplished safely to meet the defined end state.**

CNL has prepared an Overview DDP for the wider WL site closure project. The Overview DDP provides the overall plan for decommissioning of the WL site, including WR-1, and has been periodically revised to include adjustments to the plan. This Overview DDP is part of a larger body work that supports the CNL application for a decommissioning licence for the WL site. It is supported by additional technical information, including but not limited to the CSR, EIS, DSAR, and CNL policies, programs and procedures. Furthermore, CNL has also prepared a DDP specifically for WR-1 that will be maintained throughout the project lifetime and is also subject to CNSC review and approval prior to being implemented.

11) **<u>Final decommissioning Plan</u>** - **Prior to the conduct of decommissioning actions, a final decommissioning plan shall be prepared and submitted to the regulatory body for approval.**

The final decommissioning plan and supporting documents shall cover the selected decommissioning strategy; the schedule, type and sequence of decommissioning actions; the waste management strategy applied, including clearance, the proposed end state and how the licensee will demonstrate that the end state has been achieved; the storage and disposal of the waste from decommissioning; the timeframe for decommissioning; and financing for the completion of decommissioning. If the final decommissioning plan includes new technologies and concepts for decommissioning, the licensee shall demonstrate that such methods are safe and effective. Interested parties shall be provided with an opportunity to examine the final decommissioning plan can provide comments prior to its approval.

A DDP has been be prepared to address the requirements of a final decommissioning plan including each of the items listed above. Development of the DDP is done in accordance with the WL Closure Project Quality Assurance Manual, CSA N294, the WL site licence and the WL Licence Conditions Handbook, and guidance from recent project experience and regulatory input.

12) **<u>Conduct of decommissioning actions</u> – The licensee shall implement the final decommissioning plan, including management of radioactive waste, in compliance with national regulations.**

Decommissioning techniques shall be selected such that protection and safety is optimized, protection of the environment is ensured, the generation of waste is minimized and any potential negative impact on the storage and disposal of waste is minimized.

All operations at CNL, decommissioning or otherwise, are subject to approval by the CNSC. Approval is granted via a site licence that summarizes the CNSC accepted policies, programs and key procedures that govern the work processes at WL. All work is performed in accordance with the policies, programs and key procedures identified, including Radiation and Environmental Protection, Occupational Health and Safety, Waste Management, Security and Quality Assurance. Any changes to these policies, programs or key procedures are submitted, if required, to the CNSC for review and/or acceptance prior to being implemented to perform work.

CNL has been performing decommissioning work at WL under a CNSC-issued Decommissioning Licence since 2002. Since 2002, the CNSC has verified that CNL is performing its work in a safe and compliant manner. All work necessary to decommission WR-1 will comply with these accepted practices to ensure protection of workers, the public and the environment, safe and optimized waste management and minimization.

The ISD of WR-1 is subject to an environmental assessment, and must demonstrate safety and protection of people and the environment through a decommissioning safety assessment that complies with applicable CNSC regulatory requirements and Canadian standards, prior to receiving CNSC approval to proceed.

13) **<u>Emergency response arrangements for decommissioning</u> - Emergency response arrangements for decommissioning, commensurate with the hazards, shall be established and maintained, and events significant to safety shall be reported to the regulatory body in a timely manner.**

Establishment of appropriate emergency response measures is required as per the WL Decommissioning Licence and the Licence Conditions Handbook (Condition 10.1). CNL meets this requirement through the implementation of WL's Site Emergency Response Plan, as per the CNSC-issued site decommissioning licence.

14) **<u>Radioactive waste management in decommissioning</u> - Radioactive waste shall be managed for all waste streams in decommissioning.**

Radioactive waste that remains at the facility and radioactive waste that is generated during decommissioning shall be disposed of properly. If disposal capacity is not available, radioactive waste shall be stored safely in accordance with the relevant requirements.

The safe and effective management of radioactive wastes is mandated through the CNSC Decommissioning Licence under which all activities at WL are performed. Section 11 of the WL Decommissioning Licence mandates that CNL maintain a waste management program for WL.

CNL maintains a waste management program for WL that controls the management of all radioactive wastes generated at WL, as per the CNSC-issued site decommissioning licence.

15) **<u>Completion of decommissioning actions and termination of the authorization for decommissioning</u> - On the completion of decommissioning actions, the licensee shall demonstrate that the end state criteria as specified in the final decommissioning plan and any additional regulatory requirements have been met. The regulatory body shall verify compliance with the end state criteria and shall decide on termination of the authorization for decommissioning.**

Upon completion of ISD of WR-1, CNL will prepare an End-State Report, which will document the work performed and the end-state achieved. Further, CNL will implement an Environmental Assessment Follow Up Program (EAFP) to provide evidence that the system is performing as designed after the end-state has been achieved. The End-State Report is mandated by the WL Quality Assurance Plan and CSA N294. The EAFP is a mandated component of the environmental assessment process under the *Canadian Environmental Assessment Act, 2012*. Ongoing environmental monitoring is also a critical aspect of the WL Decommissioning Licence. CNL has previously prepared end-state reports for other work completed on the WL site, to the satisfaction of the CNSC, and will ensure the WR-1 End-State Report meets the same expectations. CNL has also been performing routine EAFP monitoring of all decommissioning work, to the satisfaction of the CNSC, to demonstrate their compliance with the criteria of the CSR.

CNL will not adjust or cease monitoring of the WL without the approval of the CNSC, and any decision to terminate, amend or transfer the Decommissioning Licence for the WL site will be made by the CNSC.

1.3 Documentation Framework and Structure

There are multiple documents in support of the Project that are closely related and contain a degree of overlap in their supporting analyses and content, specifically:

- the DSAR;
- the EIS;
- the Safety Case Report; and
- the DDP.

This **DSAR** provides a safety envelope for execution of the Project and is required to obtain a licence. The DSAR assesses safety for both the closure and post-closure phases of the Project. The DSAR is prepared in parallel with the design of the facility and is produced to support and inform the EIS and licensing.

The **EIS** documents the assessment and determination of the potential effects of the Project on the environment, including the atmospheric, geological, hydrogeological, surface water, aquatic and terrestrial environments, as well as studying the effects on land and resource use and socio-economic environment. The EIS also assesses effects of accidents and malfunctions on the environment, and how the environment may affect the Project. The EIS evaluates potential effects during the entire lifecycle of the Project, including closure and post-closure.

The **Safety Case Report** provides a comprehensive argument that ISD, post-closure activities, and long-term disposal of the WRDF can be safety undertaken. The safety case demonstrates that associated hazards and risks have been assessed, appropriate limits and conditions have been defined, and that adequate safety measures have been identified and put into place to support the Project. The document also provides the rationale for the duration of the institutional control period, although it is fully recognized that this duration will be decided in conjunction with the CNSC.

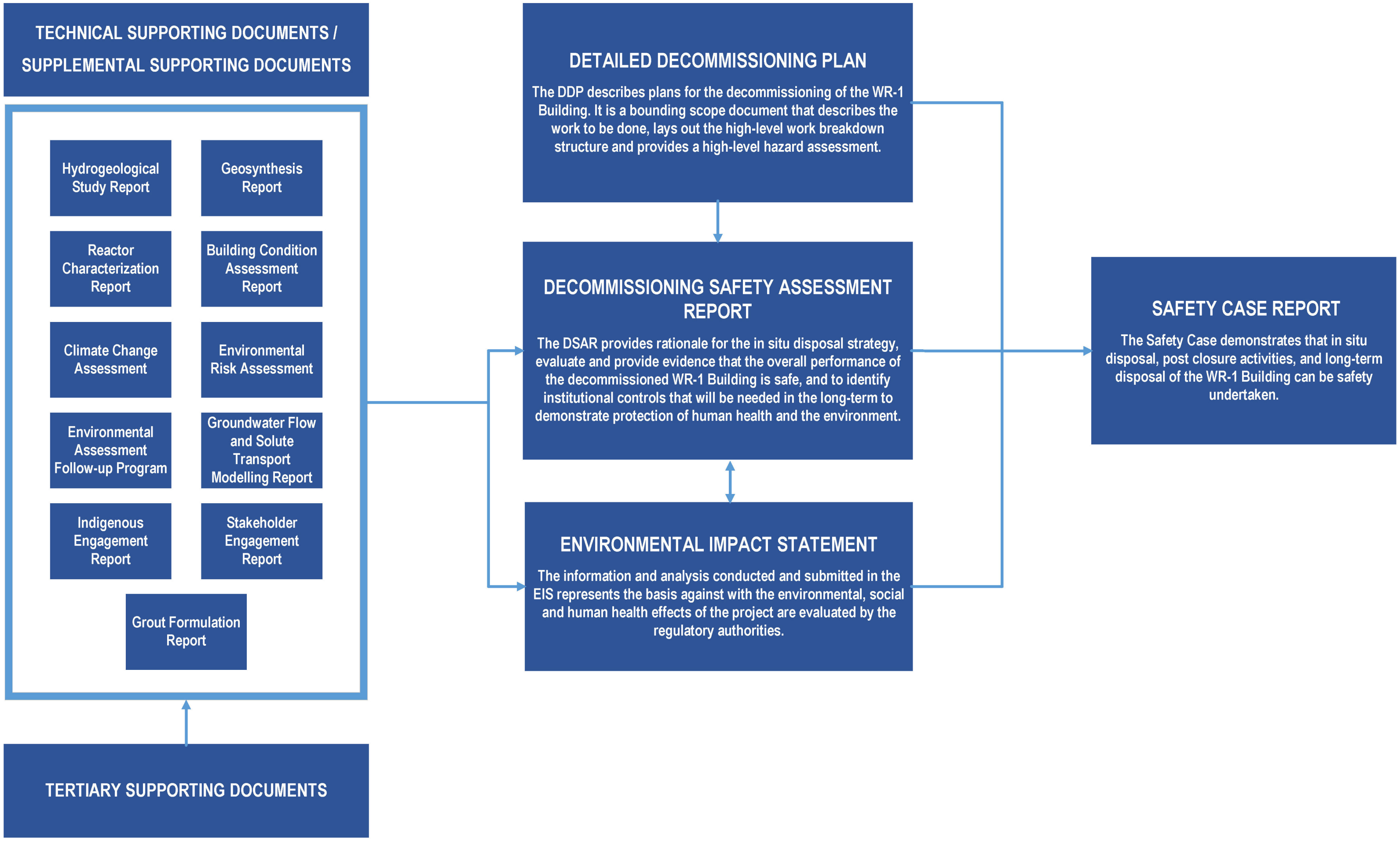
The **DDP** describes plans for the decommissioning of the WR-1 Building. It is a document that describes the work to be done, lays out the high-level work breakdown structure and provides a high-level hazard assessment.

Figure 1.3-1 illustrates the interactions between each of these documents, as well as the technical studies being completed to support these documents.

The safety strategy for the Project is sufficiently well developed at this stage to provide assurance that the overall decommissioning strategy will provide and preserve the safety functions envisaged for the WRDF. As the Project develops, the safety strategy will be continually validated and any changes to it justified in the Safety Case Report. Any evolution of the safety strategy will be carefully recorded, and the records preserved for use in the future when regulatory personnel assigned to the site and/or site staff may have changed.

The structure of the DSAR takes into consideration the guidance outlined in the *REGDOC 2.11.1, Volume III* (CNSC 2018a). Figure 1.3-2 depicts how the DSAR has been structured to address the requirements of the regulatory document.

PATH: S:\Clients\Canadian_Nuclear_Laboratories\Manitoba\99_PROJ\20145046\40_PROD\0001_DSAR\20145046-0001-G-0002.mxd PRINTED ON: 2021-11-05 AT: 11:29:38 AM



TECHNICAL SUPPORTING DOCUMENTS / SUPPLEMENTAL SUPPORTING DOCUMENTS

- Hydrogeological Study Report
- Geosynthesis Report
- Reactor Characterization Report
- Building Condition Assessment Report
- Climate Change Assessment
- Environmental Risk Assessment
- Environmental Assessment Follow-up Program
- Groundwater Flow and Solute Transport Modelling Report
- Indigenous Engagement Report
- Stakeholder Engagement Report
- Grout Formulation Report

TERTIARY SUPPORTING DOCUMENTS

DETAILED DECOMMISSIONING PLAN

The DDP describes plans for the decommissioning of the WR-1 Building. It is a bounding scope document that describes the work to be done, lays out the high-level work breakdown structure and provides a high-level hazard assessment.

DECOMMISSIONING SAFETY ASSESSMENT REPORT

The DSAR provides rationale for the in situ disposal strategy, evaluate and provide evidence that the overall performance of the decommissioned WR-1 Building is safe, and to identify institutional controls that will be needed in the long-term to demonstrate protection of human health and the environment.

ENVIRONMENTAL IMPACT STATEMENT

The information and analysis conducted and submitted in the EIS represents the basis against with the environmental, social and human health effects of the project are evaluated by the regulatory authorities.

SAFETY CASE REPORT

The Safety Case demonstrates that in situ disposal, post closure activities, and long-term disposal of the WR-1 Building can be safety undertaken.

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	ST
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
WR-1 IN SITU DISPOSAL DOCUMENT MAP

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	1.3-1

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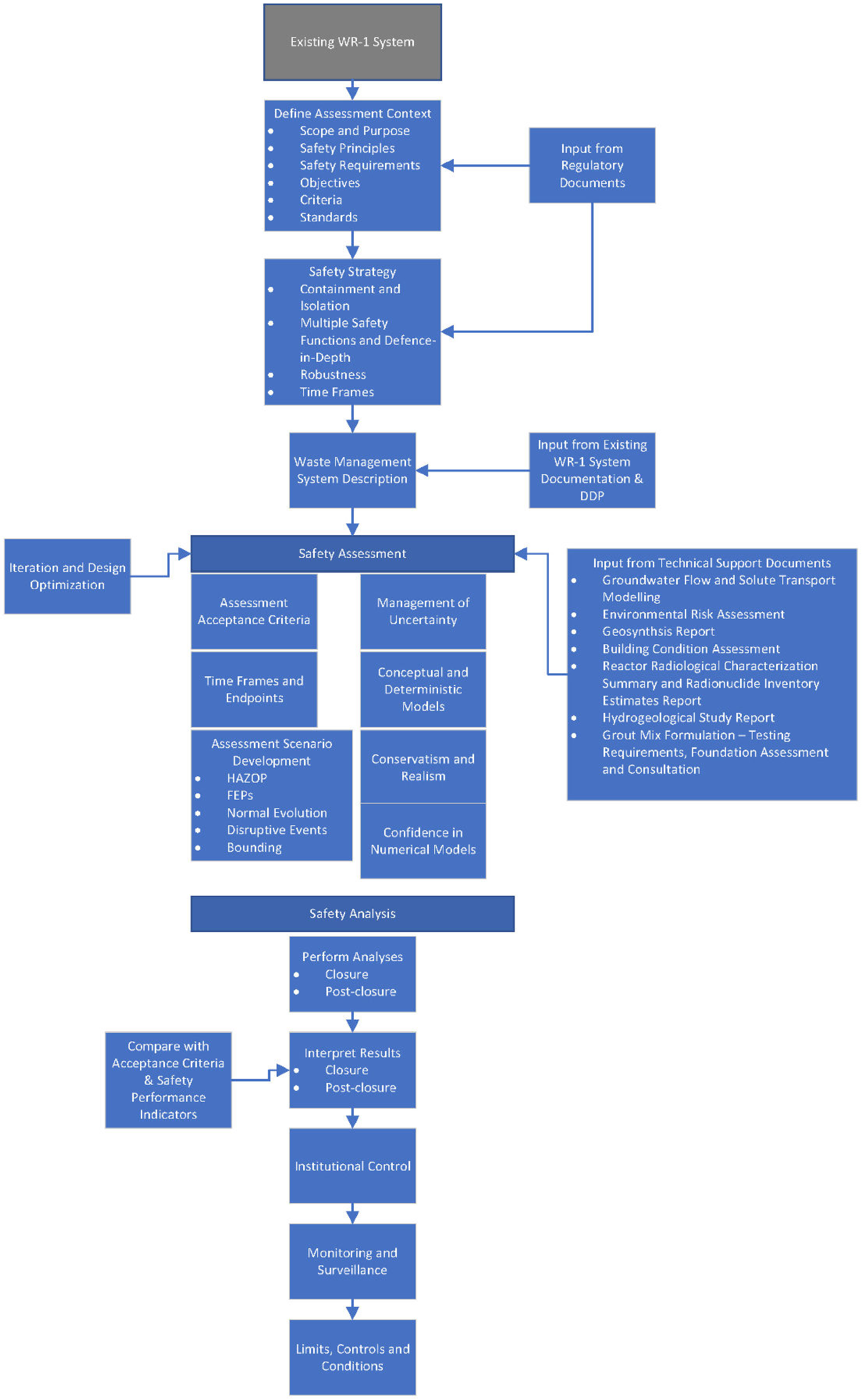

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DSAR REPORT STRUCTURE

CONSULTANT

GOLDER
MEMBER OF WSP

YYYY-MM-DD	DECEMBER 2021
DESIGNED	-
PREPARED	OR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	1.3-2

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2.0 BACKGROUND INFORMATION

This section provides background information on the environmental setting and site description, followed by an overview of the currently approved WR-1 decommissioning strategy and status. This forms the basis for the subsequent safety assessments undertaken.

2.1 Environmental Setting

2.1.1 General Location of the Project

The location of the WL site is shown on Figure 1.0-1. The WL site is located in southeastern Manitoba, approximately 100 kilometres (km) northeast of the City of Winnipeg. The Winnipeg River passes through the WL site, with a small portion of the WL site located on the west side of the river. The WL site is accessed via Provincial Highway #11 and Provincial Road #211. Entry/exit access to the main laboratory area is via an access control station at Building 401 (B401). Additional vehicle and personnel access is through control and monitoring stations located at each Controlled Area 2[3] access point. The centre of the main laboratory facility is located at approximately latitude 50°10’46”N and longitude 96°03’35”W.

2.1.2 Atmospheric Environment

Daily meteorological data from the Pinawa Whiteshell Nuclear Research Establishment (WNRE) climate station (ID 5032162) and Winnipeg Richardson International Airport (ID 5023222) were collected for the period from 1981 through to 2010. The daily average temperature in the winter season is approximately -14.3°C, while the daily average temperature in the summer season is approximately 18.0°C. The extreme minimum temperature during the 30-year period was -47.8°C while the extreme maximum temperature during the 30-year period was 37.5°C. Temperatures below -10°C have typically occurred between November and April, while temperatures above 30°C occur occasionally between May through August.

The 30-year climate normal from the Pinawa WNRE station indicates an average annual precipitation of approximately 578 millimetres equivalent (mm[eq]) for the region, with the highest precipitation occurring in the summer at 253.2 mm[eq]. The greatest extreme daily precipitation also occurs in summer at 168.4 mm[eq]. Approximately 94% of the precipitation in winter is attributed to snow. Winter extreme daily precipitation is 35 mm[eq]. The Pinawa area is generally characterized by winds predominantly blowing from the south-southeast or north-northwest directions with an annual average wind speed of 17 kilometres per hour (km/h).

The Winnipeg station (65 Ellen Street) is the only air quality monitoring station located within 100 km of the WL site and is the most representative station of the atmospheric environment regional study area (RSA[4]); therefore, it represents the background for non-radiological indicator compounds monitored at that station. The existing concentrations are below the respective provincial and federal criteria for each indicator compound, suggesting that the region has generally good air quality.

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[3] Controlled Areas are defined as site areas in which normal working conditions, including unplanned events, require personnel to follow well-established radiation protection procedures and practices. Two types of Controlled Areas are used at Whiteshell Laboratories: Controlled Area 1 and Controlled Area 2 (CNL 2020c). In a Controlled Area 2, activities and facilities that pose a radiation and/or contamination exposure hazard are permitted.

[4] The RSA is defined as the area within which the potential effects of a project may interact with the effects of other projects. The RSA varies depending on the environmental component assessed. The atmospheric environment RSA is defined as a 12 km by 12 km square that encompasses the WL site and all sources of emissions of the ISD of WR-1.

2.1.3 Geologic and Hydrogeologic Environment

For a detailed description of the regional and local geology and hydrogeology of the WL site, refer to the reports titled *WR-1 Hydrogeological Study Report* (Dillon 2018) and *Geosynthesis for WR-1 Environmental Impact Statement* (CNL 2021a).

2.1.3.1 Topography and Geomorphology

The flat topography coupled with the post-glacial immature drainage throughout the area, results in a dominance of very poorly drained conditions and development of extensive areas of organic soils. Better drained soils (6% of the area) occur adjacent to the Winnipeg River and its tributary channels and as minor inclusions in areas of greater relief adjacent to rock outcrops in the eastern part of the municipality. Imperfectly drained soils occupy 20% of the area. Drainage of local areas has been improved for development of infrastructure related to roads, and the WL site.

The WL site, in general, is mixed forest and generally flat with topography rising slightly to the east near the WL landfill several kilometres to the east of WR-1. The WL main campus is relatively flat (0% to 1% slope). The WL main campus is vegetated with grass that is continually maintained and has few trees. The ground level (Level 600 of the WR-1 Building) of WR-1 is at 266.7 metres above sea level (masl). To the west of the main campus, the ground slopes gently (7% to 8% slope) westwards towards the Winnipeg River. The Winnipeg River is located approximately 500 metres (m) to the west of the WR-1 location at approximately 255 masl, and the riverbanks rise approximately 13 m to the level of WL main campus. Most of this rise occurs at the riverbanks. A break in topography occurs at the west bank of the Winnipeg River, with an approximate water level elevation of 255 masl. The river bottom is near the bedrock surface at an elevation of about 252 masl.

The Winnipeg River flow in general is northwestward from the Lake of Woods to Lake Winnipeg. This river is 235 km long from the Norman Dam at Kenora, Ontario to its mouth at Lake Winnipeg. Its watershed is mainly in Canada (106,500 square kilometres [km^2]) and extends into norther Minnesota by approximately 20,000 km^2. The watershed stretches to the height of land about 100 km west of Lake Superior.

The flow of water in the river is controlled by the Lake of the Woods Control Board through a control structure at the start of the Winnipeg River. Two dams in Ontario and six hydroelectric dams in Manitoba provide further regulation and controls of the flows. The regulation of the Winnipeg River has increased the size of lakes along the route of the river; however, dams on the Winnipeg River are run-of-river systems and have limited storage capacity. Upstream, from the WL site is Natalie Lake, the forebay of the Seven Sisters Dam. Immediately downstream of the Seven Sisters Dam, the Winnipeg River is joined by a tributary, the Whitemouth River. The two rivers meet in an area of exposed bedrock. The Winnipeg River channel turns northward past the Seven Sisters Dam. South of the Highway 211 bridge, frequent rock outcrops and shallows are notable in the years of lower water. Near the WL site north of the Highway 211 bridge, the channel becomes somewhat narrower. River depth charts indicate that shelves are present downstream from the Highway 211 bridge on the east side of the river. Upon reaching a widening of the river downstream from the WL site near the town of Lac du Bonnet, the river reaches depths of up to approximately 27 m. The river formed post glacially and cuts through overburden with its base in bedrock. The variable depths in the Winnipeg River near the WL site suggest a series of rapids that has been flooded by hydroelectric development based upon the flooded shelves on the east side of the river. This appears to be supported by historic records including paintings that appear to indicate setting reminiscent of the now flooded narrower central river channel.

The influence of seismicity on the geomorphology of the WL site is anticipated to be negligible due to the relatively low likelihood of seismic events occurring in the area (see Section 2.1.3.4 Seismic).

2.1.3.2 Surficial Geology

The regional surficial geology WL site area comprises of extensive deposits of till and both sandy and clay based, glaciofluvial and glaciolacustrine materials. Deposits of glacial deposited material are found as till deposits, which is poorly sorted material ranging from boulder and cobbles to fine grained material. Predominantly sandy and/or clayey tills are observed throughout the western portions of the region but are less widespread in the central and eastern portion of the region (Betcher et al. 1988), where they are generally confined to bedrock depressions between bedrock outcrops. End-moraine and outwash complexes (comprising of mostly sand and gravel) are evident just west of the Winnipeg River. In general, the Winnipeg River divides the area into two basic subregions with regard to overburden geology: (i) calcareous tills to the west, and (ii) sandy tills and glaciofluvial deposits to the east (Guthrie and Scott 1988). Finer glaciolacustrine deposits are evident along the drainage depressions of the Winnipeg River and Pinawa Channel and the Lee River.

In 2015, boreholes for seven monitoring well nests were drilled (consisting of 4 to 6 monitoring wells per nest) surrounding the perimeter fence of the WL site (KGS Group 2016). Based on the borehole logs from these locations the surficial geology in the WL site area is similar to that described in the regional setting (McPherson, 1968), consisting of (from the bedrock upwards):

- *Glacial Till (also referred to as Basal Sand and Basal Till)*. The glacial till overlies the bedrock throughout the majority of the regional geological setting. This unit varies from a silty coarse sand till (in the main WL area) to a clean medium to coarse sand (in the area of the WMA), and boulders are common above the bedrock surface. In the WMA, this unit has been found to vary in thickness from 1 to 7 m thick, whereas in the main WL area, this unit varies in thickness from 3.6 to 8.3 m. This unit is referred to as “basal sand” (Dillon 2018) due to the increased sand content observed in this unit in other areas of the site (primarily the WMA). Based on its grain-size distribution and hydraulic conductivity characteristics within the area of WR-1 it is not considered to be representative of sand. However, for consistency with previous work it is referred to herein as a “basal sand”.
- *Glacio-Lacustrine Clay (also referred to as Clay Till)*. The Glacial Till is overlain by a clay till unit containing sand and silty sand seams. The lower portion of the Clay Till is derived from the Glacial Till. In the WMA, this unit has been found to vary in thickness from 2 m to 5 m (AECL 2008). In the area of WR-1, this unit varies in thickness from 3.0 m to 7.0 m. The clay till is generally thinnest in the central portion of the central WL area and thickens to the northwest and southwest towards the Winnipeg River.
- *Transitional Glacio-Lacustrine Clay (Clay) and Glacio-Fluvial and Glacio-Lacustrine Sandy Silt (Interbedded Silt and Clay)*. A glacio-lacustrine clay unit overlays the clay till unit throughout the study area. This unit is transitional, with the lower portion more laminated with silty interbeds, and the upper portion more massive. A thin surficial interbedded silt and clay unit overlies these clays. These units have been grouped given their similar properties and relative thinness of the surficial unit. In the WMA this unit has been found to vary in thickness from 2 to 8 m and is thickest in the lagoon area (AECL 2008). In the main WL area, this unit is relatively uniform in thickness, varying from 5.5 to 7.3 m.

Noted differences at the WR-1 site from observations at the Waste Management Area (WMA) include the existence of the upper organic complex, increased clay thickness, and reduced sand content in the deepest basal sand unit. Surficial geology at the plant site is shown in cross-sections on Figure 3-2 and Figure 3-3 of the *WR-1 Hydrogeological Study Report* (Dillon 2018). The hydrostratigraphic cross-section depicted on Figure 3-2 extends from the river to the west, to the WMA. The hydrostratigraphic cross-section depicted on Figure 3-3 shows the localized hydrostratigraphy, between the river to the west, to monitoring well Nest 2, located upgradient and east of the WR-1. While there is some discontinuity in the basal till (the unit was not observed at 15-6A or 16-8A to the west of WR-1), the unit extends broadly across the site, with increasing fine-grained sediment content westwards, near the Winnipeg River.

2.1.3.3 Bedrock Geology

The 2016 field investigations identified bedrock at depths varying between 14 and 19 m below ground surface, which is consistent with the undulating topography observed at surface outcrops in the area. Bedrock observed was consistent with local and regional bedrock geological records. Bedrock was observed to consist predominantly of feldspar-rich granite. Fractures were observed within the upper 10 m of bedrock. Stratigraphic cross-sections showing site stratigraphy extending from the bedrock wells installed near the WMA to the Winnipeg River are shown on Figure 3-2 and Figure 3-5 of the *WR-1 Hydrogeological Study Report* (Dillon 2018).

Previous drilling activities to bedrock at the WMA and recent drilling activities both observed large boulders present within the basal sand aquifer that directly overlies the bedrock surface. Several deep boreholes have been drilled into bedrock near the WMA and indicate that the upper 200 m of bedrock is relatively unfractured. However, the uppermost zone of Precambrian bedrock (upper 10 m) and a second zone (20 m to 30 m) have been found to contain a higher frequency of fractures.

2.1.3.4 Seismic

There are ancient faults identified in and near the Lac du Bonnet batholith that may contribute to local features and perhaps river orientation; however, there is no recent activity in the region. Based on a detection level of 2.5 on the Richter scale, the WL area and the southern two-thirds of Manitoba are aseismic. Detailed information on earthquakes that have occurred in Canada is contained in publications of Earthquakes Canada of Natural Resources Canada and their predecessor organizations.

A seismic zoning map for Canada has been developed based on these studies and is used in the National Building Code of Canada (NBCC) 2015 (NRCan 2019). The seismic hazard maps are derived from statistical analysis of past earthquakes and from advancing knowledge of Canada’s tectonic and geological structure. On the maps, seismic hazard is expressed as the most powerful ground motion that is expected to occur in an area for a given probability level. Contours delineate regions likely to experience similarly strong ground motions. Earthquakes Canada provides information on recent and historical earthquakes. The results support the NBCC classification that seismic activity has not been noted or recorded in Manitoba over nearly a 400-year period.

A seismic hazard analysis was completed for the WL site. In 1995, the NBCC placed the WL site (and all of Manitoba) within a Seismic Zone 0, a zone that has a probability of exceedance of 0.0021. The peak ground acceleration (PGA) (i.e., the maximum acceleration that a rigid structure would experience if it was located on bedrock) data from NBCC was considered for the years 2005, 2010, and 2015. The trend since 2005 is that for every probability of exceedance, the seismic hazard at the WL site has decreased (NBCC 2015). For the 1 in 10,000-year probability of exceedance, the PGA is approximately 0.10. Comparatively, the 0.10 PGA represents a light (almost moderate) earthquake for which one would not normally expect structural damage for NBCC designed building and components. There could be some non-structural damage such as fine cracking on non-ductile non-structural elements (e.g., plaster or drywall). Structures that could experience damage due to this size of earthquake are those located on soils that are susceptible to amplification and/or have quite low natural frequencies. These structures may have characteristics that make them susceptible to earthquakes (e.g., tall structures will no bracing or shear walls). Conventionally designed structures using ductile materials (e.g., steel or reinforced concrete) following good engineering practices that incorporate bracing/shear walls, symmetric geometry, and low horizontal eccentricity are not likely to be damaged by this level of earthquake (CNL 2018a).

2.1.3.5 Liquefaction

Liquefaction occurs when vibrations or water pressure within a mass of soil cause the soil particles to lose contact with one another. As a result, the soil behaves like a liquid, has an inability to support weight and can flow down even gentle slopes. This condition is usually temporary and is most often caused by an earthquake vibrating water-saturated fill or unconsolidated soil. Liquefaction is possible for cohesionless soils for earthquakes of magnitudes 4 to 6, which can produce ground shaking levels up to VIII on the Modified Mercalli Intensity. Liquefaction is therefore not deemed to be an issue for the WL site due to the aseismic conditions of Eastern Manitoba. The soil properties of the WL site also indicate that liquefaction would not be an issue. In general, the high plastic clayey overburden soils present at the site are not susceptible to cyclic liquefaction (KGS Group 2019).

2.1.3.6 Hydrogeology

Groundwater recharge primarily occurs in the local topographic high located approximately 3 km east of the WL main campus (where the geological conditions at surface allow for greater infiltration). Groundwater flow across the site follows topography and is predominantly east to west towards the Winnipeg River for all hydrostratigraphic units. At all locations, groundwater elevations were highest in the clay unit and lowest in the combined basal till/upper bedrock unit, indicating a downward direction of groundwater flow (Golder 2021).

The recharge and discharge locations along the groundwater flow path are controlled by climate, topography, and other hydraulic factors such as the variations in the permeability and/or thickness of the stratigraphic units. Horizontal groundwater flow is anticipated to be dominated by the more permeable basal till unit immediately above the bedrock, and the fractured bedrock zones.

One zone of recharge, two zones of discharge, and two transitional areas have been identified for the WL site. The central discharge area has been identified at the WMA, the lagoon area was observed to be in a recharge position, and the landfill is situated in a primarily recharge condition due to its proximity to the uplands recharge area. The recharge and discharge conditions across the WL site are largely dependent on the properties and flow conditions that occur in the basal (sand and till) units – such as unit thickness variability, lateral groundwater flow pattern, and hydraulic conductivity.

There is an upward component of groundwater flow from the basal sand unit into the overlying lacustrine clays in the WMA. At the WMA, the groundwater elevations at depth are nearly continuously greater than the elevation of the water table. Groundwater flows from depth toward the water table; therefore, the WMA is located in a groundwater discharge area.

The upward discharge of groundwater observed at the WMA was not observed at the WR-1 site in the 2015, 2016 or 2018 field program results. The data for the WL main campus suggest there is a horizontal component to groundwater flow through the basal till unit and shallow bedrock, westward towards the Winnipeg River. As well, there is evidence of a general downward component of groundwater flow through the overburden units. The upward flow in the WMA is attributed to the lower permeability and decreased dimension of the basal sand unit to the west of the WMA.

The lagoon area is between the WMA and the WL main campus to the north. Vertical downwards groundwater flow dominates over lateral flow at the water table at the lagoon. Lateral groundwater flow conditions are observed at both the water table and basal sand unit. Flow at the water table is radial from the lagoon cells and downwards to the deep zone, where it then moves laterally from east to west.

Site-wide groundwater flow conditions were compared by compiling water table elevations with data from September 2015, September 2016 and September 2018. Near WR-1, site-wide data were taken from September 2015, and compiled with 2016 water table conditions. Contours of water table elevations are presented on Figure 3-7 of the *WR-1 Hydrogeological Study Report* (Dillon 2018). The compiled data predominantly shows decreases from the northeast to the southwest across the WL site for the water table elevation, which was also observed in 2016. Site-wide groundwater elevation contours for the shallow bedrock were also generated using September 2016 and September 2018 groundwater level observations, as shown on Figure 3-8 of the *WR-1 Hydrogeological Study Report* (Dillon 2018). These compiled data also show decreasing groundwater elevations from northeast to southwest across the WL site as was observed in 2016.

Detailed evaluation of groundwater elevations in the WR-1 area was completed using the data set from the September 2016 monitoring event and updated data from September 2018. Flow conditions in 2018 were compared against the flow maps from 2016. In each of the clay and clay till units, horizontal groundwater flow is generally in a westerly direction, influenced by subsurface heterogeneity and vertically downward flow.

Across each stratigraphic unit, greater depths to water levels were noted at Nest 2, located generally upgradient from the WR-1, indicating that the groundwater elevations in the wells at this location may be influenced by activities at WR-1. This hydrogeologic condition is particularly evident in the basal till and shallow bedrock (Dillon 2018) for 2016 and 2018 conditions. A sump at the 200 Level in the WR-1 Building operates at an annual range of 8,800 to 13,400 m^3, implying typical continuous flow rates of between 0.27 to 0.42 L/s. The water collected by this sump comes from the weeping tiles and collects surface water moving downwards from the surface through the disturbed soils surrounding the WR-1 Building. The possible presence of features of higher hydraulic conductivity between the WR-1 Building and Nest 2 is suggested based on the influence on groundwater levels at Nest 2, although this was not particularly evident in the borehole logs.

Complex networks of fractures likely control groundwater flow in the shallow bedrock, and the distribution of the three monitoring wells in the shallow bedrock may not facilitate full resolution of groundwater flow patterns. During the last two sampling events (August and September 2018), the groundwater levels in bedrock wells were generally stable but lower compared to previous sampling events. Overall, the groundwater conditions in 2018 are consistent with the observations from 2016 monitoring data.

The gradients from 2016 generally agree with the calculated gradients from the 2018. In the 2018 data, there is a transient reversal in the gradient in Nest 2, but this may be attributed to the influence of seasonal effects or the WR-1 sump.

Single well response (packer) testing was completed in the upper portion of the bedrock at locations within the local study area (LSA; Dillon 2018). Interpreted hydraulic conductivity values ranged from less than 1E-7 metres per second (m/s) at two of the test locations to 2E-6 m/s at the third location. This is generally consistent with previous assessments completed within the RSA that have interpreted the upper bedrock to be within the range of 5E-12 m/s to 6.5E-7 m/s (Stevenson et al. 1996). Based on these data, the upper portion of the bedrock (in contact with the overburden) is the primary aquifer within the LSA.

The wells installed on the WL site were sampled for groundwater chemistry (Dillon 2018). Initial sampling showed sulphate levels ranging from 180 micrograms per litre (µg/L) to 2,500 mg/L. This is similar to values from the groundwater well testing performed at the WMA from 2007 to 2017. The values of sulphate reach levels that are known to promote sulphate attack on concrete. The calcium, magnesium and pH are also similar to the chemistry seen in groundwater wells at the WMA.

There is no notable radionuclide in the initial water samples taken on the WL site (Dillon 2018). Only one well located to the northwest of the WR-1 showed a slightly elevated alpha reading of 41 parts per billion. Slightly elevated alpha readings are expected, as local well waters within the Canadian Shield contain naturally occurring uranium; therefore, elevated alpha reading are not unexpected as they are consistent with measured background levels for the region.

2.1.4 Surface Water Environment

The Winnipeg River passes through the WL site, with a small portion of the WL site located on the west side of the river. The Winnipeg River is the dominant hydrological feature of the area. The Winnipeg River flows from the Lake of the Woods and the English River system of Northwestern Ontario and drains to Lake Winnipeg located northwest of the WL site (Figure 1.0-1). The Winnipeg River is classified as a medium-sized lowland river (AECL 2001a). The total drainage basin of the Winnipeg River is approximately 15,000,000 hectares (ha).

A decrease in ground elevation of 83 m from the Manitoba/Ontario border to Lake Winnipeg through a series of falls and rapids has resulted in extensive hydroelectric exploitation of this river. Six electric generating stations are present on the Winnipeg River, whose discharge rate is now largely controlled by these hydroelectric dams, which precludes any short-term correlation between precipitation and river flow. The two most relevant stations are the Seven Sisters Generating Station, which is approximately 7.5 km upstream of the WL site and McArthur Generating Station, which is approximately 26 km downstream.

Near the WL site, the river is approximately 300 m wide and flows in a northerly direction at a velocity of approximately 0.3 m/s. Flow rates measured at the nearby Seven Sisters Falls Hydroelectric Generating Station (approximately 7.5 km upstream of the WL site) typically vary between 600 cubic metres per second (m^3/s) to 1,800 m^3/s, with a record low at 125 m^3/s and as high as 2,800 m^3/s (CNL 2016a).

The flow of the river is usually lowest in late summer/early fall and highest in late spring/early summer. This follows typical temporal river patterns with water levels affected by spring snow melt and large precipitation events in spring and fall. The dams on the Winnipeg River provide some storage capacity, preventing extreme flooding and extreme low water levels. The stage-discharge for the Winnipeg River adjacent to the WL site demonstrates that there is a relationship between flow and water level at higher flow and water levels, but this changes when flow is below 1,200 m^3/s and elevation is below 255 m. As there are no extreme low water levels at low flows, it is inferred that the downstream dam, McArthur, is likely influencing water elevation in low flow conditions.

Historically, liquid effluent from the Active Liquid Waste Treatment Center (ALWTC) was discharged to the Winnipeg River via the Process Sewer at the sewer outfall located approximately 8 m offshore in approximately 5 m of water (AECL 2001a). The ALWTC started demolition in 2021 and liquid effluent is now treated at the source in equivalent systems prior to discharge to the Winnipeg River. The Winnipeg River Task Force (1995) looked at potential sources of Winnipeg River water quality degradation near the community of Sagkeeng. The Task Force found that WL has not had an adverse effect on water quality in the Winnipeg River for downstream communities (AECL 2001a).

The concentration of the dominant radionuclide associated with the discharge, caesium-137, in downstream river water was an average of 0.005 becquerels per litre (Bq/L) in 2015 compared to the Canadian Drinking Water Quality Guideline of 10 Bq/L. The values for caesium-137 at the WL site are similar to upstream values and lower downstream values, suggesting no effect on caesium-137 from WL operations in 2017. The upstream and downstream activities of strontium-90 were also similar, suggesting no effect from releases from WL site. The tritium activity detected upstream of the WL site was similar to the downstream samples and values are below the Maximum Acceptable Concentration in drinking water.

During the sediment investigation conducted in 2000, the sediment downstream of the outfall was determined to be erosional, with a clay bottom covered in parts with sand, gravel, cobble, and boulders (AECL 2001b). The investigation also indicated that a small area adjacent to the outfall had elevated radionuclide levels, but these levels were below the threshold that would cause adverse effects to human or ecological health. Therefore, during the discharge of the 1990s and 2000s, the accumulation of radionuclides in sediment had negligible environmental effects.

2.1.5 Aquatic Environment

The principal aquatic habitat near the WL site is the Winnipeg River. The Winnipeg River passes through the WL site, approximately 500 m the west of the WR-1 Building. In addition, there are several small, isolated ponds on the WL site that are fed by local runoff and intermittent streams that flow primarily during the spring (AECL 2001a). The streams are associated with gullies that dissect the clay plains along the banks of the Winnipeg River. Beaver (*Castor canadensis*) ponds are a common feature on the WL site; however, ponds rarely persist for more than a few years, affording limited quality aquatic habitat. There is also a sewage lagoon, situated north of the main plant site that harbours aquatic plants and animals, as well as man-made ditches that convey water during spring runoff, but are generally dry in summer.

The Winnipeg River supports a diverse fish community and affords spawning, rearing and foraging habitats. A total of 61 native fish species are reported for the river (Stewart and Watkinson 2004) and two species at risk are known to be present in the river within the vicinity of the WL site: Lake Sturgeon (*Acipenser fulvescens*) and Carmine Shiner (*Notropis percobromus*; COSEWIC 2019). Fish occurring within the study reach include species that are primarily resident (present year-round), as well as migratory (passing through). Fish habitat is generally similar throughout the area.

The benthic macroinvertebrate community within a riverine system is mostly comprised of both infauna and epifauna. Typically, infauna are burrowing taxa that live in the sediment, whereas epifauna live on the sediment surface. Both types of macroinvertebrates are important food sources for fish. Benthic fauna included protozoa, ostracods, nematodes, oligochaetes, leeches, mysids (opossum shrimp), crayfish, amphipods, mollusks (snails), bivalve clams (e.g., mussels) and immature stages of aquatic insects. The latter include dipteran larvae such as chironomids and *Chaoborus*, dragonflies nymphs, mayflies nymphs, caddisflies nymphs, true bugs, and aquatic beetles (AECL 2001a).

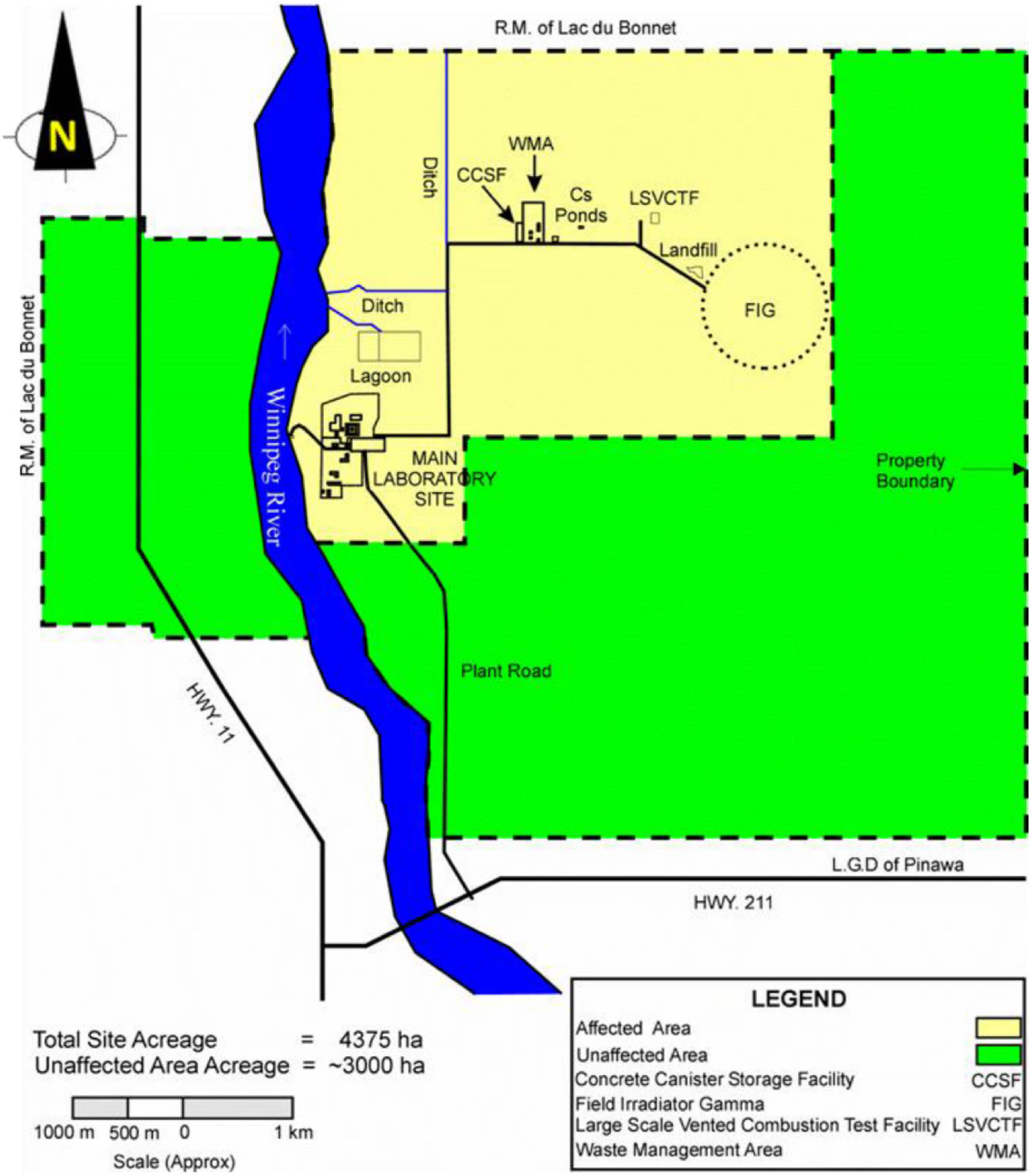
Benthic invertebrate studies were undertaken on the Winnipeg River in the vicinity of the site by AECL (1973). The abundance and total beta activity of benthic organisms were determined during 1966 and 1967 upstream and downstream of the liquid effluent outfall. Tubificid worms, chironomid larvae, and mayfly (*Hexagenia* spp.) nymphs were the most abundant species among a total of 20 benthic taxa collected. The radioactivity concentration ratios (total beta activity of organism/total beta activity of water) of these species were approximately 2. The range of individual diversity indices (d) obtained during the study was 2.45 to 2.94, indicative of mesotrophic conditions.

Aquatic plants such as bulrushes, cattails and wild rice have been identified to a depth of about 1 m along the shores of the Winnipeg River. The Winnipeg River also supports a diverse assemblage of algae (phytoplankton) (AECL 2001a).

CNL undertakes annual monitoring of radioactivity in fish flesh (muscle and tissue) collected from the Winnipeg River upstream and downstream of the WL site (CNL 2016b). Average concentrations of caesium-137, potassium-40 and gross beta activity found in fish flesh were slightly greater for Walleye at downstream locations compared to upstream (CNL 2018b). However, for the White Sucker, on average radionuclide levels were slightly lower downstream of the WL site compared to upstream. Overall, no significant differences or trends in radionuclide concentrations were apparent among areas over the last five years. In 2017, the total incremental dose due to fish ingestion was 4.82E-05 millisieverts per year (mSv/a) for adults (CNL 2018b). This was equivalent to about 0.00482% of the annual regulatory limit of 1 mSv/a for members of the public.

2.1.6 Terrestrial Environment

The WL site is approximately 4,375 ha and is segregated into Affected and Unaffected Areas (Figure 2.1.6-1). Affected land is defined as areas where nuclear development, operations or supporting activities are conducted and includes land potentially affected by such activities (AECL 2001a). The balance of the WL site (approximately 3,000 ha) is designated as the Unaffected Area. The Unaffected Area contains land that has no radionuclide history and is not affected by site’s nuclear operations. This was confirmed by a radionuclide verification survey performed during the summer of 2000 (AECL 2001a).



CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
WHITESHELL LABORATORIES AFFECTED AND UNAFFECTED AREAS

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.1.6-1**

REFERENCE(S)

1. OBTAINED FROM CANADIAN NUCLEAR LABORATORIES, WHITESHELL LABORATORIES DETAILED DECOMMISSIONING PLAN: VOLUME 6 -WHITESHELL REACTOR #1: BUILDING 100, FIGURE 2-2 WL AFFECTED AND UNAFFECTED AREAS, 2016

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There is a relatively high degree of diversity in terrestrial habitat within the WL site. The WL site is primarily under treed cover, consisting of a mixture of wetlands and forests of broadleaf, mixed and coniferous stand types. A large area contains a complex of bog, fen and swamp wetlands spanning the center and east portions of the WL site, from north to south. Black spruce dominates large portions of this wetland habitat with understories of tamarack (*Larix* sp.), willow sp. (*Salix* sp.), blueberry (*Vaccinium* sp.), common Labrador tea (*Rhododendron groenlandicum*), horsetail sp. (*Equisetum* sp.) and mosses. A ridge of well-drained sandy soil within the wetland area contains jack pine dominated forest stands. To the west of the wetland area, forests dominated by black ash and poplar sp., are present on poorly drained clay plains (AECL 2001a). There are also areas of unutilized farm fields and areas where vegetation height is controlled, that contain a mixture of regenerating grasses and shrub species. There is a broadleaf forest dominated by trembling aspen on the west half of the WL site, with gullies and ravines adjacent to the Winnipeg River.

White-tailed deer (*Odocoileus virginianus*) are reportedly common, with established wintering on-site areas, and beaver (*Castor canadensis*) activity is common within the gullies adjacent to the Winnipeg River (AECL 2001a). The white-tailed deer is an important species for traditional use by Indigenous peoples and recreational game species in the area (AECL 2001a). Other commonly observed mammal species within the terrestrial environment RSA[5] include black bear, red fox, and groundhog (also known as woodchuck; *Marmota monax*). Commonly observed amphibian and reptile species in the area include northern leopard frog (*Lithobates pipiens*) and gartersnake (*Thamnophis* sp.), and there have also been observations of snapping turtle and western painted turtle (*Chrysemys picta bellii*).

Within the WL site, the potential for hibernacula in exposed bedrock that typically forms caves (i.e., karst topography with limestone, dolomite and gypsum-containing minerals) was coarsely assessed as low because these mineral types are not present. There is also no exposed bedrock within the RSA. A bat survey conducted in 2015 at the WL site indicated that bats were not roosting within buildings at the site, but rather can be found roosting in the forested areas of the site.

Bird migratory staging areas are present on and near the WL site. The Winnipeg river is an important migratory corridor for many bird species including: common loon (*Gavia immer*), red-necked grebe (*Podiceps grisengena*), horned grebe (*Podiceps auritus*), double-crested cormorant (*Phalacrocorax auritus*), American white pelican (*Pelecanus erythrorhynchos*), Bonaparte’s gull (*Chroicocephalus philadelphia*), common tern (*Sterna hirundo*), Caspian tern (*Hydroprogne caspia*), lesser scaup (*Aythya affinis*), greater scaup (*Aythya marila*) and bald eagle (*Haliaeetus leucocephalus*). Observations of federal species at risk have included trumpeter swan (*Cygnus buccinator*), barn swallow, loggerhead shrike (*Lanius ludovicianus*), and red-headed woodpecker (*Melanerpes erythrocephalus*). Automated recording unit surveys in 2018 confirmed the presence of Canada warbler (*Cardellina canadensis*) in a recording unit on the east side of the LSA.

2.1.7 Human and Ecological Health

Background radiation (i.e., ambient radioactivity) is present in the environment due to natural and anthropogenic sources independent of WL operations, including air, soil, food, water, aquatic sediments and plant or animal tissue. The background radiation dose varies greatly, both spatially and temporally. The main natural sources of radiation are cosmic rays; naturally occurring radionuclides in air, water and food; and naturally occurring radionuclides in the soil, rocks and building materials used in homes (CNSC 2013). Naturally-occurring radionuclides such as uranium, potassium and thorium are present in soils, rocks and building materials.

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[5] The terrestrial environment RSA is the 3,710 ha portion of the WL site on the east side of the Winnipeg River.

These naturally occurring radionuclides also contribute to the background gamma radiation dose. The average background radiation dose reported for Winnipeg is 4.1 mSv/a (CNSC 2013). Naturally-occurring radionuclides are also in plants, animals and water from surrounding soils and rocks. Humans ingest these foodstuffs and receive an internal radiation dose. Radon gas, a product of the decay of uranium in soil, is inhaled and contributes to the internal radiation dose.

CNL reports the results of the Environmental Monitoring Program for the WL site each year to the CNSC. The Environmental Monitoring Program data are collected to verify that radiation doses to members of the public as a result of the operations of the WL site meet the principle of ALARA. The 2019 dose assessment showed the radiation dose to the public from WL operations was 2.30E-06 mSv/a for airborne effluent and 8.70E-05 mSv/a for liquid effluent (CNL 2020a). The 2019 total adult dose of 8.93E-05 mSv/a represents 0.009% of the effective dose limit of 1 mSv/a for members of the public. The 5-year average adult dose from 2015 to 2019 was 2.8E-06 mSv/a for airborne effluent and 5.8E-05 mSv/a for liquid effluent. The 5-year average total adult dose of 6.0E-05 mSv/a represents 0.006% of the effective dose limit. The dose to members of the public is predominantly from liquid effluents.

The five-year average adult dose due to fish ingestion from 2014 to 2018 was 4.4E-05 mSv/a (CNL 2020a). In 2019, the estimated adult dose due to fish ingestion was 6.9E-06 mSv/a for adults (CNL 2020a). This was equivalent to approximately 0.0007% of the annual regulatory limit of 1 mSv/a for members of the public.

2.1.8 Land and Resource Use

The site occupies 4,375 ha of land that was either privately owned or Crown land before the establishment of WL. Currently, AECL, a federal government crown corporation, holds the title to the WL site lands. Access to these areas has generally been restricted for security and safety.

The WL site occupies four different land use designations in the Local Government District of Pinawa (LGD of Pinawa 2004) including general agriculture, heavy industrial, recreational/commercial, and natural area. The WL site is located near several wildilfe management and protected areas, including Pinawa Dam Provincial Park, Whitemouth Falls Provincial Park, Whiteshell Provincial Park, Pinawa Provincial Park, Lewis Bog Ecological Reserve, and Lee River Wildilfe Management Area. Other land interests include trapping and outfitting concessions, mineral and quarry reserves, agricultural lands and forestry.

The WL site is in eastern Manitoba west of the Whiteshell Provincial Park on the Winnipeg River. The region is popular for cottagers and campers as it is the gateway for access to a number of Provincial Parks, including Old Pinawa Dam Provincial Park, Nopiming Provincial Park and the Whiteshell Provincial Park. The Winnipeg River, including the stretch from the WL site downstream to the Town of Lac du Bonnet is lined with residential developments, which include year-round and seasonal residences.

Outfitters are located in and around Pinawa, the Town of Lac du Bonnet, Point Du Bois, and Eriksdale. These outfitters provide black bear, whitetail deer, wolf, and waterfowl hunts, while others offer local trophy fishing trips and fly-in fishing expeditions. The WL site is located in Game Hunting Area 26 and white-tailed deer hunting in the area around the WL site, especially northwest of the Project, is popular. Upland game birds and migratory waterfowl are also hunted, with a population of rough grouse and sharptail grouse noted in proximity to the site. Hunting within the WL site is prohibited due to safety concerns.

There are no registered traplines (RTL) in the WL site. RTL 23 runs along the southern border of the site. The holder of the RTL is active and traps a variety of species, including marten, red fox, otter, fisher weasel, muskrat and mink. There are no other RTLs that share a boundary with the WL site. The majority of RTLs are east of the WL site in the Whiteshell Provincial Park.

There are quarry permits to the northeast and southwest of the WL site and several quarry withdrawals in the area, including near the northeast boundary of the WL site. There are no other mineral claims or leases in proximity of the WL site.

Land to the north and west of the WL site has been zoned by the Regional Municipality of Lac du Bonnet for agricultural purposes. The zoning designations allow a full range of agricultural activities, however, only existing livestock operations can be expanded (RM of Lac du Bonnet 2019). Other areas west of the Winnipeg River have been designated for mixed use, including residential and rural uses. To the south and west of the WL site, in the Regional Municipality of Lac du Bonnet is the Agassiz Provincial Forest. Forestry has historically been important to the region, but there has traditionally been little forestry activity near the WL site (AECL 2001a).

A review of the registered archaeological site data provided by the Historic Resource Branch shows the presence of 434 sites within 5 km of the Winnipeg River spanning from the Manitoba/Ontario border to the mouth at Lake Winnipeg. While there are no documented archaeological sites within the WL site, it should not be assumed that sites are not present, as the Sieg Serpent Site (formally called Sweet Creek Petroform) is located approximately 1,700 m south of the property (AECL 2001a). One provincially recognized historic site, the Pinawa Dam Provincial Heritage Park, is located on the Pinawa Channel/Lee River, within 5 km of the Winnipeg River.

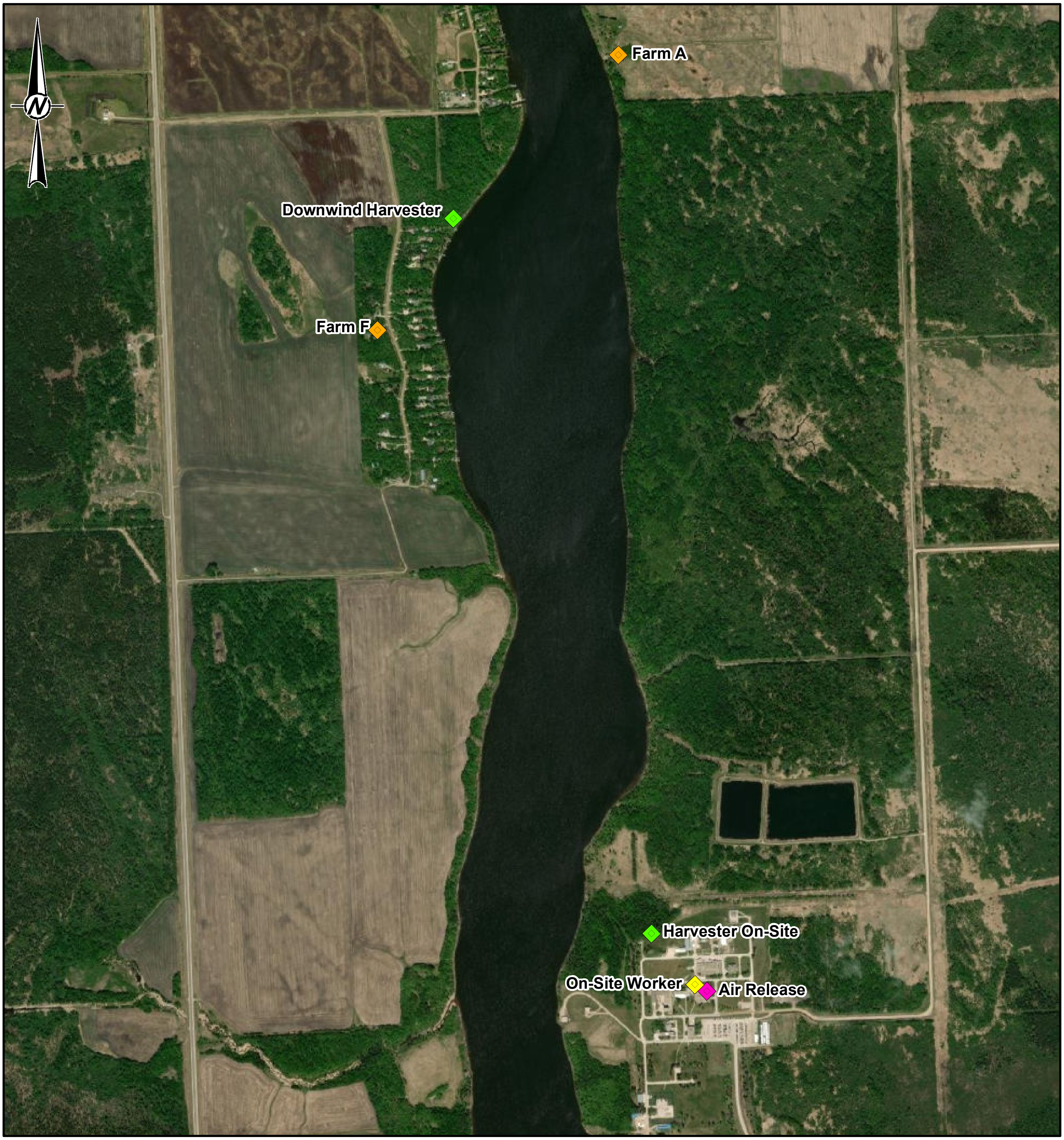
The Winnipeg River has historically provided a network of travel routes for local communities (Petch 2005). Present day traditional land and resource use activities, including trapping and wild rice harvesting, continue to supplement the diet and income of local Indigenous communities in the area. Wild rice harvesting was originally for domestic purposes but eventually became a commercial economy.

2.1.9 Socio-economic Environment

The nearest permanent residents are approximately 2 km from the Project and are located along the Winnipeg River north (Farm A) and northwest (Farm F) of the WL site (Figure 2.1.9-1). The nearest population centres are the Village of Lac Du Bonnet and the Local Government District of Pinawa. Lac Du Bonnet has a population of approximately 1,100 and is about 9 km north of the WL site. Pinawa has a population of approximately 1,400 and is about 10 km east of the WL site. The City of Winnipeg is approximately 100 km to the southeast of the WL site.

The Project is located on Treaty 3 land, while the overall WL site extends west of the Winnipeg River into Treaty 1 land. Communities that form part of these Treaties and Treaty 5 have historical and current traditional land uses with the area. The Project is also located in the homeland of the Manitoba Métis Nation, as represented by the Manitoba Métis Federation.

Three Indigenous communities are in proximity to the WL site. The Ojibway community of Sagkeeng First Nation (also known as Fort Alexander, Manitoba) is located on the shore of the Winnipeg River at Lake Winnipeg approximately 50 km northwest of the WL site. The Little Black River First Nation is located approximately 60 km northwest of the WL site, in proximity to Sagkeeng First Nation. The Brokenhead Ojibway Nation (also known as Scanterbury, Manitoba) is located along the shore of the Brokenhead River approximately 50 km west of the WL site. All three Indigenous communities include people that identify as Métis; however, no people identifying as Inuit were recorded in these communities.



REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
HUMAN RECEPTOR LOCATIONS WITHIN THE VICINITY OF THE WHITESHELL LABORATORIES DURING CLOSURE PHASE

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.1.9-1**

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2.2 Site Description

The site description provides the information required to understand the quality and quantity of data available to characterize the site, sources of exposure, exposure pathways, and human and ecological receptors.

2.2.1 Site Overview and Historical Context

The WL site was established in the 1960s by AECL to conduct nuclear research to demonstrate the organic-cooled reactor concept using heavy water (D_2O) as the moderator. The site originally included WR-1, an organic-cooled reactor, which was brought on-line in 1965. The organic-cooled reactor program was eliminated in the early 1970s to focus on the heavy water-cooled Canadian Deuterium Uranium (CANDU) reactor system. Development of programs including the Nuclear Fuel Waste Management Program, SLOWPOKE Demonstration Reactor, CANDU Reactor Safety research projects and accelerator projects, maintained WL as a diverse centre for nuclear research. Many other support facilities were required over the years to support the research programs. These included the WMA, the Concrete Canister Storage Facilities (CCSF), and ALWTC in 1963; Hot Cell Facilities (HCF) in 1965; the Immobilized Fuel Test Facility (IFTF) in 1984; the Van de Graaff Accelerator in 1970 (upgraded in 1979); and the Neutron Generator Facility in 1975.

In the mid-1990s, research programs at WL were discontinued as a result of the federal program review process that reduced funding to nuclear research. The federal government examined various alternatives for the WL site and recommended privatization. Attempts to attract a private owner to take over the facility were unsuccessful. Subsequently, the Government of Canada made the decision in 1998 to close the WL site.

Certain operations at the site are presently in various stages of operational shutdown, or decommissioned. Experimental work expected for processing of active liquid wastes was concluded in the Shielded Facilities cleanup, and removal of research equipment has also been completed. Both the Neutron Generator Facility and the Van de Graaff Accelerator have been decommissioned.

2.2.2 Site Facilities

The decommissioning plan for the WL site encompasses its facilities, buildings and land. The decommissioning of all the site facilities is addressed by the CSR (AECL 2001a) and the existing Decommissioning Licence (NRTEDL W5 8.00/2024). As previously mentioned, some of the WL site facilities have already been decommissioned. Section 2.2.2 Site Facilities provides a summary of the status of the remaining facilities that will be decommissioned under the existing Decommissioning Licence.

2.2.2.1 Nuclear Facilities

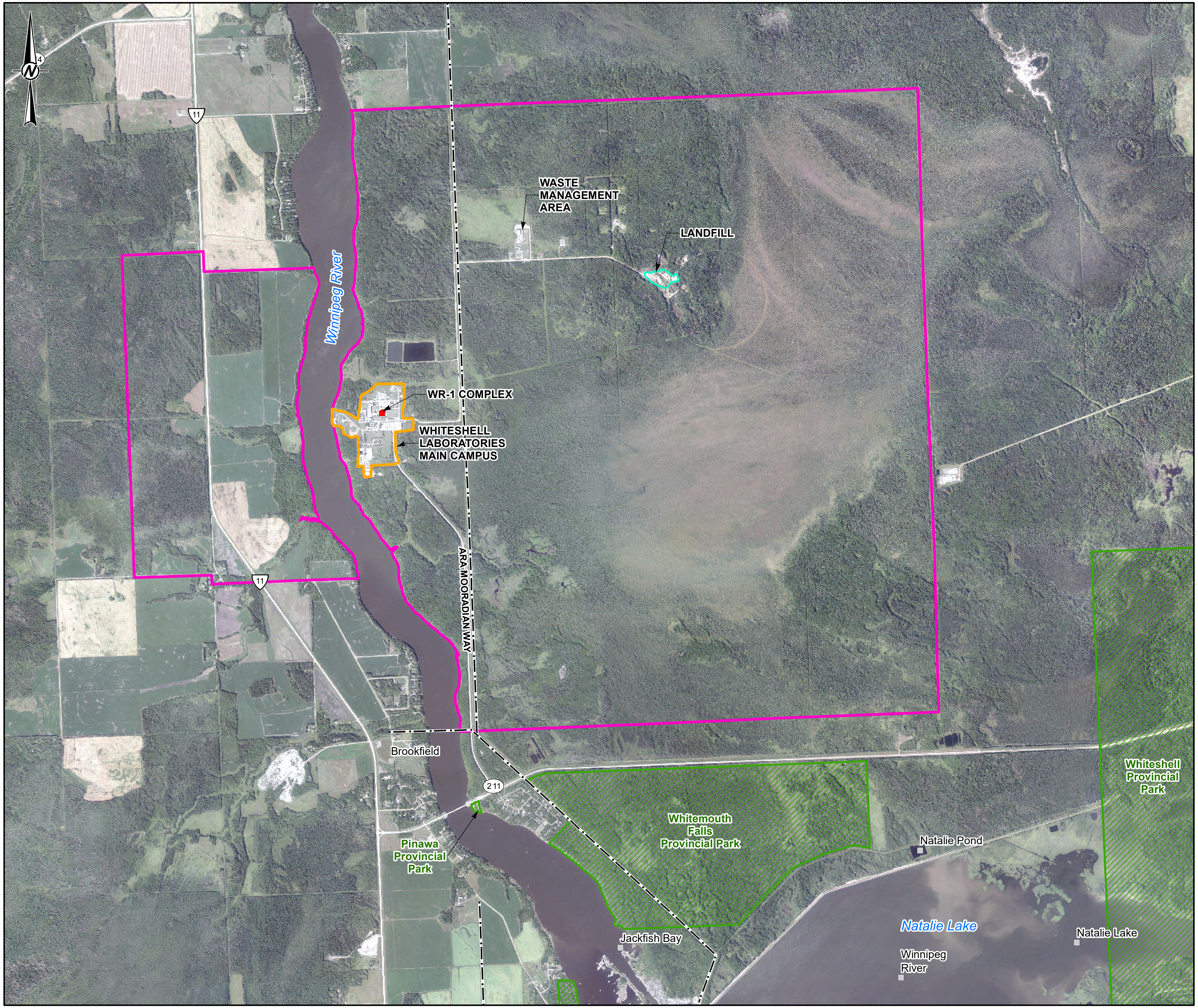
2.2.2.1.1 Whiteshell Reactor 1 Complex and Building

The WR-1 Complex is located entirely within the WL site (Figure 2.2.2-1 and Figure 2.2.2-2), and includes the WR-1 Building extending two levels above-grade (Levels 600 and 700) and five levels below-grade (Levels 100 to 500) (Figure 2.2.2-3). The east and service wings house office space and supporting facilities. Major components of the WR-1 Building include the reactor vessel, shielding and experimental loops, which are described in more detail in Section 2.3.1 WR-1 Building System Description.

All reactor systems are below-grade, except for the primary coolant pumps and heat exchangers, which are contained in a shielded room on the reactor hall floor, and the primary intake fans. Access to the four lower levels of the WR-1 Building is restricted, while the upper three levels provide office and laboratory space. A portion of Level 500 also has restricted access.

The reactor core spans Level 300 and Level 400, with lower access on Level 200 and upper access on Level 500. The heavy water moderator systems are located in the lowest portion of the structure on Level 100. The Primary Heat Transport (PHT) system extends from the lower reactor vessel access areas on Level 200 up to the Primary Loop Room (Figure 2.2.2-4).

The reactor was permanently shut down in 1985 and placed in a secure shutdown state in preparation for decommissioning. The preliminary decommissioning activities included defueling the reactor, transferring all irradiated fuel store in the fuel storage bays to storage in the CCSF, and removing heavy water and transferring to Chalk River Laboratories for storage. Organic coolant was drained from the reactor cooling circuits and transferred to the WMA for processing. The system was not flushed; therefore, it is known that some organic coolant is present within the system. Reactor control systems were isolated and removed. Building services systems are still maintained in operating mode. This preliminary decommissioning work decreased potential hazards from the facility, reducing the monitoring and surveillance requirements for a deferment period.



LEGEND

WHITESHELL LABORATORIES

- 🟥 WR-1 COMPLEX
- □ WHITESHELL LABORATORIES SITE
- □ WASTE MANAGEMENT AREA
- □ WHITESHELL LABORATORIES MAIN CAMPUS
- □ LANDFILL

BASE FEATURES

- ■ CITY/TOWN
- -·- TRANSMISSION LINE
- ▨ PROVINCIAL PARK

0 1,000 2,000

1:40,000 METRES

NOTE(S)

REFERENCE(S)

1. BASE DATA - CANVEC AND MLI, 2016
2. PROJECT DATA - CNL, 2016
3. PROJECTION: TRANSVERSE MERCATOR DATUM: NAD 83 COORDINATE SYSTEM: UTM ZONE14N

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
LOCATION OF THE WR-1 COMPLEX

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	CGE
PREPARED	CGE/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	2.2.2-1

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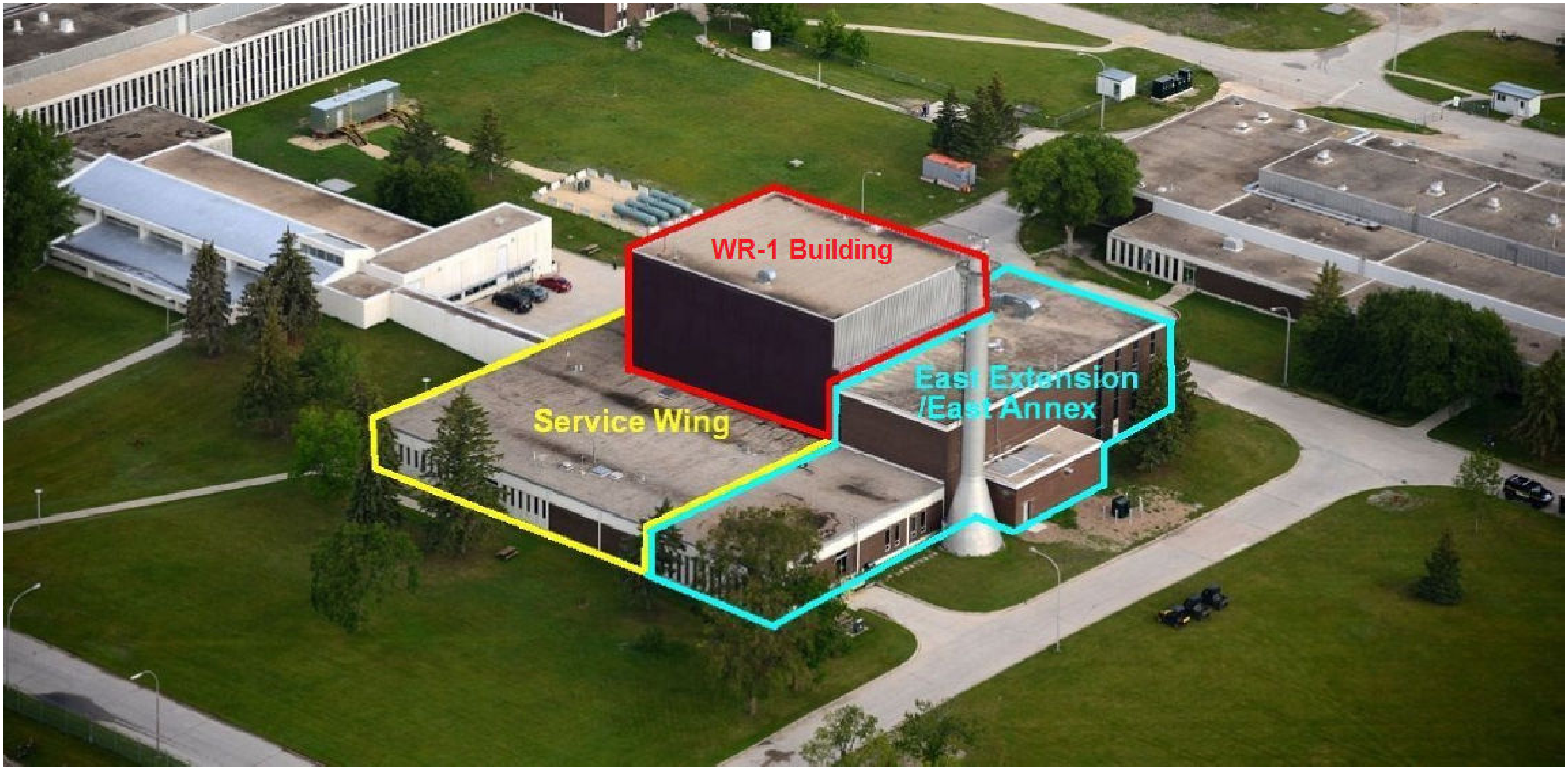

NOTE(S)

REFERENCE(S)
1. OBTAINED FROM CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	-
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

TITLE
WR-1 COMPLEX

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.2.2-2**

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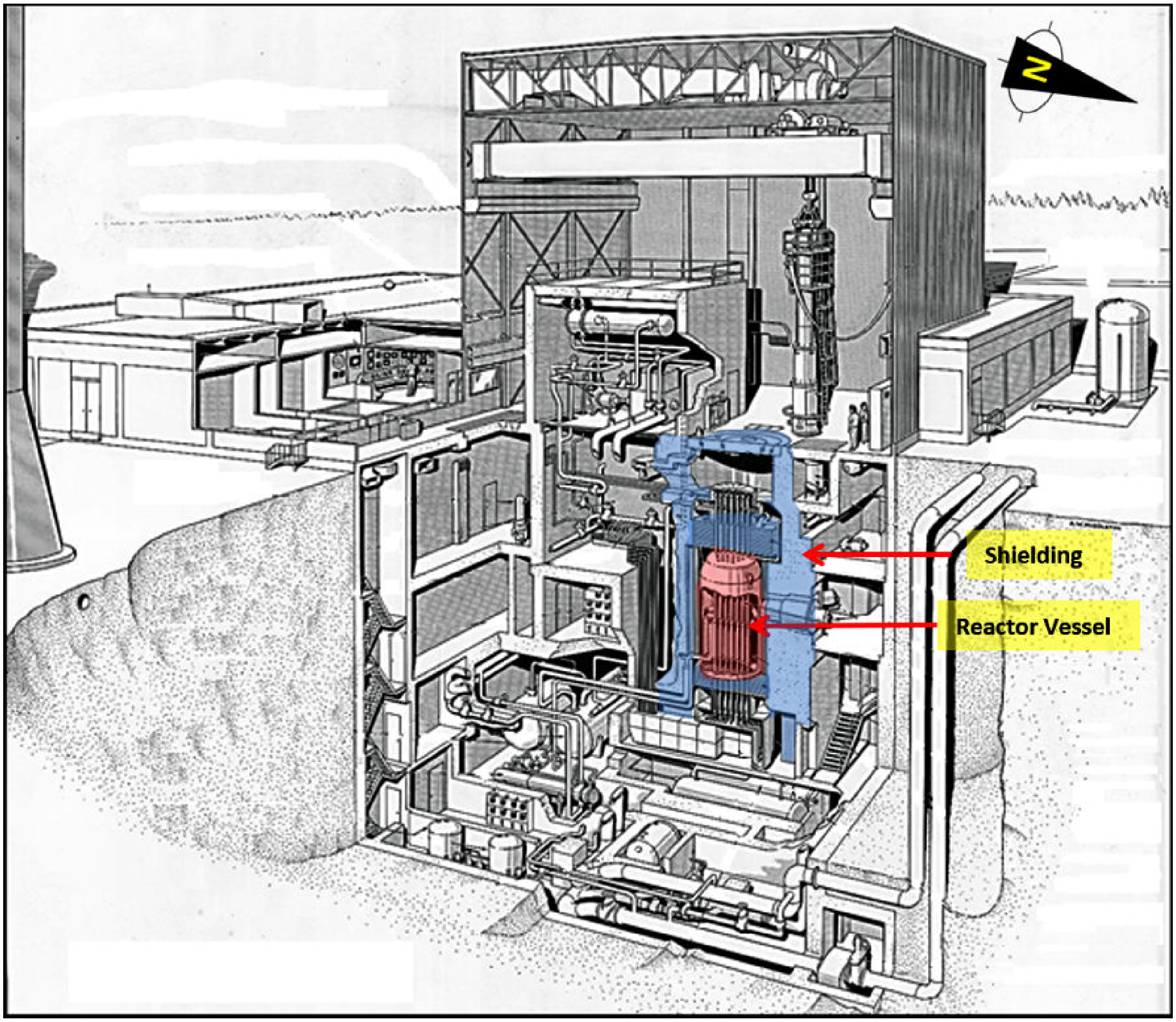

REFERENCE(S)
1. CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
WR-1 BUILDING - REACTOR VESSEL AND SHIELDING

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.2.2-3**

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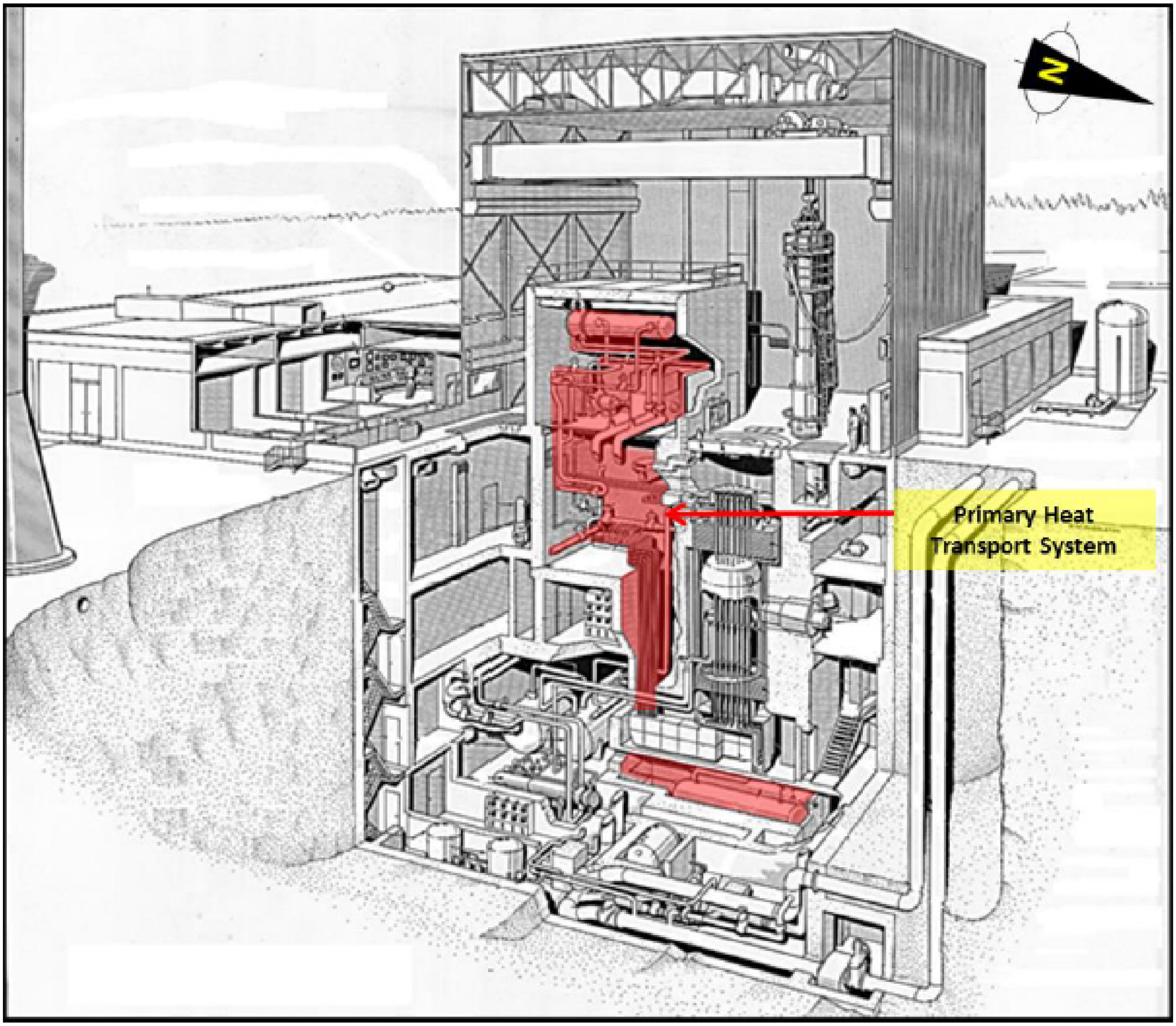

REFERENCE(S)

1. CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

WR-1 BUILDING – PRIMARY HEAT TRANSPORT SYSTEM

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.2.2-4**

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2.2.2.1.2 Reactor Building and Shielded Facilities

The Shielded Facilities include the HCF and the IFTF, both of which form the west extension of the Research and Development Building (B300). The HCF began operation in 1965 and was used to provide shielded, remote handling facilities in support of the CANDU Reactor Safety research programs. The IFTF began operation in 1984 and was used to provide space and facilities for a wide range of experiments using radioactive materials in support of the Canadian Nuclear Fuel Waste Management and CANDU Reactor Safety research programs.

The ALWTC, which is located in Building 200 (B200), began operation in 1963, receiving low level liquid waste effluent from operating nuclear facilities (WR-1, Shielded Facilities, B300 research laboratories, Building 411 laundry/decontamination). The liquid effluents were transferred via underground piping connecting existing facilities to the ALWTC. The ALWTC included a now out of service intermediate level liquid waste processing system that concentrated the waste stream originating from the Shielded Facilities. The resulting concentrate was solidified and stored at the WMA. The ALWTC started demolition in 2021 and liquid effluent is now treated at the source in equivalent systems prior to discharge to the Winnipeg River.

2.2.2.1.3 Concrete Canister Storage Facility

The Concrete Canister Fuel Storage Program was developed at WL to demonstrate that dry storage is a feasible alternative to water pool storage for irradiated reactor fuel. Due to the success of the demonstration program, concrete canisters have been used at WL since 1975 to store irradiated fuel; there are currently 16 canisters in use. Used fuel is stored in several types of steel baskets in the canisters at the CCSF. Used fuel includes WR-1 fuel bundles, CANDU power reactor bundles and fuel fragments stored in Element Storage Cans from various sources. The CCSF is composed of two storage areas: (1) the main canister site adjacent to the WMA, and (2) the demonstration canister site within the WL main campus.

2.2.2.1.4 Waste Management Area

The WMA is approximately 148 m by 312 m and has been in operation since 1963. Radioactive wastes are categorized into the following three levels according to AECL procedures defined in Barnard et al. (1985):

- Low Level Radioactive Waste (LLW), which consists of contaminated materials containing a mixture of activation and fission products, includes used lab-ware, gloves, shoe covers, wipe paper, and mops. As of the end of 2015, the WMA contained a total volume of 20,647 cubic metres (m^3) of LLW. This LLW is stored in Trenches 1 to 23, LLW Bunkers 1 to 6, Buildings 431 to 433, and Building 923 and Soil Storage Compound. The total accumulated LLW activity is estimated to be 333 terabecquerel (TBq) (CNL 2015a, Table 22.1).
- Intermediate Level Waste (ILW), which is typically composed of scrap metal materials from experiments, filters, and radioactive liquid waste that has been solidified. This waste is stored in the standpipes and in the ILW bunkers in the WMA with High Level Waste (HLW) and hazardous chemicals. The total accumulation of ILW in the WMA is approximately 1,400 m^3. It is estimated that approximately 3 metric tonnes (t) of HLW, primarily irradiated reactor fuel, is stored with the ILW, both in standpipes and in the ILW bunkers (AECL 2001a, Table 4.6).
- High Level Waste (HLW), consisting of irradiated reactor fuel, is stored in the CCSF. The CCSF provides storage for 25 metric tonnes (t) of irradiated reactor fuel (AECL 2001a).

Surface water and groundwater monitoring are conducted in relation to the WMA. Surface drainage water samples would provide the first indication of any abnormal activity levels attributed to the WMA or the CCSF. The frequency of surface water sampling is controlled by the amount of spring runoff and the amount of rainfall throughout the spring-to-fall period. Refer to Section 2.2.3.2 Off-site Contamination, for details on surface water sampling. Groundwater samples are also collected in the spring and the fall from wells located around the WMA.

2.2.2.2 Radioisotope Facilities

2.2.2.2.1 Building 300

Building 300 (B300) was the primary research laboratory for the WL site, which provided support to the full range of nuclear research and development programs conducted at WL. Most of the building was used to provide general laboratory work areas and contained 68 laboratories designed to handle various levels of radioactivity. The south end of the building is a high bay area that supports experimental activities requiring large areas and significant head room. The RD-14M experimental loop is located in the south high bay. Research program work remains in progress in RD-14M utilized by the Reactor and Applied Sciences division. The laundry services and testing equipment that used to be housed in the Decontamination Centre (B411) have been relocated to the B300 complex. Although part of the B300 complex, the Shielded Facilities are listed as a separate facility.

2.2.2.2.2 Decontamination Centre

The Decontamination Centre (B411) is currently being decommissioned. It provided a decontamination service for maintaining research and development experimental rigs, equipment, and tools. It also provided a laundry service for radiologically contaminated clothing. The decontamination area contained eight fume hoods and the work area was designed to accommodate a broad range of contaminated equipment cleanup. The laundry contained four fume hoods to accommodate sorting of contaminated clothing, and laundry equipment consisted of six industrial washing machines and four dryers.

2.2.2.3 Buried Services

Buried services run through the entire site and include drainage, district heating, electrical, fire and process water, and domestic water systems. The most significant buried services from a decommissioning perspective are the three types of drainage systems:

- Sanitary drains, which collect wastewater from toilets, showers, and the sinks, and discharge the wastewater to the site sewage lagoon.
- Low-level aqueous radioactive waste collection drains, which collect wastewater containing radiological and non-radiological contaminants. The wastewater was pumped through double walled pipes to tanks in the ALWTC. The low-level tank waste was sampled and if radioactivity levels were acceptably low, the wastewater was pumped to the process drain/storm sewer at a maximum rate of 8 L/s. Any leaks in the active lines would be contained within the outer wall of the transfer pipes and flow to leak collection points (manholes or sumps) located at low points along the route.
- Storm drains, which collect cooling water from experimental facilities, site runoff water, low-level radioactive liquid waste following sampling and monitoring, inactive effluent from non-active building sump floor drains and laboratory sinks, and process water that is used to maintain a minimum flow at the outfall for a flow measurement. The storm drain water is discharged via the outfall to the Winnipeg River.

The aqueous radioactive waste collection system was replaced by the currently in-service existing double pipe system in the mid-1980s. The old system had failed and leaks from some lines adjacent to the ALWTC had occurred. The area was partially remediated through removal of excavated soil; however, in subsequent years, the vegetation in the spill area was found to have elevated levels of beta and gamma emitting radioactivity, in particular caesium-137 and strontium-90. Routine monitoring of the area is maintained to provide an indication of mobility that would require early remediation.

2.2.3 Contaminated Lands

2.2.3.1 On-site Contamination

The affected lands are those within the WL site that are contaminated, potentially contaminated or affected by nuclear operation and are more than 1 m away from buildings. Decommissioning of land within 1 m of the buildings is considered part of the decommissioning of the building. The affected lands may contain contamination because of proximity to facilities and unusual occurrences. The primary area of concern is the ALWTC.

Surface contamination attributed to release from the HCF exhaust stack was detected in 1971 and 1972. The releases were at low-levels and there is no detectable contamination remaining in these areas. Leakage to topsoil has occurred as a result of active drain line failures, particularly in the ALWTC area, where three incidents released about 65 gigabecquerel (GBq) of mixed fission product contamination. About 9 GBq of caesium-137 and strontium-90 are estimated to remain in the ground and the area is routinely monitored. There is no indication of contamination movement from the area.

The caesium ponds were located directly east of the WMA. The ponds were developed to study the distribution of dose received by organisms living at the water-mud interface. In the 1960s, 0.5 curie (Ci) of caesium-137 was injected into the area. These ponds have been remediated and the contaminated soil removed.

2.2.3.2 Off-site Contamination

Off-site contamination from the WL operation has occurred in two areas. Routine releases (well within regulatory limits) and spill incidents (1977 organic coolant release) have resulted in localized contamination of the Winnipeg River sediment. The north property ditch and the natural drainage creek (North Ditch/Creek) northwest of the AECL site boundary was contaminated as the result of a human error (inadvertent pumping of contaminated water from the storage bunker). Both these areas are well characterized and monitored, and included, as relevant, in the environmental assessment. In the following subsections these two areas are briefly described.

2.2.3.2.1 Winnipeg River Sediments

The routine releases changed distinctly over the operational history of the WL site. Prior to 1985, the WR-1 was operating. From start-up to 1985, releases were higher than after 1985 and the radionuclide mixture was characteristic of an operating reactor. After 1986, the releases decreased progressively to the present, and there was a shift in the mixture of radionuclides. Some of the radionuclides reported have relatively short half-lives and they all have different environmental mobilities. As a result, only a few of the radionuclides are still detectable in the river sediments.

Liquid effluent from the low level liquid waste treatment system is still discharged to the Winnipeg River via the process sewer outfall located about 8 m offshore in 5 m of water. As well, small quantities of radionuclides are released to the Winnipeg River from two ditches: Ditch #8 and Ditch #9. Ditch #8 drains the land north of the WMA north to the site boundary and beyond. Water diverted from the east around the WMA flows west in Ditch #9 into the Winnipeg River (Figure 2.2.3-1). A weir directs water flowing around the WMA and CCSF to Ditch #9.

An automatic sampler continuously samples the outflow of the outfall, proportionally to its rate of flow. A weekly screening sample (4 L), representative of effluent release from the outfall during the preceding week, is collected and submitted for gross beta analysis and scanned by gamma spectrometry. Monthly composite samples are gathered for analysis of gross alpha, gross beta, tritium, radiostrontium and other radionuclides by gamma spectrometry. The outflow from the sewage lagoon is continuously sampled during discharge, and the resulting composite sample is analyzed for gross alpha, gross beta, and radiostrontium, as well as scanned by gamma spectrometry.

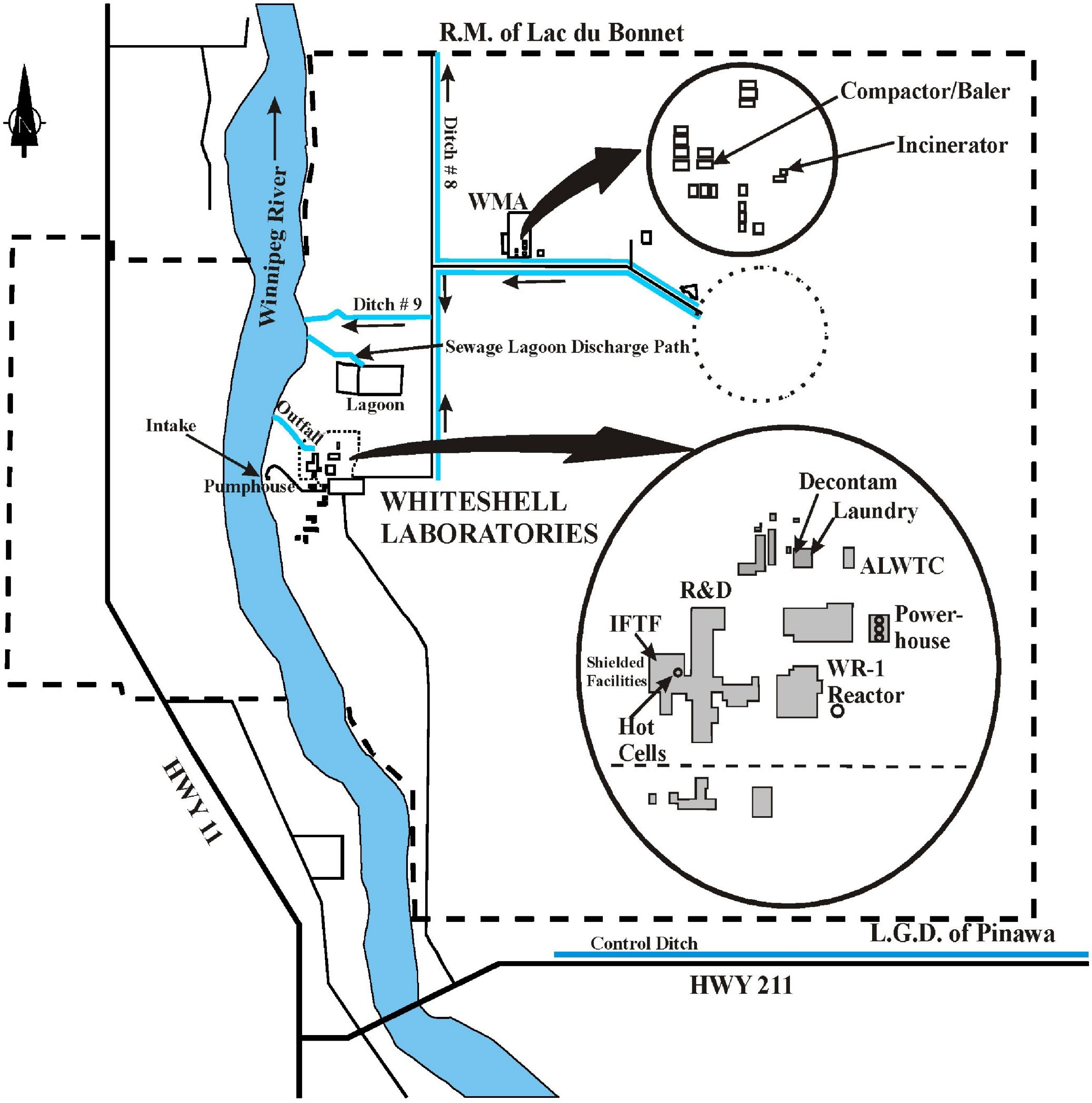
From 2012 to 2017, river bottom sediments were collected from 12 locations along the Winnipeg River, ranging from 0.8 km upstream to 13.1 km downstream of the outfall. In 2017, the gross alpha activity detected in the samples ranged from 247 Becquerels per kilogram (Bq/kg) to 405 Bq/kg (CNL 2018b). The alpha activity in these samples is due to naturally occurring isotopes such as uranium-238 or thorium-232 and their progeny, as both upstream and downstream samples contained similar levels. In addition, gamma spectrometry of the samples confirmed the presence of uranium and thorium progeny, all samples are below the Nuclear Substance and Radiation Devices Regulations Clearance Level of 1,000 Bq/kg for all of natural origin (CNL 2018b).

The gross beta activity in the sediment includes contributions from naturally occurring potassium-40 and caesium-137. Most of the beta activity for all locations continues to be from naturally occurring potassium-40. The gross beta activities measured in 2017 for most sediments are within the range of values observed over the previous 5 years, with no trends being observed (CNL 2018b). All samples collected in 2017 were below the Nuclear Substance and Radiation Devices Regulations Clearance Level of 100 Bq/kg (CNL 2018b). Sample collection is difficult as the sediment is largely impenetrable at the sampling locations due to the erosion created by the river current. On-going evaluation of the river sediments is covered by the EAFP for the WL site.

2.2.3.2.2 North Ditch/Creek

A spill incident at the WMA in 1979 led to fission product contamination of a 2 km ditch system, including the west ditch, the north ditch, and a small creek. The creek is located in the public domain north of the WL main campus, and discharges to the Winnipeg River. A follow-up ditch sampling program indicated radioactivity was deposited through the 5 to 10 cm of clay-silt soil in the ditch system near the WMA. Surface water was present in the ditches at the time, and contamination of water flowing down the drainage system exceeded the maximum permissible concentration in drinking water for continuous consumption.

The entire ditch/creek system was surveyed to determine the immediate remediation required, and the ditch flowing west from the WMA was excavated to remove contaminated soil. Routine monitoring continues to be carried out in the ditch/creek system. One-litre samples are collected whenever there is sufficient flow to enable discharge to the Winnipeg River, including the ditch bordering Highway 211 (Control Ditch) and from areas upstream from Ditch #8 and Ditch #9. The samples are analyzed for gross alpha, gross beta, and tritium.
If the levels of gross beta exceed 10 Becquerel per litre (Bq/L) the sample is submitted for gamma spectrometric and uranium analysis.



REFERENCE(S)

1. OBTAINED FROM CANADIAN NUCLEAR LABORATORIES, ANNUAL SAFETY REPORT 2015, FIGURE 11-1 EFFLUENT MONITORING LOCATIONS, 2016

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

EFFLUENT DISCHARGE LOCATION AND MONITORING LOCATIONS

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.2.3-1**

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2.3 WR-1 Building Description and Current Status

Preliminary decommissioning activities placed the facility in a safe, secure state and was categorized as a reduced site access-controlled area. The WR-1 was permanently shut down in 1985 and it was placed in a secure shutdown state in preparation for decommissioning. Preliminary decommissioning of the WR-1 commenced in 1989 and was completed in 1995. This first phase included the removal of easily mobilized radioactivity material (e.g., fuel, fluids) and decontamination of the main floor (Level 600) and first sublevel (Level 500). The last irradiated fuel was removed from the storage bays in 1993 and transferred to dry storage at the CCSF adjacent to the WL WMA. Heavy water was removed and transferred to Chalk River Laboratories for storage. Bulk organic coolant was also removed from the reactor cooling circuits and transferred to the WL WMA for incineration or solidified for storage. The organic coolant system was drained but not flushed, therefore, it is recognized that some coolant will be present within the system, including the calandria tubes. WR-1 control systems were isolated and removed. Building services systems are maintained in an operating mode.

The preliminary decommissioning work prepared the WR-1 for a deferment (surveillance and monitoring) period during which radioactivity levels would be reduced through natural decay prior to implementing further decommissioning work. At present, WR-1 is in a stable state, under a monitoring and surveillance program.

2.3.1 WR-1 Building System Description

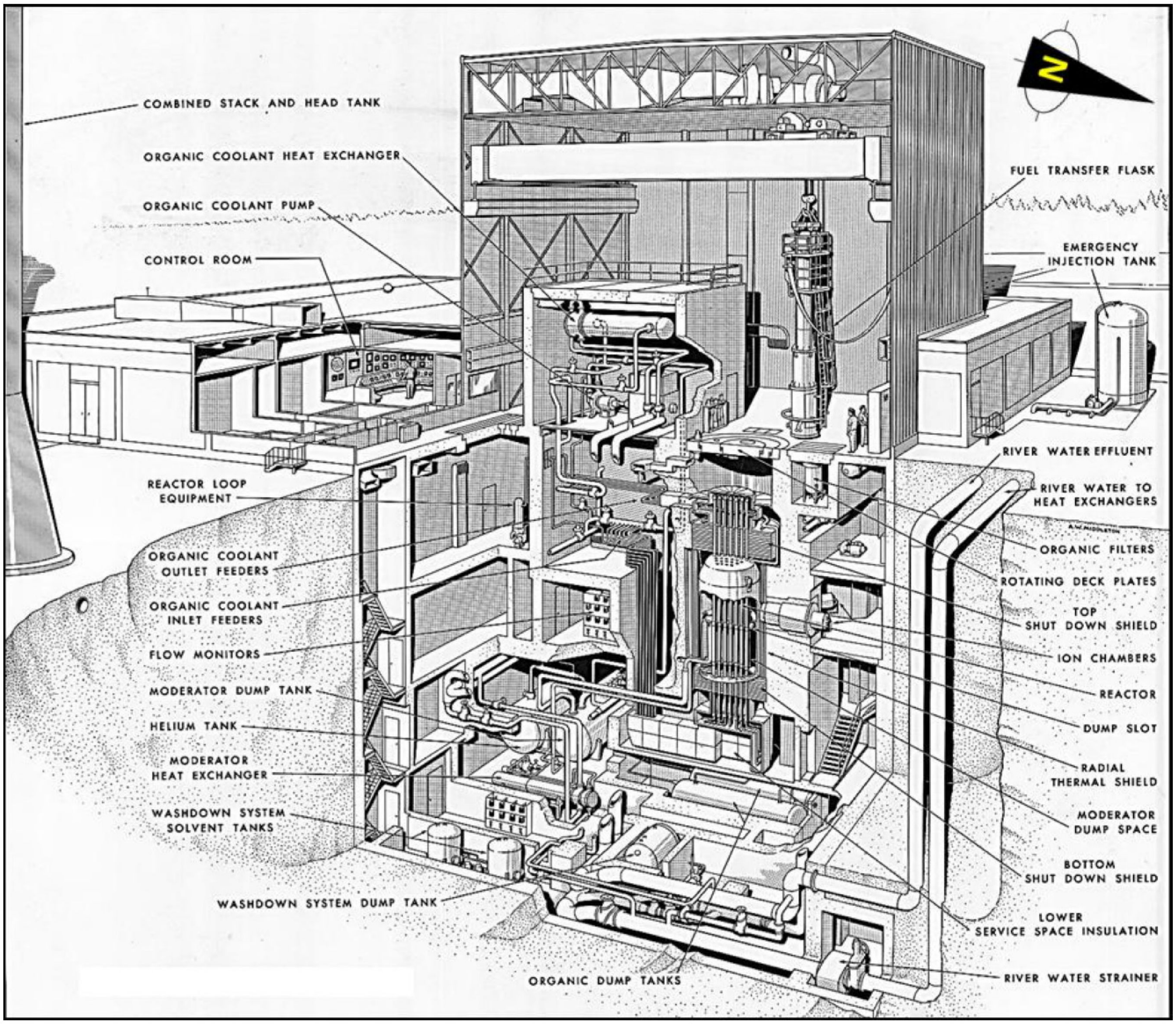
Many of the remaining systems in the WR-1 Building is permanently shut down and the systems remain largely intact. Most of the electrical power has been disconnected, but all required service systems to remain operational that are needed to maintain the facility in a safe shutdown/monitoring and surveillance state.

2.3.1.1 The WR-1 Building

The WR-1 Building consists of seven floors. The seven floors total 6,488 m^2, and five of the floors are below-grade. Level 100 (sub-basement) is located 18.5 m below the main floor (Level 600). A large portion of the WR-1 Building is located inside the Whiteshell Laboratories Controlled Area 2 (CNL 2021b). Since the reactor was shut down and preliminary decommissioning was completed, the Controlled Area has been reduced.
The WR-1 Building has been divided into two areas: the lower four levels (with restricted access) that contain shutdown reactor components (all water has been drained from systems and power disconnected), while the upper three floors have office and laboratory space, host equipment such as ventilation fans, and electrical distribution equipment still in operation for the surveillance and monitoring deferment period. Portions of Level 500 also have restricted access. The general arrangement of the WR-1 Building is shown in cross-section on Figure 2.3.1-1.

The WR-1 Building is located within the WR-1 Complex, which also includes the east annex and the service wing (Figure 2.2.2-2). The east annex is a two-storey extension that housed various systems, such as an experimental loop, a mechanical maintenance shop, and a chemical laboratory. The service wing is a single-storey extension that contained offices and the reactor control room. The east annex and service wing will be decommissioned and demolished following the original decommissioning plan (AECL 2001a).

The WR-1 Building extends both above and below-grade, and contains the reactor, PHT circuits, spent fuel storage facilities, and experimental loops. All reactor systems are located on the five levels below-grade, except for the primary coolant pumps and heat exchangers, which are contained in a shielded room protruding above the reactor hall floor.



REFERENCE(S)

1. OBTAINED FROM CANADIAN NUCLEAR LABORATORIES, WHITESHELL LABORATORIES DETAILED DECOMMISSIONING PLAN: VOLUME 6 -WHITESHELL REACTOR #1: BUILDING 100, FIGURE 2-5 WR-1 REACTOR AT WHITESHELL, 2016

CLIENT	CANADIAN NUCLEAR LABORATORIES LTD.
PROJECT	DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT
TITLE	**GENERAL ARRANGEMENT OF WR-1 BUILDING**

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.3.1-1**

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2.3.1.2 The Reactor Vessel

The reactor vessel, also known as the Calandria, shown on Figure 2.3.1-2, is a cylindrical stainless-steel tank that is surrounded by shielding. The Calandria contains 54 calandria tubes; each calandria tube surrounds a pressure tube (also known as a fuel channel). Between the calandria tube and the pressure tube was an annular space filled with carbon dioxide. The pressure tubes contained the fuel and circulating organic coolant, the calandria tubes prevented the surrounding heavy water moderator from encountering the pressure tubes. Extensions of the pressure tube beyond the calandria tubes provides connections to the coolant circulation above and below the radiation shields. The fuel from the pressure tubes and the circulating organic coolant was removed in 1985.

2.3.1.3 Shielding

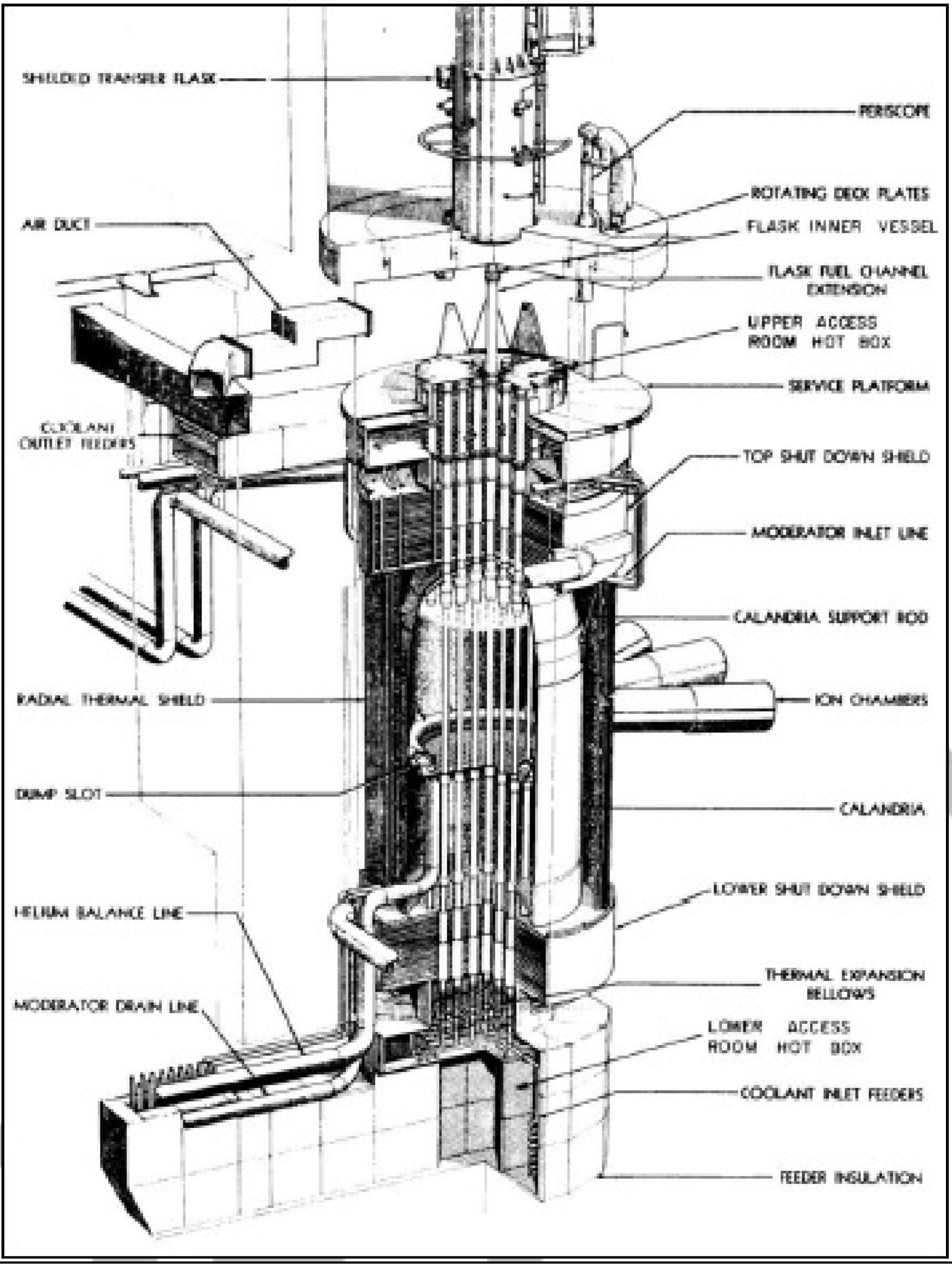
The reactor is surrounded by heavy concrete shielding (more than 2 m thick), which forms the reactor vault walls (Figure 2.2.2-3). Heavy concrete (density of 3,500 kg/m^3) was also used in the upper and lower access rooms and the shutdown shields. Piping through the concrete provided access for heavy water and helium lines and for the reactor vault exhaust duct. There are also three penetrations for the ion chambers.

Top deck plates provided an operational shield between the upper access space and the reactor hall and supported the fuel transfer flask. This shielding consisted of two rotating plates and an outer stationary ring. The plates are composed of cast steel (0.45 m-thick) topped by wood fibre hardboard (Masonite; 9 cm-thick) and a steel cover plate (0.5 cm-thick).

2.3.1.4 Heavy Water and Helium Systems

The heavy water and helium systems provided heavy water and helium to various reactor components. The heavy water system's main components included: a dump tank, a helium accumulator tank, three circulation pumps, a heat exchanger, the calandria vessel, piping and instrumentation. The helium system consists of: two helium pumps, two helium control valves, six reactor dump valves, a helium accumulator tank, five heavy water vapour condensers, a recombination unit, oxygen and helium addition stations, a sampling station, system piping, and instrumentation. The heavy water and helium systems include the following auxiliary systems:

- **Boron addition system –** The boron addition system consists of two circuits, an initial charge circuit and a continuous addition circuit. The continuous addition circuit includes a 45 L tank, metering pump, a mixing pump, a sample station, an addition station, and a cation purification column.
- **Moderator demineralizer system –** The moderator demineralizer system consists of three mixed bed ion exchangers. A shield flask was provided, if necessary, for the removal of highly contaminated ion-exchange columns. The ion exchange resins have been removed.
- **Heavy water collection and leak detection system –** The heavy water collection system consists of a stainless-steel tank, a pump, piping, and instrumentation.
- **Helium gas chromatograph –** The helium gas chromatograph consists of an analyzer, a sample conditioner, a programmer, a stream selector, and a recorder.



REFERENCE(S)

1. CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

REACTOR VESSEL (CALANDRIA)

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.3.1-2**

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2.3.1.5 Primary Heat Transport System

The PHT system, shown in Figure 2.2.2-4, was designed to remove the heat produced in the reactor core. The system was divided into three circuits (A, B, and C circuits; C Circuit in the east annex has already been removed) and the heat removed was dissipated to the Winnipeg River through conventional tube-and-shell heat exchangers using an organic primary coolant and river water for the secondary coolant. With this type of heat exchanger, one fluid runs through the tubes and another fluid flows over the tubes inside the surrounding shell to transfer heat between the two fluids. The PHT system had three similar circuits to achieve flexibility for experimental research.

There were four experimental loops in WR-1 (WR-1L2 and three fast neutron loops), and one out-of-reactor hydraulic test loop (WR-1L1 loop) in the WR-1 Building. Each in-reactor loop consisted of a fuelled test section in a reactor lattice position, with piping equipment and instrumentation located in an adjacent loop room necessary to maintain required operating conditions of flow, pressure, and temperature in the test section. A fuel position was converted to a loop by disconnecting the inlet and outlet feeders from the PHT system and connecting the feeders to the loop inlet and outlet piping.

2.3.1.5.1 Out-of-Reactor Hydraulic Test Loop

The WR-1L1 loop was a test facility capable of handling full-sized fuel channels and fuel assemblies, and consisted of a circulation pump, a pressurized pump, three test sections, three electric heaters, a make-up tank/degasifier, a condenser circuit, a purification circuit, a loop cooler, piping, and instrumentation. Organic coolant was circulated through the test section(s) at operating conditions determined by the experiment being performed.

The three test sections extended between the hatch in the reactor hall floor and the WR-1L1 loop room. Each test section consisted of an outer insulation carbon steel pipe jacket supported from the building and an inner WR-1 fuel channel. The fuel channel was sealed top and bottom to the jacket similar to the in-reactor seals to the calandria extension tubes. A carbon dioxide annulus between the outer jacket and the fuel channel provided insulation. The test sections were arranged in parallel, enabling multiple tests to be carried out simultaneously.

2.3.1.5.2 WR-1L2 Loop

The WR-1L2 loop provided a light water-cooled research facility in the WR-1 Building. The loop was designed to be operated in either the boiling water or the pressurized water mode. Distilled light water was circulated through the test section by one of two pumps. Five immersion heaters on the inlet flow regulated the temperature of the coolant entering the test section. A surge tank and a separator vessel minimized pressure fluctuations in the circuit. The separator vessel stripped the steam from the outlet flow when the boiler circuit was not in use. The steam was then passed to the degassing vessel.

A reflux boiler was used to simulate the heat removed across the boiler section of a power reactor. The condenser cooling circuit consisted of two circulation pumps and a heat exchanger cooled with standby water. A surge tank was also part of the system.

A small coolant flow was passed through a degassing circuit. A heater was located in the circuit for drying the steam so accurate steam flow measurements could be made. The degassing vessel could be operated with a hydrogen atmosphere. All vent and drain valves and bursting disc/relieve valve outlets were connected to this tank. A pump was provided for sampling the tank contents and pumping the effluent to Building 200. Pumps also supplied make-up and emergency coolant to the loop from the distilled water storage tank. A purification circuit consisting of five parallel ion exchange columns and filters were also part of the loop.

2.3.1.5.3 Fast Neutron Loops

Three fast neutron loops (WR-1L4, WR-1L5, and WR-1L6) consisted of the following main components:

- a reflux boiler;
- three centrifugal circulation pumps;
- one vertical centrifugal pressurizing pump (WR-1L6 has two pumps);
- a surge tank; and
- a test section.

Organic coolant was circulated from bottom to top through the test section and back to the reflux boiler where the heat picked up from the fuel was dissipated to the cooling water. A small bleed flow was taken from the outlet feeder, the A/B degassing tank, and pressure control. A similar quantity of degassed coolant was returned to the loop from the A/B circuit degassing pump(s). The pressurizing pump supplied the coolant to the suction of the circulation pumps. The circulation pumps were equipped with flywheels, which provided rundown cooling flow. Small motors were supplied to back up each of the main pump motors.

The WR-1 organic loops were connected to the WR-1 emergency injection system at both the inlet and outlet feeders. Emergency cooling to the test section was assured in the event of a pipe rupture occurred at any location in the loop.

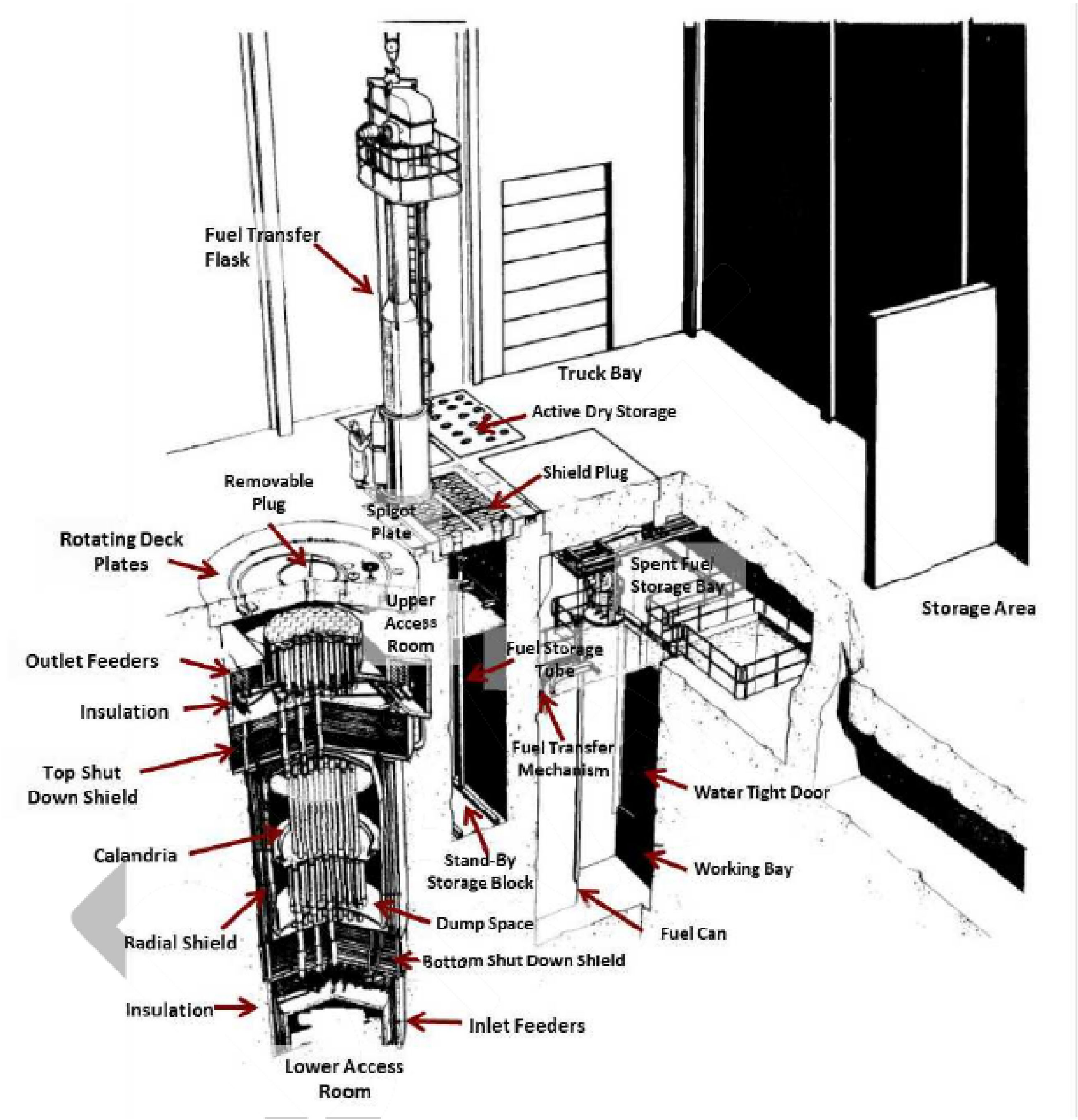
2.3.1.6 Auxiliary Systems

2.3.1.6.1 Water Supply and Drainage System

Water was required as a secondary coolant for the reactor PHT system, experimental loops, and the auxiliary systems, as well as a primary coolant for various pumps, motors, and the building services. Water was also required for the fire water, domestic water, and distilled water systems. The water supply system is sourced from the Winnipeg River via the pump house. Process and standby water were returned to the Winnipeg River through the process drainage system via the outfall station. An organic trap was provided on the discharge line to precipitate out any organic material carried along in the water being discharged into the river.

2.3.1.6.2 Spent Fuel Handling and Storage System

Throughout the operation of WR-1, irradiated fuel was removed from the reactor core with the large fuel transfer flask. Refer to Figure 2.3.1-3 illustrating the fuel handling equipment and storage bays.



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CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
WR-1 FUEL HANDLING EQUIPMENT AND STORAGE BAYS

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**2.3.1-3**

REFERENCE(S)
1. CANADIAN NUCLEAR LABORATORIES, 2017

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2.3.1.6.2.1 Fuel Storage Block

The fuel storage block provided a facility for storing and washing irradiated fuel, for storing irradiated fuel channels and experimental equipment, and for loading and unloading the fuel transfer flask. When the fuel had decayed sufficiently, it was transferred to the fuel storage bays using the fuel transfer flask.

The fuel storage block consists of a concrete-shielded, stainless steel lined pool filled with water. A carbon steel operating platform is located 0.3 m above the water level. The operating platform supports storage tubes and an operating walkway. A heavily shielded door controlled access to the block with interlocks on the fuel transfer flask to prevent fuelling operations at the block without the door being locked. The top shield over the block consists of two carbon steel plates bolted together to form a thick shield, flush with the reactor hall floor. Twenty-eight holes fitted with shielding plugs penetrate the shield and provide access ports to the storage tubes and the fuel channel storage racks below. Twenty-six storage tubes were provided: twenty-four tubes were for fuel storage, one for fuel washing, and one for loading and unloading the transfer flasks. The storage tubes are equipped with overflow drains and level instrumentation. The organic-filled tubes overflowed to the wash down system’s used-organic coolant tank. The wash tube facility consisted of a xylene filled storage tube.

2.3.1.6.2.2 Spent Fuel Station

The spent fuel station provided shielding in the reactor hall during fuel transfer operations at the fuel storage block and supported the fuel transfer flask and other equipment over the block. The spent fuel station consists of a carbon steel spigot plate recessed in the reactor hall floor, with a guide tube and a fuel transfer trolley mechanism. The spigot plate was used to position the fuel transfer flask at the station. The trolley was manually powered to move spent fuel between the transfer station and the work bay. The fuel transfer flask was interlocked at the station to prevent fuel being lowered unless the trolley was in position to receive it. The guide tube is a stainless-steel tube, extending between the spigot plate and the trolley station in the fuel transfer passage. Any organic dripping from the inner vessel was contained by the tube and collected by a drip tray at the base.

The spent fuel room, located one floor below the reactor hall, contains a concrete shielded work bay, two fuel storage bays and a fuel transfer passage. An electric crane that was used for spent fuel handling is also located there . The bays are constructed of light concrete with an epoxy lining. The underwater corners and joints were reinforced with fibreglass cloth. The storage bays are separated from the work bay by watertight gates. Inflatable seals on the stainless-steel gates provided watertight separation of the bays for draining and maintenance. The spent fuel room crane was used to transfer fuel in the storage bay area. Storage racks were provided in each bay.

The work bay provided a fuel separating mechanism, for removing the hanger rod from the fuel end, and a fuel can holding. An active fuel hanger storage facility was provided in the spent fuel room consisting of a stainless-steel tank located below a floor hatch H-24.

The spent fuel bay had a circulation system to remove decay heat from spent fuel, and irradiated equipment stored in the spent fuel bays and the fuel storage block. A circulation pump forced water from the change volume tank through a tube-in-shell heat exchanger into the bottom of the storage bays, the work bay, and the storage block. A small by-pass flow from the supply line was passed through a purification circuit containing a mixed sand and gravel bed filter, and an ion-exchange column. The water is returned by gravity to a change volume tank.

2.3.1.6.2.3 *Active Ventilation System*

The active ventilation system delivers forced air circulation throughout the WR-1 Building to provide contamination control, environment control for personnel and equipment, and the needed air exchange.

The active ventilation system was designed so that the air flows were always towards the zones having the higher contamination level to prevent the spread of contamination. The active ventilation system was sub-divided into four systems: building, cooling, control room, and the building extension. Each system had a fresh air supply, fans, filters, heating and cooling coils, and an exhaust system. All systems normally exhausted to a common stack located east of WR-1 Building. Air flow and pressure were balanced within the zones by manual volume dampers and grilles.

The incoming air was filtered and heated or cooled by preheat coils and tempering coils, and distribution fans sent air to various parts of the building. A smoke detector monitoring the exhaust duct for smoke in the exhaust gases.

The control and relay room air conditioning system drew air to replace air lost to the building active ventilation system. A circulation fan drew the air through cooling coils and a bank of filters. In the event of air borne contamination entering the control room, fresh air could be supplied to flush the room. A fan was used to exhaust directly to atmosphere through a vent in the service wing roof.

2.3.1.6.2.4 *Active Dry Storage System*

The purpose of the active dry storage system was to provide a storage facility for irradiated or contaminated hardware. The storage facility consists of a concrete shielded pit. The pit is lined with stainless steel. The facility is located in the reactor hall north of the fuel storage block. The top shield consists of an 8.34 t carbon steel plate. Twenty-seven stepped access holes with carbon steel lead filled plugs are provided in three rows.

2.3.1.6.2.5 *Fuel Wash-Down System*

The fuel wash-down system was used encase defective fuel and supply and drain organic coolant used by the fuel transfer flask, the fuel storage block, and the spent fuel storage bays. The wash-down system consists of a wash-down station, an organic storage and pumping facility, and an organic drainage facility. It consists of a top shield, a wash tube, and an organic accumulator. The wash tube is a water-jacketed stainless-steel tube with a ball valve at the top for sealing it. The external water jacket removed the decay heat from the spent fuel rod being washed.

The wash-down station was designed for washing fuel in organic coolant. Two tanks provided a reserve supply of fresh and used organic coolant. A shielded dump tank and filter unit was provided for collecting highly active or contaminated coolant from the wash tube or the fuel transfer flask.

2.3.1.6.2.6 *Transfer Flasks*

A fuel transfer flask was used to transfer irradiated fuel from the reactor to the storage block, the wash-down station, and the spent fuel station. It consists of: an outer shielding with the lifting spider, an inner vessel, mechanisms, and a cooling system. The outer shielding provided support for the flask equipment and radiation shielding for active fuel carried in it.

The inner vessel is a stainless steel, water-jacketed tube with a ball valve and snout at the lower end.
The assembly is suspended from the inner vessel hoist by a yoke attached to the upper end. The yoke travels on two screws, which move the inner vessel in the vertical plane. The stationary tube was connected to the building active ventilation system pit exhaust fan through a flexible connection during fuelling operations to remove any organic or radioactive gas released.

The cooling system consists of a stainless-steel water tank mounted on the lower platform of the flask circulation pump, connection lines to and from the inner vessel water jacket and instrumentation. Steam connections were provided to heat the cooling water or the inner vessel, if desired. In an emergency, process water could be circulated though the cooling system on a once through basis.

The fuel channel transfer flask was used to transfer irradiated fuel channels between the reactor and the storage block, and to transfer irradiated inserts and other radioactive assemblies not requiring cooling. The fuel channel transfer flask is similar to the fuel transfer flask except that all cable drives were manually operated and no instrumentation or interlocks, cooling, or ventilation were provided.

A smaller transfer flask was used to transfer fuel between WR-1 Building and the hot cells in the WL Research and Development Building. This flask consists of a stainless-steel pipe, shielded with lead. Loading and unloading was done in WR-1 in a special site at the fuel storage blocking using the reactor hall crane.

The flask maintenance station was a facility for washing and maintaining the inner vessel of the transfer flask. It was also used as a decontamination facility for the fuel handling equipment. The station is in the reactor hall northwest of the reactor. The station consists of a base plate on the reactor hall floor (to support the flask) and a shaft. A manually operated hoist served the bottom platform area. Special mechanisms were provided at the upper platform for removing the flask shutter and inner vessel assemblies.

2.3.1.6.3 Active Drainage System

The active drainage system collected liquid effluent from the various areas of WR-1 Building and groundwater from around the building base. The active drainage system included five sumps:

1) Active drainage sump A – general drainage (1.68 m^3 concrete tank);
2) Active drainage sump B – heavy water drainage (2.73 m^3 concrete tank);
3) Organic drainage sump A – organic coolant leakage (1.68 m^3 concrete tank);
4) WR-1 extension active drainage sump – general drainage WR-1 extension (3.63 m^3 concrete tank); and
5) Sub-surface active drainage sump – groundwater around WR-1 basement (15 m^3 concrete tank).

Most of the active drainage piping is welded, seamless, carbon steel pipe. The floor traps were equipped with a back water valve, which provided a ventilation barrier to prevent the spread of airborne contamination throughout the building. All rooms in the reactor area, other than those containing heavy water or organic piping and equipment, were drained to active drainage sump A. Effluent was also pumped into this sump from the active drainage sump B. Active drainage sump A held two centrifugal pumps to pump the effluent to Building 200.

The floor drains in the moderator room, boron addition room and the moderator demineralization room drained into the active drainage sump B concrete tank. A sump level monitor was located in one corner to provide early warning of water in the sump. A remotely operated sampling pump was used to sample the sump effluent for heavy water. Effluent was pumped from this sump to Active Drainage Sump A for processing or to a drumming station to collect heavy water.

The floor drains in rooms containing organic piping or equipment were connected to the Organic Active Drainage Sump A. The drains were positioned in the rooms to handle coolant spills and the effluent could be pumped into Active Drainage Sump A. The WR-1 extension active drainage sump collected the effluent from the control laboratory fume cabinets and sinks, the mechanical shop and the areas in the building extension. Sump contents were pumped into the Active Drainage Sump A.

Active Drainage Sump A is pumped out to the low level liquid waste (LLLW) treatment system. The LLLW treatment system consists of two high density polyethylene tanks, a sampling station, and two filtering units. The effluent is held in the tanks until ready for sampling. It is sampled, analyzed, filtered and adjusted as required to meet release criteria.

The sub surface drainage sump is a concrete structure located outside the north wall of the WR 1 Building. The purpose of the Sub-Surface Drainage Sump is to collect groundwater from the weeping tiles located under and around the periphery of the WR-1 Building. A network of perforated pipe, embedded in free-draining crushed stone drains the groundwater from beneath the reactor building slab on ground (bedrock) and from the exterior periphery of the foundation walls into the sump. Since no contamination is anticipated to enter this effluent from inside or outside of the WR-1 Building, this sump is typically emptied to the stormwater management system; however, it is possible to route the effluent through the LLLW Treatment System for sampling and treatment prior to release in the event of a spill outside of the WR-1 Building.

2.3.1.6.4 Emergency Injection System

The emergency injection system consists of two pressurized tanks containing fresh coolant (OS-84, also known as HB-40). Tank PO-TK1 was connected to the inlet headers of the PHT A, B, and C circuits and to the experimental loops tank and feeders. Tank PO-TK3 was connected to the outlet feeders of A, B, and C circuits through the activity monitoring system supply lines.

2.3.1.6.5 Organic Supply System

The organic supply system consists of a bulk storage tank, a dechlorination circuit, and three additional circuits. There is one additional circuit for each heat transport circuit.

2.3.1.6.6 Thermal Shield and Concrete Cooling System

The thermal shield cooling system consists of two circulation pumps, a tube-and-shell heat exchanger, a storage deaerator head tank, piping, instrumentation, and miscellaneous equipment. The concrete cooling system consists of two circulation pumps, a U-tube heat exchanger, head degassing tank, six cooling coils, piping, instrumentation, and miscellaneous equipment.

2.3.1.6.7 Compressed Air Systems

Compressed air was supplied to WR-1 Complex from the powerhouse though a carbon steel underground main, sheathed in a polyethylene protective coating. Two air receiver tanks, located on the Level 500 in WR-1 Building, provided a reserve of compressed air, which is distributed throughout the building. Mask air was supplied from the instrument air receiver tank. A pressure reducing station supplied mask air through a humidifier bank. A second pressure reducing station was available if the pressure regulator could not supply the necessary volume.

2.3.1.6.8 Heating and Cooling Systems

The heating system provided process and building heat in WR-1 Building. High temperature, high pressure water (HHW) was supplied from the powerhouse. There were two main supplies, one for the main building and one for the building extension. The system consists of several closed loop, circulating heated propylene glycol solutions or water through in-duct heater coils, room heater units, or wall convectors. The HHW boilers in the powerhouse were taken offline and now the same heating water system in the WR-1 Building is heated from three separate banks of electric heaters that operate at lower pressures. Two banks of electric heaters supply heated water to the main portion of the WR-1 Building and the third to the east annex. The main building glycol system consists of a closed loop in which a propylene glycol solution is circulated by one of two pumps. The system includes a heat exchanger and pre-heater coils. Obsolete equipment associated with the heating and cooling systems have been partially removed.

2.3.1.6.9 Activity Monitoring System

The activity monitoring system provided early detection and identification of failed fuel. The active monitoring system consisted of 19 Geiger tube detectors, which monitored the activity of the outlet coolant from the individual fuel channel outlet feeders and three Geiger tube detectors that monitored the activity of the three-primary heat-transport circuits. When a fuel failure occurred, the Geiger detector monitoring the affected fuel site would show an activity increase identifying the suspected failure as being in one of three fuel rods. Recorders commenced continuous scanning when an activity alarm occurred.

2.3.1.6.10 Fire Protection System

Fire protection was provided in WR-1 Building by: (1) the fire detection and alarm system, (2) the pressurized carbon dioxide fire prevention system, (3) the fire water system, and (4) the organic leak and smoke detection system. Self-restoring fire detectors were located throughout the building. The pressurized carbon dioxide fire prevention system would detect any rapid temperature rise in the hot box areas in the upper and lower accesses, which could be indicative of an organic coolant leak. This carbon dioxide system consisted of leak detection instrumentation, and a carbon dioxide storage and dousing system. The fire water system provided an automatic sprinkler system for all areas in WR-1 Building where a fire hazard existed and provided automatic sprinklers plus manual open-head or “fog nozzles” in areas where organic fires or organic vapour concentration might occur. The organic leak and smoke detection system consisted of photoelectric cell and light source units, which monitored the exhaust air. The fire detection and alarm system and the fire water system remain operational, but the other systems have been permanently shut down.

2.3.1.6.11 Electrical Distribution System

The WR-1 Building substation was fed by circuits from the Number 1 bus, and the WR-1 Loop substation was fed by circuits from the Number 2 bus. The loop substation has been removed. Emergency or standby electrical power was provided by four sets of diesel-driven generators located in the powerhouse. Highly reliable direct current (DC) power was supplied by rectifying the normal Class IV alternating current (AC) power or the standby Class III AC power. Lead-acid batter banks provided backup power whenever the AC supply to the rectifiers was lost. Only one battery bank remains in the system.

The WR-1 Building and ventilation stacks were connected to grounding electrodes adjacent to WR-1 Building by conductors. All non-electric equipment capable of building up a static charge were connected through grounding straps to a building grounding system, which consisted of stranded bare copper conductor embedded in the walls and floors. The static grounding circuit was connected to a ground mat provided outside and adjacent to WR-1. The metal enclosure of all electrical equipment was grounded by a fully insulated conductor with some current carrying capacity as the largest supply conductor to the equipment being grounded. The grounding conductors were connected at the electrical distribution system substation to ground buses, which were connected to the ground mat provided outside and adjacent to WR-1.

2.3.1.6.12 Annunciator System

The annunciator system consisted of alarm units with lighted, engraved windows, and plug-in logic circuit boards mounted in cabinets. A tone generator and associated loud speaker were mounted in each of the two control rooms to provide audible “alarm” and “clear” annunciator for the alarm units in the respective control rooms.

Eighteen annunciator panels were located in the main control room, each panel contained either 18 or 27 alarm units. The remainder of the alarm units were located on panels containing related equipment where possible. The loop control room also had a variety of annunciator panels; however, the loop control room has been removed from the WR-1 Building.

2.3.1.6.13 Radiological Monitoring System

The AEP 2180 fixed radiation monitor was used in WR-1 to monitor external gamma dose rates in the work area. It provided local indication with an alarm unit mounted in the main control room. All exits from WR-1 and accesses between Zone 1 and Zones 2/3/4 had hand and foot contamination monitor stations. An audible local alarm sounded if contamination was detected. Fixed radiation monitors also annunciated in the control room when dose rates in a work area exceeded the alarm set points.

2.3.2 WR-1 Building System Status

Many of the remaining systems are permanently shut down but are largely intact. Table 2.3.2-1 provides a summary of the operational status of the WR-1 systems and identifies the radiological hazards of concern for each system. The required service systems remain operational to maintain WR-1 in safe shut down and storage and surveillance state. Routine operation of these systems is documented in currently approved procedures. Building services and systems remaining in operation for the WR-1 Building include: Fire and Domestic Water Supply and Drainage Systems, Active Drainage System, Active Ventilation System, Compressed Air System, Fire Protection Systems, Electrical Distribution Systems, Annunciator System, and Contamination Monitoring System.

Since WR-1 was shut down and preliminary decommissioning activities completed, many of the areas within the WR-1 Building have been reclassified and are no longer designated within a Controlled Area, which includes most of the WR-1 rooms on Level 600 and Level 700. Some rooms that have been decommissioning and decontaminated at Level 400 and lower have systems within them that have not been completely decontaminated (some ventilation and drainage systems; WR-1 pipes). Other hazards include asbestos insulating materials, residual organic coolant in loops and building structures and service systems. Administrative controls are in place to restrict access to these areas.

The remaining hazards in the WR-1 Building are largely radiological, with asbestos and organic coolant remaining in some reactor components. The available information on the radiological status is primarily based on post-operation surveys, the end-state survey completed after the preliminary decommissioning work, and measurements in the reactor core taken in 2019 (CNL 2020b). Furthermore, a comprehensive characterization campaign was performed during 2017 and 2018 to address data gaps and to provide quantitative estimates of residual radionuclide content remaining within WR-1 systems (CNL 2020b).

Table 2.3.2-1: Summary of WR-1 Building Systems Status

Systems	Operational Status[(a)]	Key Hazardous Materials[(b)]
Major Components		
Reactor Vessel	Permanently shut down	**AP** **Tritium**
Shielding	Operational	**AP** (Primary) – The shielding in close proximity to the reactor core, mainly the thermal and biological shields **FP** – Cross-contamination. Highest levels are expected to be the shielding in the upper and lower access rooms **Lead** **Potassium hydroxide**
Heavy Water System	Permanently shut down	**Tritium** (Primary) **CP** – Corrosion products possible from activation of structures part of the heavy water system in the vicinity of the reactor core and transported to the rest of the system
Helium System	Permanently shut down	**Tritium** (Primary) **CP** – Contamination likely in helium sample coolers Other parts of the system have the possibility of being cross-contaminated
Primary Heat Transport (PHT) System	Permanently shut down; Circuit C (Room 528) dismantled and removed; other systems partially removed	**FP** (Primary) **CP** **AP** – in the sections that is in the reactor core (i.e., the pressure tubes) **Organic coolant**
Out-of-Reactor Hydraulic Test Loop (WR-1L1 Loop)	Permanently shut down	Some cross-contamination from PHT expected **Organic coolant**
WR-1L2 Loop	Permanently shut down	**CP** (Primary) **FP** – Light water-cooled loop. This loop is expected to have high level of corrosion products relative to the rest of the WR-1 systems
Fast Neutron Loops	Permanently shut down, loop 1L6 removed	**FP** (Primary) **CP** – 1L4 and 1L5 are contaminated (Class 1). 1L6 has been removed **Organic coolant**
Auxiliary Systems		
Water Supply and Drainage System	Operational	Supply water is a MARSAME Class 3. **FP** – Process drain system is known to be contaminated (contains an organic trap)
Fuel Storage Block	Permanently shut down	**FP** (Primary) **Organic coolant** **Xylene**
Spent Fuel Station	Permanently shut down	**FP** (Primary)

Table 2.3.2-1: Summary of WR-1 Building Systems Status

Systems	Operational Status[(a)]	Key Hazardous Materials[(b)]
Auxiliary Systems (cont'd)		
Active Dry Storage/ Fuel Wash-Down Systems	Permanently shut down	**FP** (Primary) **CP** – Fuel wash-down system coolant drainage facility is a MARSAME Class 1. Other related structures are a MARSAME Class 1/2
Fuel Transfer Flasks	Permanently shut down	**FP** – Internal contamination Organic coolant
Fuel Channel Transfer Flask 21 Ton Transfer Flask	Operational	**FP** – Internal contamination Organic coolant
Moderator Demineralizer System	Permanently shut down	**Tritium** (Primary) **CP**
Heavy Water Collection System	Permanently shut down	**Tritium** (Primary) **CP**
Helium Gas Chromatograph	Dismantled and removed	**Tritium** - Possible cross-contamination.
Boron Addition System	Partially removed	**Tritium** **Boron**
Ventilation System	Operational	Active ventilation system is a MARSAME Class 1; remaining system likely clean
Active Drainage System	Operational	**FP** **Tritium** **CP** **Mercury**
Emergency Injection System	Mostly dismantled and removed	**FP** – Possible from cross-contamination from PHT system
Thermal Shield and Concrete Cooling System	Permanently shut down	**AP** – Possible due to the close proximity to the reactor core **Tritium** – Trace amounts from activation of lithium for pH control
Service Air System	Operational	None expected
Instrument Air System	Operational	None expected
Mask Air System	Permanently shut down	None expected
Heating and Cooling System	Obsolete equipment partially removed; system remains operational	Likely clean, possibility of leaks from contaminated systems, most likely coolant leaks in heat exchangers in the process heating system
Activity Monitoring System	Permanently shut down	**FP** (Primary) **CP** Likely cross-contamination from PHT **Organic coolant**
Fire Protection System	Operational	None expected
Electrical Distribution System	Operational	No internal contamination expected
Annunciator System	Operational	No internal contamination expected
Radiological Monitoring System	Operational	**FP** (Primary) **CP**

Table 2.3.2-1: Summary of WR-1 Building Systems Status

Systems	Operational Status[(a)]	Key Hazardous Materials[(b)]
Auxiliary Systems (cont'd)		
Degassing and Particulate Removal System	Partially removed (C Circuit; other selected equipment)	**FP** (Primary) **AP** **Organic coolant**
Purification System	Partially dismantled and removed	**FP** (Primary) **AP**
Relief Exhaust System	Mostly dismantled and removed	**FP** (Primary) **CP**
Organic Supply System	Partially removed (Bulk Storage Tank S0-TK5)	**FP** (Primary) **CP** – Cross-contaminated from used organic coolant **Organic coolant**
Nitrogen Supply System	Mostly dismantled and removed	None expected
Neutron Power System	Permanently shut down	**AP** – Portions in the vicinity of the reactor core are likely to be activated **FP** – Possibility of cross-contamination
Protective Systems	Mostly dismantled and removed	None expected
Reactor Power Regulating System	Permanently shut down	Any contaminated parts of this system will remain inside the reactor core
Flux Detectors	Dismantled and removed	**AP**

(a) "Partially removed" means some specific equipment has been removed, but some equipment, pumps and motors remain. "Mostly dismantled" usually means equipment removed, but associated piping remains.

(b) Radiological Hazard definitions

AP = Activation products – Neutron activated structures and components. These will be in close proximity to the reactor core.

FP = Fission products.

Tritium = A weak beta emitter resulting from the activation of Heavy Water. Has the same chemical characteristics as water.

CP = Corrosion products possible from activation of particles that pass through the reactor core and then get transported to the rest of the system outside of the core

MARSAME = Multi-agency Radiation Survey and Assessment of Materials and Equipment. Class 1: Class 1 materials and equipment are impacted materials and equipment that have, or had, the following: (1) highest potential for, or known, radionuclide concentration(s) or radioactivity about the action level(s); (2) highest potential for small areas of elevated radionuclide concentration(s) or radioactivity; and (3) insufficient evidence to support reclassification as Class 2 of Class 3 materials and equipment.

Class 2: Class 2 materials and equipment are impacted materials and equipment that have, or had, (1) low potential for radionuclide concentration(s) or radioactivity above the action level(s); and (2) little or no potential for small areas of elevated radionuclide concentration(s) or radioactivity.

Class 3: Class 3 materials and equipment are impacted materials and equipment that have, or had, (1) little, or no, potential for radionuclide concentration(s) or radioactivity above background; and (2) insufficient evidence to support categorization as non-impacted.

PHT = primary heat transport.

3.0 PROJECT DESCRIPTION

The objective of the Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. The Project activities are limited to the ISD of the WR-1 Building. Other decommissioning activities are encompassed by the existing Comprehensive Study Report (AECL 2001a) and approved under the existing Decommissioning Licence (NRTEDL-W5-8.00/2024) for the WL site.

3.1 WR-1 Decommissioning Plan

The overall decommissioning approach for the WL site was to remove facilities entirely from the site except for LLW trenches located in the WMA and contaminated sediments at the WL outfall in the Winnipeg River (AECL 2001a). Decommissioning was proposed to be completed in a phased approach over a 60-year period. Three phases were proposed that would be followed by institutional control:

- Phase 1 (approximately 5 years) activities would be completed for nuclear and radioisotope buildings and facilities to place them in a safe, secure interim end-state. The Van de Graaff Accelerator and the Neutron Generator were completely decommissioned.
- Phase 2 (approximately 10 years) activities would include regular monitoring and surveillance of all buildings and facilities. Most activities would focus on the WMA. Most of the waste management facilities would be placed into a passive operational state, meaning that no further waste would be added, but facility monitoring would continue. Interim processing, handling, and storage facilities would be established to accommodate requirements for the decommissioning and/or monitoring and surveillance periods.
- Phase 3 (approximately 45 years) activities would be directed to bring the site to an end-state that would fulfil all pertinent regulatory and national policy requirements. The timing and sequence of decommissioning activities would be determined largely by the availability of disposal facilities and by the age and condition of engineered structures and buildings.

CNL is proposing an ISD approach to the decommissioning of WR-1. The below-grade reactor systems, components and structures will be permanently disposed in situ. The above-grade structures will be demolished and removed using traditional demolition methods. During decommissioning, equipment from the PHT system that is currently located on the ground-level reactor floor will be placed below-grade for ISD. A concrete cap and engineered cover will then be constructed over the below-grade structure to resist intrusion and divert precipitation and surficial runoff. The ISD approach is designed to control the rate of release of nuclear and hazardous substances from the WRDF and retain the waste away from people and the environment.

The selection of the ISD approach was based on the safety, environmental, technical and economic factors. The ISD approach is a safe option, reducing the risk to workers compared to dismantling, and providing long-term safety to the public and the environment. The ISD approach also has the least reliance on undefined future disposal options or technologies.

3.1.1 Project Schedule

CNL plans to start decommissioning activities for the WR-1 Building in 2022. The WRDF will be turned over to institutional control in 2026, which is assumed to last for a minimum of 100 years during which environmental monitoring and surveillance activities will be carried out. This timeframe is consistent with requirements for other near surface disposal projects (range 100 to 300 years), including similar projects under CNSC jurisdiction. However, it is recognized that institutional control will continue until the CNSC agrees it is no longer needed. The proposed overall schedule, including duration, for the Project is provided in Table 3.1.1-1.

Table 3.1.1-1: Project Schedule

<table>
<tr><th colspan="2">Description</th><th>Planned Schedule</th></tr>
<tr><td colspan="3"><i>Regulatory Schedule</i></td></tr>
<tr><td colspan="2">Environmental Impact Statement submitted to the Canadian Nuclear and Safety Commission (CNSC) for review</td><td>September 2017</td></tr>
<tr><td colspan="2">Detailed Decommissioning Plan submitted to CNSC</td><td>September 2017</td></tr>
<tr><td colspan="2">Public Review Period of Environmental Impact Statement (EIS)</td><td>October-December 2017</td></tr>
<tr><td colspan="2">Updated EIS submitted to the CNSC</td><td>January 2022</td></tr>
<tr><td colspan="2">CNSC, Federal and Provincial Review of updated EIS and responses to information requests/comments</td><td>January 2022 - September 2022</td></tr>
<tr><td colspan="2">Final EIS submitted to the CNSC</td><td>October 2022</td></tr>
<tr><td colspan="2">CNSC review and acceptance of Final EIS and preparation of CNSC EA Report</td><td>October – March 2023</td></tr>
<tr><td colspan="2">CNSC Licence Hearing (tentative)</td><td>April 2023 (Part 1) and July 2023 (Part 2)</td></tr>
<tr><td colspan="2">Environmental Assessment Acceptance (tentative)</td><td>August 2023</td></tr>
<tr><td colspan="3"><i>Decommissioning Schedule</i></td></tr>
<tr><td colspan="3"><i>Closure Phase</i></td></tr>
<tr><td rowspan="2">1</td><td>Preparation for In Situ Disposal</td><td>2022–2025</td></tr>
<tr><td>WR-1 Deactivation and Segregation Complete</td><td>December 2023</td></tr>
<tr><td rowspan="2">2</td><td>Grouting of Below-grade Systems and Structures</td><td>2025</td></tr>
<tr><td>In Situ Disposal Grouting Complete</td><td>January 2025</td></tr>
<tr><td rowspan="2">3</td><td>Removal of Above-grade Structures</td><td>2024-2025</td></tr>
<tr><td>WR-1 Building Demolition and Decommissioning Complete</td><td>February 2025</td></tr>
<tr><td>4</td><td>Installation of Concrete Cap and Engineered Cover</td><td>June 2026</td></tr>
<tr><td>5</td><td>Final Site Restoration</td><td>2026</td></tr>
<tr><td rowspan="2">6</td><td>Preparation of Institutional Control</td><td>2026</td></tr>
<tr><td>Site Turnover for Institutional Control and Monitoring</td><td>October 2026</td></tr>
<tr><td colspan="3"><i>Post-closure Phase</i></td></tr>
<tr><td>7</td><td>Institutional Control</td><td>2025 – 2126 (minimum)</td></tr>
<tr><td>8</td><td>Post-institutional Control</td><td>Beyond 2126</td></tr>
</table>

Note: Dates are subject to change pending receipt of environmental assessment and licensing approvals.

The Project involves two phases:

- **Closure Phase:** This phase includes the preparation and implementation of ISD, which includes preparation for ISD, grouting of below-grade structures and systems, removal of above-grade WR-1 structures and systems, installation of the concrete cap and the engineered cover, implementation of environmental controls, and final site restoration. These activities are expected to occur from 2022 to 2026.
- **Post-closure Phase:** The post-closure phase has two discrete periods, institutional control and post-institutional control.
  - Institutional control is estimated to last a minimum of 100 years during which long-term performance monitoring and maintenance activities will continue, to demonstrate compliance with the safety case assumptions. Passive controls such as access restrictions (e.g., physical barriers/fencing, signage, and land title instruments/deed restrictions) will remain in place until the end of the institutional control period. Although the duration of institutional control is estimated at a minimum of 100 years, it is recognized that it will continue until the CNSC agrees institutional controls are no longer needed. It is assumed that institutional control eventually collapses, and knowledge of the facility is lost. There is much uncertainty around when this would occur, if at all. For the purposes of the assessment, it is assumed to occur 100 years after closure of the facility.
  - Post-institutional control occurs after year 2126 and continues indefinitely; however, the timeframe defined for the assessment of potential effects, as part of the normal evolution of the Project's safety assessment is 10,000 years (See Section 5.3). This timeframe (i.e., 10,000 years) encompasses the phase in which peak effects (i.e., doses) are anticipated. During the post-institutional control period, some passive controls will still be in place including the limited footprint, the WRDF composition being relatively imperviousness and made of material of no economic value, and any remaining land use restriction acting to reduce the likelihood of a human intrusion event.

The following sections describe the general process for WR-1 ISD, and are presented in their general order of execution. However, there are opportunities to carry out portions of the work in parallel. As such, the actual order of execution may be somewhat more complex than the description implies. While the order of execution may vary, the overall scope of activities is not expected to vary significantly except as required to maintain safe work execution, and to provide long-term safety to the public and the environment.

3.1.2 Project Activities

The decommissioning activities proposed as part of the Project include:

- preparation for ISD;
- grouting of below-grade structures and systems;
- removal of above-grade WR-1 Building structures and systems;
- installation of concrete cap and engineered cover over grouted WR-1 Building area;
- final site restoration;
- preparation for institutional control; and
- institutional control.

These activities are described in the next few sections.

3.1.2.1 Preparation for In Situ Disposal

Preparation for ISD will involve a combination of sealing penetrations in the exterior walls, penetrating internal walls and equipment to provide flow paths for grout, displaced air, and dissipated heat, and establishing temporary infrastructure to support grout production and placement.

3.1.2.1.1 Deactivation of the Building

The first stage of decommissioning WR-1 Building will be transition to a ‘cold and dark’ state, in which all building services, including HVAC, electrical supply, water supply and drains, and data services are disconnected, and the building is completely de-energized. Temporary services will be installed to support safe entry into the building, and to permit physical decommissioning work to be carried out, including lighting, emergency signals, ventilation, sump water collection, and electrical power for tooling. The goal of this step is to reduce the risk of cutting into pressurized or electrified systems during grout preparation work and allow most building materials to be penetrated, cut or removed as appropriate.

Deactivation of these services may require the following steps:

- isolation of building electrical supply;
- draining of all devices containing ozone-depleting substances by qualified personnel, and removal of hazardous waste materials, as necessary;
- removal of asbestos and other hazardous material;
- removal of radioactive liquids from drain lines;
- removal of remaining surplus equipment (e.g., cabinets, sinks);
- removal of utility services, systems and major components (e.g., heating boilers, chiller);
- characterization packaging and transfer of radioactive and hazardous waste material for management in an approved waste storage facility or transfer to an off-site processing facility; and
- installation of temporary services to support safe execution of the decommissioning work.

3.1.2.1.2 Seal Building Penetrations

While the exterior walls of the below-grade portion of the WR-1 Building are intact, there are several locations where penetrations exist to allow mechanical and electrical services to enter the building. As part of closure activities, any perforations in the foundation will be filled and sealed. The penetrations will be sealed with an engineered plug so that the outer wall of the below-grade portion of WR-1 is a continuous and uninterrupted barrier to mitigate releases to the environment. Any services, systems, piping or ducting penetrating the exterior foundation walls will be cut away from the wall and the space will be filled with grout to provide an additional ‘grout break’, further limiting its potential to be a groundwater pathway.

3.1.2.1.3 Create Grout Flow Paths

To permit grout to fill the below-grade systems to the extent practicable, it may be necessary to penetrate interior walls, piping systems, or tanks. For interior walls, pathways may be created between rooms to allow flow of grout into them, as well as to allow air and grout curing heat to flow out. For piping systems and tanks, penetrations may be made at specific locations to allow grout to further penetrate into tanks and piping systems to:

1) reduce buoyancy loads from empty tanks and large pipes surrounded by liquid grout; and
2) further improve the grout flow into specific piping systems.

3.1.2.1.4 Supporting Infrastructure

CNL plans to assemble a batch mixing plant or smaller equipment on-site to provide consistent quality, and timely application, of grout. Grout produced by the batch plant would be transferred by trucks or pumped from the batch plant through piping to a number of placement points throughout the building. A water tank, piping, power, material silos, staging areas, and settling ponds may also need to be constructed for the Project.

3.1.2.1.5 Targeted Remediation

Some hazardous materials may be removed, to reduce the levels of contaminated materials within the building prior to ISD. This effort will be limited to materials that are easily accessed and present a relatively low hazard to workers to remove. This will help to further reduce the levels of hazardous materials left within the structure for encapsulation during ISD, and keep exposures to workers, the public and the environment ALARA. Examples of materials planned for removal include:

- all PCB-containing materials above exemption quantities;
- easily removable lead that is not currently being used for shielding;
- asbestos as necessary to access systems to perform decommissioning; and
- liquid organic coolant, as practical.

A survey of the building did not identify any major sources of additional hazardous material, but for environmental assessment purposes it was assumed that a conservative amount of these hazardous materials is left in place when grouting. If additional hazardous materials are discovered during decommissioning, they will be assessed for removal prior to grouting. This step will help to further reduce the levels of hazardous materials left within the structure for encapsulation during ISD.

3.1.2.2 Grouting of Below-grade Structures and Systems

Filling of structural void spaces for ISD serves the purpose of stabilizing the structure to prevent subsidence and immobilizing remaining contaminants. The introduction of fill materials, such as grout, also impedes infiltration of water and limits inadvertent intruder access to the WRDF (US DOE 2013). Grout is used for filling void spaces and because of its flowable nature, it can be introduced into the void spaces of most structures easier than traditional fill materials. In addition, the grout is used to impede the migration of contaminants out of the confines of the structure.

Grout is also used to provide shielding for workers filling areas and/or components that contain high radiological source terms. The density of cement-based grout can provide dose reduction to gamma radiation fields, allowing workers to perform other tasks in the vicinity while maintaining exposure within ALARA guidelines (US DOE 2013).

The grout has been designed to achieve the required physical properties. The design takes into account the effects of using local fill materials (e.g., sand and gravel) and the materials the grout will interact within the WR-1 below-grade structure (e.g., aluminium). Multiple grout formulations, or adjustments to the base formulation may be necessary to achieve complete filling of the below-grade structure, but all formulations or adjustments will adhere to the same minimum requirements to ensure the end-state performs as expected.

Grouting of the below-grade structure will be carried out in stages. The structure will be filled to eliminate as many void spaces as is reasonably achievable. The placement of the grout will be completed using an engineered fill schedule (i.e., grouting plan). Multiple lifts of grout will be executed to systematically fill the reactor systems and the below-grade structure. The maximum lift size (depth of fresh grout) will be determined for each room based on the structural properties of the room, and the presence of equipment that could be crushed, filled or dislodged if grout is poured too quickly. Each lift of grout will be given sufficient time to cure before additional grout is poured. Smaller lifts may be used in specific areas, for example to fill targeted voids. Quality control measures on grouting operations will be implemented to ensure all requirements for the grout are met and the final product will perform as expected.

3.1.2.3 Removal of Above-grade WR-1 Building Structures

The above-grade WR-1 components and portion of the building will be dismantled and removed after grouting has been completed. The main reactor hall, the concrete room that contains the PHT system, the reactor hall bridge crane, and the ventilation stack will be demolished. Recyclable materials will be segregated and recycled where practicable. Materials that cannot be separated easily from hazardous materials such as asbestos will be sent for off-site storage/disposal. No hazardous material or equipment from outside the WR-1 Building will be disposed of inside the WRDF as part of the Project.

The balance of the WR-1 Building outside ISD is expected to contain minimal radiological contamination. The above-grade building will be decontaminated to the extent practicable to allow the bulk of the construction material to be reused, recycled, or disposed of to a conventional (non-radiological) landfill. Segregation and performance of radiological surveys (i.e., characterization of material to be disposed of) of the building rubble will be undertaken, as required. If the building has satisfactorily been cleared of radiological contamination, then no clearance of the rubble should be required unless there is a potential for cross contamination during the demolition process.

Hazardous substances exceeding release criteria will be removed and managed in accordance with CNL’s Environmental Protection (CNL 2021c) and Management of Waste (CNL 2020c) requirements. Radiological contaminated asbestos, if present, will be packaged for storage at an approved waste management facility. Radiologically clean asbestos will be removed and disposed of in accordance with *Occupational Safety and Health and Waste Management* requirements at an approved off-site landfill. Decommissioning activities will be undertaken in compliance with the site decommissioning licence requirements and executed in a manner protecting workers, the public and the environment.

The building will be dismantled and demolished in an orderly manner. For example, the building will be stripped to the bare shell, with all of the wood, plaster and room dividers removed. The metal roof will then be removed, leaving the concrete structure available for final demolition. If feasible, the building material and structure will be reused or recycled. After demolition, the area will be backfilled and graded as necessary to achieve the required drainage conditions.

Radiological clearance surveys will be performed on the soil surrounding the foundation, and subsequent remediation will be completed as required. Soil surrounding the building will be remediated if radioactivity is encountered that exceeds soil clean-up criteria established for the site end-state and land use. The soil clean-up criteria will be established by CNL, with acceptance by the CNSC, in support of the wider WL Decommissioning Project. Soil surrounding the building footprint is not expected to exceed clean-up criteria; however, if soil contamination is encountered, it would be removed and segregated using standard excavation equipment practices. Dust suppression methods (e.g., water misting, use of applicable immobilization agents on the soil surface) will be applied during excavation as required to suppress dust levels. The contaminated soils will be managed through CNL’s Waste Management Program and placed in an approved waste management facility.

3.1.2.4 Installation of Concrete Cap and Engineered Cover

After grouting has been completed and the other portions of the WR-1 Complex are demolished, a reinforced concrete cap and engineered cover will be constructed on top of the grouted area. The concrete cap can serve as a deterrent to prevent a person who might inadvertently enter the area from being able to contact building contamination (US DOE 2013). It will also resist animal and plant intrusion into the WRDF. The concrete cap will have a design life of 100 years, aligning with recommended institutional control, and require no maintenance.

The engineered cover overlays the concrete cap and subsequently the entire grouted facility. The engineered cover reduces infiltration of precipitation into the WRDF, thus reducing the mobilization of soluble contaminants. A secondary isolation safety feature is the layer of crushed rock which contributes to resistance of flora, fauna and human intrusion.

An engineered cover (earth cover) will be installed over the concrete cap to:

- direct surface water away from the WRDF and limit water infiltration;
- protect the concrete cap from the environmental elements including freezing and thawing cycles; and
- support the growth of native vegetation (grasses, shrubs) where possible.

The engineered cover will also have a design life of 100 years and will manage stormwater consistently with the surrounding topography. It will resist wind and water erosion as well as burrowing animals and tree root penetrations.

3.1.2.5 Final Site Restoration

Upon completion of the installation of the concrete cap and engineered cover, a grass seed mixture native to the area will be used to establish vegetive cover. Maintenance of the concrete cap and engineered cover includes restricting weed growth and preventing surface erosion and abrasion. The surrounding grounds that were disturbed during demolition and decommissioning activities will be graded and restored with a native grass seed mixture compatible with the surrounding area. Stormwater management features for the reclaimed WRDF footprint will have the similar physical characteristics as the natural drainage systems in the general geographic region in terms of dynamic stability, robustness, and longevity. The engineered cover of the WRDF will be graded to promote drainage from the site to the Winnipeg River.

3.1.2.6 Preparation for Institutional Control

The grouted area will be fenced with signage as part of the institutional controls. Routine surveillance of the site will likely include inspecting the concrete cap and engineered cover for subsidence, erosion and animal or other intrusions. Additional groundwater monitoring wells will be installed, as required, to monitor the performance of the WRDF. No portion of the WL Site, including the WRDF or other areas of the WL campus will be released for other use without agreement from the CNSC that regulatory control of the land is no longer required. As such, portions of the WL site including the WRDF may remain under regulatory control in perpetuity.

3.1.2.7 Temporary Supporting Infrastructure

Temporary infrastructure may be required to support the Project during closure activities, and may include:

- batch mixing plant or similar equipment for preparation of grout;
- construction trailers;
- safety and security fencing; and
- equipment paddock and lay down area.

3.1.2.8 End-State

The final end-state for WR-1 is a permanent passive waste disposal facility (i.e., the WRDF) that applies a Defense-in-Depth strategy through the use of numerous barriers. The primary pathway for release of contamination from the system is by groundwater that has infiltrated into the sub-surface structure, picked up contamination, and then carried it out of the sub-surface structure. Each layer of the WRDF provides an additional measure to prevent and mitigate the release of contaminants to protect the public and the environment. The layers of defence against contaminant release include reactor system components, grout, internal walls, outer foundation walls, the local geosphere, a concrete cap and engineered cover, and active environmental monitoring. Combined, they form a rigorous system of barriers to provide long-term safety to the public and the environment.

Monitoring of the WRDF will continue to support the areas of institutional control, as well as any other requirements identified in the EAFP for the WL site. Wherever possible, existing programs will be adapted to meet the objectives of verifying the accuracy of the predictions made by the Project's environmental assessment and to determine the effectiveness of mitigation.

Following the completion of decommissioning work, an end-state report will be prepared. The report will describe the decommissioning work that has been performed, the outcome of that work, the results of the monitoring surveys that were performed and the interpretation of those results. The end-state report will be submitted to the CNSC to demonstrate that the intended end-state has been achieved in accordance with the DDP.

3.1.2.9 Post-Closure Activities

Future use of the WL site will depend on the ability of AECL to release parts of the site for unrestricted use upon completion of the Project. CNL is developing the WL Closure Land use and End-state Plan, along with appropriate criteria for site remediation and clean up activities, including the WRDF. The Plan defines the post-closure end-states, the post-closure land use classifications and allocation, and the physical release criteria that must be met at the site closure. These end-state definitions, land use classification and allocation, and physical release criteria are applicable to all project decommissioning activities being carried out under the WL Closure Project. Following completion of the work, the lands, including any remaining infrastructure, will enter long-term care and maintenance in accordance with the institutional control requirements. The responsible owner of the site (AECL) will be responsible for the provision of funds for the follow-up monitoring program. These costs are included in CNL's submission of a decommissioning financial guarantee and accompanying cost estimate that has been submitted to the CNSC as per the WL site licence.

In general, affected areas will be remediated to meet the WL preliminary soil cleanup and the non-radiological and radiological clearance and release criteria in accordance with the target end-state of the associated land use category (CNL 2019a). However, some lands will have restrictions of the future use and/or development of the land to ensure there are no adverse effects to the safety assessment and modelling assumptions associated with the ISD of infrastructure and facilities (i.e., the WR-1 and potentially the WMA). These areas will require ongoing controls including institutional control, access restrictions, and performance monitoring. Through the regulatory process, stakeholders will have an opportunity to provide input that could help shape future use of the WL site.

Cleanup and release criteria guidelines will be derived to protect human and key ecological receptors that sustain normal activities. Generic land use scenarios are envisioned based on how the land is used and on how sensitive and dependent the activity is on the land. Sensitivity to contamination increases among ecological or human health components most dependent on land use activities. Key biological receptors and exposure pathways are identified for each land use to protect soil quality and maintain activities performed on these lands. Recognizing differences in analyzing human health and ecological issues, soil quality guidelines for each non-radiological (e.g., heavy metals, organics) and radiological hazard are developed for both ecological and human receptors.

The post-closure phase has two separate periods: institutional control and post-institutional control. Institutional controls are requirements placed on the licensee by the CNSC for the long-term safety of a decommissioned facility. The period of institutional control will continue until the CNSC issues a Licence to Abandon in accordance with Class I Nuclear Facility Regulations [98128] (which the CNSC may or may not do).

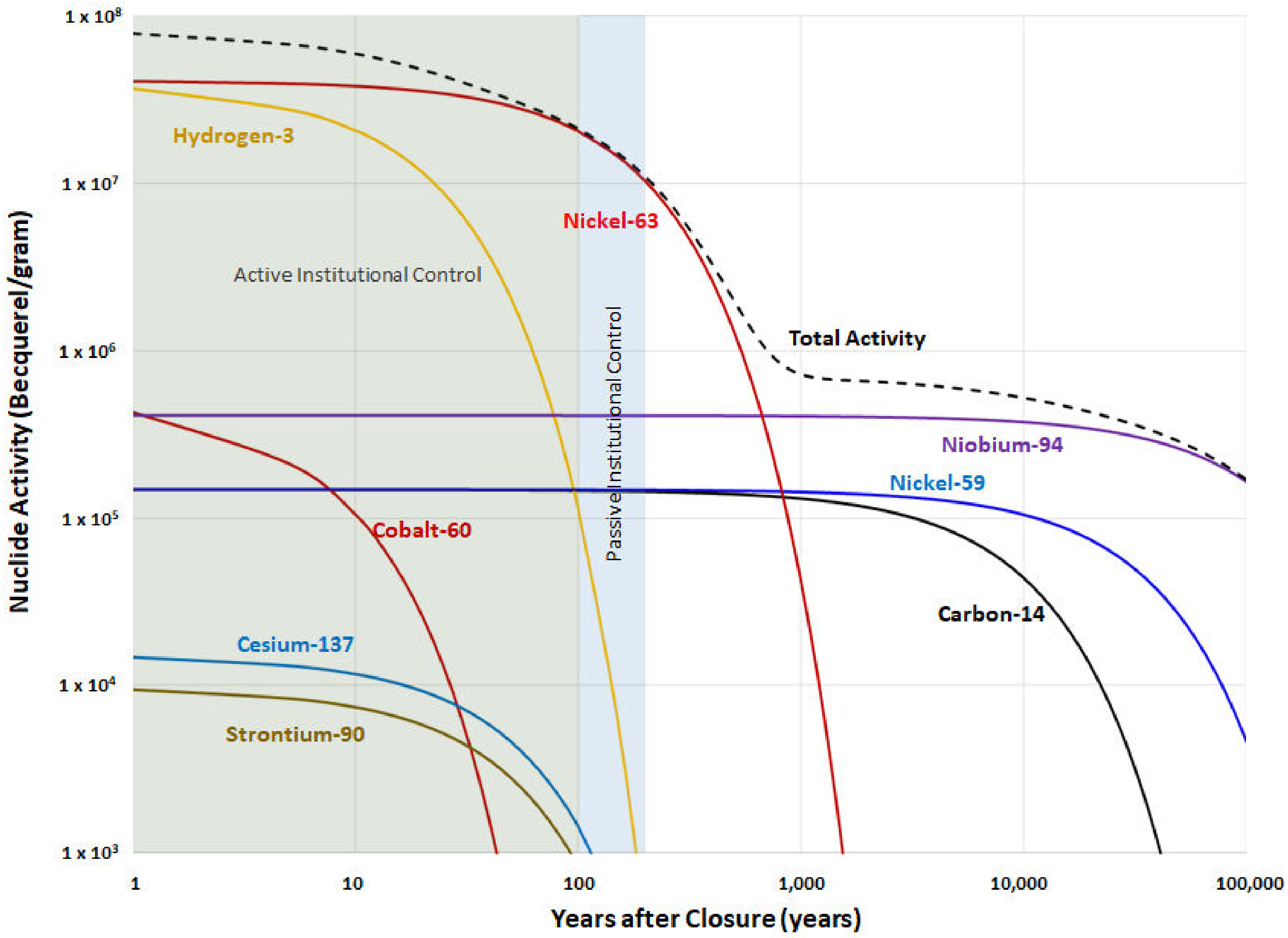
During institutional control, long-term performance monitoring and maintenance actives will continue. CNL operates an EAFP for the WL site that will be revised to include activities to manage monitoring for the Project. It will reflect the priorities and requirements that are necessary to sufficiently assess the ongoing performance of the WRDF. Since the groundwater flow at the well nest surrounding the WRDF (Golder 2021) is downward toward the bedrock, contamination releases from the WRDF, would be expected to move downward with the groundwater. As such, the monitoring program will focus on groundwater contamination, though other sampling methods may also be included such as short-term air monitoring or vegetation samples to confirm that the EAFP is comprehensive and appropriate.

Institutional control will continue until the CNSC agrees it is no longer needed. This is consistent with similar United States Department of Energy (US DOE) projects such as:

- Feed Materials Production Center in Ohio;
- Mound Plant in Ohio; and
- Rocky Flats Plant in Colorado.

For assessment purposes, a period of 100 years (2026 to 2126) of institutional control was selected. This timeframe is based on the results of the groundwater flow and solute transport model, and the expected quantities of contaminants within the WRDF over time. For prominent contaminants of concern, such as tritium and cobalt-60, the total activity of these nuclides remaining in the WRDF after 100 years quickly decreases to zero (see Figure 3.1.2-1). During the 100-year period, the peak release rate and dose rate from those contaminants are also expected to have occurred. Sampling during institutional control will verify these short-term results or signal the need for intervention. Additionally, if unexpected quantities of contaminants are found, the sensitivity analysis of the model can be used to provide an explanation of what might have happened by finding the scenario results closest to the conditions found.

It is recognized that institutional control could extend for hundreds of years beyond 2126; however, to assess the effects of an institutional control failure a specific duration was required. The 100-year period is a reasonable duration for the failure assessment given the results of the groundwater flow and solute transport model. Post-institutional control phase is assumed to occur after the year 2126.



REFERENCE(S)

1. CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
ACTIVITY CONCENTRATION IN THE WHITESHELL REACTOR DISPOSAL FACILITY OVER TIME

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**3.1.2-1**

3.1.3 Waste Classification, Inventory and Characterization

The DDP (CNL 2021b) provides a list of the radiological zoning designation for the rooms in the WR-1 Building, along with a description of the potential for contamination present, and an estimate of the contamination level and general dose rates expected in each area. This information will inform the detailed work plans that will be developed to include best industry practices and experience with the aim of limiting risks to workers, the public and the environment from ISD hazards and to be compliant with the site's ALARA policy.

Source characterization work was previously completed for the existing and potential future conditions within the WR-1 Building. CNL documents provide a summary of the estimated inventories of radionuclides (CNL 2020b) and non-radionuclides (CNL 2017a) remaining within the different reactor systems.

3.1.3.1 Radiological Hazards

Information on the radiological status and radionuclides of interest is based on post-operation surveys, the end-state survey completed after preliminary decommissioning work, and measured levels taken in the reactor core in 2011. Based on these surveys, the potential for any significant release of contamination during the remaining decommissioning work is expected to be manageable.

3.1.3.1.1 Reactor Vessel

Most of the remaining radioactivity in the WR-1 Building is situated in the reactor vessel, including the calandria, fuel channels, thermal shield, and biological shield. The combined material is estimated to contain a total radionuclide content of 1.3E+15 Bq, based on 1994 calculations corrected for decay to 2012. A radionuclide assessment showed that the fields in the core area were largely due to the fuel channels. The most abundant isotopes are nickel-63 (79% of activity), cobalt-60 (17% of activity), iron-55 (3.6% of activity), nickel-59 (less than 1% of activity), niobium-94 (less than 0.5% of activity), and carbon-14 (less than 0.5% of activity) (CNL 2021b).

3.1.3.1.2 Systems and Components

The radiological hazards in the WR-1 Building include potential exposure to radionuclides associated with irradiated reactor fuel, activated reactor components, and tritiated heavy water. Corrosion/activation and fission products may also be found in process equipment.

The predominant contaminants outside the calandria are expected to be caesium-137 and strontium-90. The total radionuclide inventory inside the process equipment is estimated at 1.3E+13 Bq, decay corrected to 2012.
The predominant contaminants in the systems and components, such as the PHT system, are caesium-137 (73% of activity), strontium-90 (26% of activity, and cobalt-60 (1% of activity). Lower amounts of americium-241, and tritium are present, with tritium being restricted to the Helium Heavy Water System. Trace amounts of ruthenium-106, caesium-137, caesium-134, cobalt-57, and cerium-144 are also found. The gamma dose rate is expected to be 99% caesium-137 and 1% cobalt-60. For more detail on the radiological hazards refer to the DDP (CNL 2021b).

The components of the PHT system, the auxiliary organic and gas systems, and the fuel wash down system are expected to be coated with used organic coolant. The residual organic is potentially contaminated from fuel failure events and corrosion products. Radiological hazards require exposure controls and any special handling procedures will be assessed during work planning activities.

3.1.3.1.3 WR-1 Building

Fuel failures in the WR-1 occurred, resulting in fission, as well as activation/corrosion products being released throughout the reactor core and PHT system. For example, a significant fuel failure event occurred when 13 assemblies failed prematurely because of delayed hydride cracking and another when damage occurred to the spent fuel bundles in the fuel bundle storage block. Table 3.1.3-1 identifies WR-1 Building rooms with elevated radiological hazards; and further detail is provided in Section 7.1.2.1.1 Dose Constraints.

Table 3.1.3-1: Building 100 Rooms with Elevated Radiological Hazards

Room	Description	Zoning	Comments
103	■ Drain Tank Room: ■ Primary Heat Transport System, Spent Fuel Handling and Storage System, Active Drainage System	R3C2	■ 50 microsieverts per hour (µSv/h) average room gamma dose rates with localized elevated fields ranging from 50 to 550 µSv/h. 2.0 millisieverts per hour (mSv/h) near contact hot spots. ■ Free of removable surface contamination.
104	■ Degassing Room: ■ Primary Heat Transport System, Heating and Cooling Systems	R3C3	■ 50-100 µSv /h average room gamma dose rates with localized elevated fields of 550 µSv/h. 6 mSv/h near contact hot spots. ■ Low level removable surface contamination.
201	Lower Access Room	R4C3	■ 250 µSv/h average room gamma dose rates. 1 mSv/h near contact hot spots. ■ Low level removable surface contamination.
301	Flask Maintenance Low Level Room	R3C3	■ 1 mSv/h average room gamma dose rates. 50 mSv/h near contact hot spots (stored waste can). ■ Generally free of removable surface contamination.
302	■ Degassing Room: ■ Primary Heat Transport System	R3C3	■ 30 to 80 µSv/h average room gamma dose rates with localized elevated fields of ~100 µSv/h. 10 mSv/h near contact hot spots. ■ Generally free of removable surface contamination.
409	Surge Tank & Pipe Shaft Room: Primary Heat Transport System.	R3C3	■ 50 µSv/h average room gamma dose rates. 400 µSv/h near contact hot spots. ■ Generally free of removable surface contamination.
410	■ 1L1 Loop Room: ■ WR-1 1L1 Experimental Loop	R3C3	■ 10 µSv/h average room gamma dose rates with localized elevated fields ranging from 20 to 180 µSv/h. No hot spots. ■ Generally free of removable surface contamination.
501	Upper Access Room	R3C3	■ 50 µSv/h average room gamma dose rates. 2 mSv/h near contact hot spots. ■ Low level removable surface contamination.
504	■ Auxiliaries Room: ■ Thermal Shield Cooling System	R3C3	■ 10 to 40 µSv/h average room gamma dose rates. 200 µSv/h near contact hot spots. ■ Generally free of removable surface contamination.
506	■ Header Room: ■ Primary Heat Transport System	R3C3	■ 50 µSv/h average room gamma dose rates with localized elevated fields 80 to 160 µSv/h. 2 to 40 mSv/h near contact hot spots. ■ Low level removable surface contamination.
537	1L5 Loop Room: Fast Neutron Loops	R3C3	■ 10 to 50 µSv/h average room gamma dose rates. 4 mSv/h near contact hot spot. ■ Generally free of removable surface contamination.
538	1L4 Loop Room: Fast Neutron Loops	R3C3	■ 10 µSv/h average room gamma dose rates with localized elevated fields of 50 µSv/h. No hot spots. ■ Low level removable surface contamination.

Table 3.1.3-1: Building 100 Rooms with Elevated Radiological Hazards

Room	Description	Zoning	Comments
539	■ 1L2 Loop Room: ■ WR-1L2 Experimental Loop Fast Neutron Loops	R3C3	■ 100 µSv/h average room gamma dose rates. 600 µSv/h near contact hot spots. ■ Moderate level removable surface contamination.
540	■ 1L2 Sample Station & Transmitter Room: ■ WR-1L2 Experimental Loop	R2C3	■ 2 µSv/h average room gamma dose rates. 80 µSv/h near contact hot spots. ■ Low level removable surface contamination.
541	■ 1L2 Auxiliary Room: ■ WR-1L2 Experimental Loop	R2C3	■ 0.5 µSv/h average room gamma dose rates. 15 µSv/h near contact hot spots. ■ Low level removable surface contamination.
601	Caged Storage Area	R2C3	■ 0.2 to 5 µSv/h average room gamma dose rates. 50 to 800 ■ µSv/h near contact hot spots (on stored flasks). ■ Generally free of removable surface contamination.
602	A & B Primary Pumps Room: PHT Circuit A, B, C main heat exchangers	R2C3	■ 50 to 200 µSv/h average room gamma dose rates with localized elevated fields of 1.2 mSv/h. 100 mSv/h near contact hot spot. Hot spot and local elevated fields associated with a stored waste can. ■ Low level removable surface contamination.

Source: CNL 2021b.

3.1.3.2 Non-radiological Hazards

Non-radiological hazardous materials in WR-1 Building include:

- Asbestos Containing Materials (ACM; friable [contains more than 1% asbestos by weight or area and can be crumbled by the human hand] and non-friable [material that contains more than 1% asbestos and cannot be crumbled under hand pressure] asbestos containing materials);
- residual organic coolant (HB-40 hydrogenated terphenyl used as reactor coolant, also known as OS-84) in the PHT system, unknown volumes potentially present in some of the tanks, and some calandria tubes;
- lead-based paint and lead shielding;
- PCBs in fluorescent light fixture ballasts;
- small quantities of mercury in thermostats and switches;
- mould; and
- other hazardous chemicals (Section 3.1.3.2.6 Other Hazardous Chemicals).

Hazardous substances removed during preparation of ISD will be managed in accordance with CNL's Environmental Protection and Waste Management requirements.

3.1.3.2.1 Asbestos

Friable ACM in mechanical insulation will be found in the WR-1 Building. The term friable is applied to a material that can be readily reduced to dust or powder by hand or moderate pressure. Therefore, ACMs that are friable have a much greater potential for airborne release when disturbed.

In 2014, much of the asbestos was removed from the non-restricted access areas of the WR-1. The restricted access areas of the WR-1 Building still contain ACMs that will remain for encapsulation within the ISD envelope (i.e., asbestos has been removed from Level 500 and above and minimal disruption of material will occur during dismantling required prior to grouting).

3.1.3.2.2 Lead

Lead-based paint is present within the WR-1 Building at the following locations:

- Room 648 and 665 – yellow painted stripes on the floor;
- Room 690 – yellow paint on hoist;
- Room 516 – yellow painted caution stripes on floor; and
- Room 513 – yellow painted caution stripes on floor and on hand railing.

Lead is also present in other forms: lead shielding, lead-based coatings, lead-glass windows, and solder.

3.1.3.2.3 Polychlorinated Biphenyls

Any remaining light ballasts in fluorescent light fixtures suspected of containing PCBs found in the will be removed in accordance with CNL’s Hazardous Waste Program. Further investigations will be performed, and suspected light ballasts will be removed.

3.1.3.2.4 Mould

Minor amounts of mould were observed within the WR-1 Building, on pipe insulation, ceiling tiles, piping and concrete column forms found in the crawlspaces.

3.1.3.2.5 Organic Coolant

The components of the PHT system, the auxiliary organic and gas systems, and the fuel wash down system are expected to be coated with used organic coolant. Fresh organic coolant is not flammable at ambient temperature, but is combustible with a flashpoint of around 170°C. However, the properties of the coolant change during irradiation with viscosity increasing and the length of flexibility of the molecules decreasing.

The quantity of HB-40 is a conservative value used to account for any undiscovered amounts; however, the residual value of organic coolant is expected to be less than the quantity provided in Table 3.1.3-2. The residual organic coolant is expected to be found as viscous liquid, sludge or as a dried coating, particularly in cold spots in pipes and tanks and in system elbows, joints, and pumps.

3.1.3.2.6 Other Hazardous Chemicals

A number of hazardous chemical and materials were used during the operation of WR-1. Table 3.1.3-2 provides a list of these materials, their expected locations within the WR-1 Building, and the estimated quantity remaining. Some items could not be directly estimated for various reasons. For example, chromium plating was used on many components without any indication of such in available documentation. In such cases, conservative assumptions have been made to provide a reasonable estimate. For several contaminants (e.g., xylene), there is no confirmation of their presence within WR-1 and their inclusion in this inventory is precautionary, to ensure the effects of the discovery of detectable quantities are not significant. Additional non-radiological constituents of potential concern (COPCs) identified during decommissioning will be assessed and remediated as needed to ensure there are no significant effects to the environment.

Table 3.1.3-2: Hazardous Chemicals and Materials within WR-1 Building

Chemical / Material	Expected Location	Description	Safety Concern	Quantity (kg)	Form
HB-40	PHT System; Auxiliary Organic System and Gas System; Fuel Wash Down system, calandria tubes, tanks	Bulk removed from the reactor cooling circuits, however, components expected to be coated with used organic coolant	■ Flammable at flashpoint around 170°C ■ Limited toxicity; standard safety precautions ■ Contamination from fuel failures and corrosion products	87,700	Liquid in system low points
Lead	Throughout the WR-1 Building	Lead shielding, lead-based coatings, lead-glass windows, and solder, paint	■ Sanding and grinding activities could mobilize	40,800	Solid, Various shielding uses (e.g., sheets, bricks)
Xylene	Spent Fuel Handling and Storage System; Wash Down System – Trapped in system low points or blockages	Used as a cleaning solvent	■ Moderate hazard and highly flammable ■ Protections – goggles, protective clothing, proper gloves, and adequate ventilation	1.9	Liquid/Vapour
Boron	Heavy Water System; Auxiliary Systems, including Boron Addition System Low points/Joints where residual solutes collect.	Added as boric acid to heavy water to control reactor reactivity	■ Long –term exposure may cause kidney damage and a risk to pregnant workers ■ Protections – goggles, protective clothing, proper gloves, and adequate ventilation	0.0009	Solids/Solutes
Palladium	Organic Supply System, Helium System - Low points/Joints where residual solutes collect	Palladium bed absorption columns used in system; columns have been removed, but other equipment may be contaminated 5% palladium on pelletized alumina used in a recombiner	■ Negligible hazard ■ Finely divided palladium metal can be pyrophoric	15.5	Solids/Solutes
Potassium Hydroxide	Chemical Addition Tank in the Concrete Cooling System - Low points/Joints where residual solutes collect	Used for pH control of cooling water	■ Strong base ■ Protection – chemical protection suit, including self-contained breathing apparatus	0.01	Solids/Solutes
Cadmium	Ion chamber Component and as plating a Fuel Storage Block	Alloy cladding component	■ Carcinogen, and development and reproductive toxicant	91.4	Solid/Plating

Table 3.1.3-2: Hazardous Chemicals and Materials within WR-1 Building

Chemical / Material	Expected Location	Description	Safety Concern	Quantity (kg)	Form
Chromium	Various Thermocouples and as Plating on various components (e.g., condenser tubes; boiler tubes)	Alloy cladding component	■ Carcinogen, and suspected respiratory toxicant	148	Solid/Plating
Mercury	Various Thermocouples and as Plating on various components (e.g., condenser tubes; boiler tubes)	Electric relay switches, industrial lighting bulbs and fluorescent tubes	■ Developmental toxicant; wide range of suspected toxic effects	0.33	Liquid/Vapour
Beryllium	WR-1 – Fuel Elements	Alloy in cladding fuel elements	■ Carcinogen, and respiratory toxicant	—	Trace solid residuals
Platinum	Flux Detectors	Used as wire in the detector	■ Limited hazard ■ Protections – standard safety precautions	—	Trace solid residuals
Magnesium Oxide	FLUX Detectors	Used as an insulator	■ Negligible hazard	—	Trace solid residuals
Gadolinium Nitrate	SLOWPOKE Demonstration Reactor Liquid Absorber Safety System	Gadolinium nitrate solution flowed into the pool to shut the reactor down	■ Moderate hazard ■ Protection – goggles, protective clothing, proper gloves, and adequate ventilation	—	Trace solid residuals
Ozone Depleting Substances	Multiple systems	Air conditioning and refrigeration systems	■ Negligible hazard ■ Environmental Concern	—	Trace solid residuals
Multiple Ion Exchange Columns	Heavy Water System, Distilled Water System, Spent Fuel Bay Circulation System, Concrete Cooling System, Boron Addition System, SLOWPOKE Demonstration Reactor Auxiliary Systems, WR-1L2 Loop, Fast Neutron Loop	Numerous ion exchange columns are incorporated into the systems of the WR-1	■ Negligible hazard ■ Specific resin type should be confirmed before removal ■ Protection – standard safety precautions	—	Trace solid residuals

PHT = Primary Heat Transport

3.1.3.3 Waste Generation and Management

The handling and disposal of waste material resulting from decommissioning of the WR-1 Building will be conducted in accordance with CNL company-wide requirements:

- Waste Characterization and Tracking;
- Waste Minimization Program Requirements;
- Management of Solid Waste;
- Management Liquid Waste; and
- WL Waste Materials Management.

Waste handling and disposal will be in accordance with the Waste Management Program (CNL 2020c). For a summary of the waste streams volumes anticipated to require management from decommissioning of the WR 1 Building, refer to Table 30 of the DDP (CNL 2021b). For a summary of the clearance criteria of materials from the WL site, refer to Table 32 of the DDP (CNL 2021b).

3.1.3.3.1 Radiological Wastes

Radiological wastes will be generated by decommissioning activities. The source of these wastes include:

- Preparation of reactor systems for ISD – contaminated personal protection controls including swipe samples and coupons, contaminated tools, and equipment will be generated.
- Dismantling of the WR-1 above-grade structure – contaminated equipment/structures will need to be handled, such as the above-grade portion of the PHT system and reactor deck plates.
- Grouting activities – contaminated Personal Protection Equipment and Clothes will be generated.

Radiological wastes, such as personal protective equipment, are not planned for encapsulation and will be managed in the WMA or transported off-site (e.g., CNL's Chalk River Laboratories in Ontario).

3.1.3.3.2 Hazardous Non-Radiological Wastes

Targeted removal of hazardous substances remaining within the WR-1 Building will generate small quantities of non-radiological hazardous wastes. Hazardous wastes will be managed in accordance with CNL's Waste Management Program (CNL 2020c) and Environmental Protection Program, and will meet all Federal, Provincial and Municipal requirements. The wastes will be shipped off-site to an appropriate hazardous waste facility or encapsulated in the same manner as radiological wastes where it is demonstrated safe to do so.

3.1.3.3.3 Clean Wastes and Likely Clean Wastes

Removal of the WR-1 above-grade structure and decommissioning of the temporary supporting infrastructure will generate clean and likely clean waste. Likely clean wastes will be monitored for radioactivity in accordance with Radiation Protection and Waste Management Program requirements to confirm that the waste is clean. Any wastes not meeting criteria for classification as clean will be managed as radioactive waste. Disposal methods for clean waste materials will meet all Provincial and Municipal requirements. The disposition options are reuse or recycle, disposal in an off-site landfill, and disposal in the WL landfill.

3.1.3.4 Transporting Waste Off-Site

The transport of radioactive waste is regulated under the *Packaging and Transport of Nuclear Substances Regulations*. Transported waste from the Project includes radioactive waste. Key activities consist of:

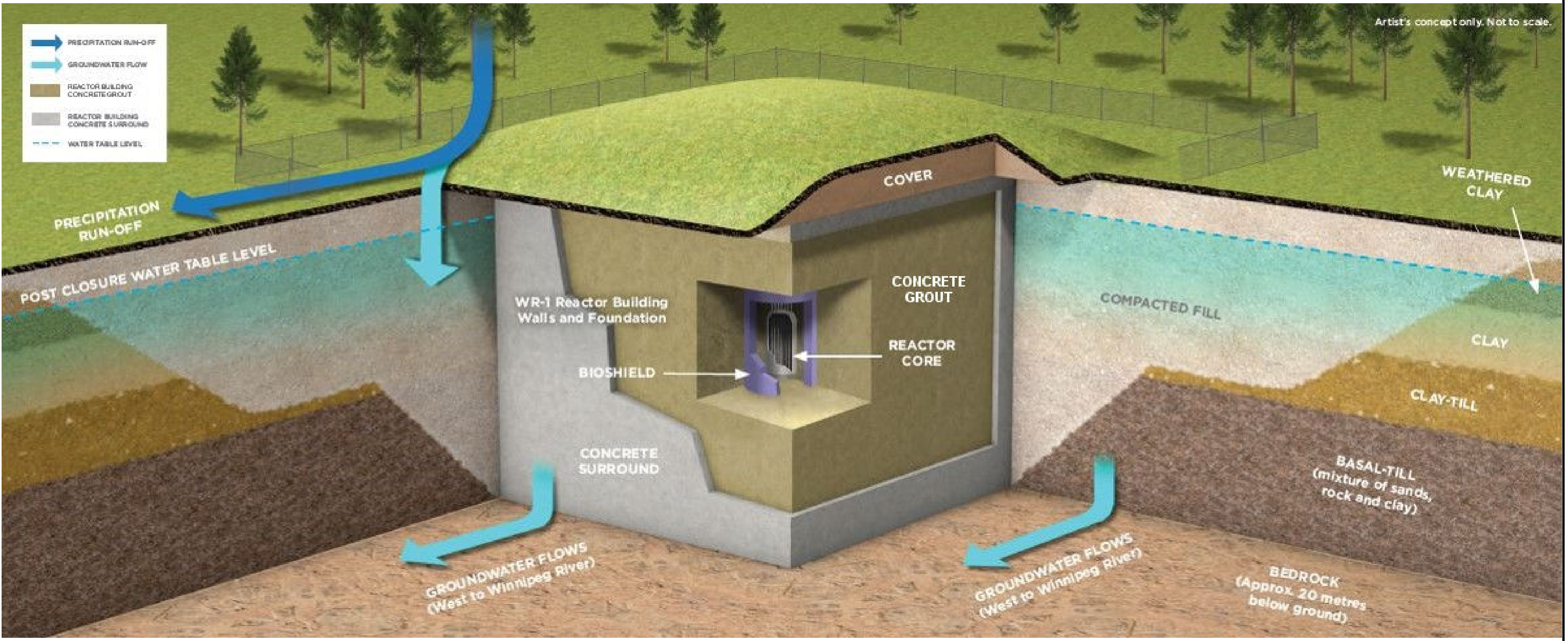
- loading the waste into an approved shipping container;
- loading the container onto the vehicle;
- monitoring the vehicle for contamination and cleaning it, if necessary; and
- driving the transport vehicle to an approved facility.

The Waste Management Program Description Document (CNL 2020d) defines the process for managing wastes from point of generation to ultimate disposition. It provides direct support and oversight to ensure waste generating activities are performed in a manner that protects the workers, the public and the environment. The Transportation of Dangerous Goods interfaces with the Waste Management Program and is responsible for the coordination and transport of all types of wastes (including radiological, hazardous and non-regulated construction and demolition material) from the Project. Specific requirements in the management of transporting radioactive wastes from the Project to Chalk River Laboratories, for long-term storage and/or disposal, include liaising with Chalk River Laboratories Waste Receiver and transportation subcontractors. Preparations for shipments of dangerous goods (including classifying the shipment and/or packages and preparing shipment documentation) will meet all applicable regulations. The main function of the Transportation of Dangerous Goods Program is to protect personnel, property and the environment from the effects of radiation and hazardous materials during transport. This is accomplished by establishing and maintaining requirements and procedures necessary to facilitate the safe transport of dangerous goods and non-regulated waste materials from the Project.

3.1.4 Multilayered Barrier System

The proposed ISD approach to decommission WR-1 relies on a number of barriers, which passively resist release of contaminants (Figure 3.1.4-1). The above-grade structures will be demolished, and the majority of the wastes will be recycled or disposed of in appropriate waste disposal facilities. Some of the demolished above-grade structures will be placed within the WR-1 Building. A concrete cap and engineered cover will then be constructed over the below-grade structure to deter intrusion and protect the structure from water. After the closure phase, the remaining structure will be called the WRDF. The WRDF will contain and isolate the waste from the environment until a reduction in hazard has occurred due to radioactive decay. Institutional control is expected to be maintained for a minimum of 100 years. The WRDF will contain and isolate the waste from the environment until a significant reduction in hazard has occurred due to radioactive decay. Additional detail regarding how the principle of defence in depth has been applied to the Project is provided in Section 6.0.

WR-1 EXCAVATION, FOUNDATION AND GROUNDWATER FLOW - PROJECTED POST-DECOMMISSIONING



NOTE(S)

REFERENCE(S)
1. OBTAINED FROM CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	-
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

TITLE
WR-1 IN SITU DISPOSAL SYSTEM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**3.1.4-1**

3.1.5 Safety-related Systems

The safety-related systems are those that are required to maintain the WR-1 Building in a safe state during the storage with surveillance phase of the Project, as well as during the decommissioning execution phase. These systems will be degraded, replaced, or removed as decommissioning work progresses from storage with surveillance to final disposal. As grout is poured, ventilation or active drainage systems will only be required until the room is filled. Upon completion of grouting the entire structure, these systems will no longer be required, and will become non-functional due to grouting. Electronic fire detection systems will be replaced by fire patrols and other administrative controls to terminate the system connections and isolate the facility. During the institutional control phase, the fire detection system will no longer be required as a fire affecting the WRDF is not considered a risk, leaving the reactor vault structure as the only safety-related system. The reduction, replacement or cessation of any safety related system will be performed in compliance with the WL site licence and Licence Conditions Handbook and supporting programs and procedures.

These safety-related systems are only required during the demolition/grouting phase of the project. Once the below ground structure is filled with grout, and the above-ground structures will be decontaminated and removed, the potential for doses in excess of 1 mSv greatly decreases due to the passive nature of the disposal facility isolating and containing the inventory. None of these safety-related systems are required post closure.

3.1.6 CNL Management System and Quality Assurance

Compliance programs are in place that translate legal and related requirements into processes or program requirements appropriate for the WL site. The compliance programs establish a common set of work practices and procedures so that work is performed consistently across all CNL sites. These programs were initially designed for an operating nuclear facility with control systems to handle a much larger (several orders of magnitude) radionuclide inventory than are expected from the Project. Equivalent programs and control systems will remain during the closure phase and will be augmented as necessary according to the DDP so that any effects from the Project activities would be handled in a controlled and effective manner (CNL 2021b).

All Project activities will be managed under the CNL Corporate Management System, which is required under Licence Condition 1.1 of the Decommissioning Licence (CNSC Licence No. NRTEDL-W5- 8.00/2024). Work will be conducted in accordance with CNL's approved policies, programs and procedures for the safety of workers, the public and the environment. These procedural documents satisfy various program and licensing requirements such as CSA N286-12. CNL's Corporate Management System is based on a comprehensive framework that covers all aspects of the management of the business. Effective corporate governance is achieved through the establishment and implementation of controls that are integrated into the Corporate Management System. An important feature of the Management System is the Nuclear Performance Assurance Review Board (NPARB). This board provides a comprehensive mechanism for executing the Site Licence Holder's functional oversight of activities (processes and HSSE programs and facilities important to continued licencing of the WL site). Meeting on a quarterly basis, the NPARB reviews performance and effectiveness of CNL's processes, programs and nuclear facilities to identify opportunities for improvements and the need for change. All NPARB recommendations are managed and tracked to completion through CNL's Corrective Action Program.

CNL applies their Corporate Management System to administer, and continually improve operations to provide sustained confidence that nuclear safety and security is assured. The Quality Assurance program for decommissioning at the WL site is based on Canadian Standards Association (CSA) N286.6-98 (CSA 2003), Decommissioning Quality Assurance for Nuclear Power Plants, and is aligned with the CNL's Management System Manual. The Quality Assurance Program is applicable to management, engineering, technical analysis, operations, and other work carried out in support of the management of radioactive waste for the Project. The CNSC staff have reviewed and accepted the WL Decommissioning Quality Assurance Plan as meeting the requirements of CSA N286.6-98. CNL continues to conduct decommissioning activities at WL in accordance with this program.

To meet CNL’s strategic requirements, the Project will:

- contain the radioactive contamination until it has decayed to levels that do not present a risk to the public and environment; and
- meet the following CNL Compliance Programs during closure and as relevant into post-closure:
  - Radiation Protection (closure);
  - Environmental Protection (closure and institutional control);
  - Emergency Preparedness (closure and institutional control);
  - Waste Management (closure and institutional control);
  - Occupational Safety and Health (closure and institutional control);
  - Nuclear Criticality Safety (closure);
  - Physical Security (closure and institutional control);
  - Nuclear Materials and Safeguards Management Compliance (closure);
  - Operating Experience (closure and institutional control);
  - Pressure Boundary (closure);
  - Transportation of Dangerous Goods (closure); and
  - Fire Protection (closure).

CNL’s Emergency Preparedness, Radiation Protection, Environmental Protection and Occupational Health and Safety Programs and associated procedures are in place to assist in the response to radiological and non-radiological incidents. Incident response and mitigation procedures and capabilities are maintained for all facilities, processes and activities with identified environmental aspects. Response and mitigative actions to anticipated environmental incidents are addressed in facility/operation/building emergency procedures. The WR-1 Building has an existing emergency procedure that will be modified as the facility changes – the WL Emergency Operations Centre Operating Procedure. This procedure conforms to the legislative and regulatory requirements as outlined by CNSC’s *REGDOC 2.10.1 Nuclear Emergency Preparedness and Response* (CNSC 2016), and the Federal Nuclear Emergency Plan. In accordance with these requirements, this procedure, as part of WL’s broader Emergency Management framework, serves to provide for the protection of life, property and the environment in the event of an abnormal condition or emergency situations affecting the WL site or surrounding area.

4.0 SAFETY STRATEGY

The safety strategy refers to the approach that will be taken to comply with the safety objectives and principles, to comply with regulatory requirements, to confirm that good engineering practice has been adopted, and to optimize safety and protection. In accordance with section 4.26 of IAEA SSG-23, the safety strategy comprises an overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case.

The ISD approach provides a permanent, passive decommissioning end-state, and incorporates proven technologies and best industry practices, including documented experience from the IAEA and other similar international facilities. The safety of the Project post-closure is provided by means of passive features so that there is no need for active management, which is in alignment with IAEA Requirement 5 of SSR-5 (IAEA 2011). The performance of the natural and engineered barriers provide for safety following the decommissioning of the WR-1. In addition, institutional controls, including restrictions on land use, and a program for monitoring will be completed in the post-closure phase to help ensure the safety of the public and the environment. Institutional controls will also contribute to safety by preventing or reducing the likelihood of human actions that could inadvertently interfere with the WRDF or degrade the safety features.

As outlined in Section 4.1 Safety Objectives, the intended safety objectives are the containment and isolation of the waste. Although the Project will result in long-term institutional controls for a small portion of the WL site, the remaining land is safe and appropriate for other use. This remaining land may be transferred to other parties following engagement with stakeholders, Indigenous peoples and the public. Future uses/zoning have not been determined, but it is assumed that the land will meet Canadian Council of Ministers of the Environment (CCME) land use criteria. As the Project further develops, the safety strategy will be continually validated, and any changes justified in the safety case.

4.1 Safety Objectives

The overall objective of the Project is to decommission WR-1 Building in a manner that meets end-state criteria and aligns with current international best practices, including the protection of present and future generations, and the avoidance of imposing undue burden on future generations (IAEA 2006). The proposed ISD approach includes three main objectives:

- apply international best practices to safely decommission the WR-1 Building while providing protection to the human and ecological environment by:
  a) limiting the need for interim storage by reducing deferment periods where appropriate;
  b) limiting releases of radiological and other hazardous substances from the facility;
  c) demonstrating that the potential effects on the environment from the Project are within acceptable limits and in compliance with applicable regulation and do not nullify obligations previously committed to in the existing Comprehensive Study Report (AECL 2001a); and
  d) demonstrating the long-term safety of the Project through consideration of the site characteristics and engineered design features, including implementation of a long-term monitoring and surveillance program for the WL site.
- apply CNL and international safety design principles to minimize radiation doses to the public and workers (e.g., meeting the ALARA principle); and

- reduce risk to workers during the decommissioning phase by avoiding and minimizing industrial hazards.

For the Project, the intended safety objectives are the containment and isolation of the waste. The associated concept for long-term waste management is based on the passive containment and isolation of the waste within the WRDF. Containment is achieved through robust design based on multiple barriers providing defence-in-depth. The IAEA SSR-5 *Disposal of Radioactive Waste* (IAEA 2011) guidance states that multiple safety functions be provided, such that safety does not depend unduly upon any single safety function; and that if one barrier does not perform as intended, there are further barriers to compensate, and maintain the safety of people and the environment. Isolation is achieved through proper site selection and, as necessary, institutional controls to limit access and land use. The concept of isolation involves essentially two aspects: physical separation of the waste from the accessible environment; and isolation of systems, structures and components performing safety functions from disturbing effects.

Discussion of how these safety objectives is met is presented in subsequent sections. In accordance with IAEA SSR-5 *Disposal of Radioactive Waste* (IAEA 2011) guidance, the Project is designed to provide containment by controlling the rate of release of nuclear and hazardous substances from the WRDF, and isolation by limiting access to the waste by people and the environment. Additionally, in accordance with IAEA SSR-5 guidance, the Project is designed to slow the dispersion of radionuclides in the geosphere and biosphere, and to provide isolation of the waste from factors that could degrade the integrity of the WRDF. Various elements of the Project, including the physical components and CNL's Management System procedures and programs, contribute to meeting these safety objectives over different timescales.

4.1.1 Containment

As indicated in IAEA SSR-5 "Disposal facilities are not expected to provide complete containment and isolation of waste over all time; this is neither practicable nor necessitated by the hazard associated with waste, which declines with time." For the WRDF, containment is achieved through using combined inherent natural and engineered barriers to provide safety following the decommissioning of WR-1. A key part of the safety assessment is evaluating the performance and effectiveness of the containment lines of defence, while addressing uncertainty in the properties and performance of the system and barriers. During the post-closure phase, the pumping from the WR-1 Complex sumps will cease, and the groundwater elevation will recover to a new equilibrium resulting in a portion of the WRDF being below the water table, enabling a transport mechanism for radiological and hazardous materials. Thus the main objective to ensure that the release rates for hazardous and radiological contamination are below regulatory and safety limits through application of quality assurance in the assessment and establishment of the engineered materials that will act as a containment barriers. Systematically, the natural and engineered barriers consist of the reactor core and bioshield, grout fill, building foundation, concrete cap and engineered cover, and finally the geologic surround. This concept is shown schematically on Figure 3.1.4-1. The nature of the reactor components waste form in the WR-1 provides a key barrier to the release of contamination into the environment as well.

4.1.1.1 Reactor Components

The reactor core components (combined calandria and fuel channels), although sources of contamination, function as barriers providing contaminant isolation and containment. The majority of the remaining contamination in WR-1 is located within the piping and tanks that make up the reactor systems (primarily in the calandria and fuel channels). The contamination is both on the internal surfaces (surficial contamination), as well as embedded in the material itself (activated components). In some cases, the components themselves are the contaminant (e.g., lead). The reactor core (combined calandria and fuel channels) is considered a barrier to contaminant release as the activation products within the components are only released as the component corrodes over time. These components are the initial barrier and must first breakdown through corrosion and dissolution for contamination to be released to any groundwater. No contamination within them will be released prior to their corrosion and dissolution. The effectiveness of the bioshield and non-core reactor components as barriers was conservatively disregarded for the purposes of the Safety Assessment.

4.1.1.2 Grout

The grout will fill the majority of the remaining ISD facility below-grade, as well as the contaminated reactor system components (Figure 3.1.4-1). The primary purpose of grouting of the facility is to stabilize the structure and prevent subsidence over time. The safety case for WRDF containment is built on the conservative assumption that the only barrier function the grout provides to radionuclide release is the hydraulic conductivity that controls the rate of groundwater movement through it.

Grout is not expected to completely penetrate and fill every void space, as the existence of voids and cold joints is not detrimental to the overall safety of the WRDF. Instead, the structure will be filled to eliminate as many void spaces as is reasonably achievable. Smaller diameter pipes and conduit will not be drilled or cut unless they penetrate through the outer foundation walls. Larger pipes or ducts will be drilled or cut to allow grout to more easily flow into and fill them. The overall fill design will target the elimination of transport pathways within larger diameter pipes or ducts by cutting or ‘air gapping’ pipes and ducts (i.e., a physical space cut in piping so that when grout fill is placed, that space is filled with grout and will provide a 'grout break' in that pipe and limit its potential to be a groundwater pathway). This will further limit the effects of potential voids. The placement of the grout will be completed using an engineered fill schedule (i.e., grouting plan). Multiple lifts of grout will be poured to systematically fill the below-grade structure.

The DSAR and solute transport modelling are based on target properties of the cured grout. CNL has specified the minimum properties for the grout that will be used in WR-1 (Table 4.1.1-1). These properties were based on the industry best practices and are either required for effective grout installation or were used in the safety assessment modelling (EcoMetrix 2021) to confirm protection of the public and the environment.

Table 4.1.1-1: Target Physical Properties of Cured Grout

Property	Target	Basis
Bleed Water after 24 hr (vol.%)	0	Eliminate need for liquid removal
Maximum Temperature Rise during Curing	<25°C difference between grout interior and exterior	Manage effects of heat of hydration
pH	<13.5 for bulk fill grout	Compatible with materials and contaminants in most of the rooms to be grouted
Compressive Strength	>3.4 MPa at 28 days	Non-structural grout, needs only to support its own mass
Effective Porosity (vol.%)	<0.6	Used in solute transport model
Dry Bulk Density (kg/m^3)	2,100	Used in solute transport model
Hydraulic Conductivity (m/yr)	<0.03	Used in solute transport model

Vol. % = percent volume; MPa = megapascal.

The most significant property of the grout used in the solute transport modelling is the hydraulic conductivity. CNL has specified that the bulk fill grout used in the decommissioning of WR-1 must have a hydraulic conductivity of less than 9.5E-10 m/s (CNL 2017b). The safety case of the proposed ISD does not require that the reactor vault be grouted or that all voids and systems are filled; therefore some void spaces will remain after grouting is complete. The existing exterior foundation walls provide a sufficient barrier to releases, and including grout in the assessment would not considerably increase the effectiveness of WRDF containment. Recognizing that not all of the areas within the building will be fully filled by the grout, and some voids will remain after final grout placement, a higher value of 5.0E-08 m/s was selected for the hydraulic conductivity of grout in the WR-1 assessment model (i.e., a factor of fifty times greater than the maximum value specified by CNL).

In the long-term (i.e., thousands of years), it is expected that the grout will degrade, and the hydraulic conductivity will increase as a result of this degradation. As noted in the literature, there are many factors that contribute to the degradation of grout over time, and the ability to model its performance is limited as a result of the uncertainty associated with these factors (Walton et al. 1990, Clifton et al. 1995). Contributing factors to degradation include sulphate and magnesium attack (leading to expansion and disruption of the cement), reinforcement corrosion through chloride attack, leaching, carbonation, alkali aggregate reaction, freeze/thaw and cracking. The extent to which degradation of concrete will occur as the result of these contributing factors is dependent on the environmental conditions surrounding the concrete, which are uncertain (Walton et al. 1990; Clifton et al. 1995). For the WRDF assessment, a step function was assumed (with linear transitions in between steps) to simulate the anticipated increase in hydraulic conductivity of the grout as degradation progresses. Due to the uncertainty associated with the degradation of the grout over time, this concept was explored in the context of a sensitivity analysis. The degradation is assumed to occur as a step function over the first 2000 years and the grout will reach its fully degraded hydraulic conductivity value of 5.0E-07 m/s by year 2000 (Table 4.1.1-3). The degraded grout hydraulic conductivity value was chosen to match that of the highest value of the surrounding geological units. For comparison, the geological conditions are anticipated to remain consistent until glaciation occurs (at least 60,000 years from present).

The grout mix has been designed to achieve the required physical properties (Table 4.1.1-1) as well as necessary fresh properties to enable efficient delivery and placement into the void spaces of the WR-1 below-grade structure. The design considers the effects of using local fill materials (e.g., sand and gravel) and the materials the grout will interact with in the WR-1 below-grade structure (e.g., aluminium and lead). The grout design includes guidance on appropriate quality control measures to be applied during mixing and placement of the grout.

Initial performance requirements and a supporting test plan were prepared by Savannah River National Laboratory (SRNL 2018). CNL engaged a vendor to develop a grout formulation that meets or exceeds the requirements specified by Savannah River National Laboratory, using locally available materials. A similar grout design process (where an existing formula was adapted to use local materials) has already been successfully performed by CNL (Golder 2018a, 2019a). The formulations have been tested to validate their performance against the required and assumed properties, to confirm they perform as well as or better than estimated in the solute transport model, prior to the installation of any grout into WR-1 (Golder 2019a).

Preliminary screening of the grout formulation using various combinations of locally available materials was completed, with many of the formulas easily meeting all the key target criteria. The primary candidate for the bulk fill formula was carried forward for a second phase of testing (i.e., Stage 2 testing) and refinement. The current bulk fill grout formulation is given in Table 4.1.1-2.

Table 4.1.1-2: Grout Fill Formulation

Material	Quantity per m^3
Portland Cement	89 kg
Fly Ash	297 kg
Sand	1,570 kg
Gravel	0 kg
Water	232 kg
Polycarboxylate Polymer	1.77 L
Diutan Gum Based Viscosity Modifying Admixture	260 g

4.1.1.3 Building Foundation

4.1.1.3.1 Internal Walls

Internal building walls and floors may provide an additional barrier between sections of grout; however, for conservatism this is not relied on in the safety analysis. While penetrations exist in these interior walls (to allow services to pass between rooms), they are mostly sealed for operational purposes, such as fire-stopping. Any remaining penetrations will be plugged by grout during the grouting process and are small in relation to the walls themselves. The internal walls and floors are largely painted or otherwise coated to seal the concrete to protect it from wear, provide traction, or for preventing internal contamination of the concrete. These coatings provide a waterproof barrier in many locations and will limit the speed at which water may move in to, and out of, the structure. This will further limit the speed at which the system components or grout may degrade, and the rate at which the contamination can leave the grouted areas. The sealants on these walls must first degrade to allow more prominent water movement in the materials before degradation of the concrete can begin. Concrete degradation, like the grout, is expected to occur over longer time periods, and occur gradually.

For the purposes of the groundwater flow modelling the internal structure within the WRDF is not relied on for limiting the transport of solutes or the degradation of materials. The groundwater flow model considers the grout within the WRDF and all components therein as a uniform material with bulk hydraulic properties.

4.1.1.3.2 Existing Building Foundation

Following decommissioning of the underlying crawl space and above-grade portion of the PHT system, the foundation walls and floor slab for the central reactor portion of the WR-1 Building will remain to form a barrier for preventing the release of contaminants from the WRDF. The building foundation was constructed with 3,500 psi reinforced concrete with a variable thickness ranging from approximately 0.5 m to 0.6 m for the vertical walls and approximately 1.3 m for the floor slab. As part of closure activities, any perforations in the foundation will be filled and sealed. Analysis of core samples taken from the building foundation resulted in estimated hydraulic conductivity values ranging from 1.6E-11 m/s to 9.8E-11 m/s (Golder 2019a). The value used to represent the foundation in the modelling assessment was approximately five times higher than the maximum measured value (i.e., 5E-10 m/s; Golder 2021).

The external walls or concrete surround of WR-1 Building provide a discrete, continuous, engineered barrier to the release of contamination. Penetrations in the exterior walls are given engineered seals to provide long-term performance. Like the internal walls, the exterior walls largely have sealants that restrict water movement. As with the internal walls and grout, degradation of the exterior walls is expected to occur over long periods of time and occur gradually.

4.1.1.4 Concrete Cap and Engineered Cover

After grouting has been completed and the other portions of the WR-1 Complex are demolished, a reinforced concrete cap will be constructed on top of the grouted area. The concrete cap can serve as a deterrent to prevent anyone who might inadvertently enter the area from being able to contact building contamination (US DOE 2013). It will also resist animal and plant intrusion into the WRDF. The cap will be made of reinforced concrete and is planned to be at least 850 mm thick in all areas. It will have a design life of 100 years, aligning with duration of institutional control, and will not require maintenance.

An engineered cover (earth cover) will be installed over the concrete cap to:

- direct surface water away from the WRDF and limit water infiltration;
- protect the concrete cap from the environmental elements including freezing and thawing cycles; and
- support the growth of native vegetation (grasses, shrubs), where possible.

Design of the concrete cap and engineered cover will also provide:

- erosion protection to reduce the rate of erosion due to action of surface water and wind;
- subsidence protection so that settlement of the foundation soils can be withstood;
- structural stability, specifically designed to withstand potential severe weather events and seismic events; and
- intrusion barrier to prevent accidental intrusion (e.g., excavation or drilling).

4.1.1.5 Geological and Hydrogeological Characteristics

The surrounding geosphere provides natural barriers for long-term safety during post-closure as the WRDF will be located below-grade (see Section 2.3.1.1 The WR-1 Building). At the outer edge of the building foundation, there is the backfill surrounding the WR-1 Building and beyond the backfill is bedrock overlain by unconsolidated silt and clay deposits. The geological setting for the WRDF is described in Section 2.1.3 Geologic and Hydrogeologic Environment. The soil conditions at the WR-1 provide an additional barrier to the release of contamination into the environment. The local soils are primarily clay-based and provide a natural barrier to groundwater movement. The soils and underlying bedrock provide a final barrier to groundwater movement, with relatively low groundwater velocities (around 5 m per year) and the ability to chemically sorb contaminants to further reduce or delay their concentrations in any surface water releases.

4.1.1.6 Post-closure Monitoring

The most informative method of environmental monitoring for the performance of the WRDF during post-closure is the monitoring of the groundwater surrounding WRDF. Groundwater monitoring provides verification that the WRDF, and the barriers to release, are performing their function as expected. Monitoring also provides the best system of detecting something unexpected may have occurred, and as well provide data necessary to make decisions with regard to potential mitigating actions.

4.1.1.7 Evaluation of Adequacy of Containment

The WRDF will meet the safety criteria required. The degradation of the concrete cap and engineered cover as represented in the groundwater flow modelling assessment results in the infiltration rate and groundwater flow through the WRDF incrementally increasing over time (Table 4.1.1-3).

Table 4.1.1-3: Infiltration and Simulated Flow Rates through the WRDF Components Over Time

Time Following Decommissioning (yr)	Model Parameters				Simulated Flow (m^3/d)		
	Grout and Cover K (m/s)	Foundation K (m/s)	Cover Recharge (mm/yr)	Backfill Recharge (mm/yr)	Grout	Foundation	Backfill
0	5.0E-08	5.0E-10	8.0E-01	2.0E-01	6.2E-03	1.1E-02	1.67E-01
500	1.00E-07	1.00E-09	1.6E+00	2.0E-01	1.2E-02	2.2E-02	1.69E-01
1,000	2.0E-07	2.0E-09	3.2E+00	2.0E-01	2.4E02	4.3E-02	1.73E-01
2,000	5.0E-07	5.0E-09	8.0E+00	2.0E-01	5.3E-02	9.3E-02	1.80E-01
5,000	5.0E-07	5.0E-08	8.0E+00	2.0E-01	1.2E-01	2.4E-01	1.86E-01
10,000	5.0E-07	5.0E-07	8.0E+00	2.0E-01	1.3E-01	2.8E-01	1.93E-01

K = hydraulic conductivity; m^3/d = cubic metres per day' m/s = metres per second; mm/yr = millimetres per year.

Table 4.1.1-3 provides an evaluation of the timeframes for which the degraded performance of each barrier occurs. The adequacy of the defence-in-depth is evaluated through the safety assessment during the normal evolution and disruptive scenarios and takes into consideration the simulated flow rates through the WRDF. As passive features, each barrier is assumed to degrade over the post-closure timeframes.

As defined in *IAEA SSG-23, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste* (IAEA 2011) the containment safety function is considered robust, if demonstrated through the safety assessment, it can adequately slow radionuclide migration to a level below the applicable acceptance criterion.

The grouting formulation and design of the concrete cap and engineered cover (planning and documentation) will adhere to CNL's Management System Documents including design authority and design engineering program description (CNL 2020e) and program requirement (CNL 2020f).

4.1.2 Isolation

Isolation is achieved through the combined natural and engineered barriers, which provide safety following the decommissioning of the WR-1. Isolation is provided through the location and design of the Project, by filling much of the WR-1 ISD envelope with grout and placing a reinforced concrete cap over the footprint of the building, making potential human access more difficult. Natural barriers are provided by the location of the WR-1 Building. For example, being well beneath the low permeability soil surrounding the WR-1 and the downward hydraulic gradient will prevent any potential contamination from moving upwards to the typically occupied surface and accumulating in soil and transferring to the environment in the immediate vicinity of the WRDF.

Engineered barriers include the concrete cap and engineered cover as described in Section 4.1.1 Containment. The concrete cap and engineered cover will serve in part as an intrusion barrier, and the obvious man-made nature of the materials will warn future generations of the potential danger of the WRDF. The concrete cap and engineered cover are common features of waste repositories. They are designed to prevent human intrusion by providing a hard barrier to actively resist drilling and excavations should someone bypass the earth cover. Should the concrete cap and engineered cover be breached by human intrusion the wastes are further isolated through the grout used within the cavities of the WRDF, as described in Section 4.1.1.2 Grout and Section 3.1.2.1.3 Create Grout Flow Paths.

In addition to natural and engineered barriers, institutional controls are established to protect humans and the environment. In accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), institutional controls limit the residual risk of the facility after it has been decommissioned. Institutional controls can have active measures (requiring activities on the site such as water treatment, monitoring, surveillance and maintenance) and passive measures (e.g., that do not require activities on the site, such as land use restrictions, or markers). Both active and passive measures contribute to isolation. For additional information on institutional controls refer to Section 10.0 Institutional Control.

As defined in *IAEA SSG-23, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste* (IAEA 2011) the isolation safety function is considered robust, if demonstrated through the safety assessment, it can adequately slow radionuclide migration to a level below the applicable acceptance criterion.

4.1.3 Robustness

The robust nature of the Project relies on several elements:

- demonstration of the robustness of the individual barriers and their safety function;
- evaluation of the concept of defence-in-depth (i.e., the presence of multiple diverse safety functions to provide that the overall performance of the multi-layered ISD system does not rely on a single safety function);
- verification and good engineering practices have been applied; and
- demonstration that safety can be achieved by passive means.

The robustness of the WRDF is demonstrated by evaluating a range of conservative bounding scenarios, using a suite of different calculation variations, and using conservative models and data inputs. A systematic and transparent process for developing and analysing models to evaluate post-closure safety has been used to build confidence in the assessment. The assessment approach provides the capability to explore and evaluate alternative assumptions within a range of scenarios. This enables uncertainties to be explored and facilitates the optimization of the Project design.

Model uncertainty is further addressed by using defensible models that are supported by guidelines or standards, where possible. Measurement data, where available, were used as inputs to calculations, for performing comparisons between results, or for limiting the range of variations for a scenario. When measured data were unavailable, data obtained from literature/guidance documents was used. The parameter values chosen are conservative and defensible, and due to the inherent conservatism, they serve to test the robust nature of the Project. Resulting calculations are therefore more likely to overestimate the risks from the WRDF. Criteria have been defined to prevent an unacceptable level of risk to people or the environment. The robust nature of the Project is further demonstrated by applying deliberate conservative assumptions to the models.

4.2 Safety and Design Principles

As described in Section 4.1 Safety Objectives, one of the key objectives of the Project is to reduce radiological and non-radiological exposure to the public and workers and to limit the risk of industrial hazards (e.g., working at heights). The CNL and international safety and design principles to be applied to the Project include:

- defence-in-depth principle;
- ALARA principle; and
- nuclear safety culture.

Safety and design principles are used so that off-site releases meet the constraint dose limit, the public limit, and the ALARA principle. In addition to these principles, the design and implementation of the Project will also use Canadian and international best practices and safety fundamentals, including those from the IAEA and the CNSC. The safety strategy for the Project draws upon the experience and lessons learned from the decommissioning of other similar facilities. While ISD has not yet been completed in Canada, the US Environmental Protection Agency (US EPA) and the US DOE have recognized encapsulation as a decommissioning option since the 1970s (US DOE 2013). The ISD approach has been used or is planned to be used on a variety of projects in the United States including:

- two large reactors (P and R) and their ancillary facilities at the Savannah River Site;
- fuel processing facilities at Idaho National Laboratory; and
- the below-grade portion of several small reactors' facilities at Idaho National Laboratory and one at Savannah River Site.

These projects were completed under a combination of US DOE authority and US EPA and State regulations. Experience with these projects determined that ISD, in many cases, is the safest, timeliest and most cost-effective decommissioning option compared to demolishing and excavating the entire facility and transporting the rubble to a radioactive waste landfill (US DOE 2013). Principles of good engineering practice that have been applied in the design of similar projects have been considered in the Project design. The materials and construction techniques to be used are well understood, and that knowledge gained from similar applications (e.g., at Savannah River Site) confirms these materials are well suited for their intended purpose.

4.2.1 Defence-in-Depth Principle

The ISD approach is designed to control the rate of release of nuclear and hazardous substances from the WRDF and retain the waste away from people and the environment. The design considers possible events that could degrade the integrity of the WRDF. It is recognized that there are inevitable uncertainties associated with predicting the performance of the WRDF over a long-time scale (i.e., thousands of years); therefore, safety is established through adequate defence-in-depth and verified through long-term environmental monitoring during institutional control.

The key aspect of the defence-in-depth principle is the provision of multiple layers of protection against abnormal events. In other words, the safety performance of the WRDF Project is not dependent on any single safety function. When defence-in-depth is applied to all activities during the life cycle of the Project, it provides protection against a wide range of events (e.g., anticipated operational occurrences, accident conditions, equipment failure, or human error within the facility) and from events that originate outside the WRDF (e.g., forest fires or floods).

Defence-in-depth consists of two components: 1) equipment and administrative features that provide preventative or mitigation to a degree proportional to the hazard potential; and 2) integrated safety management programs that control operations.

The application of the defence-in-depth principle to abnormal events requires that no single human or equipment failure would result in an unacceptable hazard to the workers in the facility, the public, or the environment. This principle is centred on several barriers of protection with ascending levels of importance: accommodation, mitigation, and prevention. The level of application corresponds to the level of risk posed by the postulated event.

To effectively control a hazard, the same basic approach has been taken based on the following hierarchy of hazard control principles:

- eliminating the hazard;
- reducing/replacing the hazard;
- isolating the hazard;
- controlling the hazard;
- personnel protective equipment and clothing;
- policies and procedures; and
- appropriate documentation.

Of these seven principles, the most effective is to eliminate the hazard, and wherever possible this will be the preferred method of hazard control for the Project. When a hazard cannot be eliminated, the remaining principles are implemented to varying degrees to provide an acceptable level of defence-in-depth.

The defence-in-depth principle is used to compensate for potential mechanical and human failure and unexpected occurrences. A series of barriers will prevent or reduce the likelihood of a radioactive release to the environment. For human errors, prevention is achieved by a combination of process design and administrative controls (e.g., training and procedures), as well as by establishment of a strong safety culture.

4.2.2 As Low As Reasonably Achievable Principle

The ALARA principle requires that the exposures to people shall be as low as reasonably achievable, social and economic factors being taken into account. CNL’s ALARA program will be followed and the essential elements include:

- demonstrated management commitment to the ALARA principle;
- implementation of ALARA through design, organization and management, selection and training of personnel, oversight of the Radiation Protection Program, resources, and documentation;
- establishment of nuclear safety culture;
- planning and control of all work;
- application of task-specific dose and dose-rate radiological control hold points; and
- performance of regular operational reviews.

In addition to CNL's ALARA program, the CNSC's *Regulatory Guide G-129 Keeping Radiation Exposures and Doses "As Low as Reasonably Achievable"* (CNSC 2004) is taken into consideration in the Project design. This document outlines approaches to achieving ALARA through the management control over work practices, qualification and training for personnel, control of occupational and public exposure to radiation, and planning for unusual situations.

The ALARA principle for the Project implementation will be achieved by:

- implementing zoning and access control measures;
- maintaining adequate shielding for structures and waste packages with high radiation fields;
- providing process equipment segregation;
- establishing radiation alarms;
- continuous monitoring; and
- implementing operator training and approved procedures.

Optimal protective measures against the hazards of ionizing radiation will be reached when further reductions in radiation doses are outweighed by the additional efforts and costs required for their implementation. This principle applies to all phases throughout the life cycle of the Project, from decommissioning and closure to post-closure, and is a particularly important consideration when developing the decommissioning procedures.

4.2.3 Nuclear Safety Culture

CNL has adopted the Institute of Nuclear Power Operations' nuclear safety culture definition (INPO 2004)[6]. The following principles and traits are well recognized to contribute to a healthy nuclear safety culture.

- **Personal accountability** – All individuals take personal responsibility for safety.
- **Questioning attitude** – Individuals avoid complacency and continuously challenge existing conditions and activities in order to identify discrepancies that might result in error or inappropriate action.
- **Effective safety communication** – Communications maintain a focus on safety.
- **Leadership safety values and actions** – Leaders demonstrate a commitment to safety in their decisions and behaviors.
- **Decision-making** – Decisions that support or affect nuclear safety are systematic, rigorous and thorough.
- **Respectful work environment** – Trust and respect permeate the organization.
- **Continuous learning –** Opportunities to learn about ways to ensure safety are sought out and implemented.
- **Problem identification and resolution –** Issues potentially affecting safety are promptly identified, fully evaluated and promptly and adequately addressed (corrections are commensurate with the potential consequences).

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[6] Nuclear safety culture" is defined as the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.

- **Environment for raising concerns** – A safety-conscious work environment is maintained where personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment or discrimination.
- **Work processes** – The process of planning and controlling work activities is implemented so that safety is maintained.

CNSC's *REGDOC-2.1.2 Safety Culture*, *Management System: Safety Culture* (CNSC 2018a) provides a definition of safety culture and highlights general safety culture requirements that apply to all licensees. The document describes the expected and suggested criteria for licensees to self-assess, establish corrective action plans, and report on safety culture. This document was considered in the development of the safety assessment completed for the Project.

5.0 ASSESSMENT APPROACH

The assessment approach was selected to provide reasonable assurance that the Project and the management of radioactive waste are consistent with all applicable requirements. This section includes a description of the general approach used to demonstrate safety over the long-term, confidence in the results, and how the approach addresses the principles of radioactive waste management in CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The assessment approach uses a combination of complementary analyses at various levels of detail.

Central to the assessment approach is the development of assessment acceptance criteria, timeframes, and scenarios. The assessment approach applied was selected to provide reasonable assurance that the Project and the management of radioactive waste are consistent with applicable requirements. This section describes the approach used to demonstrate safety over the long-term, confidence in the results, and how the approach addresses the principles of radioactive waste management identified in CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), and demonstrates the safety of the WRDF during the closure and post-closure phases.

The assessment strategy includes the following:

- implementation of scientific and engineering principles;
- facility design selection and alternative options evaluation;
- defining defensible assessment acceptance criteria and endpoints;
- identification of assessment timeframes;
- identification of assessment scenarios with consideration of the factors that are important to long-term safety;
- development of bounding scenarios to show the limits of the potential effect during the closure and post-closure phases using conservative calculations that intentionally over-estimate the potential effect; and
- completion of deterministic calculations, with sensitivity assessment to demonstrate understanding of the assessment model, reflecting the long-term timeframes and reflecting associated data uncertainty over the assessment timeframes.

The criteria and scenarios demonstrate protection of people and the environment from both radiological and non-radiological hazards and encompass the time period when maximum effects are predicted. A wide range of scenarios are considered to develop an understanding of the system and provide a thorough safety case for the WRDF. These scenarios include the normal evolution of the site (i.e., likely conditions to be experienced), as well as disruptive events that encompass less likely conditions (i.e., upset conditions). Scoping assessments provide a general understanding of the overall system to identify aspects that are critical to safety; whereas bounding assessments provide the limiting estimates of the system performance. Evaluation of limiting values from disruptive events and identification of aspects critical to safety, provide a useful check on the long-term assessment calculations and improve confidence in the prediction of safety.

For the safety assessment, the Project is overlaid on the existing environment, then systematically, potential variation in the performance of engineering and administrative controls, environmental conditions, and human population dynamics are considered. Several analyses are undertaken to identify the credible incidents to be examined: HAZOP, Accidents and Malfunction Analysis, and FEPs Analysis. An Environmental Risk Assessment (ERA) is completed to define exposure predictions for the Normal Evolution Scenario in accordance with CSA N288.6 (CSA 2012). Additional risk assessment work is completed to encompass disruptive events as part of the safety assessment. The exposures predicted are compared to assessment acceptance criteria to determine if the Project can be safely completed and will be protective of the public and environment over the long-term.

The safety assessment is central to the safety case and evaluates the overall performance of the WRDF and any effect on human health and on the environment. The evaluation includes potential long-term effects arising from radioactive waste or residual contamination. The assessment uses a pathways analysis, based on the expected evolution of the WL site and the WRDF, to predict:

- contaminant release;
- contaminant transport;
- receptor exposure; and
- potential effects resulting from the exposure.

The best available information on the site characterization, existing source term inventory, and the Project design have been used in the safety assessment (i.e., the assessment is based on an appropriate level of understanding of the disposal system and its potential behaviour). To manage uncertainty, an appropriate degree of conservatism was integrated into the development of modelling inputs and assumptions for quantifying contaminant releases, contaminant transport and receptor exposure. In some cases, this included simplifications of inputs to make them more amenable for inclusion in assessment models. A sensitivity analysis was completed to illustrate the robustness of the solute transport modelling. Relevant outcomes from the sensitivity analysis were carried forward into the ERA as bounding scenarios and residual effects were analyzed for their potential to affect human and ecological health. Additional details on the models used for the safety assessment are provided in Section 5.5 Conceptual and Deterministic Models.

The net effect of all assumptions within the safety assessment is that it depicts a conservative representation of long-term effects and risk and intentionally over-estimating future consequences. This provides an additional margin of safety for situations where the results of the safety analyses are considered as indicators of safety, rather than accurate predictions given the long timeframes.

5.1 Scientific and Engineering Principles

Applicable principles of good engineering practice are applied in the design of the WRDF, and the materials and construction techniques foreseen for the Project are well understood. Knowledge gained from other projects world-wide confirms that these materials are well-suited for the intended decommissioning approach. To demonstrate this, the guidance given in IAEA SSG-23 has been applied, in particular by making observations, testing hypotheses, assessing reproducibility, and performance of peer reviews.

Data used for the safety assessment has been developed from results of waste characterization activities and obtained from a number of sources, including published literature and reference documents or as design data from the WR-1 Building and CNL documents. Site-specific data and information are used where possible. When unavailable, data have been selected from recently published, relevant information sources. Where assumptions have been made, the validity of the assumptions and appropriateness of the data selection was confirmed by appropriate peer reviews.

Peer reviews are performed as part of the WL Quality Assurance program's review process implemented through CNL's procedure *Functional Instruction Document Processes to Create, Capture and Use Records* (CNL 2019b), using a formalized system to capture such review. At WL, this process provides for the proper application of scientific and engineering principles. The review considers the technical details and the associated justification.

This also includes checking that the work addresses relevant regulatory standards and applies national and international best practice guidance.

The structure of the WR-1 is constructed below-grade, built of reinforced concrete and set on bedrock. Reinforced concrete, and its design and application are well understood. The design also makes use of well-understood construction techniques. Specifically, the use and placement of grout inside the facility, and in the construction of the concrete cap that forms part of the engineered barrier. The engineered barrier has also incorporated lessons learned from other waste disposal projects, such as the design of municipal landfills with respect to the use of geomembranes and construction of earthen mounds. Similar facilities around with world have been studied so that sound engineering principles and lessons learned have been incorporated into the design of the WRDF.

The appropriate Canadian and international standards have been identified to confirm that robust scientific and engineering principles are applied rigorously to the design and construction of the WRDF. Specifically, the requirements of the following standards and guides have been identified for the design:

- CNSC REGDOC-2.11.1 Vol. III Assessment the Long-term Safety of Radioactive Waste Management;
- CSA N292.3-14 Management of Low- and Intermediate-Level Radioactive Waste;
- CSA N294-09 (R2014) Decommissioning of Facilities Containing Nuclear Substances;
- CSA A23.3-12 Design of Concrete Structures;
- IAEA Safety Standard SSG-29 Near Surface Disposal of Radioactive Waste;
- IAEA Safety Standard SSG-31 Monitoring and Surveillance of Radioactive Waste Disposal Facilities; and
- IAEA Safety Standard SSR-5 Disposal of Radioactive Waste.

The closure phase of the Project requires that application of what would be considered as standard demolition and construction techniques. These include the removal of large structures, rubblizing concrete, strategically removing asbestos, operating heavy equipment, and pouring concrete/grout, while providing dust suppression and ventilation control. Each of these activities will be performed in accordance with the DDP and emphasis will be placed on the following:

- structural assessment will be performed so that all demolition work is performed in accordance with safe work practices and civil/engineering requirements;
- appropriate methods will be used during demolition process to provide support to structures so that unintended structural failure does not occur; and
- a plan for demolition of structural components will be developed and followed.

For post-closure, the principles of good engineering practice are implemented by considering the design, by employing these principles to slow the degradation of the WRDF, containing the waste, and maintaining safety performance of the facility. Engineering and design requirements for disposal are based on both containment and isolation features of the WRDF. Robust containment is achieved because the design uses multiple barriers to passage of contaminant, providing defence-in-depth, and providing a period of institutional control. In addition, the grout design has been developed based on experience gained during ISD of reactor facilities at the US DOE Savannah River Site. Conservatism is built into the facility design by assessing normal evolution and disruptive event scenarios.

5.2 Assessment Acceptance Criteria and Endpoints

Central to the assessment strategy is the development of assessment acceptance criteria. The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL (CNSC 2020a), including Action Levels and compliance programs. Further, the 50 years of operating experience at the WL site provides important context for the acceptance criteria for decommissioning the WL site. This includes the establishment of proven work and effects management plans, as well as monitoring programs.

The following sections provide context on the endpoints for the assessment criteria. Assessment endpoints are derived from regulatory requirements and international guidance as applicable (Section 1.2 Regulatory Requirements and Guidance Documents). Assessment endpoints are based on a comparison of the assessment results against the criteria to be met for radiological and non-radiological effects on human and non-human biota. Criteria to be met are established based on regulatory limits and objectives, or where such limits do not exist other scientifically justifiable benchmarks. Complementary safety indicators, such as barrier performance, can be used to enhance the confidence in the assessment results.

The endpoint for the Project is defined as the time when maximum radiological effect occurs as suggested by CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The maximum radiological effect is determined to be the peak dose received by the most exposed person.

5.2.1 Human Health

5.2.1.1 Radiological

Whiteshell Laboratories maintains a Radiation Protection Program, which provides a management framework and processes that are designed to confirm that radiation exposures arising are maintained below regulatory dose limits and are kept ALARA. The operational limits for the WL site are specified in the Licence Conditions Handbook for WL and encompass decommissioning activities.

CNSC Radiological Dose Limits

As part of normal operations, nuclear reactors release, in a controlled manner, small quantities of radioactive substances into the atmosphere and geosphere. These releases are regulated and carefully monitored by the CNSC. Doses received by members of the public from routine releases at nuclear reactors are too low to be measured directly. Therefore, the public dose limit of 1 mSv/a was used which is considered to be protective of human health. This limit applies to the WL site as a whole, within which the Project is encompassed.

The statuary dose limits for members of the public as set out in the CNSC *Radiation Protection Regulations* are presented in Table 5.2.1-1.

Table 5.2.1-1: Dose Limits for Members of the Public

Type	Application	Annual Dose Limit (mSv/a)
Effective Dose	—	1
Equivalent Dose	Skin	50
Equivalent Dose	Lens of Eye	15
Equivalent Dose	Hands and Feet	50

mSv/a = millisievert per year.

While Table 5.2.1-1 lists a number of dose limits, the limiting dose restriction implemented at the currently operational WL site is the 1 mSv/a Effective Dose limit. The WL site derived release limits (DRLs) presented in this document are therefore based on the 1 mSv/a Effective Dose limit.

Whiteshell Laboratories Operating Derived Release Limits

As part of normal operations, the WL has calculated derived release limits (DRLs) (CNL 2016c) to confirm releases from existing facilities are below the regulatory public dose limit. Derived release limits are required because radioactive materials released into the environment through gaseous, particulate, and liquid effluents from nuclear power plants, can expose members of the public to low radiation doses via external and internal pathways. External exposure occurs from direct contact with radionuclide-contaminated ground surfaces, or by immersion into contaminated water and air; internal exposure occurs through the inhalation of contaminated air and/or consumption of contaminated foods or water. Such radiation doses to members of the public are subject to statutory limits, which are set out in Sections 13 and 14 of the CNSC *Radiation Protection Regulations* (shown in Table 5.2.1-1). Operating DRLs for the WL site (including the Project) were established in accordance with CSA *N288.1-08: Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities* (CSA 2014).

Whiteshell Laboratories Administrative Levels and Action Levels

The DRLs would result in a cumulative annual dose of 1 mSv to members of the public if the facility released radioactive material at the DRLs. In addition to the regulatory limit of 1 mSv/a, nuclear facilities are required to establish internal operating constraints as part of their ALARA programs.

Nuclear facilities maintain their own internal operating targets, which equate to approximately 1% of the specified DRLs. As outlined in the CNL's Environmental Protection Program, Administrative Levels and Action Levels have been established as internal operating targets. An Administrative Level is a CNL internal reporting level for radioactive emissions through an individual effluent stream, established to provide timely warning of abnormal radioactive emissions, with the intent of aiding in the application of the ALARA process (i.e., avoiding and an Action Level). An Action Level is a level of radioactive emissions that if reached may represent a loss of control of performance for the facility. Releases above this limit must be investigated and reported to CNSC staff. These levels are documented in CNL's *Derived Release Limits for AECL's Whiteshell Laboratories* (CNL 2016c) and will be applied to the Project as necessary throughout closure.

In addition, although DRLs are expressed as an annual release limit, the weekly and monthly rates of release are further controlled. For gaseous releases, the maintained weekly limit is the annual DRL divided by 52 weeks, while liquid monthly release limits represent the annual DRL divided by 12 months. Weekly airborne releases and monthly liquid releases are compared to the respective weekly and monthly DRLs and are reported to the CNSC on an annual basis.

Operational Environmental Monitoring

Under the *Nuclear Safety and Control Act*, licensees of nuclear facilities are required to implement an environmental monitoring program to demonstrate that the public and the environment are protected from emissions related to the facility's nuclear activities. The monitoring program's results are submitted to the CNSC to demonstrate compliance with applicable guidelines and limits, as set out in regulations that oversee Canada's nuclear industry.

CNL maintains a comprehensive site Environmental Monitoring Program for the WL site (CNL 2019c). This program is consistent with guidelines in the National Standard of Canada, CAN/CSA-288.4-10. The purpose of the program is to confirm that radiation doses caused by releases of radioactive material in site effluent remain below the annual dose limit for members of the public, as specified in the regulations made pursuant to the *Nuclear Safety and Control Act*. The program includes the sampling of several environmental media and sample locations serving as trend indicators. Monitored environmental media include ambient air, precipitation, surface and groundwater, (including the Winnipeg River), vegetables, fish, game and river sediment.

WR-1 Post-closure Radiological Release Limits

For the post-closure phase, there is the potential for radiation exposure and environmental effects over time periods far into the future. Some effects may be assumed to occur, for example, gradual leaching of radionuclides into groundwater and subsequent migration through environmental media and transfer to humans. Therefore, the safety assessment must predict the behaviour of the WRDF for time periods of hundreds or thousands of years (see Section 5.3 Timeframes).

As per CNSC's *REGDOC-2.11.1, Volume III*, the long-term safety assessments of the WRDF will provide a reasonable assurance that the regulatory radiological dose limit for public exposure (currently 1 mSv/a) will not be exceeded (CNSC 2018a). However, to account for the possibility of exposure to multiple sources and to help confirm that doses resulting from the facility being assessed are ALARA, an acceptance criterion that is less than the regulatory limit will be used. This establishes that all releases combined will not cause a member of the public to receive a dose in excess of the public dose limit (1 mSv/a).

For design optimization of the closure and post-closure phases, the International Commission on Radiological Protection recommends a design target, referred to as a "dose constraint", of no more than 0.3 mSv/a. While the dose constraint is used as a design target in the optimization process, it is not used as a limit for compliance.

For the Project, the established total dose constraint of 0.25 mSv/a was applied to the public from all radionuclides and all pathways for the long-term expected evolution of the site (CNL 2016a). This post-closure radiological release limit (0.25 mSv/a) is conservative, representing approximately a 17% decrease from the International Commission on Radiological Protection recommended design target dose constraint of 0.3 mSv/a. This constraint (below the 1mSv per year limit) is adopted in accordance with REGDOC 2.11.1 (CNSC 2019) and SSG-29 (IAEA 2014a), to account for other potential exposure sources for the public.

For predicted effects outside of the Normal Evolution Scenario, international guidance is considered. If human intrusion is expected to lead to an annual dose of less than 1 mSv/a to those living around the site, then efforts to reduce the probability of intrusion or limit its consequence are not warranted. As outlined in IAEA SSR-5 (IAEA 2011), if human intrusion is expected to lead to a possible human dose of more than 20 mSv/a to those living around the site (i.e., disruptive events), then alternative options for waste disposal are to be considered (e.g., source term reductions). Deterministic effects will be prevented if effective whole-body annual dose exposures are limited to 20 mSv/a. If the predicted range of doses is 1 to 20 mSv/a then reasonable efforts are warranted to provide mitigations reduce the probability of intrusion or limit its consequence.

5.2.1.2 Non-radiological

The primary concern for the Project is radioactivity; however, some hazardous non-radiological materials still remain in the WR-1 Building including:

- asbestos (friable [contains more than 1% asbestos by weight or area and can be crumbled by the human hand] and non-friable [material that contains more than 1% asbestos and cannot be crumbled under hand pressure] containing materials);
- organic coolant (hydrogenated terphenyl [HB-40] used as reactor coolant, also known as OS-84) in the PHT system and tanks;

- lead-based paint and lead shielding;
- PCBs after fluorescent light fixture ballasts still remaining in the WR-1 Building have been remediated the presence of PCBs within the WRDF will be limited [e.g., paint];
- small quantities of mercury in thermostats, switches and active drain lines; and
- mould.

With the exception of HB-40, the identified hazardous substances are routinely addressed in construction projects. For details pertaining to the location of these non-radiological hazards within the WR-1 Building refer to the DDP (CNL 2021b). During the closure phase, the exposure pathway to humans is through inhalation. During the post-closure phase, the exposure pathways to humans is through effects to groundwater and ultimately surface water.

Public exposure criteria adopted for airborne non-radiological hazardous substances are shown in Table 5.2.1-2.

Table 5.2.1-2: Exposure Criteria for Airborne Non-Radiological Hazardous Substances

Non-Radiological Hazardous Material	Applicable Air Quality Guideline (AAQC) (μg/m^3)	Reference
Potassium Hydroxide	14	Ontario AAQC (corrosion)(a)
Boron	120	Ontario AAQC (particulate)(a)
Lead	2	Manitoba AAQC (MAC)(b)
Xylene	730	Ontario AAQC (health)(a)
Palladium	10	Ontario AAQC (health)(a)
Chromium	0.5	Ontario AI (health)(a)
Cadmium	2	Manitoba AAQC (MAC)(b)
HB-40	500	OSHA(c) TWA/10
Mercury	2	Ontario AAQC (health)(a)

a) MOE 2012

b) Manitoba Conservation 2005

c) OSHA 1989 Time Weighted Average (TWA) of 5 mg/m^3 for hydrogenated terphenyl cited in Material Safety Data Sheet (MSDS) for HB-40 (Eastman Chemical Company 2015)/

Other non-radiological COPCs have been identified as potentially remaining in the WR-1 system, such as asbestos and mould. Mould and asbestos are hazards that are routinely addressed at CNL within approved procedures that outline the process for safely performing work on or near these materials (CNL 2017e, 2019h). Following these procedures ensures these materials are managed within the required regulations and limits and in accordance with standard practice.

The organic coolant, HB-40, consists mainly of hydrogenated terphenyl (74% to 87%), with smaller fractions of partially hydrogenated terphenyls and terphenyl. Occupational Safety and Health Administration (Section 142.3 of the Labour Code) permissible exposure limit for hydrogenated terphenyl is 5 mg/m3 (OHSA 2017) (refer to Section 4.1.3 Selection of Exposure pathways of the ERA [EcoMetrix 2021]).

For liquid effluent, the maximum predicted concentration in groundwater at the Winnipeg River were compared to the following criteria:

- Health Canada Canadian Drinking Water Quality Guidelines (Health Canada 2017);
- Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011);
- CCME Water Quality Guidelines for Protection of Aquatic Life (CCME 1999);
- Ontario Provincial Water Quality Objective, which are assumed by Ministry of Environment and Climate Change to be protective of human health (MOEE 1994); and
- British Columbian Water Quality Guidelines (BC MOE 2017).

HB-40 does not have federal or provincial drinking water quality guidelines; however, Weeks (1974) derived a safe drinking water concentration for workers of 232 mg/L. This is based on a minimal effect concentration of 250 milligram per kilogram per day (mg/kg/d) in mice, divided by 100-fold safety factor. A safe HB-40 drinking water concentration for members of the public at the WL site is estimated at 8.8 mg/L based on a toxicity reference value (TRV) of 0.25 mg/kg/d (250/1000), a drinking water intake rate of 2 litres per day (L/d) and body weight of approximately 70 kg.

Non-radiological Risk Characterization

Risk will be quantified for each human receptor category based on the calculation of a hazard quotient (HQ) for non-carcinogenic COPC. The HQ is calculated for each receptor for each hazardous (non-radiological substance), using toxicity reference values. If the HQ for a non-radiological COPC is less than 0.2 per medium, then no adverse effects are likely as concentrations are below levels that are known to cause adverse effects. If the HQ is below 0.2 per medium, it may be inferred that adverse effects are not likely. The potential effects are those associated with the benchmark exposure level if an HQ exceeds 0.2, and further assessment is required. In general terms, an increase in exposure above the benchmark level is associated with an increase in risk. As the magnitude of the HQ increases so does the potential for environmental effects, the likelihood of the effect depends on the magnitude of exposure and the endpoint used to assess effects.

5.2.2 Non-human Biota Protection

5.2.2.1 Radiological

For the protection of non-human biota from radiation exposure, the primary concern is the total radiation dose to the organisms resulting in deterministic effects. The development of benchmarks for radiation protection of non-human biota is not as mature as the development of benchmarks for hazardous substances, due to the historic assumption that protecting humans from radiation is sufficient to protect the environment. However, benchmark values for mean radiation doses to non-human biota have been derived for various types of organisms (IAEA 1992).

Radiation Benchmarks

Aquatic and terrestrial biotas may receive radiation doses from exposures to radioactivity in the atmosphere, surface water, soil, and groundwater. The criteria for assessing the potential effect of the Project on non-human biota are detailed in the ERA (EcoMetrix 2021). As recommended by the CSA N288.6-12 standard, the following radiation dose benchmarks (CSA 2012, UNSCEAR 2008) were selected:

- 9.6 milligray per day (mGy/d; 0.4 milligray per hour [mGy/h]) for aquatic biota (fish and benthic invertebrates); and
- 2.4 mGy/d (0.1 mGy/h) for terrestrial biota (includes birds and mammals with riparian habits).

These are total dose benchmarks, therefore, the dose compared to these benchmarks is the sum of the doses from each radionuclide of concern. As a dose constraint, doses to aquatic and terrestrial biota are to be at least two orders of magnitude below the criteria.

Risk Characterization

For radionuclides, a total radiation dose from all radionuclides is calculated for each receptor, and this total dose is compared to the radiation dose benchmarks. If the total radiation dose is less than the radiation dose benchmark, then no adverse effects are likely as doses are below levels that are known to cause adverse effects.

5.2.2.2 Non-radiological

Non-radiological COPCs were identified by comparing the maximum concentration of each non-radionuclide in each medium measured at the WL site to appropriate guidelines for the protection of ecological receptors. Where appropriate guidelines were not available, upper background concentrations were used as the screening criteria. Non-radionuclides with maximum concentrations exceeding the guideline values or screening criteria were identified as COPCs and assessed further to characterize risk.

Non-radiological benchmarks for aquatic biota are expressed as water or sediment concentrations. The benchmarks for terrestrial biota are expressed as soil concentrations, for plants and soil invertebrates, or as doses for high organisms, such as birds and mammals.

Risk characterization will involve calculation of a HQ for each COPC and receptor species, in each relevant exposure area. The HQ is calculated as an exposure concentration or dose, divided by a benchmark concentration or dose below which no adverse effects are expected.

For assessment of non-human biota, if the HQ for a non-radiological COPC is less than one, then no adverse effects are likely as concentrations are below levels that are known to cause adverse effects. If the HQ exceeds one, it may be inferred that adverse effects to individual are possible. The potential effects are those associated with the benchmark exposure level if an HQ exceeds one for non-human biota, further assessment is required to evaluate the potential for effects at a population level. In general terms, an increase in exposure above the benchmark level is associated with an increase in risk. As the magnitude of the HQ increases so does the potential for environmental effects, the likelihood of the effect depending on the magnitude of exposure and the endpoint used to assess effects.

5.3 Timeframes

Various timeframes are indicated within the safety assessment. These timeframes serve different purposes in the safety assessment and have different definitions. The following timeframes are used within the safety assessment.

- Assessment timeframe – The time over which the effects of the project are assessed and the Normal Evolution Scenario is defined.
- Design life – The time over which an engineered component will perform to its minimum specifications. All design lives are completed within the assessment timeframe.
- Barrier lifetime – The time over which a component degrades from fully functional, to a fully degraded final state. This period encompasses any defined design life, and a period after the design life, where the component is no longer meeting the original minimum specification, but also is not fully degraded. All barrier lifetimes are completed within the assessment timeframe.
- Glaciation timeframe – The estimated time until onset, and completion of the next glacial advance and retreat at the project site. This is independent of the assessment timeframe.
- Modelling timeframe – The output of the groundwater flow, solute transport and dose models are provided for a period of 500,000 years after closure. This timeframe is selected to provide confidence that the models have captured the peak effects of the Project. This timeframe is independent of the assessment timeframe and provides no bearing on the development of the Normal Evolution Scenario.
- Closure phase – The time during which physical construction of the WRDF is occurring. Expected to last approximately 3 years
- Post-closure phase – The time after construction of the WRDF is complete, which includes institutional control and post-institutional control:
    - Institutional control period – The time during which the CNSC or other authority having jurisdiction requires oversight of the WRDF through a licence or other regulatory means. For the purposes of the assessment, it is assumed to last a minimum of 100 years after closure of the facility, during which long-term performance monitoring and maintenance activities will continue, to demonstrate compliance with the safety case assumptions.
    - Post-institutional control occurs after the assumed loss of institutional control (~year 2125) and continues indefinitely.

The assessment timeframe is 10,000 years. The timeframe is established in compliance with the CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), which requires that, "the assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompass the period of time when the maximum impact is predicted to occur." Per Section 7.4 of CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a) there is no time limit associated with the statutory objective to "prevent unreasonable risk, to the environment and health and safety of persons…" (*Nuclear Safety and Control Act*, 9(a)(i)). Instead, the determination of the appropriate time period is part of the assessment process.

The approach taken to determine the assessment timeframe accounted for the following elements:

- hazardous lifetime of the contaminants associated with the waste;
- duration of the operational period (before the facility reaches its end-state);
- design life of engineered barriers;
- duration of institutional control; and
- frequency (probability) of natural events and human-induced environmental changes (e.g., seismic occurrence, flood, drought, glaciation, climate change).

The following sections provides rationale for the selection of the 10,000-year timeframe and how each of the criterion listed in the REGDOC 2.11.1 has been met and supports the proposed 10,000-year timeframe.

Hazardous Lifetime of the Contaminants

CNL has determined the hazard level of each contaminant in the WR-1 by calculating an annual dose rate, or HQ for the expected exposure pathways. For radionuclides, the total dose rate is the sum of dose contributions from all radionuclides; and the maximum total dose rate occurs within the 10,000-year assessment period. For most non-radionuclides, the peak HQ also occurs during the 10,000-year assessment period.

Some non-radionuclides, such as lead, reach their peak over millions of years; however, the peak HQ over millions of years is lower than the acceptable HQ and is a result of reaching maximum solubility of lead in groundwater. REGDOC 2.11.1 states “In some cases, only the magnitude of the maximum impact, independent of time, may be sufficient for the assessment (e.g., bounding assessments using calculations based on solubility constraints)”. For the WR-1 safety assessment, as the peak HQs for these non-radionuclides are constrained by their solubility limit, it is appropriate to remove the time dependency from their assessment.

Further, for all radionuclides and non-radionuclides, the peak release rate, independent of time, is used to assess the impacts on receptors. This approach removes the time dependency of the assessment during selection of an assessment timeframe, and permits the selection of a 10,000 year assessment timeframe.

Duration of the Operational Period

The effects of the construction and closure of the Project on the environment are assessed separately prior to evaluation of the 10,000-year post-closure assessment period. The timeframe for closure is 3 years and institutional control is assumed to last a minimum of 100 years. In addition to this, the WR-1 has had at least 30 years of storage with surveillance, during which time the long-term performance of the facility structures forming the engineered barrier has been studied. This provides similar experience as would be obtained during the operation of a waste disposal facility prior to closure. The relatively short period (i.e., 133 years) does not contribute significant additional time to the assessment timeframe of 10,000 years.

Design Life of Engineered Barriers

Criterion #3 is the consideration of the life of the engineered barriers and providing that the assessment timeframe considers the effects of changes or degradation of those barriers on the assessment outcomes. To meet Criterion #3, the assessment may employ one of three approaches:

1) Assume barrier life cycle is complete within the assessment timeframe;
2) Assume barrier properties are set to the most bounding conservative value in the barrier life cycle irrespective of time; or

3) Assume barrier life cycle exceeds hazardous lifetime of materials.

There are four main engineered barriers with the WRDF: the wasteform itself, the grout, the existing building walls and foundation, and the concrete cap and engineered cover. In the case of the cover, grout and foundation, approach #1 was taken and the assumed lifetime of the cover, foundation, and grout fall within 10,000 years. The assumed barrier lifetime for the cover and grout is 2,000 years, and for the foundation it is 10,000 years. The assessment assumes that these barriers degrade via a step function (with linear interpolation between steps), increasing in hydraulic conductivity and ending with natural soil conditions.

The release of contaminants from the wasteform is a slow process controlled by the dissolution of metal components. The rate of dissolution or degradation of the wasteform is dependent on the groundwater chemistry, the surface area of the wasteform and the corrosion resistance of the wasteform. For the wasteform, approach #2 was taken. The safety assessment does not examine the changes in groundwater chemistry over time, and instead uses the more conservative long-term chemistry (neutral pH) as the initial condition. The release rate from the wasteform is corrosion controlled. As a result, there is no change in the wasteform release rate over time, and the release rate from the wasteform is bounded.

Duration of Institutional Controls

The post-closure phase has two discrete periods: institutional control and post-institutional control, as described in Section 3.1.1 Project Schedule. The site is expected to remain under institutional control for a minimum of 100 years to provide a means to confirm the continued safe and effective function of the Project following site closure. During institutional control, long-term performance monitoring and maintenance activities will continue through to 2125 to demonstrate compliance with the safety case assumptions. In the assessment, it is assumed that human intrusion and disruptive events would be prevented during institutional control. Passive controls such as access restrictions (e.g., physical barriers/fencing, signage, and land title instruments/deed restrictions) will remain in place until the end of institutional control. A summary of the Project schedule is provided in Table 3.1.1-1.

This timeframe is consistent with that required for other near surface disposal projects (range of 100 to 300 years), including a similar project under CNSC jurisdiction. Examples include:

- Centre De La Manche Disposal Facility (operated by the Agence Nationale pour la Gestion des Déchets Radioactifs [ANDRA; French National Radioactive Waste Agency] in France) (Chino et al. 1999);
- L'Aube (operated by ANDRA in France) (Potier 1998);
- Rokkasho Low-level Radioactive Waste Disposal Centre (operated by Japan Nuclear Fuel Ltd in Japan) (Bergström et al. 2011); and
- Deep Geological Repository (proposed by Ontario Power Generation) (NWMO 2011).

It is recognized that institutional control will continue until the CNSC agrees institutional controls are no longer needed. The assessment assumes a minimum of 100 years of institutional control. The assessment also assumes that institutional controls cannot be relied upon as long-term barrier to the release of contaminants. Instead, institutional controls are looked at as a short-term barrier, and a mean to verify performance of the WRDF in the short-term and to provide additional confidence in the long-term safety assessment. Institutional controls are assumed to fail after 100 years; therefore, these falls within the proposed assessment timeframe of 10,000 years.

Frequency of Natural and Anthropogenic Changes

Another key consideration in the development of the assessment timeframe is the frequency of natural events and human-induced environmental changes (e.g., seismic occurrence, flood, drought, glaciation, climate change). Seismic effects are assessed through a set of conservative disruptive event scenarios occurring within the 10,000-year assessment timeframe, including accelerated engineered barrier degradation, localized fast pathways, human intrusion (exploratory drilling), and the inclusion of a fracture model. Climate change has been previously accounted for through specific scenarios, including the river level fall/discharge to shore case for drought conditions, and the erosion case to represent floods. Current understanding of the long-term effects of global warming indicated that the next ice age will not occur for 100,000 years as a result of anthropogenic climate change, at a time when the effects of an event will be insignificant compared to the peak dose rates expected during the 10,000-year assessment timeframe. By accounting for all of these scenarios early in the assessment timeframe, when the inventory is larger and has not decayed, the results of the later events are bounded.

5.4 Assessment Scenario Development

Safety assessments are systematic processes to verify that applicable safety requirements are met throughout the life of a project. Central to the safety assessment, a wide range of scenarios are considered to develop an understanding of the system and provide a thorough safety case for a project. The scenarios selected for detailed assessment are those most likely to occur (i.e., expected events) or those relatively unlikely to occur but could have major consequences. In accordance with IAEA 2014, the selection of scenarios for detailed assessment for the Project are justified and, where appropriate, supporting evidence is provided. This is to confirm the effective use of extensive assessment efforts and to obtain a design for the Project that best protects human health and the environment.

Scenario development considered all reasonable future states of the site and the biosphere. Scenario development does not try to predict the future, rather it aims to demonstrate the importance of sources of uncertainty, providing meaningful illustrations of future conditions to assist decision makers. The scenarios demonstrate protection of people and the environment from both radiological and non-radiological hazards and encompass the time period when maximum effects are predicted. For post-closure, the findings of the safety assessment delineate necessary institutional controls (e.g., determination of the timeframe required for monitoring and surveillance of the WL site to provide the protection of the public and the environment). Consistent with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), the scenarios developed for the Project include the expected evolution of the Project (i.e., the Normal Evolution Scenario) and those low-probability, disruptive events that could occur leading to upset conditions (i.e., bounding scenarios; refer to Table 5.4.3-1).

Disruptive events were identified and evaluated using different approaches, depending on the phase of the Project. For the closure phase, hazard identification was performed through a HAZOP (Appendix B) for the planned activities for the Project, which includes potential accidents and malfunctions. For the post-closure phase, an evaluation of FEPs was performed to comprehensively and systematically identify all factors related to the safety of the WRDF. If it is determined that a factor cannot be ruled out and could play a significant role in the performance of the WRDF, this factor is carried forward into the identification of potential performance scenarios for the safety assessment. Further, these factors are also used to select which disruptive events will form the bounding scenarios, meaning the events with the smallest predicted margin to the assessment acceptance criteria during the post-closure timeframe.

5.4.1 Hazard Identification

5.4.1.1 Overview

Proactive methods are used during the Project planning to identify potential risks. Through design and/or mitigation these risks are addressed thereby avoiding or limiting the potential for environmental effects, worker injury (harm) and/or costly operational disruptions. The purpose of a HAZOP is to identify potential hazards and associated conditions and it is suited to projects or processes that are well-defined and are at the detailed design phase (e.g., design specifications and detailed procedures are available). A HAZOP was completed for the project at the early design stage to identify high-level risks and scenarios that could occur. This was performed via a “what-if” workshop where potential scenarios and conditions were identified (Appendix B). The objective of the HAZOP was to identify potential hazards and associated conditions that could arise during the proposed decommissioning activities of the WR-1 Building (i.e., closure phase), causing harm to workers.

The HAZOP process was organized according to the proposed Decommissioning Work Packages, summarized below:

- Building Systems;
- Prepare Reactor Systems;
- Grout Installation; and
- Building Demolition and Site Remediation.

The WL site operating experience over the past 50+ years is important context for this assessment, particularly the past 25+ years of experience conducting decommissioning activities. It is recognized that hazard levels are expected to be lower than those encountered during historic operations of the WR-1.

Further, an accidents and malfunctions report (ISR 2016) was prepared to identify credible events which could arise during the WR-1 closure phase, causing potential harm to people or the environment. The “What-if” questions and scenarios developed during HAZOP were incorporated into the accidents and malfunctions report where appropriate. Potential effects from the accidents and malfunctions, as well as factors which may mitigate the effects are also identified. The report determined that the consequences of these accidents and malfunctions were anticipated to be significantly lower than those identified by the CSR (AECL 2001), the effects would be localized to the decommissioning site, and are sufficiently mitigated by existing Management System Programs (see Section 3.1.6); specifically emergency response/management and environmental protection programs. These programs form the basis of the CNSC-issued site decommissioning licence. They are assumed to be sufficiently comprehensive to support this Project, and any residual hazards from the potential events are used as the basis for assessment in Section 7.0 (i.e., Table 7.1.1-1).

Decommissioning of the WR-1 Building is essentially confirming the current Occupational Health and Safety (including CNL’s Radiation Protection Program) and Environmental Protection controls and measures at site adequately encompass the decommissioning activities, with particular focus on the identification, definition and management of waste streams. A comprehensive program for the management of wastes arising from decommissioning work at the WL site has been developed (Section 3.1.3.3 Waste Generation and Management).

5.4.1.2 Methods

To supplement the decommissioning strategy and rationale provided in the DDP (CNL 2021b), a HAZOP was developed through (1) documentation review and (2) completing a 'What-If' workshop (Appendix B). Information considered included:

- CNL documentation for the WL facilities, including annual safety reports and environmental monitoring reports;
- industry documentation for similar decommissioning projects; and
- lessons learned from previous decommissioning work, including relevant Operating Experience (OPEX) used to plan work execution.

Brainstorming by a team of technically diverse professionals resulted in the identification of potential hazards to be considered for the closure phase, as well as potential accidents and malfunctions. An initial estimate of the consequence, likelihood, and mitigating factors were developed for each hazard or 'What-if' scenario presented.

5.4.1.3 Results

While none of the activities to be completed are unique in concept to ISD, there are elements considered non-routine for the WL site and limited in experience in application relating to reactor decommissioning. These include the following:

- potential displacement of internal radiological contamination by grout;
- consideration for buoyant forces on grouted materials and equipment (e.g., tanks);
- chemical reactions between grout and reactor components (e.g., hydrogen generation from aluminium with high alkalinity grout);
- hazardous grout constituents from a respiratory perspective, suspended in air (silica or cement dust); and
- potential mobilization of fixed organic coolant during disturbance activities (e.g., grout curing heat).

In addition to conventional radiological and non-radiological hazards associated with building decontamination and demolition, ISD activities present several additional hazards that will be considered during work planning including:

- construction hazards (e.g., equipment movement, grout sequencing, and excavation);
- chemical hazards;
- biological hazards;
- thermal consideration relating to grout curing; and
- conventional occupational accidents:
  - hearing and eye protection issues;
  - confined space work;
  - excavation and trenching;
  - material handling; and
  - increased potential for slips, trips and falls.

These potential hazards are considered routine and will be addressed through work planning and review of the Health and Safety Plan and the contingency plans. As well, various non-radiological and radiological hazards that may be encountered during ISD include:

- high radiation fields and exposure time required in all radiation fields (i.e., ALARA);
- workplace contamination;
- environmental contamination;
- radioactive sources not identified in the inventory;
- leaking radioactive sources or loss of shielding;
- leaks of contaminated liquids generated during decommissioning;
- piping systems or tanks containing radioactive liquids or sediments;
- disturbance of buried pipes with contaminated liquids that may not be intact;
- accumulation of radioactive particles in ventilation and filtration systems;
- accumulation of a significant radioactive inventory (e.g., contaminated and activated tools and materials/ waste);
- clean areas being contaminated due to loss of containment (e.g., opening of closed systems, such as ventilation ducts and tanks; however, contaminant immobilization will be conducted as required); and
- decontaminated zones being re-contaminated due to personnel movement and material handling.

These hazards are not ‘new’ to the WL site; however, they are considered to be non-routine. Based on the preliminary decommissioning work package breakdown, no ‘new’ activities will need to be addressed in this assessment for the closure phase. However, a review of these hazards will be done, once detailed work planning information is available (e.g., decontamination, grouting, and dismantling plans and schedules). Work plans will be developed specifically to include best industry practices and experience to limit risks to workers, the public and the environment from ISD hazards and to be compliant with the site’s ALARA policy.

5.4.2 Features, Events and Processes

To define the range of potential future conditions and scenarios, the analysis considers numerous factors that could affect performance. These factors are referred to as Features, Events and Processes (FEPs). A FEP is a feature, event, process or other factor that could directly or indirectly influence the long-term safety and performance of the disposal system. A Feature is a prominent or distinctive part or characteristic of the Project or its environment (e.g., engineered cap), an Event is a change or complex of changes located in a restricted portion of time and space (e.g., rainfall), and a Process is a phenomenon marked by gradual changes that lead towards a particular result (e.g., climate change; IAEA 2004).

5.4.2.1 Methods

A FEPs list (Appendix C) was completed as part of the safety case establishment for the Project.
The development of the FEPs list is performed through a comprehensive and systematic examination of the Project activities and components to identify all factors that may be relevant to the safety of the Project.
The development of the list is based on experience and subject matter expertise, as well as international guidance including the Nuclear Energy Agency’s *Features, Events, and Processing for the Disposal of Radioactive Waste* international database (NEA 2000) and the International Atomic Energy Agency Improvement

of Safety Assessment Methodologies for Near Surface Disposal Facilities FEPs list (IAEA 2004). As well, previous Chalk River Laboratories waste disposal projects with similarities to this Project were used for this analysis (AECL 2014, 2013).

The FEPs to be considered were classified as Internal or External to the Project. External FEPs are those beyond the control of Project execution, originating outside the Project. External FEPs include geological processes and events, climatic processes and events, and future human interaction. Internal FEPs include engineered control features, subgeological surround, surface environment, human behaviour, source-term characteristics, solute transport factors, and exposure pathway factors.

Following the development of the FEPs list, a screening analysis was completed to determine the applicability of each potential FEP on the safety of the Project. Specific FEPs were screened out if:

1) FEP is not applicable to the waste types to be encountered, Project design, or environmental setting;
2) there is an extremely low likelihood that the FEP would occur; and/or
3) the FEP would have low consequence and negligible impact (CNSC 2018a).

The FEPs that were not screened out were carried forward into the safety assessment with the relevant factors encompassed in the assessment scenarios (refer to Table 5.4.3-1).

The FEPs developed for the closure phase were influenced by the “What-if” questions raised during the HAZOP exercise (see Section5.4.1). Most of these FEPs were addressed through existing procedures that would be in place during closure work that would mitigate the risks or uncertainty introduced by a specific FEP. Therefore, these FEPs were excluded from the safety assessment. The FEPs included in the safety assessment for the closure phase were related to exposure to airborne emissions from closure activities (Section 7.0).

The FEPs that were carried forward into the post-closure phase assessment of the WRDF guided the development of the groundwater flow and solute transport model and the ERA. They helped determine the parameters to be included in the models, as well as key events and processes to be modelled through the sensitivity analyses.

The “Features” portion of the FEPs informed the key parameters and features of the existing structures, waste forms and inventory, the surrounding environmental setting, climatic setting, hydrogeological setting, geological setting, and human habitation and land use patterns that needed to be included in the overall model. The “Events” portion of the FEPs provided specific types of occurrences that are likely or unlikely to occur, that can affect the long-term safety and performance of the WRDF. The “Processes” portion of the FEPs provided guidance on modelling the long-term safety and performance of the WRDF. These events and processes are evaluated as part of the expected evolution (Normal Evolution Scenario, Section5.4.3.1) or through Disruptive Events (Section 5.4.3.2) or Bounding Scenarios (Section 5.4.3.3).

The FEPs determine the parameters and equations of the models; uncertainty in the parameters and equations determine the scenarios to be assessed. The Normal Evolution Scenario include site-specific values, or reasonably conservative values for FEPs that occur or are likely to occur during the assessment timeframe (Section 5.3). Disruptive Events and Bounding Scenarios use deliberately extreme values for FEPs that are shown to influence the model outcomes through the sensitivity cases. .Sensitivity cases use a range of values for parameters to understand the relative significance of a particular FEP to the overall system performance.

5.4.2.2 Results

The comprehensive list of FEPs that could potentially be relevant to the safety case of the Project for the post-closure phase is provided in Attachment A. The FEPs were defined as being excluded (not relevant) from the safety assessment or were included. For those FEPs included in the assessment, the applicability of a FEP to the assessment scenarios, or if it was captured as part of the sensitivity analysis, was identified. The following categories were used to describe how a FEP was considered in the Normal Evolution Scenario, Disruptive Events/Bounding Scenario, and Sensitivity Analysis.

Table 5.4.2-1: FEP Categories Used in the Safety Assessment

Category	Description
1 – Normal Evolution	The analysis related to the FEP was completed using available data with reasonable ranges.
2 – Disruptive Events/Bounding Scenarios	The analysis related to the FEP was completed using available data with extreme value ranges.
3 – Sensitivity Analysis	The analysis related to the FEP was completed with variability on input parameters.

5.4.3 Assessment Scenarios

Consistent with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), the FEPs not screened out were used to develop scenarios that are classified into those considered in the expected evolution of the Project (i.e., the Normal Evolution Scenario) and those low-probability events that could occur leading to upset conditions (i.e., bounding scenarios for disruptive events).

The safety assessment uses the best available information on the Project and its surrounding environment to form the basis of the mathematical models developed specifically to represent the system. These have been used to calculate results that can be compared with relevant Canadian criteria and standards, as well as inform on uncertainties and identify the most important aspects of the system. These uncertainties and important aspects are captured using scenarios, which are summarized within the groupings below:

- Normal Evolution Scenario;
- Disruptive Events (for which bounding scenarios are selected);
- Defence-in-Depth Cases; and
- Sensitivity Analysis Cases.

The Normal Evolution Scenario is a description of the most likely, expected, evolution of the WRDF and its surrounding environment. The Normal Evolution Scenario accounts for the expected degradation of the engineered barriers over the post-closure phase. It starts immediately following the closure phase, to capture the entire post-closure phase (i.e., institutional control and post-institutional control periods). It is assumed that the WRDF will degrade over time due to mechanical stresses and chemical reactions.

Disruptive events are variants on the Normal Evolution Scenario, designed to address uncertainties that have arisen during the definition of scenarios and conceptual models. Each disruptive event is described with scenario-specific assumptions. Bounding scenarios are then identified to represent the “worst case”, with consequences greater than the other disruptive events considered. The following disruptive events were assessed in the safety assessment:

- **Unsealed Borehole:** Considers the consequence of an enhanced permeability pathway for contaminated groundwater to affect downgradient receptors (i.e., the Winnipeg River).
- **Well in Plume:** To confirm the long-term safety of future generation, the assumption was made that after institutional control a human receptor (On-site Farm) would obtain drinking water from a well, established half-way between the WRDF and the Winnipeg River.
- **Localized Failure of the WRDF:** A localized failure that results in an enhanced hydraulic connection through the WRDF containment barrier and an associated elevated exposure of human and non-human receptors near the WL site to contaminated groundwater.
- **WRDF Barrier Failure:** Failure of the WRDF concrete foundation resulting in an open fracture. Considers that the grouting fabrication and/or installation is inappropriate, or that the long-term performance deteriorates rapidly due to unforeseen or underestimated physical, chemical, and/or biological processes.
- **Human Intrusion:** The specific human activity considered is exploratory drilling. Exposure to (1) drill crew at the wellhead, (2) residents near to the site, (3) core transportation personnel, and (4) laboratory technicians were all considered possible.
- **Glaciation:** Considers an eventual glacial advance causing a full excision and surface distribution of the WRDF radionuclide inventory and exposure of humans resettling in the area following glacial retreat, approximately 140,000 years from closure.
- **Seismicity:** Considers the potential effects from a seismic event during the 10,000 year assessment timeframe. This disruptive event is encompassed by the WRDF Barrier Failure event as a potential outcome of a larger than design basis earthquake.
- **Liquefaction:** Considers the potential for liquefaction of soils surrounding the WRDF.

In addition to Normal Evolution Scenario and disruptive events, the safety assessment also considers other types of scenarios, including Defence-in-Depth Cases and Sensitivity Analysis Cases:

- **Defence-in-Depth Cases** are aimed at building confidence in the performance of the WRDF after closure. These cases examine the extent to which the Project depends on key engineered barriers and what would happen if those barriers were not present. This group of scenarios therefore involves hypothetical combinations to analyze the barriers in the system. Each scenario involves a change in one or more parameters related to a particular barrier; by comparing the results to those of the Normal Evolution Scenario, the influence of the barrier is tested. Refer to Section 6.0 Defence-in-Depth for the In Situ Disposal System for the scenarios considered in the Defence-in-Depth case.
- **Sensitivity Analysis Cases** are used to directly examine the effect of important uncertainties in the models and data used to represent the system. As many modelling aspects can in practice be expressed through parameter values, sensitivity cases focus on using alternative parameter value choices. The alternative parameter values that are assigned need not represent specific bounds on uncertainty (as in some cases these cannot easily be established); in other words, the alternative parameters need not necessarily be the “highest” or “lowest” possible values. Rather, they are used to test the effect of uncertainty; for example,

if parameter x is increased by a factor of 10, by what factor does the dose increase? Refer to Section 6.0 Defence-in-Depth for the In Situ Disposal System for the sensitivity cases considered assessment.

Table 5.4.3-1 presents the scenarios identified as part of the safety assessment.

Key considerations in identifying the assessment scenarios included:

- the hydrogeological investigations characterizing the site (i.e., groundwater flow) and the corresponding hydrogeological surround that will mitigate contaminant transport;
- the ISD envelope (i.e., the grout formulation and the design of the concrete cap and engineered cover) that will contribute to containment, including longevity and integrity;
- the source terms of the radiological and non-radiological inventories remaining;
- contaminant fate and transport modelling;
- the post-closure timeframe;
- climatic processes;
- future human activity; and
- stakeholder and Indigenous people’s feedback (e.g., lifestyle and dietary survey, human and ecological receptors).

The Normal Evolution Scenario is a reasonable extrapolation of present-day site features and receptor lifestyles, and it includes the expected evolution of the site post-closure. The expected longevity and integrity of the subsurface geological surround, including the WRDF, is encompassed by the Normal Evolution Scenario, while potential failure of the subsurface geological surround is assessed by the bounding scenarios. The Normal Evolution Scenario considers natural conditions such as floods or forest fires, as well as extreme conditions such as climate shift, expected to occur within the assessment timeframe. Disruptive events postulate the occurrence of very unlikely events that could lead to high-risk conditions, for example, loss of containment. These disruptive events are assumed to occur within the timeframe defined for assessment of potential effects (10,000 years), which encompasses the phase in which peak effects (i.e., doses) are anticipated. Disruptive events encompass bounding and non-bounding scenarios.

The Normal Evolution Scenario is described in more detail in Section 5.4.3.1 Normal Evolution Scenario. The disruptive events considered for the Project are described in Section 5.4.3.2 Disruptive Events, and the disruptive events selected as bounding scenarios are described in Section 5.4.3.3 Bounding Scenarios.

Table 5.4.3-1: Normal Evolution Scenario and Disruptive Events Considered in the Decommissioning Safety Assessment

Scenario		Description		Key Scenario Assumptions	Solute Transport Model Result
Normal Evolution Scenario		The expected long-term evolution of the Project and the site following closure. The scenario includes the consideration of probable features, events and processes, such as forest fires, and flooding.		■ Based on the timeframe for the assessment the Normal Evolution Scenario includes extreme conditions such as climate shifts ■ Groundwater flow and solute transport conditions are representative of base case conditions ■ An On-site Farm was not considered reasonable for the Normal Evolution Scenario. The WR-1 site will be under institutional control for the first 100 years of post-closure, which will physically restrict residential use of the site, including any farming activities. After institutional control, the WRDF site will be designated for commercial or industrial land use.	Base Case Simulation with model output provided as mass loading rates to the Winnipeg River.
Disruptive Events	**Bounding Scenarios**	WRDF Barrier Failure	Open fracture in the foundation of the WRDF; unconfined failure of waste isolation mechanism.	■ A significant void occurs resulting in non-conformance of the WRDF due to the failure of the concrete building foundation or the long-term performance deteriorates rapidly due to unforeseen or underestimated physical, chemical, and/or biological processes. ■ Presence of void represented in the groundwater flow model as a 2 m-wide zone of enhanced hydraulic conductivity (10,000 times higher than the hydraulic conductivity of the foundation as specified in the model) across the full width of the WRDF. ■ Results in incomplete encapsulation of the contaminated waste. ■ Human and ecological receptors exposure pathways would be consistent with the Normal Evolution Scenario.	Scenario 3 (WRDF Barrier Failure) – with model output provided as mass loading rates to the Winnipeg River.
		Well in Plume	A well in the groundwater plume half-way between the WRDF and the Winnipeg River is used for drinking water by the on-site farm.	■ A farm is established on-site (On-site Farm) ■ Groundwater flow and solute transport conditions are representative of base case conditions. ■ Same as the Normal Evolution Scenario, except that the On-site Farm has a well in the groundwater plume from WRDF and is used for drinking water.	Scenario 16 – Half Pathway Length, with model output provided as groundwater concentrations.
		Human Intrusion	Human intrusion into the WRDF by an exploration borehole.	■ Considers the drilling of an exploration well into the WRDF and into the ISD waste. ■ Groundwater flow and solute transport conditions are representative of base case conditions. ■ Contaminated waste would be brought to the surface during drilling and becomes mixed with clean material during excavation. Conservatively this waste would be assumed to be left on surface, as well as transported for testing. ■ Current drilling exploration best practices and standards are not followed during the intrusion.	Base Case Simulation with model output provided as dissolved and solids (total) concentrations within WR-1.
	Non-bounding Scenarios	Localized Failure of the WRDF	Perforation of the WRDF barrier; localized failure of waste isolation mechanism.	■ Considers a localized failure in the grout encasement. ■ Small excess voids or a relatively moderate void occurs resulting in non-conformance of the WRDF. ■ To capture worst-case, it is assumed that incomplete encapsulation within the ISD results in a localized failure of the WRDF. ■ Human and ecological receptors exposure pathways are consistent with the Normal Evolution Scenario.	No equivalent solute transport modelling simulation was completed because it is bounded by the WRDF Barrier Failure Bounding Scenario.
		Unsealed Borehole	Insufficiently sealed or substantially degraded site investigation or monitoring borehole.	■ Considers a deep borehole on the WL site not being properly sealed prior to abandonment, or the degradation over time of a currently sufficiently sealed well.	No equivalent solute transport modelling simulation was completed because it is bounded by the Human Intrusion Bounding Scenario.
		Glaciation	Substantial perforation or excision of the WRDF and removal of the concrete cap and engineered cover.	■ As the current climate trend will likely delay the glacial period until 100,000 years after present, the scenario of human inhabitants returning to the area after the glacial retreat would be projected to occur 140,000 years from present. ■ The worst-case scenario is assumed to include the glacial advance having completely removed the concrete cap and engineered cover and excised the WRDF (i.e., glacial erosion), and glacial retreat having dispersed the ISD material within the surface environment. ■ Assumed that receptors consistent with the population present today would become established (i.e., consistent habits and exposure pathways).	Base Case Simulation with model output provided as total mass in WR-1 at 140,000 years.
		Seismicity	Seismic event which would damage the WRDF.	■ This scenario considers the probability of a seismic event which could damage the WRDF. The results of the seismic analysis for WR-1 (1 in 10,000-year event, PGA = 0.10) indicate that there will be no cracking or displacement of any portion of the facility.	Not applicable.
		Liquefaction	Liquefaction during a seismic event.	■ Given the aseismic conditions of Eastern Manitoba and the soil properties of the WL site, liquefaction is not anticipated to be an issue for the Project.	Not applicable.

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5.4.3.1 Normal Evolution Scenario

The Normal Evolution Scenario is the expected long-term evolution of the WL site after closure has been completed. The radiological and non-radiological inventories currently present within the WR-1 Complex will be contained for the design life of the WRDF by the low permeability components of the host geological materials and the WRDF itself. The WRDF is expected to degrade over time due to mechanical stresses and chemical reactions. Contaminants will be released from the WRDF in the future due to corrosion of the reactor components and the degradation of the ISD components. Therefore, over time contaminants will migrate into the geosphere and discharge into shallow groundwater and ultimately be realized in surface water. Non-human and human biota within the vicinity of the WL site could be exposed to these contaminants using surface water for consumption and irrigation, through fishing and swimming in the Winnipeg River, and through land use (e.g., farming, hunting, recreation, and dwelling).

Taking into consideration the half-lives of the contaminants present, the WRDF will be designed to provide the adequate containment required for radiological hazards to decay to negligible levels for effluent levels to be protective of human and non-human biota within the environment. The relevant timescale for the Normal Evolution Scenario will encompass global warming predictions.

A Normal Evolution Scenario should be a reasonable extrapolation of present-day site features and receptor lifestyles. It includes the expected evolution of the site and degradation of the WRDF (gradual or total loss of barrier function) as it ages. It does not include biological evolution of individual receptor species, which have been assumed to be static for the purpose of the safety assessment. As identified above, based on the timeframe for the assessment, the Normal Evolution Scenario includes extreme conditions such as climate shifts. Other extreme events identified as very rare, such as glaciation were analyzed separately.

An On-site Farm was not considered reasonable for the Normal Evolution Scenario, until the end of institutional control. The WRDF will be under institutional control for the first 100 years of post-closure, which will physically restrict residential use of the site, including any farming activities. After institutional control, the WRDF site will be designated for commercial or industrial land use. To account for the possibility of a farm closer to the WRDF, an On-site Farm receptor drinking river water has been included after institutional control. The use of a drinking water well on site was not considered likely due to the proximity to the river, and the local hydrogeological conditions. The use of a well for drinking water is considered as a disruptive event only (see Section 5.4.3.2.5). For the Normal Evolution Scenario, the public dose limit of 1 millisievert per year (1 mSv/a) was used, which is considered protective of human health. This limit applies to the WL site, within which the Project is encompassed.

Geomorphically relevant processes generally fall into three categories (1) production of overburden by weathering and erosion, (2) transport of material, and (3) deposition of material. Geomorphic changes may be caused by wind, waves, chemical dissolution, slope movement, groundwater movement, surface water flow, tectonism, volcanism, and glacial action. In relation to potential non-glacial events specific to the WRDF, the central landscape feature is the Winnipeg River, as rivers are a key link in the connectivity among different landscape elements. River channel migration is known to occur within a flood plain and is typically characterized as back and forth movement of several metres over a relatively short geological time span. For example, the Red River has an average migration rate range of about 0.04 to 0.08 metres per year resulting in the widening of the valley cross-section (Brooks et al. 2002) and the lower Assiniboine River has a mean rate of 0.4 metres per year. However, substantial movement has been shown over short-periods of time during activities such as damming. The Winnipeg River hosts six hydroelectric dams, as well as many lakes, with flows controlled by the Lake of the Woods Control Board. This watercourse is highly controlled and monitored and will be for the foreseeable future, therefore river migration is not considered as part of the Normal Evolution Scenario.

The sensitivity analyses completed for the solute transport modelling (Golder 2021), considered potential changes in the surrounding environment. Specifically, Scenario 1 considered a preferential pathway, which reflects a shorter travel time between WR-1 and the Winnipeg River, and considers a geological (e.g., river migration) or man-made feature that would provide an enhanced hydraulic connection through the groundwater flow system; and Scenario 13 considered a low river stage. Furthermore, glaciation is discussed in Section 5.4.3.2.6 Glaciation, representing a more significant geomorphological change.

An appropriate degree of conservatism was integrated into the supporting analyses for the safety assessment, for example:

- the source area was assumed to be equivalent to the outer extent of the grout block (i.e., the grout was not considered as a barrier);
- the source mass contained in the biological shield and reactor systems is made instantly available within the grout of the WRDF at the assumed release time (year 2035);
- for lead sources (i.e., thick plugs and shielding at thickness ranging from 2.5 inches to 5 inches) and source mass contained within the reactor it was assumed that release would occur gradually through corrosion;
- all solute mass associated with the source is instantly converted to dissolved phase and has an infinite solubility, except for HB-40 and lead, which have solubility limits of 0.8 mg/L and 0.1 mg/L, respectively;
- solute partitioning (adsorption) was only considered for the bedrock pathway;
- lateral dispersion downstream was not simulated; and
- radioactive decay, ingrowth and dispersion were accounted for.

5.4.3.2 Disruptive Events

Disruptive events are plausible but unlikely events that could affect the otherwise normal evolution of the waste disposal system. The following disruptive events were assessed in the safety assessment:

- Unsealed Borehole;
- Human Intrusion.
- Localized Failure of the WRDF;
- WRDF Barrier Failure;
- Well in Plume;
- Glaciation;
- Seismicity; and
- Liquefaction.

For the disruptive events, 1 millisievert per year (mSv/a) IAEA Lower Reference Level 20 mSv/a IAEA Upper Reference Level was used which is considered protective of human health. Deterministic effects will be prevented if effective whole-body annual dose exposures are limited to 20 mSv/a. The safety assessment also considered disruptive events where the environment could affect the Project including glaciation, seismicity and liquefaction.

5.4.3.2.1 Unsealed Borehole

Within the vicinity of the Project, several site investigation/monitoring boreholes have been developed down to and beyond the depth of the proposed WRDF. The Unsealed Borehole Disruptive Event considers the consequence of a deep borehole on the WL site not being properly sealed prior to abandonment, or the degradation over time of a sufficiently sealed borehole. In this event, contaminated groundwater may travel to a faster geosphere pathway through the borehole and therefore, arrive at the Winnipeg River faster, resulting in increased exposure to human and non-human receptors near the WL site.

The Unsealed Borehole Disruptive Event requires the failure of industry best practices and standards, including quality assurance. Further, it either requires insufficiencies in the operational and institutional control monitoring programs allowing an existing unsealed borehole to go undetected or the failure of a currently sealed borehole to develop after monitoring has ceased.

The effect was assessed during the post-closure phase, at the time of peak groundwater concentration. This is conservative as it can be assumed that the earliest inadvertent human activities could take place would be immediately after the end of institutional control (i.e., 100 years after closure), once monitoring and adaptive management of the site has ceased, Human and ecological receptor exposure pathways will be consistent with the Normal Evolution Scenario. The localized increased permeability of the geosphere would result in COPC concentrations in the Winnipeg River being slightly elevated, therefore, slightly increasing exposure to human and ecological receptors. This disruptive event was not carried forward as a bounding scenario.

5.4.3.2.2 Human Intrusion

The limited footprint and current geological information indicate that direct intrusion into the WRDF is unlikely; however, it is conceivable that after site controls are no longer effective, human activity could result in undeliberate intrusion into the decommissioned structure. The specific human activity considered for the Human Intrusion Disruptive Event is exploratory drilling. To be protective, it is also assumed that current drilling exploration best practices and standards are not followed during the intrusion; for example, procedures do not include proper containment, proper disposal of borehole material, nor proper closure of the borehole. Exposure to (1) drill crew at the wellhead, (2) residents near to the site, (3) core transportation personnel, and (4) laboratory technicians were all considered possible. Sources of exposure include gas released from the borehole, the drill core itself, and contaminated groundwater.

To occur, the Human Intrusion Disruptive Event requires failure of government controls, such as land use restrictions. Further, it requires the concrete cap and engineered cover to be insufficient at deterring human intrusion, the drill crew not being experienced enough to recognize the alien local geosphere and/or the invaluable composition of the core, the drill not deflecting around ISD barriers, and the failure of drilling exploration best practices and standards.

The assumption is the drilling of an exploration well into the WRDF and into the ISD waste. It is assumed that contaminated waste would be brought to the surface during drilling and becomes mixed with clean material during excavation. Conservatively, this waste would be assumed to be left on surface, as well as transported for testing. The conservative assumption would also be made that the well is not abandoned in accordance with best practices and standards. The drill crew is exposed over the workday through dermal contact, incidental ingestion, and groundshine. The drill crew exposure is anticipated to encompass that of transport and laboratory personnel. For the trespassers, there may also be inhalation of dust from resuspension of dried waste material; this pathway is incomplete for the driller as the material will be wet while they are at site. The on-site drill crew exposure is anticipated to encompass that of transport and laboratory personnel. Other human and ecological receptors exposure pathways would be consistent with the Normal Evolution Scenario, except that harvesters would additionally be exposed through contact with soil on-site. The localized increased permeability of the WRDF would result in COPC concentrations in the Winnipeg River being slightly elevated. The localized presence of waste on surface would result in COPC concentrations in the soil on-site being elevated, and the localized intrusion into the WRDF and waste on surface would result in a source of airborne effluent. The additional source of contamination and exposure pathways would result in COPC exposure to receptors being elevated. This disruptive event was carried forward as a bounding scenario (see Section 5.4.3.3.1 Human Intrusion Bounding Scenario).

5.4.3.2.3 Localized Failure of the Whiteshell Reactor Disposal Facility

The Localized Failure of the WRDF Disruptive Event considers the consequence of an insufficiency in the containment/isolation provided by the WRDF, as a result of a flaw in design or execution. Specifically, this scenario represents a localized failure that results in an enhanced hydraulic connection through the containment barrier and an associated elevated exposure of human and non-human receptors near the WL site to contaminated groundwater.

The assumption for the Localized Failure of the WRDF Disruptive Event is that excessive small to moderate unsealed openings, cracks or voids are present in the existing foundation, resulting in reduction of effectiveness of the foundation as a barrier. Human and ecological receptors exposure pathways are consistent with the Normal Evolution Scenario. The increased permeability of the WRDF foundation barrier will result in increases to some COPC mass loading rates to the Winnipeg River, therefore increasing exposure to receptors. This disruptive event was not carried forward as a bounding scenario as it was deemed sufficiently bounded by the WRDF Barrier Failure Scenario.

5.4.3.2.4 Whiteshell Reactor Disposal Facility Barrier Failure

The WRDF Barrier Failure Disruptive Event considers the potential for an uncontrolled release of contaminants from the WRDF into the geosphere. Specifically, the scenario considers a significant failure of the concrete foundation, or that the long-term performance of the foundation deteriorates rapidly due to unforeseen or underestimated physical, chemical, and/or biological processes. The source of exposure is due to a zone of increased permeability in a 2-m wide section (i.e., a fracture) spanning the full width of the containment barrier, and an elevated exposure to human and non-human receptors near the WL site to contaminated groundwater.

The WRDF Barrier Failure Disruptive Event was considered because the foundation was not originally designed as the primary containment barrier for an ISD facility. It is considered a containment barrier within the WRDF in the groundwater flow model based on its properties determined through the original design drawings and the building condition assessment carried out (Golder 2019b). The WRDF Barrier Failure Disruptive Event requires a significant failure of the WRDF design and/or construction process. Industry experience and expertise has been used in planning for the Project, and ISD is a proven approach. During investigation (e.g., foundation condition), design (e.g., grout fill design), and construction activities, adherence to industry best practices and standards, including quality assurance are required. For this disruptive event to occur, a significant failure of large components or multiple smaller components of work planning, and execution must fail (e.g., failure to adhere to Quality Assurance programs when reviewing the condition of the existing foundation wall). This disruptive event could also be caused by a significant external event, such as a beyond design basis; earthquake that creates significant damage to the WRDF foundation.

This disruptive event was conceptualized in the groundwater flow modelling analysis as a 2 m-wide zone of enhanced hydraulic conductivity (10,000 times higher than the hydraulic conductivity of the foundation as specified in the model) across the full width of the WRDF, resulting in incomplete containment of the contaminated waste. Human and ecological receptors exposure pathways would be consistent with the Normal Evolution Scenario. The zone of increased permeability through the WRDF foundation would result in COPC concentrations in the Winnipeg River being elevated, therefore, increasing exposure to receptors. This disruptive event was carried forward as a bounding scenario (see Section 5.4.3.3.2 WRDF Barrier Failure Bounding Scenario).

5.4.3.2.5 Well in Plume

It is not possible to predict the behaviour of people in the future with any certainty. In estimating doses to individuals in the future, the assumption is made that at some time in the distant future government failure will lead to government controls (e.g., zoning designation, land use restrictions, or orders) being ineffective, and people will be present locally and make some use of local resources (i.e., unplanned future land use). To confirm the long-term safety of future generations, the assumption was made that a human receptor (On-site Farm) has a well in the groundwater plume from WRDF and uses it for drinking water.

A well in the groundwater plume was considered not feasible until after institutional control ends (i.e., 100 years after closure). To occur, the Well in Plume Disruptive Event requires failure of government controls, such as land use restrictions; loss of knowledge of the WRDF by local residents, and that the surface land could be attractive to settlement, all of which together is unlikely within the 100-year institutional control period.

The Well in Plume Disruptive Event is the same as the Normal Evolution Scenario, except that the On-site Farm has a well located in the overburden half-way between the WRDF and the Winnipeg River and is used for drinking water. Water for other purposes, including bathing and irrigation of garden crops is taken from the Winnipeg River near the site because water yield rates from the hypothetical well were estimated to be very low. Calculations of well capacity were completed based on the methods in Driscoll (1995) for an overburden well (0.051 m radius) located in the basal till unit (i.e., the overburden unit with the greatest capacity for water production). For a well situated in this unit pumping at its maximum capacity it is reasonable to assume that the flow to that well would be governed by the average aquifer properties due to its radius of influence. Under these conditions the estimated well capacity is 0.02 cubic metres per day (m^3/d). Therefore, the well cannot be used for purposes other than drinking because the well capacity is too low.

This conclusion is supported by observations during routine groundwater sampling campaigns at boreholes on the WL site. In both 2018 and 2019, groundwater sampling of the basal sand unit boreholes downgradient of WR-1 were incomplete due to an inability to obtain sufficient water for sampling (CNL 2019e, 2020g). This reinforces the conclusion that drinking water wells are unlikely downgradient of the WRDF due to a very low potential well capacity to support the needs of any potential future human receptor. This disruptive event was carried forward as a bounding scenario (see Section 5.4.3.3.3 Well in Plume Bounding Scenario).

5.4.3.2.6 Glaciation

The climate of the Earth has been marked by glacial periods where dense ice-covered significant portions of the planet to interglacial periods, such as the present condition, when glaciers retreat to the poles. Based on geological records, it is anticipated that at the end of the current interglacial period, glaciation would occur covering the WL site (most of North America) with a thick, crushing sheet of ice that would be present for possibly tens of thousands of years. During glaciation, human inhabitants would not be present; however, glaciation would radically alter the local and regional geography. The consideration of geomorphology encompasses depositional features such as moraines, eskers, and proglacial lakes, as well as erosional features such as excision into bedrock. Therefore, the conservative evaluation of the potential Glaciation Scenario assumes the exposure of the

ISD waste to the surface environment and needs to be evaluated to determine potential risk to human and non-human biota.

Typically, glacial cycles are assumed to be approximately 100,000 years, with the glaciation phase lasting approximately 90,000 years and the deglaciation phase lasting approximately 10,000 years (Peltier 2011; Clark et al. 2009). The earth is currently in an interglaciation period, meaning that it is between ice ages. It is estimated that the current period, called the Holocene, began approximately 11,700 years ago (Clark et al. 2016). Records suggest that previous interglaciation periods have lasted anywhere from 10,000 to 20,000 years (Berger et al. 2003). Global warming, however, is anticipated to elongate the interglacial period and postpone the next glacial event by tens of thousands of years. The global warming projected until the year 3000 (i.e., 0.6°C to 7.8°C over 1,000 years) represents a much higher warming rate than the rate seen at the end of the last glacial period (4°C over an estimated 8,000 years) (EIS Section 10.4 Climate Change). This corresponds to a higher rate of increase in atmospheric carbon dioxide (CO_2) concentration than observed in previous periods (Clark et al. 2009; Berger et al. 2003).

Coupling of climate change models and glaciation models predicts a relatively long interglacial period of about 55,000 years (Berger et al. 2003) to 100,000 (Peltier 2011). Peltier (2011) also notes that if the concentrations of greenhouse gases remains similar to the present, another glacial event is unlikely due to the increased surface warmth. However, projections for atmospheric concentrations beyond 3,000 years, let alone 100,000, after present are uncertain. Therefore, the potential for a glacial event should not be discounted.

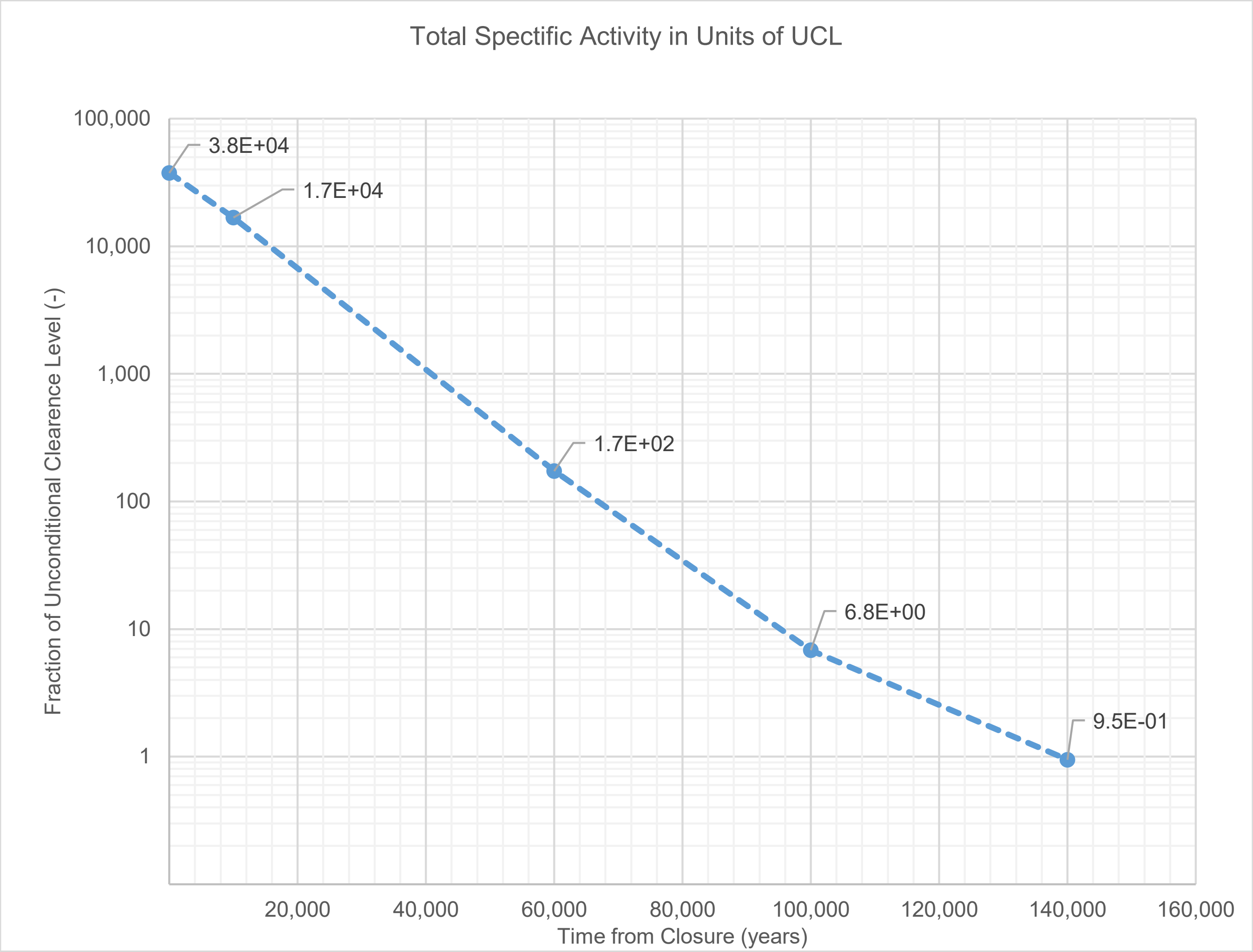
Project design will employ the best available grout technology and quality assurance to extend the permanence of the WRDF beyond the cessation of the institutional control period . The WRDF may not withstand the effects of glaciation. Glaciation is expected to occur within this area in accordance with the natural glaciation cycle established. This glaciation is expected to occur approximately 100,000 years from present time, but as noted above, this return may be delayed or not happen at all. If glaciation occurs, the area will first enter into a permafrost condition and then be covered with a thick sheet of ice for tens of thousands of years, based on glaciation studies and data for this region.

The worst-case scenario is assumed to include the glacial advance having completely removed the concrete cap and engineered cover and excised the WRDF (i.e., glacial erosion), and glacial retreat having dispersed the ISD waste within the surface environment. It is recognized that glaciation radically alters the geography (e.g., forming drumlins). As the sheet of ice expands the accumulation of snow and ice crushes and abrades both surface rock and bedrock and erosional landforms result (e.g., u-shaped valleys). When the ice sheet retreats the crushed rock picked up and carried along during the advance is left behind in the path of its retreat, creating depositional landforms (e.g., eskers). Examples of possible changes include a riverbed being significantly altered during advancement, or during retreat deposited material damning a riverbed, or large chunks of ice being left behind to form glacial lakes or ponds (i.e., kettle lakes).

To ensure a conservative estimate of dose consequences, it has been assumed that human inhabitants would return to the area no later than 140,000 years from the present. At the time of this projected re-habitation, glaciation would have dispersed the remaining radioactive materials, which represent a fraction of the present-day radioactivity at the facility, and incorporated glacially-derived soil and rock materials such that remaining components of the WRDF would no longer be intact. The remaining activity of the remaining materials will have been reduced due to the natural radioactive decay and leaching of the materials via transport in groundwater over time. The amount of residual radioactivity remaining at the time of re-habitation is less than the Unconditional Clearance Level defined by CNSC (as detailed in Section 5.4.3.2.6.1). The remaining long half-lived activation products at the time of rehabilitation would be of an activity similar to naturally occurring geological settings in the region, including the Kasmere Lake surficial uranium deposit. A comparison of the activity in the ISD to natural analogues is provided in Section 5.4.3.2.6.2.

This assessment aims to provide meaningful illustration of future conditions to inform the Project design. During glaciation, following glacial advance, it could be expected that the groundwater conditions in the vicinity of the WL area and the infiltration of water into the WRDF could change; for example, groundwater flow rates may decrease in the event of extensive permafrost conditions or increase as a result of glacial thaw. However, this cannot be predicted with an acceptable degree of certainty. For the purposes of this assessment groundwater flow conditions were assumed to be maintained at a consistent rate throughout the glacial period. It is likely that there would be no human receptors present during the glaciation period as the environment would not be able to sustain a human population. Predicting what the environment will look like after glacial retreat, along with predicting the lifestyles and living conditions of the human inhabitants of Manitoba after the cessation of an ice age, 140,000 years from now, is fraught with uncertainty. An appropriate level of conservatism has been incorporated to account for these uncertainties, in addition to considering multiple lines of reasoning.

The simulated activity remaining within the WRDF is plotted as a function of CNSC's Unconditional Clearance Levels (UCL) on Figure 5.4.3-1 based on the simulated mass of the reactor components and grout (i.e., assuming the remaining solute mass is distributed throughout the grout, which is consistent with the solute transport modelling assumptions). As shown on the plot, the activity remaining within the WRDF is estimated to be above the Unconditional Clearance Levels for approximately 140,000 years following closure. This period spans the early estimate for the onset of glaciation (i.e., 60,000 years), the expected onset of glaciation (i.e., 100,000 years) and roughly corresponds to the expected time of glacial retreat.



REFERENCE(S)

1. CANADIAN NUCLEAR LABORATORIES, 2017

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

TOTAL SPECIFIC ACTIVITY REMAINING AS A FRACTION OF UNCONDITIONAL CLEARANCE LEVELS

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**5.4.3-1**

People living on or near the reactor contents 140,000 years from now would be expected to receive radiation exposure through the dominant pathways of:

- inhalation or ingestion of radioactive material; and/or
- direct external gamma radiation exposure.

The formation of a glacial lake over the WRDF would limit the critical pathways for exposure (i.e., inhalation, ingestion and direct contact). Whereas following retreat, the ISD material is excised and spread within the surface environment, and that 140,000 years from present, receptors would become established and may be exposed (assumed to have consistent habits and exposure pathways).

In estimating doses to individuals in the future, the assumption that human inhabitants could potentially use exposed sediment from the Winnipeg River for agriculture purposes was considered. It is recognized that this scenario is unlikely given current population habits and surface geology. It was considered, however, as sediment is often enriched soil and attractive for agriculture purposes, and in light of pending climate change, the credibility of the scenario was assessed. Based on climate change projections for the region, droughts are expected to decrease in the Project region in the near- (2011 through 2040) and mid-term (2041 through 2070), and in the long-term (2070 and beyond) it is anticipated that wet areas will continue to become wetter. Therefore, this was not deemed to be a credible scenario, as climate conditions anticipated do not support the occurrence of an increased frequency or extent of exposed riverbed sediment.

The total mass of metal wastes within WR-1 was estimated at 880 Megagrams (Mg) (CNL 2015b). Recent refinements to the waste estimate indicate the mass of reactor metal wastes will be approximately 900 Mg (CNL 2021b); however, the previous estimate of 880 Mg was retained for conservatism in the assessment. This includes the reactor core (combined calandria and fuel channels), the PHT system and the various other radioactive systems that comprise WRDF. The reactor core (calandria and fuel channels) accounts for the majority of the original source term. By the time the glacier retreats, only a few long-lived radionuclides will remain in the waste. The long-lived radioactive materials will generally be found within the reactor core and the biological shield, while the remaining long-lived radioactive materials will be distributed within the PHT system and some of the ancillary circuits. All of these components will be encased within the concrete foundation and the grout used to fill major void spaces during decommissioning.

After closure, the WR-1 Building sump pumps will be turned off and water table will equilibrate, resulting in a portion of the WRDF to be located below the level of groundwater in the area. It is expected that there will be substantial oxidation and leaching of the metals in the process systems after being immersed in groundwater for 100,000 years prior to the glaciation event. Following the hypothetical excision of the waste during glaciation retreat (140,000 years), the remnants of the reactor and the grout that enclosed it would not be expected to maintain structural integrity. Rather the metals and the remaining radioactivity (140,000 years) would be distributed as an undefined mass, mixed with the contents of the current reactor vault. In addition, all of this would be mixed with the gravel and soil normally associated with glacial deposits. The assumption is, however, that the contents of the reactor vault would be deposited somewhere on the surface.

The assumed exposure of the ISD waste on the surface was evaluated to determine the potential risk to human and non-human biota. This is considered through multiple lines of reasoning in the subsequent sections using CNSC Unconditional Clearance Levels (Section 5.4.3.2.6.1 Comparison with Unconditional Clearance Levels) and comparisons with natural analogues (Section 5.4.3.2.6.2 Natural Analogue).

5.4.3.2.6.1 Comparison with Unconditional Clearance Levels

As described in Section 5.4.3.1 Normal Evolution Scenario, it is anticipated that the WRDF will eventually deteriorate over time allowing the release of the solutes contained in the biological shield, PHT system, and reactor components to the interior of the grouted structure, and eventually to the geological pathway (groundwater), which provides transport to the downstream environment. Given the uncertainty associated with the future glacial environment (e.g., formation of permafrost, increased flow from glacial melt) it was assumed that groundwater conditions in the vicinity of WR-1 would be unchanged in the glacial and post-glacial setting. As such, leaching from WR-1 was assumed to continue throughout the glacial period. Table 5.4.3-2 shows the calculated radioactivity remaining after being subjected to natural radioactive decay and groundwater leaching for 140,000 years. The corrosion process can be expected to affect the structural strength of the process equipment remaining in place but much of the corroded equipment will still be present after 140,000 years. The dose calculations in Table 5.4.3-2 assume the remaining radioactivity is mixed with the 880 Mg (CNL 2015b) of corroded WR-1 components. As a conservative assumption, the potential dilution of the radioactivity within the glacial deposits has not been considered. The predicted activity concentration for the long-lived radionuclides is then compared to CNSC clearance levels.

The CNSC has established Unconditional Clearance Levels (or concentrations) at which a person may safely abandon or dispose of a radioactive substance. The Unconditional Clearance Levels may be found in Schedule 2 of the CNSC *Nuclear Substances and Radiation Devices Regulations* (SOR/2000-207). These Unconditional Clearance Levels correspond to a probable exposure of 10 μSv/a to a resident member of the public residing in the vicinity and include consideration of doses from the inhalation and ingestion of radioactive materials and from external gamma radiation. As described in the *Nuclear Substances and Radiation Devices Regulations*, where several different radionuclides are present, one must calculate the quotient obtained by dividing the activity concentration for each radionuclide by the Unconditional Clearance Level for that radionuclide and sum the fractional quotients thus obtained. The evaluation is provided in Table 5.4.3-2.

Table 5.4.3-2: Comparison of Radioactivity Remaining After 140,000 Years with CNSC Unconditional Clearance Levels Assuming Leaching of Radioactivity from WRDF in Groundwater

Radionuclide	Total Initial Activity (Bq)	Activity Remaining after 140,000 years (Bq)	Fraction remaining after 140,000 Years	Specific Radioactivity (Bq/g)	CNSC Unconditional Clearance Level (Bq/g)	Fraction of Unconditional Clearance Level (%)
Calcium-41	1.40E+08	2.12E-10	0.00%	2.41E-19	1	0.00%
Carbon-14	2.99E+12	3.53E+02	0.00%	4.01E-07	1	0.00%
Chlorine-36	4.20E+03	8.22E-11	0.00%	9.34E-20	1	0.00%
Iodine-129	2.80E+05	8.13E+02	0.29%	9.24E-07	0.01	0.01%
Neptunium-237	1.21E+06	1.88E+04	1.55%	2.14E-05	1	0.00%
Nickel-59	8.30E+12	9.28E+09	0.11%	1.05E+01	100	10.55%
Niobium-94	3.00E+12	7.35E+07	0.00%	8.35E-02	0.1	83.55%
Plutonium-239	6.36E+09	3.32E+05	0.01%	3.78E-04	0.1	0.38%
Technetium-99	1.30E+08	2.40E+05	0.18%	2.72E-04	1	0.03%
Uranium-234	1.27E+07	3.83E+04	0.30%	4.35E-05	1	0.00%
Uranium-235	1.60E+06	5.30E+03	0.33%	4.01E-07	1	0.00%
Uranium-238	1.24E+07	3.63E+04	0.29%	4.97E-09	1	0.00%
					Total	**94.52%**

Bq = Becquerel; Bq/g = Becquerels per gram.

As presented in Table 5.4.3-2, the total remaining activity concentration is 95% of the CNSC Unconditional Clearance Level. This corresponds to a probable annual dose of 9.5 μSv for people living near or on the wastes 140,000 years from now, which is much less than the public dose limit of 1 mSv/y. Hence, under this scenario, the dispersal of the wastes in the WRDF after the next glaciation cycle is anticipated to lead to an acceptable radiation exposure to future inhabitants of the area. The calculation of remaining activity assumed that the WRDF and surrounding geological environment would be intact following glaciation and hydrogeological conditions would be consistent throughout the glacial and post-glacial periods (i.e., leaching would continue). As previously acknowledged, it is possible that glacial activity would extract and disperse the WRDF and surrounding unconsolidated deposits over a wide area, which would result in alternative exposure scenarios.

5.4.3.2.6.2 Natural Analogue

The IAEA and CNSC guidance recognize that due to the very long time periods involved for a disposal facility, there are uncertainties in the assessment. Ways to enhance confidence in the safety features and provide an understanding of the disposal system include testing and evaluation of barrier materials and the use of natural analogues. In Section 5.4.3.2.6.1 Comparison with Unconditional Clearance Levels, the specific radioactivity is compared to the CNSC Unconditional Clearance Level. In this section, the specific radioactivity is compared to natural analogues.

Many naturally occurring ore bodies contain elevated concentrations of radionuclides. Unlike naturally occurring subsurface deposits, the WRDF has been designed to provide multiples lines of defence to control the rate of release of nuclear and hazardous substances from the WRDF and retain the waste away from people and the environment. However, it is feasible that during the next glaciation cycle the engineered cap and geological surround will undergo accelerated erosion leading to the loss of containment. It is assumed that the metals and the remaining radioactivity would be distributed as an undefined mass, mixed with the gravel and soil normally associated with glacial deposits. The existing ore bodies provide a point of comparison for evaluating the potential health risks to human and non-human biota of ISD material becoming dispersed within the surface environment.

Three natural analogues were considered, as detailed below.

The Maqarin Site

Maqarin is located in north-east Jordan, near the border with Syria, in the river valley of the Yarmouk River. The valley is deeply incised allowing a good view of the stratigraphy. The Maqarin natural analogue is a well-documented analogy for cementitious radioactive waste engineered barriers. Numerous studies have been done to evaluate cement evolution, and high pH leachate development and potential consequences (e.g., Khoury et al 1992).

The geological composition of the Maqarin site differs from the future location of the WRDF in that the Maqarin site contains large concentrations of organic matter, whereas the WRDF would be embedded in a soil/clay matrix containing limited quantities of organic matter (McPherson 1968). The groundwater at the Maqarin site was found to have a pH of 12.5. This is consistent with the expected porewater pH that was made for the selection of corrosion rates for the WRDF (though more conservative neutral pH corrosion rates were used in the solute transport modelling assessment). As such, the Maqarin site is considered an appropriate natural analogue to use as a comparison to the environmental conditions that will be experienced by the grout and concrete at the WRDF.

Natural Radioactivity in Soils and Rocks

The radioactivity released from the WRDF will not be unique in the sense that low levels of radioactivity are prevalent throughout the Earth's crust (CCME 2007a). Natural surface soils in Canada generally register uranium values in the range of 0.5 to 10 ppm. Soils with these levels of radioactivity are widely distributed throughout Canada. In addition, relatively high concentrations of metals (including uranium) occur naturally in Canadian soils, stream sediments, and water bodies as a result of naturally occurring bodies of ore.

There are a number of near-surface uranium deposits located in Canada, including:

- British Columbia – Prairie Flats, Sinking Pond, Stinkhole Prospect, North Wow Flat;
- Manitoba – Kasmere Lake;
- New Brunswick – Oromocto Lake, Whooper Swamp;
- Nova Scotia – TA Bog; and
- Yukon – Partridge Lake (IAEA 1984).

Generally, these near-surface uranium deposits were deposited after the last ice age (Jones 1990); therefore, they are relatively young. The naturally occurring radionuclides within the deposits have very long radioactive decay times, therefore, these deposits have not been in place long enough to generate radioactive daughter products (Tixier and Beckie 2001). In other words, there are relatively no short-lived radionuclides present, and the long-term potential hazard from the WRDF 140,000 years from now will be similar in terms of specific radioactivity to the surficial uranium deposits that naturally exist today. Surficial uranium deposits are formed at or within a few metres of the surface; therefore, these naturally occurring deposits can be used as analogues for qualitative estimation of the potential effects of the waste in the WRDF becoming exposed to the surface environment 140,000 years from now.

In 2007, the CCME reviewed environmental levels of radionuclides in soil, groundwater and vegetation in several locations with subsurface uranium deposits, including Prairie Flats (CCME 2007a). The Prairie Flats deposit is located just south of the Town of Summerland (southwest of Kelowna and northwest of Penticton on Okanagan Lake) and is recognized as a large and complex deposit (IAEA 1984). It underlies a hay field in an area where year-round the water table is maintained at less than 1 metre below ground surface and the site is intersected by a series of drainage ditches and underground culverts. The annual precipitation rate in the region is approximately 400 millimetres (mm) to 700 mm, most falling in the winter months, leading to considerable spring runoff (IAEA 1984).

Measured vertical hydraulic gradient indicates an upward discharge of groundwater into the shallow peat and clay unit. This deposit is estimated to be up to 10,000 years old, with ongoing deposition from upwards groundwater flow and it is estimated that 230 t of uranium are deposited in the top 3 m of soil within the peat and clay unit as triuranium octoxide (U_3O_8) (Tixier and Beckie 2001), with local uranium concentration in the surface layer exceeding 1,000 parts per million (ppm) (IAEA 1984). A typical natural deposit will contain several million tonnes of ore, which is larger than the quantity of radioactive waste within the WRDF (IAEA 1984). Since glacial retreat the Prairie Flats deposit is estimated to accumulate 23 kg/yr (Jones 1990).

As stated above, unlike naturally occurring subsurface deposits, the WRDF has been designed to provide multiples lines of defence to control the rate of release of nuclear and hazardous substances from the WRDF and retain the waste away from people and the environment. As compared to the Prairie Flats, there are other natural analogues that could be chosen that are more closely representative of the intact WRDF (i.e., massive uranium

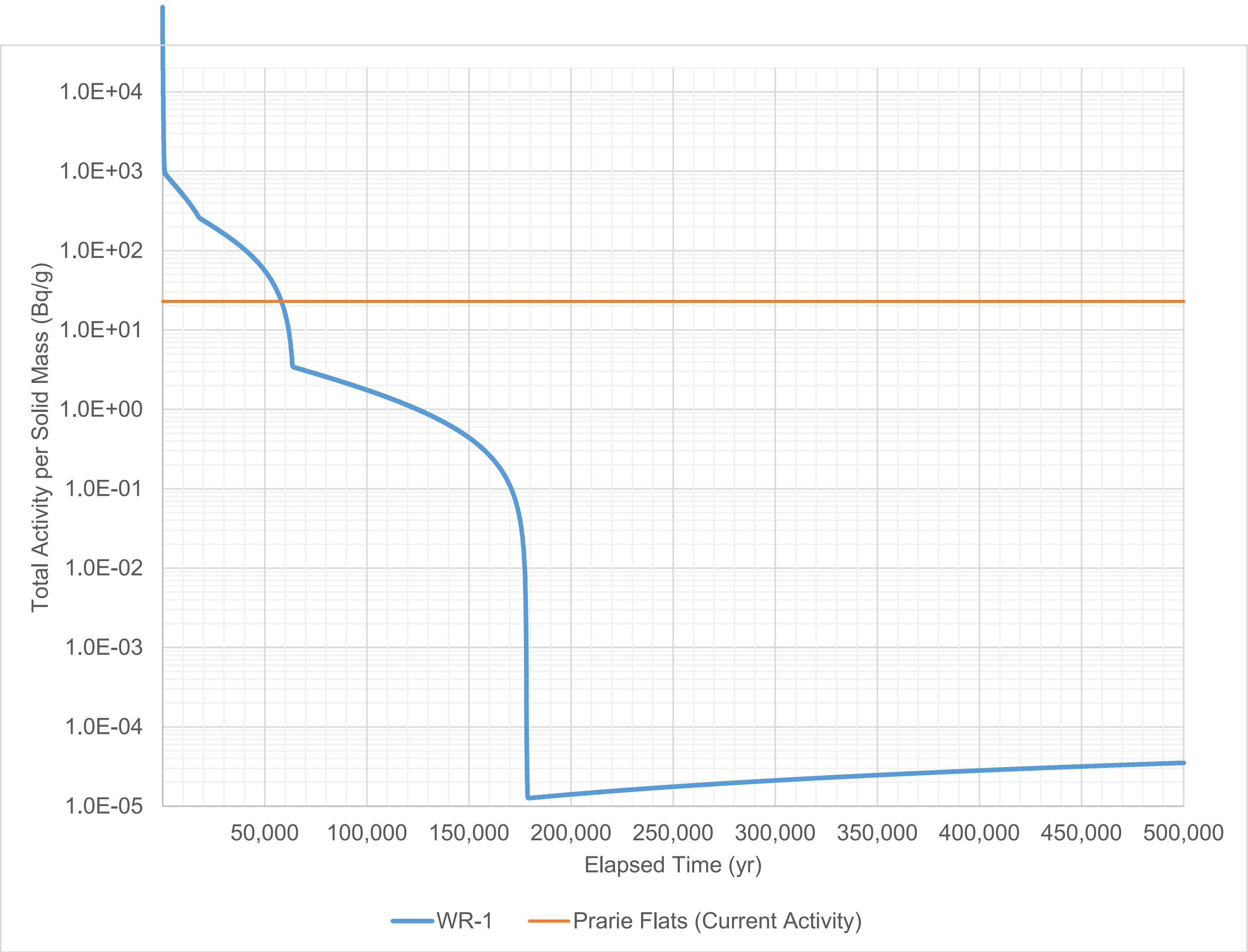
bearing rocks). However, the Prairie Flats deposit is more representative of the state of the WRDF after glacial retreat (i.e., relatively loose material distributed within the surface environment).
Further, the environmental setting of the Prairie Flats deposit also provides an appropriate comparison for the Project (e.g., for downward gradient to groundwater flow).

The greatest concentration of uranium in the Prairie Flats exceeds 1,000 ppm. A concentration of 500 ppm of uranium-238 corresponds to 0.5 g of uranium-238 per kg of soil, or an activity concentration of approximately 6 Becquerel's per gram (Bq/g) (Levinson et al. 1984). In a surficial uranium deposit, uranium-238 exists with other isotopes of uranium and thorium as well as their progeny, with uranium-234, thorium-230 and radium-226 trending towards equilibrium with uranium-238 (IAEA 1984), depending on the age of the deposit. If we consider only the primary long-lived isotopes of the uranium-238 decay chain (uranium-238, thorium-230 and radium-226), this translates to a total specific radioactivity of about 23 Bq/g when the radionuclides within the uranium-238 decay chain are in secular equilibrium. For comparison, the concentration of radioactivity within the reactor vault following the end of the glaciation period (estimated to be approximately 140,000 years from present) was calculated to be about 11 Bq/g. In native deposits mobile progeny, including radium-226 and its daughter radionuclides, leach out at very low concentrations over the centuries and generally do not accumulate within deposits.

The decrease in specific radioactivity within the WRDF as a function of time is depicted on Figure 5.4.3-2 (Golder 2021). Figure 5.4.3-2 shows that by the time the glacier retreat occurs the radioactivity content in the vicinity of WR-1 will have decayed to levels less than what is typical for surficial uranium deposits in Canada. The grout block activity intersects the Prairie Flats activity after approximately 60,000 years elapsed time. This includes the progeny of long-lived uranium and thorium isotopes.

While the levels of environmental radioactivity attributed to releases from the WR-1 site are comparable to the levels of radioactivity occurring naturally, it is noted that many of the radionuclides present within the reactor vault are artificially produced and not naturally occurring. As described in Table 5.4.3-2, every specific radionuclide has a unique detriment (or hazard) attributable to it. Similarly, the 14 naturally occurring radionuclides within the uranium-238 decay chain each present a unique detriment (or hazard). Hence a simple comparison of the specific environmental radioactivity of various radionuclides is not appropriate. The standard method for directly comparing the environmental hazards from the artificial radionuclides within the WR-1 to the naturally occurring radionuclides present everywhere in the Earth's crust, is to consider the dose to members of the critical group.

Radiological consequences to a hypothetical exposure group settling in the vicinity of the WL site area after the glacial retreat will be bound by the current levels of exposure to members of the public living in the vicinity of surficial uranium deposits. In 2007, the CCME concluded that environmental levels of radionuclides at several locations containing subsurface uranium deposits, including Prairie Flats, met regulatory guidelines for the protection of the health of human and non-human biota, and that "no adverse effects are expected." Experience has shown that a sound knowledge of the potential radiological effects associated with the presence of these natural deposits has generally resulted in no measurable effect on human health (CCME 2007b).



REFERENCE(S)

1. GOLDER, 2017

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

TOTAL ACTIVITY PER SOLID MASS (BQ/G) OVER TIME FOR THE WRDF AS COMPARED TO THE PRAIRIE FLATS NEAR SURFACE URANIUM DEPOSIT IN BRITISH COLUMBIA, CANADA

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**5.4.3-2**

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5.4.3.2.7 Seismicity

In 1995, the NBCC placed the WL site (and all of Manitoba) within a Seismic Zone 0, a zone that has a probability of exceedance of 0.0021. The PGA data from NBCC was considered for the years 2005, 2010, and 2015. The trend since 2005 is that for every probability of exceedance, the seismic hazard at the WL site has decreased. For the 1 in 10,000-year probability of exceedance, the PGA is approximately 0.10. Comparatively, the 0.10 PGA represents and earthquake of about Moment Magnitude 4.5. This is considered a relatively small earthquake for which one would not expect structural damage for NBCC designed buildings and components. There could be some non-structural damage such as fine cracking of non-ductile non-structural elements (e.g., plaster or drywall). Structures that could experience damage due to this size of earthquake are those located on soils that are susceptible to amplification and/or have quite low natural frequencies (comparatively high mass and/or low stiffness). These structures may have characteristics that make them susceptive to earthquakes (i.e., tall structures with no bracing or shear walls, strongly asymmetric geometry, torsionally sensitive [centre of mass is far from the centre of stiffness], or constructed from brittle materials [e.g., unreinforced masonry]). Conventionally designed structures using ductile materials (structural steel or reinforced concrete) following good engineering practices that incorporate bracing/shear walls, symmetric geometry, and low horizontal eccentricity are not likely to be damaged by this level of earthquake. The results of the seismic analysis for WR-1 (1 in 10,000-year event, PGA = 0.10) indicate that there will be no cracking or displacement of any portion of the facility.

5.4.3.2.8 Liquefaction

Liquefaction occurs when vibrations or water pressure within a mass of soil cause the soil particles to lose contact with one another. As a result, the soil behaves like a liquid, has an inability to support weight and can flow down even gently slopes. This condition is usually temporary and is most often caused by an earthquake vibrating water-saturated fill or unconsolidated soil. Liquefaction is therefore not deemed to be an issue for the WL site due to the aseismic conditions of Eastern Manitoba. Cone penetrometer testing of the soil properties of the WL site also indicate that liquefaction would not be an issue. In general, the high plastic clayey overburden soils present at the site are not susceptible to cyclic liquefaction (KGS Group 2019).

5.4.3.3 Bounding Scenarios

As indicated in Section 5.4.3.2 Disruptive Events, three disruptive events were identified as “worst case”, with consequences greater than the other events considered, and carried forward as bounding scenarios. Each of these bounding scenarios is described below.

5.4.3.3.1 Human Intrusion Bounding Scenario

The Human Intrusion Disruptive Event comprises the drilling of an exploration well into the WRDF and into the ISD waste. This disruptive event was carried forward as a bounding scenario because it would result in the direct interaction of workers and the public with the waste material.

For this bounding scenario, it was assumed that a four-inch (10.16 cm) exploration borehole was drilled through the concrete cap and engineered cover, grout, concrete structure, and ISD waste (19 m from ground surface to bedrock), and the material encountered was brought to surface, handled by the drill crew and dumped on the ground. The portion of contaminated material in the reactor mixed with the clean material (i.e., soil) was estimated to be 0.947. Concentrations were multiplied by this portion to represent dilution of waste mixing with clean cover soil when brought to the surface. Pathways relevant to exposure from an exploratory borehole include dermal contact, incidental ingestion, and groundshine. The driller (adult) was assumed to be exposure over a period of 1 hour while drilling the borehole. Further, once the drill crew has left, inadvertent trespassing results in exposure to human receptors. For the trespassers, there is the potential for inhalation of dust from resuspension of dried waste material; this pathway is incomplete for the driller as the material will be wet while they are at site. For evaluation of radiological effects, an Adult, Child, 1-year old Infant, and 3-month old Infant (CSA N288.6-12) were considered. For non-radiological effects, an Adult and Toddler (Health Canada 2010) were considered. In both evaluations, they were assumed to spend time at the drill location daily (one hour per day).

This bounding scenario is assessed in Section 8.6.1 Human Intrusion.

5.4.3.3.2 Whiteshell Reactor Disposal Facility Barrier Failure Bounding Scenario

In the sensitivity analysis completed to illustrate the robustness of the model (Golder 2021) assumptions regarding the effectiveness of the WRDF (in terms of containment and isolation of contaminants) were evaluated for each of the WRDF barriers. The WRDF Barrier Failure Disruptive Event represents an open (2 m-wide) fracture in the building foundation (refer to Table 5.4.3-1). This disruptive event was carried forward as a bounding scenario because it represents the most significant deterioration of the foundation as a barrier that was evaluated as a part of the simulations of potential future conditions.

The foundation floor and walls for WR-1 Building were specified in the Normal Evolution Scenario post-closure simulations as a 1 m-thick concrete or equivalent barrier with a uniform hydraulic conductivity. The potential change in groundwater flow rates through the building materials in the event of a failure of the foundation were evaluated. For this simulation the groundwater flow model was reconfigured to have a 2 m-wide zone of enhanced hydraulic conductivity (5E-06 m/s or 10 times the hydraulic conductivity of the groundwater transport pathway) within the foundation floor. This scenario is bounding of the Localized Failure of the WRDF Disruptive Event as shown in the results of Sensitivity Case 3 (see Section 6.4). Under this scenario, as was the case with the Normal Evolution Scenario, the grout is not considered to be a barrier (the source mass was distributed throughout the grout).

The failure of the foundation resulted in a minor change in the ground water flow rates in the vicinity of the WRDF as the local hydrology and availability of groundwater is controlled primarily by the flow through the adjacent hydrostratigraphic units. This scenario resulted in an increase to the early-time flows through the grout, which produced minor increases to peak mass loading rates at the bedrock pathway outflow location relative to the base case. Only those solutes with zero sorption experienced significant increase in peak mass loading rate, and in all cases the increase was 8% or less. In general, the model results were not sensitive to the presence of a local failure (Golder 2021).

The sensitivity to model results was also evaluated for additional aspects of the WRDF barriers, including the model representation of the engineered cover (Scenario 2) and timescales associated with breakdown of the foundation and grout (Scenario 8). The alternative representation of the engineered cover resulted in rates of infiltration through the WRDF that significantly exceeded the surplus infiltration for the WL site and were therefore considered unrealistic. In addition, the adjustments to timescales associated with barrier degradation resulted in generally lower solute mass loading rates. As such these scenarios were not carried forward as bounding scenarios.

Simulations were also completed to provide a basis for the level of protection provided by the foundation and to support the evaluation of defence-in-depth principles as a part of the WRDF safety assessment (e.g., Scenario 15 in Golder 2021 where the complete and instantaneous removal of the foundation barrier was evaluated), though these are not considered to be realistic bounding scenarios compared to other scenarios evaluating barrier degradation.

5.4.3.3.3 Well in Plume Bounding Scenario

A well in the groundwater plume was considered not feasible until after institutional control ends (i.e., 100 years after closure). During institutional control, long-term performance monitoring and maintenance activities will occur to demonstrate compliance with the safety case assumptions; therefore, this disruptive event is unlikely during institutional control. The Well in Plume Bounding Scenario is identical to the Normal Evolution Scenario, except that a groundwater well has been established for the purposes of drinking water for the on-site farm and that results are only presented after 100 years of institutional control. This well is assumed be half-way between the WRDF and the Winnipeg River within the centre of the groundwater flow path from the WRDF, while in the Normal Evolution Scenario drinking water for the on-site farm is obtained from the Winnipeg River. As previously described in Section 5.4.3.2.5 Well in Plume, the capacity of an overburden well at the WRDF would be limited, and therefore, water for other purposes (i.e., bathing and irrigation of garden crops) is taken from the Winnipeg River near the WL site. This bounding scenario is assessed in Section 8.6.3 Well in Plume.

5.5 Conceptual and Deterministic Models

Conceptual models were used to illustrate the performance of facilities under varying conditions and provide an analytical, quantitative analysis of performance. The conceptual model developed for the Project represents the environmental setting and the conceptual design of the WRDF (including mitigation to protect against radiological and non-radiological hazards associated with the equipment and infrastructure to remain in place). In accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), the robustness of the model reflects the complexity of the facilities and the site setting, and the potential hazards present to workers, the public and the environment. As well, the degree of conservatism applied to the model accounts for uncertainty related to factors such as the simplification of the site description and the use of surrogate information to address gaps in site-specific information. What is deemed appropriate in terms of robustness for each model developed is based on operational experience and expertise (including CNL, as well as third parties), professional opinion, and the informed judgement of interested and affected organizations and specialists (CNSC 2018a).

A deterministic model, which uses single-valued input data to calculate a single-valued result, was compared with the assessment acceptance criteria. Plausible variations of the input data values are accounted for by additional individual deterministic calculations using different values of input parameters. In this way, the response of the model outputs to variations in input data were determined, referred to as sensitivity analyses documented in the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021). Evaluation of limiting values from bounding assessments and identification of aspects critical to safety provide useful checks on the long-term assessment calculations and improve confidence in the predictions of safety. Deterministic calculations, with sensitivity assessment, relied on numerical software models, specifically for groundwater flow modelling, solute transport modelling and ERA. These are further described below.

According to CNSC's guidance for applying probabilistic safety assessments for existing facilities as per *REGDOC-2.4.2 Probabilistic Safety Assessment (PSA) for Nuclear Power Plants* (CNSC 2014), the requirement does not apply unless they have been included in whole or in part, in the licence or the licencing basis. For WR-1, such analysis is not required according to the licencing basis. Wherever possible, the safety of the WRDF is based on a deterministic design using the defence-in-depth concept (see Section 4.2.1 Defence-in-Depth Principle). Uncertainty in the assumptions and approach to the assessment have been evaluated in the context of sensitivity analyses. Sensitivity analyses identify which uncertain input parameters are most likely to affect the assessment outcomes, and conservative and bounding values have been incorporated as input parameters for those important parameters, in place of probabilistic analysis.

5.5.1 WR-1 Groundwater Flow and Solute Transport Modelling

To support the DSAR, groundwater flow and solute transport models were developed and used to simulate the current and future (post-closure) conditions of the WL facilities. Output from these models fed into an ERA (EcoMetrix 2021). The groundwater flow and solute transport modelling completed for the Project is described in detail in the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021).

5.5.1.1 Groundwater Flow Modelling

A three-dimensional numerical groundwater model was constructed and calibrated to represent the base case of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostratigraphic units and groundwater flow boundaries. MODFLOW 2005[7] (Harbaugh 2005) was used to complete the groundwater flow simulations. MODFLOW is a multi-purpose three-dimensional code developed by the United States Geological Survey for groundwater flow simulations. It is modular in nature and uses the finite difference formulation of the groundwater flow equation in its solution.

Visual MODFLOW® (Version 4.6.0.156)[8] was used in the numeric flow engine for the simulations presented *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021). . The MODFLOW NWT solver was used to solve the groundwater flow equations (Niswonger et al. 2011). MODPATH[9] (Pollock 1989), a companion code to MODFLOW, was used to complete the particle tracking analyses necessary to illustrate the groundwater flow paths from the WR-1 Complex. A software verification report for Visual MODFLOW is provided in the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021).

The groundwater flow model was calibrated through refinement of the material properties (e.g., hydraulic conductivity) of the hydrostratigraphic units and groundwater flow boundary conditions (e.g., surficial recharge) until an acceptable match was obtained between simulated and observed groundwater elevations and flow rates. Calibration targets were primarily comprised of measured water levels at multi-level groundwater monitoring wells and estimates of groundwater inflows to the WR-1 Building sump. The surficial recharge applied to the model is consistent with the observations from hydrologic studies of the WL site in that infiltration represents a minimal component of the overall water balance. Based on the calibration statistics in combination with the general patterns of groundwater flow and check on groundwater inflow to the sump and overall water balance, the model is considered to provide a reasonable match to observed conditions at the site. The calibrated groundwater model is subsequently used as the basis for construction of the forecast groundwater model. The influence of uncertainties in the key hydrogeological parameters is explored in the sensitivity analyses (Golder 2021).

During the closure phase, the WR-1 Building sumps will be removed, and the building will be filled with grout. When the sumps are removed, groundwater elevations in the building area will rise, eventually resulting in saturation of the lower portion of the grout within the building. The calibrated model was subsequently adapted to include the grout, the building foundation and sump removal. Representation of the grout and building foundation considered the degradation of these materials that is expected to occur over the assessment timeframe, as reflected by applying progressive incremental increases in their hydraulic conductivity.

Predictive simulations were completed using the model to evaluate the post-closure groundwater conditions in the vicinity of the WRDF. This included evaluation of the rates of groundwater flow through the grout and foundation, and delineation of the groundwater flow paths from the decommissioned WR-1 Building to the ultimate

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[7] Software supplier is US Geological Survey.

[8] Software supplier is Waterloo Hydrogeologic.

[9] Software supplier is US Geological Survey.

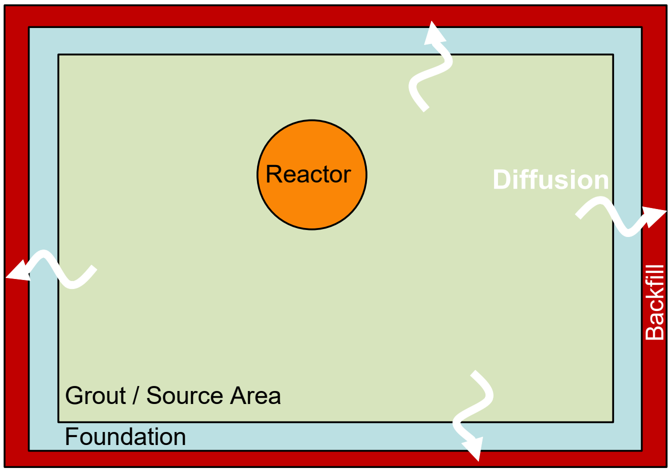
discharge location in the downgradient environment. The conceptual transport pathway from the reactor to the ultimate discharge location is illustrated on Figure 5.5.1-1. As shown on Figure 5.5.1-1, the assessment considers advection and diffusion from the source area (the grout) through the sides and base of the foundation. Diffusion through the top of the source area was not considered as the anticipated groundwater elevation in the grout following closure is at or below this level. Note that the groundwater flow and solute transport models conservatively assumed that the complete solute source inventory is situated and transported within the saturated zone. As such, the release and transport of mass does not occur in the unsaturated zone (i.e., the case, should it occur, where the water table is situated below the top of the source area [grout] following decommissioning).

Upon arrival at the outer edge of the foundation wall or floor, mass is transported through the backfill (which surrounds the foundations of the WRDF), and then into the upper bedrock. The groundwater flow rate through the bedrock pathway was specified to be equal to that of the backfill pathway for all stages of the forecast simulations. With the assumed bedrock porosity of 0.01 this translates to a groundwater travel time between the bedrock and the Winnipeg River of approximately 100 years

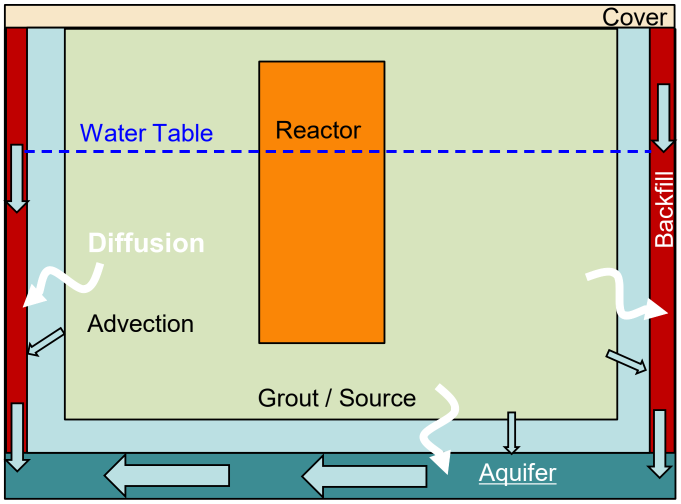
To address the uncertainty associated with the base case model configuration, a sensitivity analysis was completed, which involved perturbation of some of the key model input parameters and comparison of their relative influence on the model results. Additional scenarios were completed using the model that considered possible alternative future site conditions that are distinct from the base case, for example the water level and location of the Winnipeg River relative to the WR-1 Building (refer to Section 5 of the *WR-1 Groundwater Flow and Solute Transport Modelling Report* [Golder 2021]).

Source Area Conceptualization

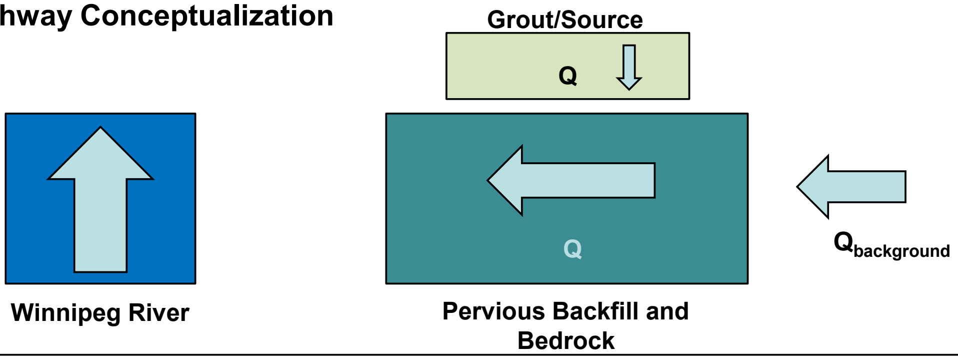
WR-1 Building
Plan View



WR-1 Building
Section View
(west-east)



Pathway Conceptualization



CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
WR-1 GROUNDWATER FLOW AND SOLUTE TRANSPORT MODELLING REPORT

TITLE
SCHEMATIC ILLUSTRATION OF THE SOLUTE TRANSPORT PATHWAY

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	NFB
PREPARED	NFB
REVIEWED	KL
APPROVED	KL

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**5.5.1-1**

NOTES
1. GROUNDWATER FLOW RATES THROUGH THE PATHWAYS (DENOTED BY Q) DETERMINED BY THE POST-DECOMMISSIONING GROUNDWATER FLOW MODELS – REFER TO GOLDER, 2021
2. THE SOURCE MASS IS ASSUMED TO BE UNIFORMLY DISTRIBUTED THROUGHOUT THE GROUT;
3. BACKFILL CONCEPTUALIZED AS A 3M THICK SKIN SURROUNDING THE SIDES OF THE BUILDING FOUNDATION (TOTAL OF 3,192 M3)
4. CROSS-SECTIONAL AREA OF THE BEDROCK PATHWAY ADJUSTED THROUGHOUT SIMULATION BASED ON FLOW THROUGH PATHWAY DIVIDED BY GRADIENT (0.003) AND K (5X10-7 M/S)
5. LENGTH OF THE BEDROCK PATHWAY (DETERMINED THROUGH PARTICLE TRACKING): 500 M

S:\Clients\Canadian_Nuclear_Laboratories\Manitoba\99_PROJ\20145046\40_PROD\0001_DSAR\20145046-0001-G-0018.mxd

IF THIS MEASUREMENT DOES NOT MATCH WHAT IS SHOWN, THE SHEET SIZE HAS BEEN MODIFIED FROM:
25mm
0

5.5.1.2 Solute Transport Modelling

An analytical approach was adopted to complete the solute transport modelling. GoldSim®[10], a commercially available, flexible, object-oriented computer program, was selected as the software package to complete the analytical calculations. The GoldSim software is fully documented in the Main Users Guide (GTG 2014a), and the Contaminant Transport Module Users Guide (GTG 2014b). A software verification report for GoldSim is provided as an attachment to the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021).

The solute transport model was configured so that solute mass is tracked from the source area, through the subsurface pathways (identified based on the results of the groundwater flow modelling), to its ultimate discharge location at downgradient receptors (i.e., the Winnipeg River). The process of advective and diffusive mass release and solute transport in groundwater were simulated within an integrated GoldSim model. A detailed description of the model and its working assumptions and mechanisms is documented in the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021).

The WR-1 source was defined in the solute transport model based on source characterization work completed by CNL (2020b) of the reactor (the core and ancillary systems), biological shield, and substances that will remain within the WR-1 Building following closure. The source term was comprised of a total of 85 solutes, including decay products from radionuclides identified as a part of source characterization work. The ISD of WR-1 will involve permanently filling the interior of the WR-1 Building with grout. For the purposes of this assessment the source area was assumed to be distributed uniformly throughout the grout block, even though the source inventory is located within the reactor itself. This modelling assumption effectively eliminates the grout as a barrier to transport by extending the source term all the way to the exterior wall.

Because the level of radioactivity remaining after a given period will depend on the initial level of the radioactivity and the decay rate of each particular radionuclide, consideration was given for the selection of a “release time” associated with the WR-1 radionuclide mass inventory. This corresponds to when the groundwater level in the soil surrounding the WRDF recovers and saturation of the grout occurs. A period of 10 years was assumed for complete resaturation of the grout block.

Solute mass from the reactor sources was released in the model based on conservative assumptions regarding the corrosion rates of the various reactor components (e.g., fuel channels, calandria, shielding), whereas non-reactor-based source mass (e.g., mass within the biological shield) was instantaneously released. Once released, mass is subject to advection and diffusion through the grout and foundation. Upon reaching the edge of the building foundation the mass continues through the backfill materials surrounding the WR-1 Building and is transported to the primary groundwater flow pathway within the shallow bedrock. The solute transport model takes into consideration the effects of adsorption within the bedrock pathway using solute partitioning coefficients.

Results of the solute transport modelling were provided in terms of mass loading rates (expressed as grams or Becquerels per year) at the outlet of the shallow bedrock pathway, which is assumed to discharge directly to the Winnipeg River.

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[10] Software supplier is GoldSim Technology Group.

5.5.2 Environmental Risk Assessment

An ERA using IMPACT (Version 5.5.1) was conducted to assess the radiological and non-radiological stressors on off-site members of the public and on-site WL workers, who were identified as potentially being exposed to low levels of airborne or waterborne contaminants (EcoMetrix 2021).

IMPACT is a modelling tool, created, maintained and supported by EcoMetrix (formerly Beak International Inc.). The IMPACT model was originally developed in 1993 as part of research projects funded by the Atomic Energy Control Board (now the CNSC). Since the initial development, IMPACT has been continuously updated to improve the interface to encompass an up-to-date understanding of the fate, transport and toxicity of metals, radionuclides, and other COPC released to the environment.

IMPACT (Version 5.5.1) is an environmental transport and pathways model. It is used to determine the annual dispersion factors from sources to receptors in the ERA (EcoMetrix 2021). This code contains a database of dose coefficients, which are progeny-inclusive dose coefficients from Annex C of CSA N288.1-14. It considers all members of the decay chain for decay and ingrowth times defined in CSA N288.1-14.

IMPACT (Version 5.5.1) aligns with the guidance for DRLs that is referred to in the CSA N288.1-14 and uses specific activity models for tritium and carbon-14 in accordance with CSA N288.1-14 and as recommended by CSA N288.6-12. .

The IMPACT model is a customizable tool that allows for the assessment of transport and fate of COPCs through a site-specific environment. In addition to site-specific receptors, the assessment model integrates the conceptual hydrogeological model that has been developed for the WL site and the site-specific COPC sources.

In ecological and human health risk assessments, the conceptual assessment model provides an illustration of the transfer of COPC through the different components of the environment. When there is a complete environmental pathway for a COPC to make it from a source to a receptor, there is the potential for exposure to the receptor and the pathway is included in the evaluation. The assessment model combines the effects of multiple pathways to provide an estimated total exposure to the receptor for each COPC, and the assessment model includes the transfer of COPCs from one environmental media to another, for example deposition from the air to soil and then transport through the food chain.

Potential risks to human receptors from radionuclides were evaluated by comparing predicted total doses to the public dose limit of 1 mSv/a, and the dose constraint for the Project of 0.25 mSv/a. For non-radionuclides, potential risks to human receptors were evaluated by comparing predicted doses to toxicity reference values (TRVs). Hazard quotients for non-carcinogenic constituents were estimated by dividing the estimated exposure by a TRV that is known to be protective. To account for uncertainty in pathways beyond Project activities it was determined that to be protective a benchmark HQ value of 0.2 per medium (e.g., water, soil, food, air) would be used for the assessment. This is consistent with the approach taken by Health Canada (2010) in their guidance on human health quantitative risk assessment.

The estimated incremental risk of developing cancer over a lifetime of exposure are compared to *de minimis* risk levels that are considered essentially negligible compared to background cancer risk. Cancer risks for contaminated sites that are considered acceptable can range from 1 in 10,000 to 1 in 1,000,000 in different jurisdictions. Health Canada (2010) considers an increase in lifetime cancer risk of 1 in 100,000 (or 0.00001) to be essentially negligible compared to the cancer risk level from all background causes in North America of approximately 4 in 10 (or 40,000 in 100,000 or 0.4).

Potential risks to ecological receptors from radionuclides were evaluated by comparing predicted total doses to the ecological dose benchmarks of 9.6 mGy/d for aquatic biota and 2.4 mGy/d for terrestrial and riparian biota. For non-radionuclides, potential risks to ecological receptors were evaluated by dividing the estimated exposure value by the toxicity benchmarks to estimate hazard quotients. A target hazard quotient of 1 was used for the assessment, consistent with guidance in CSA N288.6-12 (CSA, 2012).

ERICA (Version 1.2.1) was used as a source of biota dose coefficients. Its parameters, including dose coefficients, have been subject to validation through numerous intercomparison exercises, as described by Brown et al. (2008, 2013, 2016) and have generally compared well with other sources. The intercomparison of dose coefficients are described by Vives I Batlle et al. (2007, 2011). The external dose predictions for small mammals have been validated against dosimetric measurements (Beresford et al. 2008). The code and database are updated from time to time, as described in its documented version history.

Throughout the planning and preparation of the ERA, all staff worked under EcoMetrix' ISO 9001:2015 certified Quality Management System. All work was internally reviewed and verified. Reviews included verification of input data in the IMPACT files against source documents and verification of selected results with independent calculation spreadsheets, as well as review of report content. Comments have been dispositioned and addressed as appropriate by report revisions. The review process has been documented through a paper trail of review comments and dispositions.

In summary, the ERA uses the expected source terms of atmospheric and liquid release to predict the transport of these substances through the environment and predict the subsequent exposure and dose to the public and exposure and effects on representative ecological receptors. A detailed description of the assessment, including working assumptions and inputs into the model are provided in the ERA (EcoMetrix 2021).

5.6 Alternative Options, Iteration and Design Optimization

Alternative means for the decommissioning of the WR-1 and the design of the facility were evaluated to provide input into the selection of the preferred option and design components. The facility design and its components were also optimized using an iterative process. The following sections describe the alternatives and design options considered and the iterative process to optimize the Project design.

5.6.1 Alternative Options and Facility Design Selection

Section 2.0 (Purpose of the Project and Alternatives to the Project) of the EIS evaluates alternative means for carrying out the Project. Alternative means to decommission WR-1 were identified through internal CNL discussions and from public and Indigenous engagement activities. Environmental effects of each alternative are considered including biophysical, socio-economic, and public and worker health and safety. Four alternative means were identified:

1) **Off-site Storage/Disposal** – Complete dismantling of the reactor and removal of wastes to an off-site facility for interim storage or disposal.

2) **Status Quo –** Building and equipment remains in its current storage with surveillance state and decommissioned following a deferment period.

3) **On-site Storage –** Radioactive waste is collected into storage containers, moved to the waste management area (WMA), and stored and monitored indefinitely.

4) **On-site Disposal –** Radioactive waste is placed into a safe final disposal configuration. Disposal can be provided for all or some of the WR-1 components.

Criteria used to assess each of the alternatives were grouped into four categories: technical feasibility, economic feasibility, safety and environment effects. Each alternative was evaluated first for its technical feasibility (e.g., whether the approach has been used elsewhere and can be easily adapted to this application). For those alternatives deemed technically feasible, a comparison of economic feasibility (i.e., cost), safety and environmental effects was completed. The alternatives assessment considered safety (e.g., effects on workers and the public), as well as biophysical (i.e., groundwater, aquatic and terrestrial environments and atmospheric environment) and social (i.e., socio-economic and land and resource use) effects.

The criteria for assessment alternative means are summarized in Table 5.6.1-1.

Table 5.6.1-1: Criteria for Assessing Alternative Means of Carrying Out the Project

Category	Project Phase		Criteria
Technical Feasibility	Closure		***Is the alternative an approach and/or technology that has been successfully deployed elsewhere?*** ***Does the alternative require any tools, equipment or technologies that cannot be easily adapted to the current application (e.g., climate, location, waste type)?***
	Post-closure	Institutional Control	***Does the alternative rely on development of new facilities or technologies that do not yet exist (environmental monitoring or waste processing technologies, storage or disposal facilities)?***
		Post-institutional Control	***Does the alternative require physical human intervention/support beyond the institutional control period?***
Economic Feasibility	Closure		***Are the costs of the alternative supportable within the current funding framework?*** ■ Ongoing storage and surveillance ■ Decommissioning of the WR-1 Building ■ Transportation of waste off site
	Post-closure	Institutional Control	***Are the costs during the institutional control period well defined and sustainable for a reasonable period of assumed institutional control?*** ■ Post-closure environmental and performance monitoring ■ Construction and maintenance of additional interim or final storage facilities
	Post-closure	Post-institutional Control	***Does the alternative require an economic commitment beyond the duration of a reasonable assumed institutional control period?*** ***Are the long-term costs well defined?***
Safety	Closure and Post-closure		**What are the effects on Worker Safety?** ■ Radiological hazards during decommissioning ■ Non-radiological hazards during decommissioning ■ Industrial safety during decommissioning ■ Waste handling and transport hazards ***What are the effects on Public and Indigenous Safety?*** ■ Exposure risks to public at the WL site during closure ■ Transportation hazards

Table 5.6.1-1: Criteria for Assessing Alternative Means of Carrying Out the Project

Category	Project Phase	Criteria
Environmental Effects	Closure and Post-closure	***What are the potential effects on***: - Groundwater - Aquatic environment - Terrestrial environment - Atmospheric environment - Socio-economic environment

The technical feasibility criteria were not given a specific weight, but were given a “go, no-go” decision. All feasible alternatives were then assessed for economics and effects on worker, public and Indigenous safety, as well as environmental effects. Safety and environmental effects were weighted as follows:

- 30% worker safety;
- 30% public and Indigenous safety;
- 30% biophysical (groundwater, aquatic, terrestrial and atmospheric) environment; and
- 10% socio-economic environment.

Worker safety, public and Indigenous safety, and biophysical environment were given equal weight (30%) because it is recognized that they are inter-related (i.e., one can affect the other), and the assessment does not value the safety of one group of people over another. Socio-economic factors were given a lower weight as physical health and safety are a higher priority than socio-economic health even though both play an important role in the decision-making process.

On-site storage was eliminated from the assessment because it is contrary to the ALARA principle. It exposes workers and the public to the highest potential exposures during the decommissioning work and does not remove the wastes from the site. Interim storage requires significant additional storage infrastructure to be constructed at the WL site, which will increase the scope, hazards and costs of the work, again without providing any reduction of on-site liabilities. The assessment of the alternative decommissioning strategies for WR-1 clearly shows that for the remaining three alternatives evaluated, each can be executed safely. The recommended alternative for the decommissioning of WR-1, based on the alternative means analysis (EIS Section 2.0) is On-site Disposal. The selection of the ISD approach was based on the safety, environmental, technical and economic factors. In situ Disposal is a safe option, reducing the risk to workers compared to dismantling, and providing long-term safety to the public and the environment. The ISD approach also has the least reliance on undefined future disposal options or technologies.

Alternative options in the facility design were also evaluated and the preferred option selected. The following provides a description of the alternative options considered in the facility design and rationale for the selection of the preferred option.

One alternative strategy for the decommissioning of the WR-1 Building would be to **extend the period of institutional control** to prevent human intrusion. The effectiveness of this strategy can be evaluated by examining the extent to which the dose to the trespasser could be reduced by preventing human intrusion for a longer period (e.g., longer than 100 years).

The dose to the trespasser is calculated from the average concentration of the radioactive material in the vault, namely the concrete walls and the metals therein. The activities in the reactor vault are dominated by activated metals and carbon-14 at 100 years. The highest dose rate is associated with stainless steel. After 200 years the gamma dose greatly diminishes, but other contributors have much longer half-lives and are not as affected. Extending institutional control beyond 100 years does not substantially reduce the dose rates from these materials. Therefore, this was found to be of negligible benefit in reducing human intrusion dose. The benefits of 100 years of institutional control is more significant, as they could mitigate doses associated with the initial release of tritium from the facility. As such, extending institutional control beyond 100 years was not adopted.

One approach to controlling the flow of groundwater through the facility, and thus the release of contaminants from the WRDF, is to **reduce the inflows to the facility**. Currently, the subsurface active drainage sump pump in Room 112 performs this function, but in the model the pump ceases to function once grout has been poured into its pit. After evaluation of flows through the grout structure, it was determined that flows resulting from low hydraulic conductivity do not warrant groundwater flow control by way of active means (i.e., a pump).

Backfilling of the reactor vault with grout is a difficult task due to the various means in place to prevent access to the reactor core. Backfilling the vault with grout will also expose aluminum materials to high pH conditions and thus mean these metals could dissolve rapidly. Backfilling the vault with grout would offer a less permeable environment for the wastes, and therefore, provide some additional containment of radioactivity. This benefit is limited to the life of the grout (assumed 2,000 years), which is well short of the period over which the core will remain in place. As such the benefit of grouting the vault is limited for the additional efforts and risks required to grout it. As such, the decision was made not to grout the vault, as the WRDF is shown to be safe without it.

5.6.2 Iteration and Design Optimization

The documentation supporting this DSAR (see Section 5.9 Technical Supporting Studies) has followed an iterative process, with the results used to refine the assessment of the Project. These technical studies build on the outcomes of the previous work to adopt more realistic assumptions, progressively reduce those uncertainties and increase confidence in the projected outcomes. For example, information on the properties of the geology and overburden has been obtained throughout the design of the Project. Groundwater flow and solute transport modelling has been completed in an iterative process to include the new site-specific information obtained through detailed hydrogeological studies. In addition, refinements in knowledge relating to the source term was gained from additional studies to reduce uncertainty in the WR-1 radiological characteristics and radionuclide inventory estimates. Iterations of the solute transport modelling was completed to include this updated information.

A Building Condition Assessment (Golder 2019b) was completed to evaluate the integrity of the existing subsurface concrete foundation and bottom slab of the WR-1 prior to grouting the existing structure of the WR-1 Building. The scope of the work was a condition assessment of the exposed and accessible concrete elements of the substructure, supplemented with laboratory testing of recovered cores to establish the condition and properties of the concrete. This information was then used to validate model assumptions in the solute transport modelling and optimize design elements.

A comprehensive characterization program (CNL 2020b) was performed to address data gaps and provide quantitative, unbiased estimates of residual radionuclide content remaining within the WR-1 systems. The study (CNL 2020b) provided the necessary information to validate existing inventory estimates as bounding case scenarios and to refine the inventories used as inputs to the safety assessments completed for the Project. Overall, the technical supporting studies completed and subsequent refinements to the models were used to optimize the Project design. The three major components subject to design optimization are the grout formulation, the grout emplacement plan and the concrete cap and engineered cover.

CNL has worked with the Savannah River National Laboratories in developing recommended requirements for grout mixes, based on their previous ISD experiences. CNL developed a set of requirements for WR-1 grout based on the configuration of the facility and from groundwater modelling parameters used in the DSAR.

The Concrete Cap and Engineered Cover is being designed to meet the essential performance requirements it must in order to perform its intended functions but is also being optimised to require no ongoing maintenance, to utilize locally available materials, and to improve overall longevity. Grout placement plan would be an additional area of future optimization (going out to market for expertise) for the best grout placement strategy. CNL intends to utilize the expertise of experienced concrete placement vendors to capitalize on their experience in large scale cementitious materials applications to obtain a high quality, final product.

5.7 Management of Uncertainty

This section describes how uncertainty is managed in the safety assessment.

5.7.1 Conservatism and Realism

A fundamental part of the assessment strategy is that the safety analyses used a conservative approach to take uncertainties in data into account, and that the models used for the bounding assessment incorporated conservative assumptions. In those cases where scientifically informed knowledge and data are available, realistic assumptions are made. Additionally, where measurement data are available, it is used as input to calculations, for performing a comparison between results, or to limiting the range of variants for a scenario.

Accompanying each conceptual model is the specific information needed to describe it. A realistic and conservative approach is used to define parameters, noting that in many cases there are uncertainties. Where there are high levels of uncertainty, more conservative assumptions are implemented. Thus, it is acknowledged that the combined effect of many conservative assumptions can lead to unrealistic consequence estimates. To counteract this tendency, realistic data are used wherever possible in the safety analyses for the closure and post-closure phases to determine appropriate design requirement for the WRDF. Examples of conservative assumptions and realism used in the groundwater flow and solute transport modelling include the analyses performed for the grout formulation and the building foundation, conceptualization of the geosphere, and the source inventory of radionuclide and non-radionuclide solute mass. Validation of these assumptions are described in detail in the WR-1 Grout Formula Testing Report (Golder 2019a), the Building Condition Assessment Report (Golder 2019b), the Geosynthesis Report (CNL 2019a), and the *Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* Report (CNL 2020b). Conservatism and realism included in the ERA involved receptor characterization and time independent summing of dose peaks. The conservatism and realism included the DSAR are described below.

5.7.1.1 Assumption on the Grout

The key property of the grout used in the solute transport modelling is the hydraulic conductivity. CNL has specified that the grout used in the decommissioning of WR-1 must have a hydraulic conductivity of less than 0.03 m/year m/s. To support the selection of this value CNL provided reference materials for grout specifications that were used in the ISD of a nuclear reactor at the Savannah River site in the southern United States. A literature review of the hydraulic conductivity of other grout mixtures was completed (Golder 2018b). Recognizing that not all areas within the building will be fully penetrated by the grout, and some voids will remain after final grout placement, a higher value than that found in the at the Savannah River site and in the literature, review was selected to represent the grout in the WRDF models (i.e., a factor of 50 times greater than the maximum value specified by CNL and 50 times greater than the maximum measured value at Savannah River).

In the long-term, it is anticipated that the grout will degrade, and the hydraulic conductivity will increase as a result of this degradation. The extent to which degradation of concrete will occur is dependent on the environmental conditions surrounding the concrete, which are uncertain. For the WRDF assessment, a step function (with linear interpolation between steps) was assumed to emulate the anticipated increase in hydraulic conductivity of the grout as degradation progresses (Golder 2021). Solute portioning within the grout was assumed to be zero.

The cementitious nature of the grout will create a higher pH groundwater environment within the WRDF, which will have the effect of reducing corrosion rates for the reactor components further. It was conservatively assumed that these elevated pH conditions do not provide any benefit to further reducing the corrosion rate, thus over-estimating the corrosion rate over the grout lifetime.

5.7.1.2 Assumption on the Building Foundation

Decommissioning of the WR-1 Complex will involve removal of the service wing and east annex (including the underlying crawl space), and above-grade portions of the WR-1 Building. The excavated area will be backfilled with compact clay soil, . Following decommissioning, the foundation walls and floor slab for the central reactor portion of the WR-1 Building will remain in order to form a barrier for preventing the release of solutes within the grout block. In the absence of data, the building foundation was assumed to have a hydraulic conductivity of over 100 times higher than the values for ordinary concrete specified in literature (Cerny and Rovnanikova 2002; Arnold et al. 2009) and 16 times higher than the highest value tested in the Korean nuclear repository environment (Park and Kim 2013). As was the case with the grout, solute portioning within the building foundation was assumed to be zero.

5.7.1.3 Conceptualization of the Geosphere

The geological and hydrogeological setting of the WL site has been the subject of extensive subsurface investigations completed over a period of more than 5 decades. This includes recent investigation of the vicinity of the WL main campus where WR-1 is located. This enhanced understanding of the geological and hydrogeological environment allowed for a high degree of realism to be carried forward into the conceptual model development with respect to stratigraphy, material properties of the unconsolidated deposits and bedrock, groundwater elevations, groundwater flow directions, and hydraulic gradients.

5.7.1.4 Assumptions on the Solute Source Inventory

The estimated inventory of WR-1 and its associated systems were derived by two general methods; physical samples, and computer models (CNL 2020b). Computer models formed the basis of the inventory estimate and were supplemented by physical sample characterization data to fill gaps and validate model estimates. In all cases, the most conservative results of either the physical samples or computer models were used to derive the final inventory estimate. Each estimation method also included aspects of conservatism in their individual approaches, contributing further to the conservatism built into the final inventory estimate. This compounded conservatism is based on physical data specific to WR-1 and provides an appropriate level of realism to the estimate.

Conservatism in the computer modelling surrounds assumptions related to the homogeneity of the reactor components. The models assumed the most activated areas of the core (centre line of flux) were representative of the entire volume of materials, when in fact there is a decrease in activation of components as you move away from the centre.

Conservatism in the physical characterization included assumptions around the total surface area of reactor systems used to extrapolate a total inventory for each system. The surface area of each system was calculated using both engineering drawings and physical inspection of the installed systems. The estimate was believed to be accurate, but a safety factor of 25% was applied for conservatism.

Conservatism in the chemical properties of contaminants was also applied to ensure contaminant exposures were focussed in pathways where impacts would be largest. The models assume volatile contaminants are transported through the aqueous pathway to ensure they are available for uptake in the local food chain, and not limited to diluted, highly localized air exposures.

Additional conservatism has been included to assume that the entire solute inventory will be located within saturated transport conditions, maximizing potential releases. However, it is estimated based on groundwater flow modelling that a portion of the waste inventory will be located above the water table (i.e., in the unsaturated zone) where the effects of contaminant release through corrosion will be reduced.

5.7.1.5 Assumptions on the Receptor Characteristics

For the Farm receptors the local fractions were taken from the DRL. The DRL local fractions are conservative as they are higher than CSA default local fractions for animal products (e.g., beef, pork, eggs, venison, honey) and potatoes. However, DRL local fractions are lower than the CSA default for fruit because it is anticipated that less will grow in Manitoba. Fish is based on local anglers and wheat is zero as the locals do not mill their own grain. Vegetables are the same as CSA default location fractions.

During post-closure of the WRDF, impacts on receptors will be mitigated through effective institutional control and monitoring of the WRDF. The assessment conservatively assumes that institutional controls will occur for a minimum of 100 years and all knowledge of the facility is lost. Results of the assessment are presented for all years and assume no mitigation of impacts resulting from effective institutional controls and monitoring programs.

5.7.1.6 Time Independent Summing of Dose Peaks

For post-closure, conservative assumptions include the use of the maximum peak for a 40-year model run (i.e., assumed the same maximum loading every year) for the Normal Evolution Scenario. Some bounding scenarios presented results only after the 100-year institutional control period as these events were not considered feasible during this period (e.g., no human habitation will be permitted during institutional control). In addition, for whatever time window is looked at, it is assumed that the maximum peaks are coincident, which will result in higher doses.

5.7.2 Closure Phase

For the closure phase, site-specific measured data are used to confirm the radioactive inventory. Radiological and hazardous material inventories used are based on measurement data from the facility supported by the use of ORIGEN-S (CNL 2020b) modelled inventory as a conservative estimate. Radionuclide inventories used for the above-grade structure were obtained from measured values. Tritium in air data were obtained from actual measurement of tritium concentrations inside WR-1 between 1985 and 2018. Section 3.1.3 Waste Classification, Inventory and Characterization has identified realistic quantities of hazardous materials to be present in the WR-1 Building.

A Building Condition Assessment was completed to reduce model uncertainty by evaluating the integrity of the existing subsurface concrete foundation and bottom slab of the WR-1 prior to grouting the existing structure of the WR-1 Building. The scope of the assessment was a condition assessment of the exposed and accessible concrete elements of the substructure, supplemented with laboratory testing of recovered cores to establish the condition and properties of the concrete.

5.7.3 Post-closure Phase

Where possible, scientifically informed, physical realistic assumptions are made for processes that are understood and can be justified on the basis of results of research and/or site investigation. Where there are high levels of uncertainty associated with processes and data, conservative assumptions are applied to allow uncertainties to be bounded. The use of assumptions has been limited when possible to reduce conservatism in the models. The conservatism of the models has been further reduced by using models that have been validated against experimental datasets and the datasets themselves based on observation.

Uncertainty in the future evolution of the site and human behavior is addressed by assessing a range of scenarios that describe the potential evolution of the system. Data uncertainties were addressed by multiple deterministic calculations where some targeted sensitivity analyses were completed to explore aspects of the post-closure model that were recognized as being uncertain. The sensitivity of the system to uncertainties in groundwater flows, inventory and sorption coefficients was also examined, as described in Section 6.0 Defence-in-Depth for the In Situ Disposal System.

5.8 Confidence in Numerical Models

The majority of the safety analysis relies on the use of commercially available software to develop numerical models to quantity the various scenarios. A brief overview of the following software is provided below:

- MODFLOW-2005 was used to complete the groundwater flow simulations, as described in Section 5.5.1.1 Groundwater Flow Modelling.
- Visual MODFLOW (Version 4.6.0.156P) was used for numeric flow engine simulations, as described in Section 5.5.1.2 Solute Transport Modelling.
- MODPATH was used for particle tracking analysis, as described in Section 5.5.1.2 Solute Transport Modelling.
- GoldSim (Version 11.1) was used to complete the analytical solute transport calculations, as described in Section 5.5.1.2 Solute Transport Modelling.
- IMPACT (Version 5.5.1) was used to determine annual dispersion factors from sources to receptors in the ERA, as described in Section 5.5.2 Environmental Risk Assessment.
- ERICA (Version 1.2.1) was used to inform the results of the non-human biota assessment in the ERA (EcoMetrix 2020), as described in Section 5.5.2 Environmental Risk Assessment.
- ONEDANT code models the neutron flux in the reactor core. It provides a one-dimensional diffusion accelerated neutron particle transport. It is similar to other neutron transport codes.
- ORIGEN-S code is system module of SCALE and is used to calculate radionuclide decay, actinide transmutation, fission product buildup and decay, as well as associated radiation energies. This code is used in the *WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* (CNL 2020b).

Confidence in these codes results from having extensively testing them against a broad set of verification tests. Results generated by these codes, where possible, were compared to those results calculated by similar tools. These results are kept in quality assurance and validation and verification files.

5.9 Technical Supporting Studies

Technical supporting studies completed to support the safety assessment and increase confidence in the analyses included are described below:

- **WR-1 Groundwater Flow and Solute Transport Modelling:** A three-dimensional numerical groundwater model was constructed and calibrated using site-specific data to represent the base case of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. A solute transport model was developed to estimate the solute mass release and transport from the WRDF through the geological environment to downgradient receptors.
- **Environmental Risk Assessment:** An ERA was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota.
The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern.
- **Hydrogeological Study Report:** Detailed site-specific information on the hydrogeology can be found in this Technical Supporting Document. The work in the Hydrogeological Study Report provides baseline data for refining the conceptual hydrogeological model for the site, which is used in developing and calibrating the three-dimensional groundwater flow and solute transport model.
- **Geosynthesis Report:** For WR-1 ISD, the Geosynthesis Report is a compilation of geoscientific information that summarizes the overall understanding of site characteristics, attributes and evolution (past and future) that are relevant to demonstrating long-term performance and safety of an undertaking that relies on geoscientific information, which in the current context is the in-situ decommissioning of WR-1.This report provides information to support the identification of geoscientific data uncertainties and an assessment of their relevance to the Project.
- **Building Condition Assessment Report:** The Building Condition Assessment is part of the ISD program to evaluate the integrity of the existing subsurface concrete foundation and bottom slab of the WR-1 prior to grouting the existing structure of the WR-1 Building. The scope of the work was a condition assessment of the exposed and accessible concrete elements of the substructure, supplemented with laboratory testing of recovered cores to establish the condition and properties of the concrete.
- **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report:** This technical study report presents a summary of existing characterization information of the WR-1 that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided.
- **Grout Mix Formulation Report:** CNL has worked with the Savannah River National Laboratories in developing recommended grout mixes, based on their previous ISD experiences CNL developed a set of requirements for WR-1 grout based on the configuration of the facility and from groundwater modelling parameters used in the DSAR. These requirements and performance criteria for the grout mix formulation are discussed in this report.

6.0 DEFENCE-IN-DEPTH FOR THE IN SITU DISPOSAL SYSTEM

The key aspects of defence-in-depth are layering of defensive principles by providing multiple layers of protection against normal and abnormal events. The assessment of defence-in-depth is completed to demonstrate that multiple safety functions are implemented for the WRDF. This concept is centred on the use of independent and redundant levels of protection to compensate for potential human and equipment failures to that no single level is exclusively relied upon for ultimate safety of the WRDF.

The key method for building confidence in the WRDF is the analysis of the engineered and natural barrier performance. This provides important information on the overall envelope of safety performance and the importance of specific-design targets.

It is recognized that there will be uncertainty associated with the modelling process and the results of the predictive simulations. This uncertainty stems from limitations in the available subsurface information and can be related to variability in the soil and bedrock properties (e.g., faults and fracture zones, hydraulic conductivity, porosity) or uncertainties with the conceptual model (e.g., location of flow boundaries; recharge rates; continuity in aquitards, direction of groundwater flow; simplification of fracture flow systems.). To gain an understanding of the potential impact of this uncertainty in the forecast simulations, a series of defence-in-depth scenarios were evaluated through a sensitivity analysis that assesses the potential variability in the simulated results as a function of both conceptual model uncertainty and general uncertainty in the model input parameters. The following describes the defence-in-depth scenarios analyzed the multilayered ISD system. These scenarios are described in detail in the *Groundwater Flow and Solute Transport Modelling Report* (Golder 2021) and are summarized in Table 6-1. The scenarios as they appear in the report are listed in brackets.

Table 6-1: Summary of Evaluation of Uncertainty in the Performance of the Multilayered Barrier System

Analysis Component	Uncertainty	Evaluation of Uncertainty
Geological Pathway	■ Material properties (hydraulic conductivity) ■ Presence of preferential pathway ■ Sorption-partition coefficients applied in pathway	■ Inclusion of preferential pathway in groundwater flow model (Scenario 1). ■ Evaluation of alternative backfill and bedrock hydraulic conductivity (Scenario 10 and Scenario 11). ■ Conservative estimates of sorption-partition coefficients for base case assessment. Evaluation of upper and lower bound sorption-partition coefficients (Scenarios 5 and 6). ■ Half groundwater flow pathway length (Scenario 16).
Infiltration through Cover	Rate of infiltration	Groundwater flow through source with alternative model boundary condition for the cover (Scenario 2) and under enhanced degradation of the cover, grout and foundation (Scenario 8).
Building Foundation	■ Material properties (hydraulic conductivity) ■ Presence of preferential pathway ■ Timescales associated with degradation	■ Groundwater flow through source with WRDF barrier failure (Scenario 3) and under enhanced degradation of the cover, grout and foundation (Scenario 8). ■ Evaluation of removal of the foundation (Scenario 15).

Table 6-1: Summary of Evaluation of Uncertainty in the Performance of the Multilayered Barrier System

Analysis Component	Uncertainty	Evaluation of Uncertainty
Grout	■ Assumption that mass was distributed throughout grout with diffusion across foundation only; ■ Assumption of rates of degradation of grout; ■ Assumption of hydraulic conductivity of grout throughout assessment timeframe	■ Alternative solute transport scenario where mass is distributed in reactor only with diffusion across grout and foundation (Scenario 4). ■ Evaluation of groundwater flow through the source under enhanced degradation of the cover, grout and foundation (Scenario 8). ■ Evaluation of alternative hydraulic conductivity values to represent grout degradation (Scenario 14).
Source Term	■ Initial mass of non-radiological solutes and tritium ■ Initial mass of radionuclides	■ Base case assumes conservative estimate of radionuclide mass remaining. ■ Decay products that may not be transportable in water (i.e., gas phase) were included in the assessment. Evaluation of upper-bound mass inventory (Scenario 7 and Scenario 17).
Corrosion of reactor components	Rate of corrosion of reactor components	■ Conservative estimate of reactor corrosion rates used in base case assessment. ■ Evaluation of scenario with double the base case reactor corrosion rate (Scenario 9).
Degradation of Solutes	Rates of degradation for non-radiological solutes	■ Degradation of solutes not applied in base case model (notwithstanding radiological decay). ■ Evaluated the controls on degradation of xylene (Scenario 12).
Groundwater Receptor	■ Stage of Winnipeg River ■ Potential placement of future drinking water well	■ Evaluation of groundwater flow conditions and solute mass loading rates under low-stage condition (Scenario 13). ■ Evaluation of solute mass loadings at a drinking water well situated in the basal till unit at a location halfway between WR-1 and the Winnipeg River (Scenario 16).

6.1 Reactor Core and Bioshield Components

The majority of the remaining contamination in WR-1 Building is located within the piping and tanks that make up the reactor systems (primarily in the calandria and fuel channels). The contamination is both on the internal surfaces (surficial contamination) as well as embedded in the material itself (activated components). In some cases, the components themselves are the contaminant (e.g., lead). These system components are the initial barrier and must first breakdown through corrosion and dissolution in order for contamination to be released to any groundwater. Prior to their corrosion and dissolution, no contamination within them will be released. Breakdown of the reactor system components is expected to occur gradually over thousands of years. Corrosion rates for the reactor materials were based on estimates from literature for an aerobic environment and ranged from 1.78E-3 m/yr for aluminum to 1.0E-8 m/yr for Ozhennite and Zr-Nb alloy. Details on the selection of corrosion rates and calculation of times required for complete dissolution of each reactor component are provided in Golder's groundwater flow and solute transport modelling document (Section 4.1.3 of Golder 2021).

Due to the uncertainty associated with the breakdown of the reactor system components over time, this concept was explored in the context of a sensitivity analysis.

Timescales Associated with Reactor Corrosion (Scenario 9)

Solutes originating from the metal reactor components remaining after decommissioning were released in the solute transport model gradually over time to account for the corrosion of these materials. Rates of corrosion were assumed based on the “best estimate” values available from literature and were applied to both sides of the reactor materials. Scenario 9 was completed in order to evaluate the influence of the corrosion rates on the model results. For this simulation the corrosion rates were doubled for all materials.

Under this scenario the base case corrosion rates for the reactor components were increased by a factor of 2. The species contained in the metal of the reactor components are released congruently with the corrosion of the reactor hence the mass release rate of species from the reactor components is effectively doubled, notwithstanding the effect of decay. Species with 100% of the inventory contained in the reactor components, long half-lives, and non-sorbing properties (such as carbon-14) exhibited a 100% increase in the peak mass loading rates with negligible change in the time of peak mass loading. For species with 100% of their inventory in the reactor components but with some degree of sorption in the upper bedrock pathway (such as niobium-94, and nickel-59) the peak mass loading rate increased depending on the degree of sorption and half-life of the particular species. Species such as caesium-137 and strontium-90 showed no change with respect to a doubling of the corrosion rate of the reactor components when compared to the base case (i.e., the mass loading rates were zero), as the shorter half-lives of these radionuclides resulted in their total decay prior to release.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loading for carbon-14, one of the main project dose contributors, would result in a less than 2-fold increase in the total project contribution to public dose. For non-radionuclides, there was no appreciable difference in peak loadings or timing of peak loading. Project contribution hazard quotients would not be appreciably different from the base case.

Upper-bound Source Term Estimate for Non-radionuclide Solutes and Tritium (Scenario 7)

Factors of uncertainty were provided for the mass inventories of non-radionuclides and tritium in CNL 2020b. For this scenario, the upper end mass estimates of these non-radionuclides were applied. It was noted in the documentation for mass inventory of the non-radionuclide solutes (CNL 2020b) that the mass estimates are conservative and therefore have not been changed.

In this scenario the timing of the peak mass loading rates was essentially the same as the base case scenario. The scaling of the peak mass loading rate was proportional to the increase in mass specified in the source area (e.g., a ten-fold increase in mass resulted in a ten-fold increase in the peak mass loading rate). The exceptions to this were HB-40 and lead, which are controlled by solubility constraints and had simulated peak mass loading rates that were similar to the base case.

The doses to receptors are generally proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loading for tritium, one of the main project dose contributors, would result in a less than 2-fold increase in the total project contribution to public dose. Some non-radionuclides increased in peak loading, which would result in a proportional increase in project contribution hazard quotients.

The very low estimated Project impact from non-radionuclide solutes is such that a margin of error in inventory of several orders of magnitude would still fall within the acceptance criteria of the assessment. The confidence in the bounding nature of the inventory estimates for non-radionuclide solutes further reinforces that any uncertainty in the inventory is unlikely to result in any increase to estimated HQs from the Project.

Upper-bound Source Term Estimate for Radionuclides (Scenario 17)

Uncertainty associated with the mass inventories of radionuclides in CNL's *WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* (CNL 2020b) was addressed through Scenario 17, where an order of magnitude increase in radionuclide inventory was applied in the model (except for tritium, which was addressed in Scenario 7). Non-radionuclides were specified per the base case inventory; sensitivity to non-radionuclide inventories was addressed in Scenario 7.

The increase in radionuclide inventory by a factor of 10 generally resulted in an increase in peak mass loadings by a proportional amount for the affected solutes. For non-radionuclide solutes that are also produced through decay (e.g., copper), the increase in peak mass loading rates was not proportional to the increase in mass inventory. The simulated time to reach the peak solute mass loading rate was generally within 1% of the base case estimates.

The doses to receptors are proportional to loadings for any given radionuclide. The very low estimated Project impact from radionuclide solutes is such that a margin of error in radionuclide inventory of several orders of magnitude would still fall within the acceptance criteria of the assessment. The confidence in the conservatism of the estimated inventory values further supports the assertion that uncertainty in the inventory is not likely to result in an increase in the estimated dose rates from the Project.

6.2 Grout

As described in Section 4.1.1.2, the grout will fill the majority of the remaining ISD facility below-grade, as well as the contaminated reactor system components. The primary function of the grout is to prevent subsidence of the building over time. The safety case for the WRDF is built on the conservative assumption that the only significant aspect of the grout to its function as a barrier is the hydraulic conductivity of grout used in the groundwater flow model. Variations in hydraulic conductivity of the grout (See Scenario 8 below) affect the rate of water movement through the WRDF and thus influence the performance of the containment system; however, they do not control the overall safety of the containment system.

To confirm that the hydraulic conductivity of the grout does not control the overall safety of the containment system, Scenario 14 of the solute transport model, described in more detail below, assumes the grout rapidly degrades to match the condition of the surrounding geological layers. In the results of this scenario, there was no appreciable difference in peak loadings or the timing of peak loadings. Doses and risks from this scenario would not be appreciably different from the base case. This confirms that while the grout hydraulic conductivity does influence the overall solute transport model, it is not a controlling parameter and that its complete failure still allows the disposal system to provide protection of the public and the environment.

To support the defence-in-depth principles, the grout will also provide several benefits not explicitly assumed in the assessment models. The physical presence of grout will reduce the total amount of groundwater in the ISD envelope (See Scenario 4 below) and thus reduce the amount of water available for various chemical and corrosion reactions. The grout will also provide a favourable chemical environment due to its high pH, which will slow down the rates of corrosion of steel-based reactor vessel and various components. This reduction in corrosion due to grout pH is not included in the assessment to further provide a degree of conservatism.
The grout will also prevent easy access to the waste should human intrusion into the WRDF occur through the soil cover and reinforced concrete cap.

Inclusion of the Grout (Scenario 4)

One of the key assumptions of the base case model was that the source mass was distributed evenly throughout the grout block and that solute transport occurred through the foundation, which was conceptualized as a 1 m thick concrete material. This assumption is conservative, as the grout itself is anticipated to provide some additional separation between the source mass (most of which is confined within the metal components of the reactor) and the downstream environment. Further, the mass release from the source area is diffusion-dominated (i.e., mass released through advection represents a minor component of the overall mass loadings). This scenario explores the potential for grout to act as a protective barrier by including it as a material between the source and the foundation in the solute transport model, thereby increasing the diffusive length between the source and the backfill. For this scenario, the source is conceptualized as the main reactor area, including the reactor itself and surrounding bioshield. Under this scenario, solute mass must migrate through the grout and foundation before release to the surrounding environment occurs.

Inclusion of the grout as a barrier to solute migration generally resulted in considerably lower peak mass loading rates and significant increases to the time of peak mass loadings. For solutes with relatively short half-lives (e.g., tritium), the mass loading rates were reduced by more than an order of magnitude. These changes reflect the additional time required for solute mass to migrate through the grout prior to reaching the foundation. Exceptions to this can occur for solutes where the advective mass loading was the predominant component of the total mass released from the source area (e.g., for carbon-14, this resulted in a minor increase to peak mass loadings at the bedrock pathway outflow). Solutes such as chlorine-36 were affected by the inclusion of the grout with regard to the timing of the peak mass loading value; chlorine-36 experienced a delay of over a factor of 2 and an 81% reduction in the estimated peak mass loading. Species such as strontium-90 and caesium-137, which had no mass flux at the bedrock outflow in the Base case scenario also had no mass flux following the addition of the grout to the pathway. In general, the grout adds an additional effective barrier.

It should be noted that for this scenario the simulated groundwater flow rates through grout (as a whole) were used to represent the flow rates through the source area. In reality, the flows through the source area should be smaller to reflect the smaller footprint of the reactor.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, there was a decrease in peak loadings for tritium, a slight increase for carbon-14, and no appreciable change in loadings for the other main project dose contributors. Because carbon-14 is the primary contributor to peak dose, this would result overall in a less than 10% increase in the total project contribution to public dose, and still several orders of magnitude below the project acceptance criteria. For many non-radionuclides there was a decrease in loading, as compared to the base case, which would result in a proportional decrease in project contribution hazard quotients.

Timescales Associated with Degradation of the Cover, Grout and Foundation (Scenario 8)

In the base case post-closure solute transport model the flow rates through the cover, grout and foundation increased through time to account for degradation of these materials. These timescales were identified in Section 5.7 Management of Uncertainty as a potential source of uncertainty. As such, for Scenario 8, the time taken for each step of the degradation function was cut in half (i.e., the maximum flow rate through the building materials is reached in half the time from the base case simulation).

The simulated mass loading rates from the bedrock pathway were relatively insensitive to changes in the timescales associated with degradation applied in the model. This is a reflection of the base model configuration, where the source mass was assumed to be distributed throughout the grout. Under this setup, the downstream peak mass loading rate is controlled by diffusion of mass from the source area, and less so by the mass transported through advection. As such, the model would be more sensitive to the diffusion coefficients.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, a slight increase in the peak loading for carbon-14, one of the main project dose contributors, would result in a less than 20% increase in the total project contribution to public dose.

Rapid Grout and Foundation Degradation (Scenario 14)

The grout performance specification for hydraulic conductivity (9.5E-10 [CNL 2017b]) is approximately 50 times lower than the base case model value (i.e., a hydraulic conductivity of 5E-08 m/s). However, the site-specific long-term performance of the grout (i.e., the rate of increase in hydraulic conductivity with material degradation) is uncertain. A simulation was completed to evaluate the potential changes to mass loading rates resulting from a rapid degradation of the grout. This change was achieved by increasing the hydraulic conductivity of the grout to 5E-07 m/s after 100 years (matching the highest value of the surrounding geological units). Because groundwater flow exiting the grout must pass through the foundation to be released to the environment, the hydraulic conductivity of the concrete foundation material was also increased to 5E-07 m/s after 100 years such that it would not limit the groundwater flow through the grout. The rate of recharge through the cover was also increased to 8 mm/year, which is equivalent to the recharge at the end of the base case simulation during the final step in degradation. This scenario is representative of a case in which the grout, the effect of grouting on the foundation, and the performance of the foundation as an effective barrier is limited to the first 100 years following decommissioning.

The more rapid degradation of the grout and associated degradation of the foundation resulted in significant (two orders of magnitude) increases to the rate of flow through the grout, foundation and backfill for the period up to 5,000 years following decommissioning, after which flows in this Scenario were identical to the Base Case simulation.

Radionuclides associated with the reactor (such as carbon-14) were not sensitive to the increase in groundwater flows resulting from the more rapid degradation of the grout and foundation. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the degradation of the other barriers. For species contained only in the biological shield, such as chlorine-36, the increased flow through the grout and foundation resulted in a maximum increase in peak mass loading value by a factor of less than two. The results for strontium-90 and caesium-137 presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, there was no appreciable difference in peak loadings or the timing of peak loadings. Doses and risks from this scenario would not be appreciably different from the base case.

6.3 Internal Walls

Internal building walls and floors may provide an additional barrier between sections of grout however, this is not relied on in the safety analysis. Although penetrations exist in these interior walls to allow services to pass between rooms, they are mostly sealed for operational purposes such as fire-stopping. Any remaining penetrations are plugged by grout during the grouting process and are small in relation to the walls themselves. The internal walls and floors are largely painted or otherwise coated to seal the concrete to protect it from wear, provide traction, or for preventing internal contamination of the concrete. These coatings provide a waterproof barrier in many locations and will limit the speed at which water may move in to, and out of, the structure, further limiting the speed at which the system components or grout may degrade, and the rate at which the contamination can leave the grouted areas. The sealants on these walls must first degrade to allow more prominent water movement in the materials, before degradation of the concrete can begin. Conservatively, this is not relied upon in the assessment. Concrete degradation, like the grout, is expected to occur over longer periods, and occur gradually. Section 6.2 Grout, Scenario 14 includes uncertainties with regards to internal walls as part of the scenario, as it models the interior of the WRDF as a homogeneous material subject to overall rapid degradation.

6.4 Building Foundation

Decommissioning of the WR-1 Complex will involve removal of the service wing and east annex (including the underlying crawl space), which will be backfilled to the specifications determined by a qualified contractor and following the guidance of the WL Closure Land-Use and End-State Plan. Following decommissioning, the original foundation walls and floor slab for the central reactor portion of the WR-1 Building will remain in order to form a barrier for preventing the release of solutes within the grout block. The external walls or concrete surround of WR-1 provide a discrete, continuous, final engineered barrier to the release of contamination. Penetrations in the exterior walls are given engineered seals to provide for long-term performance. Like the internal walls, the exterior walls largely have sealants that restrict water movement. As with the internal walls and grout, degradation of the exterior walls and foundation is expected to occur over long periods of time and gradually. The base case hydraulic conductivity value for the building foundation was specified to be over 5 times higher than the highest measured value from the WR-1 building condition assessment (i.e., 5E-10 m/s in the model as compared to up to 9.8E-11m/s measured values). The assumed degradation schedule is provided in Table 4-4 of *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021), which specifies a degradation to a hydraulic conductivity of the surrounding soil over 10,000 years. Due to the uncertainty associated with the foundation degradation with time and the initial condition of the foundation, this concept was explored in the context of a sensitivity analysis.

Whiteshell Reactor Disposal Facility Barrier Failure (Scenario 3)

The foundation floor and walls for the WR-1 Building were specified in the base case post-closure simulations as a 1 m thick continuous material with a uniform hydraulic conductivity. This scenario was configured to evaluate the potential changes in groundwater flow rates through the building materials in the event of a failure of the foundation. For this simulation the base case groundwater flow model was reconfigured to have a 2 m-wide zone of enhanced hydraulic conductivity (5E-06 m/s or 10x the hydraulic conductivity of the groundwater transport pathway) within the foundation floor. This scenario is considered unlikely as the building is founded on bedrock and failure of the foundation due to further settlement is not expected.

The local failure of the foundation resulted in an increase to the early-time flows through the grout, which produced minor increases to peak mass loading rates at the bedrock pathway outflow location relative to the base case. Only those solutes with zero sorption experienced significant increase in peak mass loading, and in all cases was less than an 8% increase. In general, the model results were not sensitive to the presence of a local failure.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, there was no appreciable difference in peak loadings or the timing of peak loadings. Doses and risks from this scenario would not be appreciably different from the base case.

Removal of the Building Foundation (Scenario 15)

The base case groundwater flow model used a value of 5E-10 m/s to represent the hydraulic conductivity of the foundation. This value is approximately five times higher than the highest foundation hydraulic conductivity value measured as a part of the Building Condition assessment (Golder 2019a). However, a simulation was completed to assess the effect of a fully compromised building foundation. This simulation represents the condition where the building foundation is effectively removed at the beginning of the closure period. This is considered to be unrealistic, though has been included to provide a basis for the level of protection provided by the foundation and to support the evaluation of defence-in-depth principles as a part of the WRDF safety assessment. In the groundwater flow model this change was achieved by increasing the hydraulic conductivity of the foundation to 5E-07 m/s (matching the highest value of the surrounding geological units) for the duration of the post-closure simulation.

The compromised foundation scenario resulted in an increase to the flows through the grout, the foundation and the backfill relative to the base case. The increase in flows through the foundation were up to 14 times greater than the base case simulation at time zero, and gradually decreased to the base case value for the long term (greater than 10,000 years).

Radionuclides associated with the reactor (such as carbon-14) were not sensitive to the condition of the foundation. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the increased flow. For species contained only in the biological shield, such as Cl-36, the times to reach peak mass loading rates were reduced since the solutes found in the biological shield were immediately available for release at the time of saturation. The results for strontium-90 and caesium-137 presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loadings for tritium, one of the main project dose contributors, would result in a less than 20% increase in the total project contribution to public dose. Xylene, a COPC in the ERA, also increased in peak loading, which would result in a proportional increase in Project contribution hazard quotients.

Timescales associated with degradation of the foundation are evaluated in Section 6.2 Grout, Scenarios 8 and 14.

6.5 Local Hydrogeology

Solute release from the source (WR-1) would be transported via a groundwater pathway through the geological environment to the ultimate discharge location (i.e., the Winnipeg River). The groundwater pathway from the building foundation includes backfill (soil materials used to fill the excavation following construction of the WR-1 Building), the basal overburden unit and upper bedrock. The soil conditions at WR-1 provide an additional barrier to release of contamination into the environment. The local soils are primarily clay-based and provide a natural barrier to groundwater movement. The principal groundwater gradients in the clay soils above the basal layer and bedrock interface are vertical (i.e., downward flow) and horizontal gradients in the clay are small in comparison. Given the low permeability soils above the basal layer there is no expectation of movement upwards to the surface from the WRDF or downgradient to the river.

Groundwater velocity along the flow path is estimated to be approximately 5 m per year (translating to a travel time of approximately 100 years between the WRDF and the Winnipeg River). Sorption of contaminants along the flow path is anticipated, which would extend the travel times, retard and (for radionuclides) reduce mass loading rates at the discharge location. Due to the uncertainty associated with the local hydrogeology, groundwater pathways were evaluated in the context of the sensitivity analyses. The geosphere parameters investigated were shown to be more important for short lived or instantly released solutes. As a result, significant attention was given to measuring and characterization the local geosphere (CNL 2021a, Dillon 2018). The collected data, combined with the margin of error provided in the results, provides high confidence that the base case values are appropriate.

Hydraulic Conductivity of the Backfill (Scenario 10)

The groundwater flow model used a single value of 1E-07 m/s to represent the backfill material surrounding the WRDF. The simulated flow rate through the backfill material was used in the solute transport model to represent the flow through the bedrock geological pathway (assuming that no additional background water dilutes the mass concentrations in the pathway). For Scenario 10, the hydraulic conductivity of the backfill was increased to 1E-6 m/s and the recharge rate applied over this unit was increased from 2 to 20 mm/yr.

The higher flow rates through the backfill that occurred under this scenario generally resulted in higher peak values and earlier peak arrivals, reflecting earlier release of mass from the grout. This is due to the increase in concentration gradient between the grout and backfill, as flows through the grout decreased under this scenario (an indication the diffusion-dominated mass release process).

Radionuclides associated with the reactor (such as carbon-14) were less sensitive to changes in the hydraulic conductivity of the backfill. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the increased hydraulic conductivity. For species contained only in the biological shield, such as chlorine-36, the times to reach peak mass loading rates were reduced, since the solutes found in the biological shield are immediately available for release at the time of saturation. The results for strontium-90 and casieum-137 presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loading for tritium, one of the main project dose contributors, would result in a less than 10% increase in the total project contribution to public dose. For lead, a COPC in this assessment, the increased peak loading, would result in a proportional increase in the project contribution hazard quotient.

Preferential Pathway (Scenario 1)

Uncertainty in the geological pathways that exist between the WR-1 Building and the Winnipeg River was assessed through the inclusion of a preferential pathway in the solute transport model. Conceptually, this is intended to represent a geological or man-made feature that would provide an enhanced hydraulic connection through the groundwater flow system. The flow rate through the pathway was set to be a factor of 10 greater than the flow rate specified in the groundwater transport pathway (which was maintained in the simulation). This scenario could also represent the condition where future geomorphological changes bring the groundwater discharge location (i.e., the Winnipeg River) closer to WR-1.

The preferential pathway scenario generally resulted in earlier peak arrival, and higher peak loadings. The earlier arrival of peak mass loadings is attributed to a reduction in travel time due to the increased flow rate through the pathway. The higher peak values are attributed to an increase in the concentration gradient between the grout and the backfill as the higher flow rate “flushes” the backfill with clean water. For radionuclides with relatively short half-lives (e.g., tritium), the reduction in travel time resulted in significant increases in peak mass loading rates. Radionuclides associated with the reactor (such as carbon-14) were less sensitive to the change in travel times because of their longer half-lives. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the preferential pathway. For solutes such as chlorine-36 contained in the biological shield the time of peak mass loading rates are more affected by the preferential pathway since the solutes found in the biological shield are immediately available for release at the time of saturation.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loadings for tritium, one of the main project dose contributors, would result in a less than 30-fold increase in the total project contribution to public dose. Even with this increase, public dose is still below the 0.25 mSv/a dose constraint for the Project. Many non-radionuclides also increased in peak loading, which would result in a proportional increase in project contribution hazard quotients for COPCs.

Increased Hydraulic Conductivity of the Upper Bedrock (Scenario 11)

The bedrock pathway represents the only connection between the source area and the downgradient receptor in the solute transport model. For the base case model this was simulated as a continuous unit with a hydraulic conductivity of 5E-07 m/s. Because of the uncertainty associated with the selection of this parameter (which was originally derived through model calibration), the hydraulic conductivity of the bedrock was increased to double the base case value (1E-06 m/s) in the upper 5 m of the unit to represent an upper “weathered zone”.

Increasing the hydraulic conductivity of the bedrock had a minor to negligible influence on flows through the grout and backfill. However, the higher hydraulic conductivity of the rock resulted in higher groundwater velocities, which in turn resulted in increased peak mass loadings and earlier arrival of mass downstream for most solutes.

Radionuclides associated with the reactor (such as carbon-14) were less sensitive changes in the hydraulic conductivity of the upper bedrock. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the increased hydraulic conductivity. For species contained only in the biological shield, such as chlorine-36, the times to reach peak mass loading rates were reduced since the solutes found in the biological shield were immediately available for release at the time of saturation. The results for strontium-90 and caesium-137 presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loading for tritium, one of the main project dose contributors, would result in a less than 2-fold increase in the total project contribution to public dose. Many non-radionuclides also increased in peak loading, which would result in a proportional increase in project contribution hazard quotients.

Low River Stage (Scenario 13)

The stage of the Winnipeg River at the WL site is presently controlled by the Seven Sisters Dam, located approximately 7.5 km upstream (to the south) on the river. Future geomorphological changes to the river and dam are uncertain, and as such a simulation was completed to evaluate the potential changes to mass loadings resulting from a low river stage condition (i.e., higher gradient due to lowering of the downstream river head boundary, which controls outflow in the model). Based on the relationship between the flow at the Seven Sisters Dam and the stage of the River at the WL site in 2013 and 2014, the low flow periods at the dam correspond to a stage of approximately 254.6 masl. A value of 1.5 m was subtracted from the low stage condition to approximate a “dry river” scenario. In the groundwater flow model this was achieved by adjusting the constant head boundary condition in the post-closure groundwater flow model to an elevation of 253.1 masl (2.0 m lower than current conditions).

The low river stage simulation had a minor influence on the simulated groundwater flow rates through the grout, backfill and bedrock pathway. However, the change in hydraulic gradient in the bedrock pathway resulted in higher groundwater velocities and increased peak mass loading rates (similar to Scenario 11).

Radionuclides associated with the reactor (such as carbon-14) were less sensitive changes in the groundwater flow rates of the upper bedrock due to the effect of low river stage on the groundwater flow rate. Release of these radionuclides is governed by corrosion of the reactor components, hence limiting the effect of the groundwater flow rates. For species contained only in the biological shield, such as chlorine-36, the times to reach peak mass loading rates were reduced, since the solutes found in the biological shield were immediately available for release at the time of saturation. The results for strontium-90 and caesium-137 presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loading for tritium, one of the main project dose contributors, would result in a less than 2-fold increase in the total project contribution to public dose. Many non-radionuclides also increased in peak loading, which would result in a proportional increase in project contribution hazard quotients.

Lower-bound Sorption-Partition Coefficients (Scenario 5)

Partition coefficients specified for the bedrock pathway in the solute transport model were based on the available literature. As noted in the literature these values are dependent on the environmental factors encountered along the flow path (e.g., mineralogy, pH, competing ions). In order to account for some of this uncertainty, alternative “lower bound” partition coefficients were selected from the literature. Refer to Section 5.1 of the *WR-1 Groundwater Flow and Solute Transport Modelling Report* (Golder 2021) for details on the selection of lower-bound sorption-partition coefficients.

The lower-bound sorption-partition coefficients applied to the bedrock pathway resulted in earlier arrival of peak mass loading rates and higher peak values. For non-decaying, non-ingrowing species the timing and peak value are scalable to the change in Kd (e.g., cadmium). For radionuclides where solute mass in the pathway is dominated by the ingrowth from parent isotopes, the reduction in sorption-partition coefficients resulted in a reduction in the peak mass loading (e.g., actinium-225). For species that were modelled with no sorption the results remained unchanged with respect to either the time of arrival of peak mass loading or the value of the peak mass loading. For example, there was no change in the result for the solutes carbon-14, iodine-129, or chlorine-36. The results for strontium-90 and caesium-137 also presented no change from the base case (i.e., zero mass loading).

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, there was no change to loadings for tritium or carbon-14, the main project dose contributors. For secondary project dose contributors, there was a decrease in in polonium-210 loadings and an increase in plutonium-239 loadings, which would result in a less than 20% increase in the total project contribution to public dose. Many non-radionuclides increased in peak loading, which would result in a proportional increase in project contribution hazard quotients.

Upper-bound Sorption-Partition Coefficients (Scenario 6)

Similar to Scenario 5, upper bound partition coefficients were selected. Increases to the sorption-partition coefficients generally resulted in lowering of the peak mass loading, and time to reach the peak. It should be noted that due to the large increases in sorption-partition coefficients applied in this scenario, a number of solutes experienced peak mass arrival beyond the 500,000-year simulation period, though peak values would remain below the base case value for non-decaying, non-ingrowing solutes.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the loadings for most contaminants were much lower and would result in a much lower project contribution to public dose. Similarly, the project contribution hazard quotients for non-radionuclides would be much lower, proportional to the reduction in loadings.

Controls on the Degradation of Xylene (Scenario 12)

A conservative approach was adopted for the solute transport modelling with respect to xylene in that no sorption or degradation was applied for this radionuclide. However, the natural attenuation of xylene in groundwater is well documented and supported in the available literature (summarized in USGS [2006]). For this scenario, a typical degradation rate was selected based on the literature for xylene to account for its natural attenuation. A value of 238 days corresponds to the average time for degradation of half of the available xylene, as noted in field studies (USGS 2006). The applied degradation rate for xylene resulted in a reduction in peak mass loading rate from 4.3 g/yr in the base case to negligible mass arrival at the bedrock pathway outflow.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, there was no appreciable difference in peak loadings or the timing of peak loadings. Doses and risks from this scenario will not be appreciably different from the base case. For xylene, one of the COPCs in the ecological risk assessment, the substantial decrease in loadings would result in a proportional decrease in hazard quotients.

6.6 Concrete Cap and Engineered Cover

The concrete cap and engineered cover do not provide a barrier for release of contamination explicitly, but instead provides a means to limit additional water infiltration into the system and protect the barriers that are in place by resisting intrusion into the sub-surface structure which could potentially compromise one or more barriers. The concrete cap and engineer cover design will consider reducing the probability of unintentional intrusion by humans, plants and animals. The cover will degrade with time, much like the rest of the sub-surface structure. Figure 3.1.4-1 shows the multilayer WRDF. Due to the uncertainty associated with the design, infiltration rates were evaluated in the context of the sensitivity analyses.

Alternative Model Boundary for the Cover (Scenario 2)

In the base case, post-closure groundwater flow model, a specified rate of infiltration was applied to the cover based on assumptions regarding the cover characteristics. The infiltration rate ranged from 0.8 mm/yr. for the initial simulation to 8 mm/yr. for the final simulation stages (the rate increased to account for degradation of the cover material over time). To address the uncertainty associated with these assumptions the groundwater flow model was reconfigured to specify a constant head boundary throughout the cover to maintain a water table depth of half a metre below-ground surface for all stages of post-closure.

Results of the base groundwater flow model scenario with a constant head boundary situated in place of the concrete cap and engineered cover indicated that infiltration rates through the grout would be equivalent to approximately 135 mm/yr. of infiltration from surface. This value is larger than the maximum estimated net infiltration rate for sandy soils located in the northern portion of the WL area, which is calculated to be approximately 100 mm/yr on average for the period from 1982 through 1995 (Thorne and Hawkins 2004). The infiltration rate through the clay soils in the vicinity of the WR-1 Building are expected to be significantly (>10x) lower as compared to the infiltration through the sandy soils. As such, the infiltration rate through the cover was conservatively maintained at 135 mm/yr. for the solute transport model, and this rate was applied to all stages of degradation of the grout and foundation.

The increased infiltration through the cover zone mostly increases water flow through the permeable backfill around the WRDF, resulting in an approximately 40-fold increase in flow through the grout block of the WRDF. The 40-fold increase in flow through the grout resulted in a higher peak mass loading value and general reduction in the time of the peak value. The arrival time of the peak mass loading for species of interest (such as carbon-14) was reduced by more than half, although the peak values for these species were relatively unchanged. The peak values for radionuclides such as carbon-14 were less sensitive to the change in travel times given that the majority of the inventory for carbon-14 is contained in the reactor components, which limits their release.

The arrival times of peak mass loading for radionuclides that are contained in the biological shield or reactor systems, such as chlorine-36 and iodine-129, were affected by changes in flow through the grout, resulting in a maximum increase in peak mass loading value by a factor of approximately two. Other solutes of interest (such as the radionuclides strontium-90 and caesium-137) remained at zero mass loading. In general, solutes with shorter half-lives saw the most potential increase due to this change, as the increased flow through the system allows solutes to reach the river more quickly, reducing the relative amount lost due to decay. The increases were not considerable compared to the change in flow rate, therefore, the system relatively insensitive to changes in cover infiltration rates.

The doses to receptors are proportional to loadings for any given radionuclide or non-radionuclide. For this scenario, as compared to the base case, the increased peak loadings for tritium, one of the main project dose contributors, would result in a less than 50% increase in the total project contribution to public dose. Non-radiological COPCs lead and xylene also increased in peak loading, which would result in a proportional increase in project contribution hazard quotients.

Timescales associated with degradation of the cover are evaluated in Section 6.2 Grout, Scenario 8.

6.7 Post-closure Monitoring

The final barrier is post-closure environmental monitoring of the groundwater surrounding WR-1. Groundwater monitoring provides verification that the decommissioned WR-1, and the barriers to release, are performing their function as expected. Monitoring also provides an early warning system in the event that something unexpected has occurred and provides the data necessary to make decisions about mitigating actions required, if at all. The period of institutional control (0 to 100 years following closure) corresponds to the period when peak radionuclide activity is expected to occur (e.g., release of tritium), which will be measurable through ongoing monitoring.

7.0 CLOSURE SAFETY ASSESSMENT

This section provides the results of the safety assessment completed for the closure phase. Results for the closure assessment are provided for the radiological assessment completed for workers and the public, the non-radiological assessment completed for workers and the public, and the radiological and non-radiological assessments completed for non-human biota during the closure phase of the Project. The scope encompasses activities identified in Section 3.1.2 Project Activities, and the assessment timeframe encompasses closure activities (currently anticipated to extend from 2022 to 2026). Detailed methods and results for the closure assessment are provided in the ERA (EcoMetrix 2021).

7.1 Radiological Assessment for Workers under Normal Conditions

Based on the current work package breakdown, the HAZOP evaluation concluded that no workplace scenarios were identified as being ‘new’ and requiring safety case assessment to determine feasibility. Proposed activities are encompassed by controls, management programs, and procedures already in use at the WL site. As such, negligible release of radionuclides during closure are anticipated (i.e., exceedance of conditions experience during operations are not expected). Therefore, after grouting, contamination will be below-grade and no release and subsequent exposure modelling has been performed. During the development of detailed work plans, prior to execution of work packages, activity planning processes will be executed to confirm that activities can be carried out safely under current workplace control.

Routine hazards typically encountered are low levels (Contamination Zone 3 and lower [CNL 2021b]) of fission and corrosion products in the workplace. High levels of activation products, the presence of actinides or tritium, or working with activated components will be considered non-routine hazards. According to CSA N288.6-12, Nuclear Energy Workers who participate in a Radiation Protection Program do not require radiological assessment in the ERA because their radiation exposure is monitored, and their doses are controlled. Workers on the WL site will participate in CNL’s Radiation Protection Program. However, on-site WL workers have been assessed in the ERA for radiological exposures.

Pathways relevant to exposure to liquid effluent are considered to be “unlinked” during the closure phase, as releases to the surface environment are not expected. Proven waste management controls (e.g., secondary containment) and waste management practices will be operational during the closure phase.

7.1.1 Hazard Identification and Exposure Pathways

Radiological hazards associated with the Project include potential exposure to beta, gamma, and alpha-emitting radionuclides produced during the operational period of the WR-1 and associated facilities. These hazards primarily consist of fission products, activation products, tritium, and corrosion products. Fuel has been removed and the moderator and coolant have been drained, therefore, there is no risk of criticality occurring at the WR-1 Complex. In Table 7.1.1-1, the unique hazards and key assumptions per work package are provided.

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Table 7.1.1-1: Project Phases and Schedule

Work Package No.1	Work Package	Unique Hazards	Key Assumptions
1	**Preliminary Decommissioning:** Auxiliary Equipment Removal: 600/700 Level Labs, Solvent Recovery, Manlift, Crawlspace Remediation, Flask Removal (Fuel, Fuel Channel and Transfer Flasks)	■ No unique hazards are expected. ■ Some work will be in confined space with related safety issues to consider. ■ Some work will be in contaminated areas (e.g., crawlspace remediation). ■ Cutting of contaminated piping. ■ Heavy equipment operation. ■ Lifting and rigging required. ■ Radioactive water, organic coolant, particles and sludge may be present in some equipment.	■ Negligible radiological and industrial hazards. ■ Chalk River Laboratories will accept flasks for storage.
2	**Grout Installation**: Air Gapping/Penetrations, De-energize de-activate building systems inside ISD, Install temporary building services Primary Heat Transport System relocation, WRDF Construction (Grout Placement)	■ Tritium venting procedures will need to be implemented to provide worker safety. ■ Radioactive water, organic coolant, particles and sludge may be present in some equipment. ■ Effect on ventilation patterns when active ventilation systems are shutdown, removed or if an opening need to be cut. ■ Moderate radiation fields exist in the reactor vault, upper and lower access rooms. ■ Damaged asbestos may be present on piping or tanks that requires penetration or other preparatory work. ■ The need to remove some shielding to allow vessel preparation/grouting and use of temporary shielding to provide worker safety. ■ Airborne particulate from grout constituents may be hazardous to respiratory health and precautions will be taken to minimize dust generation and limit or prevent inhalation using respirators, as appropriate. ■ Heavy truck traffic may occur at times during delivery of grout materials. ■ Effect on ventilation patterns when ventilation pathways are filled with grout. ■ Structural reinforcements may be necessary to confirm grout pressure does not compromise the structure and blow outs do not occur. ■ Deflagration/exploration: aluminium scrap and aluminium alloys corrode when exposed to high pH (e.g., grout used at Savannah River National Laboratory has pH of 12+) producing hydrogen gas. ■ Structural failure: an unknown void space was discovered at Savannah River National Laboratory R-Reactor when a 3D CAD model was created.	■ Organic and tritium contamination expected. ■ Fragments of radioactive debris expected. ■ Grout material hazards can be mitigated by standard personal protection equipment and clothing. ■ Structure is sound and can withstand grout installation.
3	**Building Demolition and Capping:** De-energize de-activate building systems outside ISD, Removal of contaminated Services: piping HVAC, Bldg material, Building and stack Demo, Foundation Removal, Cap and Cover Installation, Site remediation and final grading	■ Work execution from heights, multiple floors above and below-grade. ■ Methods to backfill individual rooms in the lower levels need to be evaluated. ■ Radiological clearance needs to be completed well below ground prior to backfilling the excavation. ■ Heavy equipment operation. ■ Lifting and rigging. ■ Explosion during demolition.	■ Reactor Hall can be dismantled post-grouting. ■ Balance of WR-1 Above-grade building will be demolished. ■ Clearance waste may be used to fill the excavation.

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7.1.1.1 Radionuclides

The inventory of radionuclides used in this assessment is provided in Section 3 of the ERA (EcoMetrix 2021) and summarized in Table 7.1.1-2. Because the level of radioactivity present will depend on the decay rate of each particular radionuclide, consideration was given to the selection of a “release time” for the WR-1 radionuclide mass inventory. The WR-1 was permanently shut down and defueled in May 1985, and an inventory of radionuclides was developed for 50 years following the shutdown (year 2035). For the purposes of this assessment, 2035 is assumed to correspond to the “release” of mass to porewater in the grout used for the ISD (EcoMetrix 2021).

The source radionuclides in the WR-1 Building is from fission products and actinides, activation products, corrosion products, as well as radionuclides resulting from surface contamination during reactor operations. Fission products and actinides result from fuel failures releasing these products into the PHT system. Activation products would be found mainly in fuel channels, calandria vessel, thermal shields, biological shield, and structures inside the reactor vault. Corrosion products (iron-55, cobalt-60) produced in the PHT system are anticipated to be minimal due to the use of organic coolant. Surface contamination results from spills and leaks and handling of failed fuel during operations. The estimated radionuclide inventory for years 1995, 2015 and 2035 is provided in Table 7.1.1-3. The highest activity of radionuclides is expected to be limited to the reactor core as shown in Table 7.1.1-3, with small inventories of radionuclides in the reactor biological shield and PHT system.

Table 7.1.1-2: Radionuclides Associated with Main System and Components at WR-1

System / Component	Radionuclides
Reactor Core	carbon-14, iron-55, cobalt-60, nickel-59, nickel-63, niobium-94, silver-108m (minimal)
Primary Heat Transport System	caesium-137, strontium-90, cobalt-60, (small amounts of niobium-94, zirconium-95, antimony-125, europium-152, radium-226, americium-241), isotopic plutonium, technetium-99, iodine-129, curium-244, uranium-235, uranium-238
Biological Shield	carbon-14, chlorine-36, calcium-41, nickel-63, cobalt-60, europium-152
Heavy Water and Helium System	tritium, carbon-14
Corrosion products	carbon-14, chlorine-36, iron-55, nickel-63, nickel-59, cobalt-60, niobium-94
Surface contamination	caesium-137, strontium-90, isotopic plutonium, americium-241

Table 7.1.1-3: Estimated Radionuclide Inventory in Reactor System Following Shutdown

Reactor System	10 years (1995) (Bq)	30 years (2015) (Bq)	50 years (2035) (Bq)
Reactor Core	5.99E+15	1.16E+15	8.92E+14
Biological Shield	5.76E+10	n/a	1.25E+09
Primary Heat Transport System	1.86E+12	1.04E+12	6.19E+11
Total	**5.99E+15**	**1.16E+15**	**8.93E+14**

n/a = not available; Bq = Becquerel

7.1.1.1.1 Fission Products and Actinides

Fission Products are produced in the reactor fuel as a result of the nuclear fission process and are normally contained within the fuel cladding, however, fuel defects or failures can result in the release of fission products in small quantities. These small quantities are then circulated throughout the reactor core and the PHT system during the operational life of the reactor. Due to the experimental nature of the WR-1 operation, there were approximately 150 documented fuel failures in the reactor between 1966 and 1983.

Primary fission products of concern are caesium-137, strontium-90, technetium-99, iodine-129 and iodine-131. Another group of radionuclides of concern commonly associated with the nuclear fission process are certain actinides, such as americium 241, plutonium-238, plutonium-239, plutonium-240, plutonium-241 and curium-244. Actinides are the result of neutron activation of uranium-238 and the subsequent activation products and/or decay chain radioactive progeny (CNL 2021b).

7.1.1.1.2 Activation Products

Activation Products are generated when materials are subjected to neutron bombardment, therefore, primarily produced when components and structures are in close proximity to the high neutron flux of the reactor core. These components/structures absorb neutrons and as a result some of them become radioactive. Within the WR-1, Activation Products are anticipated to be present within the reactor core, the concrete biological shield, and the heavy water moderator system. Activation products may also be produced when debris from corrosion is transported through the reactor, absorbing neutrons in the process. Therefore, Activation Products would also occur within the PHT system and its ancillary systems. However, as the WR-1 used an organic based coolant minimal quantities of activation products would have been produced through this mechanism. Expected activation products of concern include cobalt-60, carbon-14, chlorine-36, iron-55, nickel-63, nickel-59, and niobium-94 (CNL 2021b), all identified as associated with the reactor core.

Principal radionuclides of concern for the activation of common structural material area as follows:

- Aluminum – carbon-14, iron-55, nickel-59, cobalt-60, nickel-63, and chlorine-36;
- Carbon Steel – carbon-14, iron-55, nickel-59, cobalt-60, nickel-63, and niobium-94;
- Stainless Steel – carbon-14, iron-55, nickel-59, cobalt-60, nickel-63, and niobium-94;
- Ozhennite – carbon-14, iron-55, nickel-59, cobalt-60, nickel-63, and niobium-94;
- Zirconium – carbon-14, iron-55, nickel-59, cobalt-60, nickel-63, and niobium-94; and
- Concrete – carbon-14 and chlorine-36.

7.1.1.1.3 Tritium

Tritium is an activation product of concern, which is produced when the heavy water moderator absorbs neutrons. Tritium is treated separately from other activation products as its properties and behaviours are different from other activation products and it is primarily associated with the moderator and associated systems. Tritium has the same chemical properties of water, so if it enters into the body, it delivers a whole-body dose because it will get distributed throughout the whole body.

The heavy water system and certain auxiliary systems, including the calandria, which is in close proximity to the reactor core, are the primary tritium hazards. All components of the heavy water system are located below the calandria vessel. Tritiated heavy water can be absorbed into the walls of pipes and tanks, and into concrete walls and floors.

Heavy water moderator was removed during preliminary decommissioning, including blowing back lines to the system hold tanks for recovery. Equipment was installed to recover residual heavy water by recirculating air through coolers, this system was purged to the active ventilation system at a controlled rate to achieve further dry out. Throughout its operation, tritium was absorbed into the metals of the system, and is now currently being released through off-gassing. For the past five years, tritium release has been consistent, confirming that tritium remains in the system. However, the current tritium release rate is well below the administrative level of 8 GBq/wk.

7.1.1.1.4 Corrosion Products

Corrosion Products are mobile activation products, produced when debris from corrosion or damage is transported though the reactor and become radioactive by absorbing neutrons. During the operation of a nuclear reactor, most metallic surfaces oxidize and form a layer of corrosion film. This layer erodes and is transported through the reactor core and is exposed to high pressures and temperatures. Corrosion products of concern are expected to be cobalt-60 and iron-55.

7.1.1.1.5 Radioactive Products Decay

Radioactive products decay to produce secondary “daughter” products, which may also be radioactive and decay in turn. Immediate, short-lived products within the decay series are not expected to remain within the environment for a significant duration and, therefore, were not considered.

7.1.1.2 Exposure Pathways

The potential effects from the ISD of the WR-1 include the release of airborne radioactive particulates produced during disconnecting of services, relocation of materials below-grade, grout preparation, and demolition activities. The potential effects from the ISD of the WR-1 include nuisance dust and fine particulates from disturbance activities including dismantling of the above-grade portion of the PHT system and grout application below-grade. Decommissioning the WR-1 Building following grouting is expected to result in negligible release of radionuclides, as CNL intends to characterize, survey, and decontaminate/immobilize residual contamination prior to demolition. Exposure pathways for on-site NEWs and non-NEWs, as well as other personnel on-site includes inhalation of air and contact with soil for outdoor workers (Table 7.1.1-4).

Table 7.1.1-4: Complete Exposure Pathways for Receptors for Exposure to Non-Radiological COPCs

Receptor	Exposure Pathway	Environmental Media
On-Site WL Worker	Inhalation	Air
	Dermal	Soil

The on-site WL worker was assumed to spend 40 h/wk and 50 wk/yr on the WL site. The worker is assumed to have no local intake since on-site drinking water would likely be obtained from the Winnipeg River. In the closure phase, contaminants are released to the atmosphere; therefore, there will be minimal effect on the Winnipeg River. In CSA N288.1-14 (2014), atmospheric releases can end up in groundwater (via infiltration through the soil) or in a small farm pond via atmospheric deposition. Deposition into a river or lake is considered negligible.

7.1.1.2.1 Grouting of Below-grade Structures and Systems

Just as decontamination activities disturbed surfaces, preparing surface for grout and grout application are expected to result in the release of radionuclides such as suspended particulate matter (SPM), particles nominally smaller than 10 micrometres (μm) in diameter (PM_{10}), and particles nominally smaller than 2.5 μm in diameter ($PM_{2.5}$). Individual radionuclide release rates and the total release rate from the reactor core, biological shield, PHT system, and active ventilation system were derived (see Section 3.1.1.2 of the ERA for detailed information [EcoMetrix 2021]).

7.1.1.2.2 Removal of Above-grade Reactor Structures

Prior to grouting, the above-grade portion of the PHT system (two heat exchangers and two outlet headers) will be dismantled and relocated below-grade. During the demolition activities, including sectioning, moving, and preparing the PHT for grouting, the release of radionuclides as SPM, PM_{10}, and $PM_{2.5}$ are anticipated. Individual radionuclide release rates and the total release rate from the PHT system were derived (see Section 3.1.1.1 of the ERA for detailed information [EcoMetrix 2021]).

7.1.1.2.3 Tritium Release during Closure

It is expected that tritium will be released from the helium and heavy water system during most of the closure activities at a rate similar to the maximum and average tritium release rates from the WR-1 Building from 2011 to 2019. A summary of maximum and average release rates from 2011 to 2019 is provided in Table 7.1.1-5 (CNL 2021b, 2019e,2018b, 2017c, 2016b, 2015a, AECL 2014, 2013). The average release rate for tritium was 1.11E+09 Bq/week (1.84E+03 Bq/s).The average release rate for tritium from 2011 to 2019 was 1.11E+09 Bq/week (1.84E+03 Bq/s). This is appropriate for the tritium release rate during demolition prior to grouting (i.e., demolition of the PHT system).

The tritium release rate to the atmosphere is expected to increase during activities associated with grouting due to vibration and heating of structures and systems, increased air flow, or venting of residual tritium. This assumption is based on a study CNL conducted on tritium releases during characterization activities associated with radiological characterization in the helium and heavy water system in 2015 (CNL 2015c).

As shown on Figure 7.1.1-1, the tritium release rate increased during characterization due to vibration and heating of the surface, and only began to decrease after characterization activities ended (CNL 2015c). The maximum weekly tritium release rate observed during characterization activities was approximately 1.28E+10 Bq/week and is appropriate to use as an expected release rate during grouting activities during the closure phase. Tritium during characterization studies was measured as total tritium. It has been assumed that tritium is in the form of tritiated water vapour – this is a conservative assumption since tritiated water partitions better to other media than elemental tritium. As indicated in CSA N288.1-14, elemental tritium is weakly absorbed by the body; therefore, any doses resulting from release of HT are due to the very small fraction, approximately 0.004% of elemental tritium that is converted to tritiated water in the human body.

It has been assumed that tritium will be released at the maximum release rate of 1.28E+10 Bq/wk for the entire 1-year duration of the grouting phase, although vibrating and heating activities are not likely to occur for the full duration (Table 7.1.1-5). This is a conservative estimate. A summary of the maximum and average predicted atmospheric tritium release rates from the WR-1 during the closure phase is provided in Table 7.1.1-6.

Table 7.1.1-5: Summary of Atmospheric Tritium Release Rates from WR-1 from 2011 to 2019

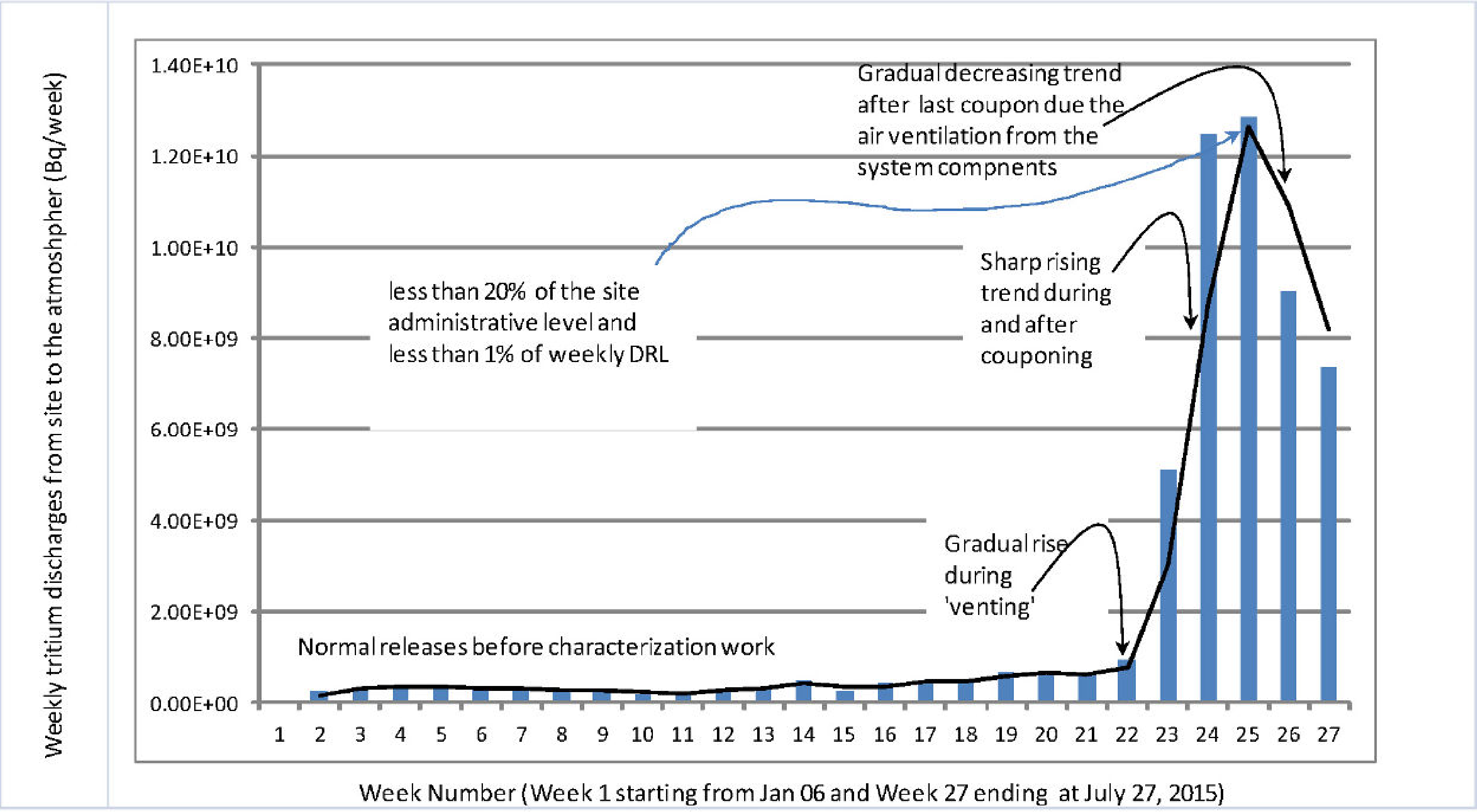
Year	Maximum (Bq/wk)	Average (Bq/wk)
2011	1.14E+09	6.01E+08
2012	5.55E+10[1]	3.66E+09
2013	1.30E+09	6.77E+08
2014	1.26E+09	6.69E+08
2015	1.28E+10	1.90E+09
2016	1.07E+09	6.24E+08
2017	6.33E+09	9.68E+08
2018	5.41E+08	2.51E+08
2019	1.61E+09	6.43E+08
2011 to 2019 Average	-	1.11E+09

Note:

1. Tritium was elevated in 2012 due to an approved non-routine release of a small volume of spent moderator water sample via the reactor stack, measured as tritium oxide (AECL 2013).

Table 7.1.1-6: Summary of Predicted Maximum and Average Atmospheric Tritium Release Rates from WR-1

Closure Activity	Maximum (Bq/s)	Average (Bq/s)
Demolition prior to Grouting	6.05E+03	1.84E+03
Grouting	2.12E+04	2.12E+04
Demolition post Grouting	None	None



REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

TITLE
WEEKLY TRITIUM RELEASE FROM STACK BEFORE AND AFTER RADIOLOGICAL CHARACTERIZATION ACTIVITIES

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	7.1.1-1

7.1.2 Planned Mitigation

All decommissioning work will be planned and executed in accordance with the Decommissioning Licence NRTEDL-W5-8.00/2024, WL Decommissioning Quality Assurance Plan, Work Planning for WL Decommissioning, Radiation Protection Program, Occupational Safety and Health Program, Environmental Protection Program, WL Emergency Plan, and Engineering Change Control. Design of and sequential removal of critical systems during decommissioning to provide fail-safe modes, reliable power supplies, maintenance programs, and defence-in-depth minimizes the likelihood of upset conditions occurring.

7.1.2.1 Workplace Planning

The scope of the Project refers to various closure activities that will be planned and executed consistent with existing operational controls, management practices, and standard operating procedures. This includes airborne and liquid effluent control, treatment, verification and release, as well as drinking water and foodstuff supply on-site. This section examines the current radiological conditions of the WR-1 Building, including the restricted access area. These are the hazards associated with the occupation and work execution. Radiation dose rate hazards in most rooms range from minimal to low, with moderate hazards being limited to only a few rooms. Areas with elevated gamma radiation levels are in the reactor core lower and upper access rooms and were associated with the PHT system and components, experimental loop, process drain lines.

7.1.2.1.1 Dose Estimates

CNL’s Radiation Protection Program requirements include the segregation of the work areas into Radiological Safety Zones (RSZ) to manage radiological hazards. The zoning is based on readily accessible external radiation fields and surface contamination levels present in the area. There are five RSZs based on dose rates and contaminations levels, as follows:

- Zone 1 – suitable for unrestricted occupancy and includes normal office areas and access corridors and washroom. The dose received by an individual from external sources of radiation during continuous occupancy should not exceed 1 mSv/a.
- Zone 2 – suitable for normal continuing occupancy, as is normally free of radioactive contamination, but may be subject to infrequent cross contamination from higher numbered zones. Zone 2 areas normally act as a buffer area between a higher numbered zone area (3 or up) that contains removable contamination and a Zone 1 area.
- Zone 3 – is considered a zone of medium occupancy, and such occupancy is subject to continual review by the Line Management and Group 1 employees. Activities generating removable surface contamination in localized areas, and for short periods of time may be permitted in a Contamination Zone 3 area. Effort should be made to eliminate removable contamination upon discovery, or as soon as practicable, based upon ALARA consideration. If the removable contamination cannot be immediately eliminated, a hazard sign should be locally posted until cleaning is completed.
- Zone 4 – is a zone of restricted occupancy, with access and work controls enforced. Entry should be infrequent and accomplished according to established procedures aimed at controlling doses.
  Such procedures should be reviewed and approved either in advance or at the time of entry, by a Group 1 employee. Exceptions shall be reviewed and approved by the responsible Radiation Protection Program Manager. Group 1 employees should be involved for all entries into Zone 4 areas through either a review of work procedures or the conduct of radiological surveys.

- Zone 5 – is a zone with a dose rate in excess of Zone 4 levels. All entries into a Zone 5 require a work permit, in which the radiological assessments and radiation protection measures are approved by a Group 1 employee, preferably a Health Physicist. All such entries required the constant attention of a Group 1 employee under a Group 1 employee direction throughout the duration of the entry.

In addition, a Supervised Area is an area in which the working conditions are kept under review by Group 1 qualified employees, but special radiation protection procedures are not normally needed. Access to a supervised area is controlled. Work with radiation sources or the storage of radioactive material are not permitted within a Supervised Area without authorization from a Group 1 qualified Radiation Surveyor or Health Physicist. Annual radiation exposures are not expected to exceed regulatory dose limits for members of the public.

In accordance with CNL's Radiation Protection Program, the WR-1 Building has been divided into the previously described RSZs. Dose rates and contamination levels for the five RSZs are summarized in Table 7.1.2-1.

Table 7.1.2-1: Classification of Radiological Safety Zones

Radiological Safety Zone	Radiation Hazard Description	Radiation Zone Criteria	Contamination Zone Criteria
		Average Whole Body Dose Rate (1 m from any surface)	General (Accessible) Removable Surface Contamination Levels
1	Very Low	≤ 0.5 µSv/h (50 µrem/h)	< maximum value for restricted use
2	Low	> 0.5 µSv/h (50 µrem/h) ≤ 10 µSv/h (1 mrem/h)	<maximum value for restricted use
3	Moderate	> 10 µSv/h (1 mrem/h) ≤ 1.0 mSv/h (100 mrem/h)	> 1 and ≤ 10 times maximum value for restricted use
4	High	> 1.0 mSv/h (100 mrem/h) ≤ 100 mSv/h (10 rem/h)	>10 times maximum value for restricted use
5	Very High	> 100 mSv/h (10 rem/h)	>10 times maximum value for restricted use; based on Health Physicist judgement, airborne contamination levels and/or external beta radiation fields present an acute hazard such that unplanned doses could realistically exceed regulatory limits or deterministic threshold level.

mrem = millirem; µmrem = micromillirem; µSv/h = microsieverts per hour; <= less than; >= more than; ≤ = equal to or less than; ≥ = equal to or more than.

Following the completion of Phase 1 decommissioning activities in 1995, WR-1 was divided into two general access areas: WR-1 unrestricted access and WR-1 restricted access. The WR-1 unrestricted access area consists of rooms that underwent decommissioning and decontamination under the Phase 1 decommissioning activities and had radiological hazards reduced to background or minimal levels meeting either RSZ 1 or 2 hazard conditions. This includes most of the rooms on Levels 500, 600 and 700. The WR-1 restricted access area is comprised of rooms that have not undergone any decommissioning activities or had remaining elevated radiological hazards following the completion of Phase 1 decommissioning consistent with either RSZ 2 or 3 hazard conditions. This includes rooms and areas on Levels 100, 200, 300 and 400, and some rooms on Levels 500 and 600. The DDP (CNL 2021b) provides a room-by-room summary of measured radiation and contamination levels. Table 7.1.2-2 highlights rooms with elevated radiological hazards. Removable surface contamination is limited to mixed fission products and actinides. All rooms are free of tritium surface contamination.

Table 7.1.2-2: Rooms with Elevated Radiological Hazards

Room	Description	Zoning	Comments
103	■ Drain Tank Room: ■ Primary Heat Transport System, Spent Fuel Handling and Storage Systems, Active Drainage System	R3C2	■ 5 mrem/h (50 µSv/h) average room gamma dose rates with localized elevated fields ranging from 5-55 mrem/h (50-550 µSv/h). 200 mrem/h (2.0 mSv/h) near contact hot spot. ■ Free of removable surface contamination.
104	■ Degassing Room: ■ Primary Heat Transport System, Heating and Cooling Systems	R3C3	■ 5-10 mrem/h (50-100 µSv /h) average room gamma dose rates with localized elevated fields of 55 mrem/h (550 µSv/h). 20-1,000 mrem/h (0.2-10 mSv/h) near contact hot spots. ■ Low level removable surface contamination.
201	Lower Access Room	R4C3	■ 25 mrem/h (250 µSv/h) average room gamma dose rates. 100 mrem/h (1 mSv/h) near contact hot spot. ■ Low level removable surface contamination.
301	Flask Maintenance Low Level Room	R3C3	■ 100 mrem/h (1 mSv/h) average room gamma dose rates. 5 rem/h (50 mSv/h) near contact hot spot (stored waste can). ■ Generally free of removable surface contamination.
302	■ Degassing Room: ■ Primary Heat Transport System	R3C3	■ 3-8 mrem/h (30-80 µSv/h) average room gamma dose rates with localized elevated fields of ~10 mrem/h (~100 µSv/h). 1,000 mrem/h (10 mSv/h) near contact hot spot. ■ Generally free of removable surface contamination.
409	Surge Tank & Pipe Shaft Room: Primary Heat Transport System	R3C3	■ 5 mrem/h (50 µSv/h) average room gamma dose rates. 40 mrem/h (400 µSv/h) near contact hot spot. ■ Generally free of removable surface contamination.
410	■ WR-1L1 Loop Room: ■ WR-1 1L1 Experimental Loop	R3C3	■ 1 mrem/h (10 µSv/h) average room gamma dose rates with localized elevated fields ranging from 2-18 mrem/h (20-180 µSv/h). No hot spots. ■ Generally free of removable surface contamination.
501	Upper Access Room	R3C3	■ 5 mrem/h (50 µSv/h) average room gamma dose rates. 200 mrem/h (2 mSv/h) near contact hot spot. ■ Low level removable surface contamination.
504	■ Auxiliaries Room: ■ Thermal Shield Cooling System	R3C3	■ 1-4 mrem/h (10-40 µSv/h) average room gamma dose rates. 20 mrem/h (200 µSv/h) near contact hot spot. ■ Generally free of removable surface contamination.

Table 7.1.2-2: Rooms with Elevated Radiological Hazards

Room	Description	Zoning	Comments
506	■ Header Room: ■ Primary Heat Transport System	R3C3	■ 5 mrem/h (50 μSv/h) average room gamma dose rates with localized elevated fields ranging from 8-16 mrem/h (80-160 μSv/h). 200-4,000 mrem/h (2-40 mSv/h) near-contact hot spots. ■ Low level removable surface contamination.
537	■ WR-1L5 Loop Room: ■ Fast Neutron Loops	R3C3	■ 1-5 mrem/h (10-50 μSv/h) average room gamma dose rates. 400 mrem/h (4 mSv/h) near contact hot spot. ■ Generally free of removable surface contamination.
538	■ WR-1L4 Loop Room: ■ Fast Neutron Loops	R3C3	■ 1 mrem/h (10 μSv/h) average room gamma dose rates with localized elevated fields of 5 mrem/h (50 μSv/h). No hot spots. ■ Low level removable surface contamination.
539	■ WR-1L2 Loop Room: ■ WR-1L2 Experimental Loop Fast Neutron Loops	R3C3	■ 10 mrem/h (100 μSv/h) average room gamma dose rates. 60 mrem/h (600 μSv/h) near contact hot spot. ■ Moderate level removable surface contamination.
540	■ WR-1L2 Sample Station & Transmitter Room: ■ WR-1L2 Experimental Loop	R2C3	■ 0.2 mrem/h (2 μSv/h) average room gamma dose rates. 8 mrem/h (80 μSv/h) near contact hot spot. ■ Low level removable surface contamination.
601	Caged Storage Area	R2C3	■ 0.02-0.5 mrem/h (0.2-5 μSv/h) average room gamma dose rates. 5-80 mrem/h (50-800 μSv/h) near contact hot spots (on stored flasks). ■ Generally free of removable surface contamination.
602	■ Primary Pump Room: ■ Primary Heat Transport System A and B circuit main heat exchangers	R3C3	■ 5-20 mrem/h (50-200 μSv/h) average room gamma dose rates with localized elevated fields of 120 mrem/h (1.2 mSv/h). 10 mrem/h (100 mSv/h) near contact hot spot. Hot spot and local elevated fields associated with a stored waste can. ■ Low level removable surface contamination.

Source: CNL 2021b.

mrem/h = millirem per hour; μSv/h = microsievert per hour, mSv/h = millisievert per hour.

Table 7.1.2-3 provides a brief hazard description for each work package related to the decommissioning of the WR-1 (refer to Section 5.4.1 Hazard and Operability Study) and estimate of doses. The radiological hazards and external radiological dose rates used for dose calculations are based on the WR-1 radiological hazard surveys. The work times for specific decommissioning work tasks were based on WL Projects resource estimates for various work groups.

Table 7.1.2-3: Hazard and Work Description and Dose Estimates for Decommissioning Tasks

Work Package	Hazard Description	Best Estimate and Upper Bounding Dose Estimates	
		Individual (mSv)	Collective (p-mSv)
1. Preliminary Decommissioning: Perform Laboratory Decommissioning; Remove the Solvent Recovery System; Remove the Manlift; Execute Radiological Crawlspace Remediation; Remove the Fuel Transfer Flask; Remove the Fuel Channel Transfer Flask; Remove the 21 Ton Transfer Flask; De-Energize Electrical Inside ISD; De-Energize Electrical Outside ISD; Deactivate Piping, Plumbing, and Drains Outside ISD	Activities under this Work Package will typically pose minimal or low radiological hazards, except for the removal of the Fuel Transfer Flask that may potentially pose low to moderate radiological hazards, which have been encountered and safely mitigated during decommissioning of other WL buildings, such as the Shielded Facilities (B300), B300 laboratories, the Decontamination Centre (B411), and the Active Liquid Waste Treatment Centre (B200). Non-radiological hazards include asbestos, lead, the potential for chemicals in residual quantities, as well as standard construction/decommissioning hazards.	1-2	4-7
2. Grout Installation: Remove Primary Heat Transport System; Install Temporary Systems Inside ISD; Install Temporary Systems Outside ISD; Penetrate Systems; Reinforce and Form ISD Openings/Weak Points; Install Grout Placement System; Pour Grout	Radiological hazards are anticipated to be encountered while: • Remediating asbestos. • Removing the Primary Heat Transport System. • Air gapping services and systems, and penetrating tanks and systems. • Sealing all penetrations in the outer ISD envelope walls. • Installing the grout placement system and preparing for grouting. Activities under this Work Package will pose radiological hazards that have been routinely encountered and safely mitigated during decommissioning of the Decontamination Centre (B411) and the B200 Low Level Liquid Waste System. Non-radiological hazards include asbestos, the potential for chemicals in residual quantities, standard construction/decommissioning hazards, and grouting hazards.	1-8	50-80
3. Building Demolition and Capping: Perform Radiological Decontamination; Perform Mercury Removal; Remove Contaminated Piping; Remove Contaminated Building Materials; Crawlspace Waste Clearance; Remove Contaminated HVAC System; Perform Soil Remediation	Decommissioning activities under this Work Package will typically pose minimal or low radiological hazards that have been routinely encountered and safely mitigated during decommissioning of other WL buildings, such as the B300 laboratories and the Decontamination Centre (B411). Non-radiological hazards include asbestos and standard construction/decommissioning hazards.	0.25-0.55	1-2

Source: CNL 2021b.

mSv = millisievert; p-mSv = person-millisievert.

A G1 RP employee shall review and approve the selection of hazard and exposure controls for radiological related decommissioning activities. The minimum level of required G1 employee approval (Radiation Surveyor, Senior Radiation Surveyor, or HP) is defined in CNL's ALARA Review and Assessment procedure (CNL 2017d). Planned work for which exposures are anticipated that will or could exceed a WL dose action level (CNL 2020h) shall not be undertaken without prior authorization from the WL RP Program Manager.

7.1.2.1.2 As Low As Reasonably Achievable Methods

During decommissioning, all radiological work will be assessed, planned and performed in accordance with the *Quality Assurance Plan Whiteshell Decommissioning* (CNL 2018c) and CNL's ALARA work planning and control procedures and WL job scope and safety analysis guidelines to determine that all radiological exposures are kept ALARA. For decommissioning the work packages will be detailed into work plans prepared to document the work scopes, anticipated hazards, and waste streams and the reference approach for safely conducting work.

7.1.2.1.3 Workplace Characterization Confirmation

Further workplace characterization will take place prior to the start and, when deemed necessary by the responsible Health Physicist, during the execution of each work package associated with radioactive systems. This will provide assurance that work will be completed safely. The objective of the characterization will be to:

- confirm/identify where contamination is present, including what systems potentially have internal contamination;
- identify the type of radionuclides present and in which system they are in;
- determine the quantity and distribution of radionuclides present;
- determine the physical and chemical state of radionuclides present; and
- determine the likely waste classification of systems, components and structures outside the ISD envelop (e.g., likely able to free-release, low-level radioactive waste, intermediate-level radioactive waste).

Characterization activities may include:

- further review of historical survey and preliminary decommissioning;
- performing scoping surveys to identify elevated fields and contaminated areas and the systems with which they are associated;
- work planning to minimize potential spread of contamination; and
- obtaining internal samples (e.g., swipes, sediment, coupons) from systems identified during scoping surveys as needing internal sampling, which will assist to validate the expected overall source term for the WR-1 and confirm the bounds of the assessment are not exceeded.

7.1.2.1.4 Additional Controls

7.1.2.1.4.1 Radioactive Waste Generation

Work shall not commence until a waste disposal pathway has been defined and available disposal capacity has been confirmed.

7.1.2.1.4.2 Elevated Radioactivity Fields

There is the potential that work related to ISD activities may be required in areas of elevated radioactivity. A dose estimate developed as part of the Detailed Decommissioning Plan for workers involved will identify specific areas where mitigative actions will take place to ensure doses to workers remain ALARA. Mitigation options include:

- careful assessment and work planning (e.g., radiation protection surveys, pre-job briefs, work permit system, and work stop procedure);
- personnel exclusion;
- use of remote techniques for ISD activities, such as decontamination, dismantling and grouting;
- selective removal of source term;
- temporary shielding and/or ventilation/filtration; and
- Personal Protection Equipment and Clothes.

7.1.2.1.4.3 Tritium

If necessary, tritium monitoring and standard vacuum drying techniques can be used prior to the grouting of reactor components that could be contaminated with tritium (CNL 2021b). It is not anticipated that any of the areas within the WR-1 Building will have airborne tritium levels that would result in a workplace concern.
All heavy water has been removed from the reactor systems and the systems have been dried. Radiological characterization of the heavy water and helium system found no evidence of any remaining heavy water.
It is recognized that there is “dry” tritium contamination on the internal surfaces of system piping and tanks that may require puncturing for grout application; however, they are not anticipated to present a significant hazard.

7.1.3 Airborne Radiological Releases

For the closure phase, the radionuclides that are considered for the Human Health Risk Assessment (HHRA) were based on operational experience at the WL site and were primarily identified in the *Derived Release Limits for AECL’s Whiteshell Laboratories* (CNL 2016c) and the *WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* (CNL 2020b). These radionuclides have been found in the WL’s airborne effluent or are reasonably expected to be found in the airborne effluent.

The estimated inventory of radionuclides derived for the timeframe of 30 years following shutdown was determined to be appropriate for the closure phase as closure will start approximately 30 years after reactor shutdown in 1985. Estimated radionuclide inventories for the reactor core, biological shield, and PHT system can be found in Table 7.1.1-3.

Only the above-ground portion of the PHT system will be disassembled. During grouting it was conservatively assumed that the entire inventory of radionuclides is made available for release with dust particulate. In reality, much of the inventory will be fixed on surfaces of WR-1 structures. Decontamination practices and contamination control techniques provide additional mitigation. Mitigation for fugitive dust emissions includes the use of contamination immobilization agents, containment, ventilation and HEPA filters to control generation of airborne emissions during decontamination or removal of contaminated systems or structures. In addition, use of dust suppression methods during building demolition or soil remediation activities will be implemented to control airborne emissions and nuisance dust during building demolition or soil remediation. Releases from dust or gaseous emissions are expected to be negligible because these particulates are anticipated to be captured in the remnant organics (oil), within the system. In addition, the structure is made of steel not concrete, which also inhibits the collection of dust and particulates on the PHT system. Regardless of the decontamination and dismantling techniques used for the WR-1, best practices will be implemented and will follow CNL's Environmental Protection Program.

Estimated radionuclide release rate for the dismantling of the PHT system prior to grouting and for grouting activities are presented in Table 7.1.3-1 and Table 7.1.3-2, respectively. Estimated tritium release rate for the majority of the closure activities was based on the average obtained from operational data collected from 2011 to 2019, which was 1.11E+09 Bq/week (1.84E+03 Bq/s). During grouting, however, vibration and heating of structures and systems is expected to increase atmospheric tritium release rate. For grouting the estimated release rate used was the maximum weekly tritium release rate observed during characterization activities undertaken between 2011 and 2015, which was 1.28E+10 Bq/wk (EcoMetrix 2021).

Table 7.1.3-1: Estimated Radionuclide Release Rate from Primary Heat Transport System

Radionuclide	Concentration in Demolition Material (Bq/g)	Release Rate (Bq/s)					
		SPM		PM$_{10}$		PM$_{2.5}$	
		Max	Avg	Max	Avg	Max	Avg
Americium-241	2.11E+02	7.79E+00	2.05E+00	7.79E+00	2.05E+00	7.79E-01	2.05E-01
Americium-243	1.91E-01	7.05E-03	1.85E-03	7.05E-03	1.85E-03	7.05E-04	1.85E-04
Caesium-137	4.83E+03	1.78E+02	4.68E+01	1.78E+02	4.68E+01	1.78E+01	4.68E+00
Cobalt-60	2.45E+00	9.03E-02	2.37E-02	9.03E-02	2.37E-02	9.03E-03	2.37E-03
Curium-244	2.11E+00	7.79E-02	2.05E-02	7.79E-02	2.05E-02	7.79E-03	2.05E-03
Europium-154	3.93E+01	1.45E+00	3.80E-01	1.45E+00	3.80E-01	1.45E-01	3.80E-02
Europium-155	4.83E+00	1.78E-01	4.68E-02	1.78E-01	4.68E-02	1.78E-02	4.68E-03
Iodine-129	2.82E-03	1.04E-04	2.73E-05	1.04E-04	2.73E-05	1.04E-05	2.73E-06
Neptunium-237	1.11E-02	4.08E-04	1.07E-04	4.08E-04	1.07E-04	4.08E-05	1.07E-05
Neptunium-239	1.91E-01	7.05E-03	1.85E-03	7.05E-03	1.85E-03	7.05E-04	1.85E-04
Plutonium-238	2.52E+01	9.28E-01	2.44E-01	9.28E-01	2.44E-01	9.28E-02	2.44E-02
Plutonium-239	6.14E+01	2.26E+00	5.95E-01	2.26E+00	5.95E-01	2.26E-01	5.95E-02
Plutonium-240	8.76E+01	3.23E+00	8.49E-01	3.23E+00	8.49E-01	3.23E-01	8.49E-02
Plutonium-241	2.01E+03	7.42E+01	1.95E+01	7.42E+01	1.95E+01	7.42E+00	1.95E+00
Silver-108m	1.63E-02	6.01E-04	1.58E-04	6.01E-04	1.58E-04	6.01E-05	1.58E-05
Strontium-90	3.12E+03	1.15E+02	3.02E+01	1.15E+02	3.02E+01	1.15E+01	3.02E+00
Technetium-99	1.31E+00	4.83E-02	1.27E-02	4.83E-02	1.27E-02	4.83E-03	1.27E-03
Uranium-235	5.80E-03	2.14E-04	5.62E-05	2.14E-04	5.62E-05	2.14E-05	5.62E-06
Uranium-238	1.25E-01	4.60E-03	1.21E-03	4.60E-03	1.21E-03	4.60E-04	1.21E-04
Total	**1.04E+04**	**3.84E+02**	**1.01E+02**	**3.84E+02**	**1.01E+02**	**3.84E+01**	**1.01E+01**

Bq/g = Becquerels per gram; Bq/s = Becquerels per second; SPM = suspended particulate matter; PM$_{10}$ = particulate matter less than 10 microns in diameter; PM$_{2.5}$ = particulate matter less than 2.5 microns in diameter; Max = maximum; Avg = Average.

Table 7.1.3-2: Estimated Radionuclide Release Rate for Grouting Activities

Radionuclide	Concentration in Grout (Bq/g)	Release Rate (Bq/s)					
		SPM		PM_{10}		$PM_{2.5}$	
		Max	Avg	Max	Avg	Max	Avg
Reactor Core							
Carbon-14	1.4E+02	1.1E+00	5.8E-01	5.2E-01	2.7E-01	7.8E-02	4.2E-02
Cobalt-60	6.3E+03	5.1E+01	2.7E+01	2.4E+01	1.3E+01	3.6E+00	1.9E+00
Iron-55	8.1E+02	6.5E+00	3.4E+00	3.1E+00	1.6E+00	4.6E-01	2.5E-01
Nickel-59	3.8E+02	3.0E+00	1.6E+00	1.4E+00	7.6E-01	2.2E-01	1.2E-01
Nickel-63	4.4E+04	3.5E+02	1.9E+02	1.7E+02	8.8E+01	2.5E+01	1.3E+01
Niobium-94	1.4E+02	1.1E+00	5.8E-01	5.2E-01	2.8E-01	7.9E-02	4.2E-02
Silver-108m	2.7E-04	2.2E-06	1.2E-06	1.0E-06	5.5E-07	1.6E-07	8.3E-08
Total	**5.2E+04**	**4.13E+02**	**2.19E+02**	**1.95E+02**	**1.04E+02**	**2.96E+01**	**1.57E+01**
Biological Shield							
Calcium-41	6.4E-03	5.1E-05	2.7E-05	2.4E-05	1.3E-05	3.7E-06	1.9E-06
Carbon-14	2.8E-03	2.3E-05	1.2E-05	1.1E-05	5.7E-06	1.6E-06	8.6E-07
Chlorine-36	1.9E-07	1.5E-09	8.2E-10	7.3E-10	3.9E-10	1.1E-10	5.8E-11
Cobalt-60	1.8E-01	1.5E-03	7.8E-04	7.0E-04	3.7E-04	1.1E-04	5.6E-05
Europium-152	3.3E-02	2.6E-04	1.4E-04	1.2E-04	6.6E-05	1.9E-05	9.9E-06
Nickel-63	2.7E-02	2.1E-04	1.1E-04	1.0E-04	5.4E-05	1.5E-05	8.1E-06
Total	**2.5E-01**	**2.02E-03**	**1.07E-03**	**9.56E-04**	**5.07E-04**	**1.45E-04**	**7.67E-05**
Primary Heat Transport System							
Americium-241	7.7E-01	6.1E-03	3.3E-03	2.9E-03	1.5E-03	4.4E-04	2.3E-04
Americium-243	6.9E-04	5.6E-06	2.9E-06	2.6E-06	1.4E-06	4.0E-07	2.1E-07
Caesium-137	1.8E+01	1.4E-01	7.4E-02	6.6E-02	3.5E-02	1.0E-02	5.3E-03
Cobalt-60	8.9E-03	7.1E-05	3.8E-05	3.4E-05	1.8E-05	5.1E-06	2.7E-06
Curium-244	7.7E-03	6.1E-05	3.3E-05	2.9E-05	1.5E-05	4.4E-06	2.3E-06
Europium-154	1.4E-01	1.1E-03	6.0E-04	5.4E-04	2.9E-04	8.2E-05	4.3E-05
Europium-155	1.8E-02	1.4E-04	7.4E-05	6.6E-05	3.5E-05	1.0E-05	5.3E-06
Iodine-129	1.0E-05	8.2E-08	4.3E-08	3.9E-08	2.1E-08	5.9E-09	3.1E-09
Neptunium-237	4.0E-05	3.2E-07	1.7E-07	1.5E-07	8.1E-08	2.3E-08	1.2E-08
Neptunium-239	6.9E-04	5.6E-06	2.9E-06	2.6E-06	1.4E-06	4.0E-07	2.1E-07
Plutonium-238	9.1E-02	7.3E-04	3.9E-04	3.5E-04	1.8E-04	5.2E-05	2.8E-05
Plutonium-239	2.2E-01	1.8E-03	9.5E-04	8.4E-04	4.5E-04	1.3E-04	6.8E-05
Plutonium-240	3.2E-01	2.5E-03	1.3E-03	1.2E-03	6.4E-04	1.8E-04	9.6E-05
Plutonium-241	7.3E+00	5.8E-02	3.1E-02	2.8E-02	1.5E-02	4.2E-03	2.2E-03
Silver-108m	5.9E-05	4.7E-07	2.5E-07	2.2E-07	1.2E-07	3.4E-08	1.8E-08
Strontium-90	1.1E+01	9.1E-02	4.8E-02	4.3E-02	2.3E-02	6.5E-03	3.4E-03
Technetium-99	4.7E-03	3.8E-05	2.0E-05	1.8E-05	9.5E-06	2.7E-06	1.4E-06
Uranium-235	2.1E-05	1.7E-07	8.9E-08	8.0E-08	4.2E-08	1.2E-08	6.4E-09
Uranium-238	4.5E-04	3.6E-06	1.9E-06	1.7E-06	9.1E-07	2.6E-07	1.4E-07
Total	**3.8E+01**	**3.0E-01**	**1.6E-01**	**1.4E-01**	**7.6E-02**	**2.2E-02**	**1.1E-02**

Table 7.1.3-2: Estimated Radionuclide Release Rate for Grouting Activities

Radionuclide	Concentration in Grout (Bq/g)	Release Rate (Bq/s)					
		SPM		PM_{10}		$PM_{2.5}$	
		Max	Avg	Max	Avg	Max	Avg
Active Ventilation System							
Americium-241	3.6E-02	2.9E-04	1.5E-04	1.4E-04	7.3E-05	2.1E-05	1.1E-05
Americium-243	2.9E-05	2.3E-07	1.2E-07	1.1E-07	5.8E-08	1.7E-08	8.7E-09
Caesium-137	5.8E-01	4.6E-03	2.4E-03	2.2E-03	1.2E-03	3.3E-04	1.7E-04
Curium-244	2.1E-04	1.7E-06	9.0E-07	8.0E-07	4.3E-07	1.2E-07	6.4E-08
Europium-154	2.7E-03	2.2E-05	1.2E-05	1.0E-05	5.5E-06	1.6E-06	8.3E-07
Europium-155	1.8E-04	1.5E-06	7.7E-07	6.9E-07	3.7E-07	1.0E-07	5.5E-08
Iodine-129	4.2E-07	3.4E-09	1.8E-09	1.6E-09	8.5E-10	2.4E-10	1.3E-10
Neptunium-237	1.7E-06	1.3E-08	7.1E-09	6.3E-09	3.3E-09	9.6E-10	5.1E-10
Neptunium-239	2.9E-05	2.3E-07	1.2E-07	1.1E-07	5.8E-08	1.7E-08	8.7E-09
Plutonium-238	3.5E-03	2.8E-05	1.5E-05	1.3E-05	7.0E-06	2.0E-06	1.1E-06
Plutonium-239	9.2E-03	7.4E-05	3.9E-05	3.5E-05	1.9E-05	5.3E-06	2.8E-06
Plutonium-240	1.3E-02	1.1E-04	5.6E-05	5.0E-05	2.6E-05	7.6E-06	4.0E-06
Plutonium-241	2.0E-01	1.6E-03	8.4E-04	7.5E-04	4.0E-04	1.1E-04	6.0E-05
Strontium-90	3.8E-01	3.0E-03	1.6E-03	1.4E-03	7.6E-04	2.2E-04	1.2E-04
Technetium-99	2.0E-04	1.6E-06	8.4E-07	7.5E-07	4.0E-07	1.1E-07	6.0E-08
Uranium-235	8.7E-07	7.0E-09	3.7E-09	3.3E-09	1.8E-09	5.0E-10	2.7E-10
Uranium-238	1.9E-05	1.5E-07	8.0E-08	7.1E-08	3.8E-08	1.1E-08	5.7E-09
Total	**1.2E+00**	**9.8E-03**	**5.2E-03**	**4.6E-03**	**2.4E-03**	**7.0E-04**	**3.7E-04**

Bq/g = Becquerels per gram; Bq/s = Becquerels per second; SPM = suspended particulate matter; PM_{10} = particulate matter less than 10 microns in diameter; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; Max = maximum; Avg = Average.

7.1.4 Estimated On-site Worker Dose

Adherence to the CNL's Radiation Protection and Occupational Safety and Health programs will provide that exposures are controlled and kept within regulated limits. Furthermore, the application of the Radiation Protection Program provides that exposures are justified and met the ALARA principle, taking social and economic factors into account. Refer to Section 5.2.1 Human Health for specific assessment criteria pertaining to Operating Radiological Regulation Criteria.

An on-site receptor (e.g., personnel leasing office/business space on the WL site) was evaluated for the closure phase. Maximum and average estimated radiation doses for the on-site receptor during dismantling activities prior to grouting are presented in Tables 4-8 and 4-9 of the ERA (EcoMetrix 2021), respectively. Based on maximum and average airborne emission rates derived for the closure phase (Section 7.1.3 Airborne Radiological Releases), maximum and average doses during grouting activities are presented in Table 4-16 and Table 4-17 of the ERA (EcoMetrix 2021), respectively. Total dose for the on-site receptor for the four scenarios examined are summarized in Table 7.1.4-1. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 7.1.4-1: Summary of Total Dose for On-site Worker during Closure

On-Site Worker Activity	Dose (mSv/a)	Percent of Dose Constraint	Percent of Public Dose Limit
Demolition - Maximum	6.04E-03	2.42%	0.60%
Demolition - Average	1.59E-03	0.63%	0.16%
Grouting - Maximum	1.80E-04	0.07%	0.02%
Grouting – Average	9.76E-05	0.04%	0.01%

Note:

mSv/a = microsieverts per year; <= less than.
A dose of 0.0025 mSv/a is equivalent to 1% of the dose constraint and a dose of 0.01 mSv/a is equivalent to 1% of the public dose limit.

The total radiation dose to all human receptors during closure activities is expected to be well below the public dose limit of 1 mSv/a, and the dose constraint for the Project of 0.25 mSv/a. Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure to radioactive releases from the Project activities. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

7.1.5 Assumptions and Uncertainty

There is uncertainty in the estimated inventory of radioactive material that could be encountered during the cutting and perforation of components, during grouting preparation activities, or dismantling of the WR-1 Building. It is recognized that unknown or greater than anticipated quantities of hazardous materials could be present. As needed, system characterization and radiological surveys will be completed to inform the development of detailed work plans.

There is uncertainty in the radiological release rates to the atmosphere, however, the estimates considered during planning are recognized as conservative. Prior to grouting, the above-grade portion of the PHT system (i.e., two heat exchangers and remaining two outlet headers) will be dismantled and placed below-grade for ISD. This is assumed to encompass a mass of 20 Mg (20.1% of the PHT system). It was assumed that the above-grade inventory of each radionuclide (20.1% of the PHT inventory) is dispersed over the mass of the demolition material (20 Mg). Therefore, the maximum and average particulate release rates in g/s were multiplied by 20.1% of the radionuclide inventory from the PHT system to estimate the release rate per radionuclide (in units of Bq/s) that could be expected. The duration of demolition activities prior to grouting is dependent on optimizing worker radiological exposure. As shutdown occurred in 1985 and decommission activities will take place 35 to 40 years after, the closest year available for the inventory was used. For the PHT system this was year 30. During detailed planning, system characterizations will be used to informed work plans. If contamination levels are low, the placement of the above-grade portion of PHT system below-grade would likely take one to two months; however, as a conservative estimate it has been assumed that this is carried out over a 6-month period. Release rates assume the radionuclides are released as SPM, PM_{10} or $PM_{2.5}$; however, PM_{10} is generally associated with demolition work.

While the grouting is assumed to take 3 to 5 months on a schedule of 5 days per week, it has been assumed that it will occur for one year. . It is estimated that 10,000 m^3 of bulk fill grout will be produced and placed. For the active ventilation system total surface contamination was assumed to be 40 Bq/cm^2. The total surface area that will remain below-grade was estimated at 665,830,000 cm^2; therefore, the total activity is 2.66E+10 Bq. The release rates were presented as SPM, PM_{10} and $PM_{2.5}$, but assuming radionuclides released as SPM gives the most conservative release rate for radionuclides. Therefore, the PM_{10} and $PM_{2.5}$ fractions were included in the SPM.

After grouting, demolition of above-grade structures are expected to result in negligible release of radionuclides to the atmosphere. It is expected that demolition of the above-grade structure will take two years following the WRDF completion.

7.2 Radiological Assessment for Public under Normal Conditions

Human receptors evaluated for both the radiological assessment for the closure phase includes off-site member of the public and those critical groups. The critical groups used were consistent with those used for dose calculations in the *Derived Release Limits for AECL's Whiteshell Laboratories* (CNL 2016c). These receptors are potentially exposed to airborne effluent from the WR-1 Building during the closure phase. These critical groups include: Farm A and Farm F (year-round occupants, with livestock) and Traditional Land User exposure through harvesting country foods (Harvester). Country foods include deer, hare and berries/plants. It was communicated through Indigenous engagement activities that wild rice and medicinal plants are harvested near the WL site. However, wild rice does not grow in close proximity to the WR-1 (excluded from assessment) and the consumption of medicinal plants would be bound in the assessment by the consumption of local berries. Refer to Figure 8.2.1-1, which depicts the location of human receptors assumed to be present within the vicinity of the Project during the closure phase.

7.2.1 Hazard Identification and Exposure Pathways

The receptors for the HHRA were selected to be appropriate for assessment of both radiological and non-radiological stressors on human health. An on-site receptor (e.g., personnel leasing office/business space on the WL site) was evaluated for the closure phase, assessed as being present during demolition activities prior to grouting activities (i.e., during dismantling of above-grade portion of the PHT system and placement below-grade).

Off-site members of the public are potentially exposed to low levels of airborne contaminants. The most affected off-site member of the public is identified as the "critical group". The critical group identified for the Project is Farm A and Farm F (year-round occupants), as well as harvesters (i.e., traditional users of the area who may be exposed through harvesting country foods). It is assumed that the harvesters spend part of their time on-site, part near Farm F, and part at an unexposed location.

Recreational users such as swimmers, anglers, and boaters that occasionally carry out recreational activities along the Winnipeg River are not considered for the closure scenario because these activities are not representative of population groups in the area.

Pathways relevant to potential exposure to liquid effluent sources are considered to be unlinked (i.e., limited by mitigation), as during the closure phase releases to surface water are not expected to occur. Therefore, exposure to atmospheric releases is the primary pathway. Human receptors on Farm A and Farm F will be exposed to contaminants in the air (inhalation, immersion), soil (incidental ingestion, groundshine), well water (ingestion, bathing), and ingestion of homegrown vegetables, fruits, and livestock, as well as ingestion from locally hunted venison. Harvesters will be exposed to contaminants in the air and through the ingestion of country foods, such as deer, hare and berries/plants. Details pertaining to the exposure pathways are summarized in Table 7.2.1-1.

Atmospheric deposition to the Winnipeg River is considered negligible. This is consistent with the Candu Owners Group (COG) DRL guidance (COG 2013) which shows (assuming a modest flow rate for a lake of 0.1 m/s and an assumed water depth of 10 m) that the transfer of radionuclides from the atmosphere to large bodies of water (including lakes and rivers) is considered negligible. Rivers have larger flow rates than lakes; therefore, the conclusion for lakes that the atmospheric deposition pathway is negligible is applicable to rivers as well.

In CSA N288.1-14 (2014) the release of radionuclides from surface soil to the atmosphere is considered negligible because transfer is predominately from the atmosphere to soil. This pathway was not included during closure for this reason.

It was noted from Indigenous engagement activities that Sagkeeng First Nation members harvest wild rice and medicinal plants in the WL site. However, wild rice does not grow in close proximity to WR-1.

CNL (2018a) also conducted an Indigenous Food Intake Survey completed by members of the Sagkeeng First Nation and Manitoba Métis Citizens to understand the types and quantities of local food consumed. The results indicate that survey participants consume animals such as wild game (e.g., moose, deer, rabbit and hare), waterfowl (e.g., duck and geese), fish, fruits and berries, and medicinal plants (e.g., weekay and cedar). Although a number of respondents indicated that they eat moose, moose are not commonly found around Pinawa and Lac du Bonnet but are typically farther north. Additionally, during the closure phase, the focus is on terrestrial pathways, since only atmospheric releases are expected. Based on these considerations terrestrial animals including hare and deer, terrestrial plants including berries, and medicinal plants including cedar and weekay are included in the assessment for the harvester. Consistent with the COG DRL guidance (COG 2013), atmospheric deposition to a large waterbody is considered negligible. Large bodies of water would include lakes and rivers (e.g., the Winnipeg River). Therefore, during closure, the harvester was not assumed to eat local fish or duck, since the relevant exposure pathways are the terrestrial pathways and not the aquatic pathways. Other considerations for the harvester (such as moose ingestion) are considered in the post closure phase where aquatic pathways are more applicable.

Full-time residency was assumed for the residents of Farm A and Farm F. The harvester was assumed to harvest on the WL site for 2 hours per week (h/wk), upwind in the vicinity of Farm A for 2 h/wk, and downwind for 2 h/wk. For Farm A, the percentage of food items in the diet that are obtained from local sources was consistent with the *Derived Release Limits for AECL's Whiteshell Laboratories* (CNL 2016c), with some modifications. Although honey is currently produced at another local farm; it was assumed that during the closure phase Farms A and F obtain their honey from their own farms; therefore, the percentage of honey obtained from local sources was considered to be 100%. Farms A and F are both livestock farms with the same characteristics as each other except for the location. Members of Farms A and F are considered to:

- Reside at the farm 100% of the time;
- Obtain the majority of their fruit and vegetables from their on-site garden;
- Obtain drinking water from an on-site well;
- Supply all of their own milk, poultry and eggs from their farm;
- Supply some of their beef and pork from their farm;
- Acquire honey from their farm;
- Obtain game (deer) meat requirement from hunting on their own property; and
- Use a backyard swimming pool filled with well water 3 months out of the year.

For the harvester, the local percentage of food intake was assumed to be 33% (for deer, rabbit, and berries/plants) based on the assumption that the harvester only spends 33% of harvesting time on-site.

Table 7.2.1-1: Complete Exposure Pathways for Receptors Exposed to Radiological COPCs during Closure Phase

Receptor	Exposure Source	Exposure Pathway	Environmental Media
Farm F and Farm A - full-time residency (Adult, 10-year old Child, 1-year old Infant (formula/milk), and 3-month old Infant formula/nursing)	Airborne Emissions	Inhalation	Air
		Ingestion	Water (well water)
			Soil (incidental)
			Terrestrial Plants (homegrown)
			Terrestrial Animals and Animal Products (livestock, game, milk, and honey)
		External	Air
			Water (well water)
			Soil
Harvester - 2 h/wk on-site, downwind, and upwind from the Project activities (Adult, 10-year old Child, and 1-year old Infant)		Inhalation	Air
		Ingestion	■ Terrestrial animals (hare, deer) ■ Terrestrial plants (berries, weekay, cedar)
		External	Air
			Soil

h/wk = hours per week.

7.2.2 Planned Mitigation

All decommissioning work will be planned and executed in accordance with the Decommissioning Licence, the WL Decommissioning Quality Assurance Plan (CNL 2018c), CNL's Environmental Protection Program (CNL 2018d), the WL Site Emergency Response Plan (CNL 2019g), the Work Planning for WL Decommissioning, CNL's Radiation Protection Program, CNL's Occupational Safety and Health Program, and CNL's Engineering Change Control.

Operational controls to protect workers, the public and the environment will still be in effect during the closure phase. This includes the WR-1 Building being in place until grouting activities have been completed, secondary containment and waste management system, and adequate ventilation and filtration system being operational, as well as standard operating procedures including the review of emergency procedures, and conduction of routine inspections of equipment, spill trays, and spill kits being used to limit the potential for radiological releases from site to the surrounding environment. As well, the public has restricted access to the WL site, and access to the WRDF will be restricted to project personnel.

7.2.3 Airborne Radiological Releases

Radionuclides are considered of public and regulatory interest, therefore, all radionuclides identified through the source-term characterization process were identified and evaluated in the HHRA (i.e., all identified radionuclides were carried forwarded into the assessment). Refer to Section 7.1.3 Airborne Radiological Releases for release rates expected during the closure phase.

7.2.4 Estimated Public Dose

Exposures to the harvester and Farm F residents were assessed. Based on maximum and average airborne emission rates derived for the closure phase (Section 7.1.3 Airborne Radiological Releases), maximum and average estimated radiation doses for the off-site receptors during dismantling activities prior to grouting are presented in Tables 4-11 through 4-15 of the ERA (EcoMetrix 2021). Estimated radiation doses during grouting activities are presented in Table 4-18 through Table 4-23 of the ERA (EcoMetrix 2021). Total dose for the harvester and Farm F receptors for the four scenarios examined are summarized in Table 7.2.4-1 and Table 7.2.4-2 for demolition and grouting stages of the Project, respectively.

7.2.4.1 Demolition Activities

Radiological doses expected for Farm F (i.e., critical group) are considered representative of those for Farm A, as radiological doses to the residents (receptors: Adult, 10-year old Child, 1-year old Infant and 3-month old Infant) of Farm F would be higher than those of Farm A. Maximum and average estimated radiation doses for Farm F residents during demolition activities are detailed in Tables 4-12 and 4-13 of the ERA (EcoMetrix 2021) respectively. Maximum and average estimated radiation dose for Farm F 3-month old Infant are detailed in Tables 4-14 and 4-15 of the ERA (EcoMetrix 2021), respectively. Maximum and average estimated radiation doses for Harvesters (Adult, 10-year old Child, and 1-year old Infant) are detailed in Tables 4-10 and 4-11 of the ERA (EcoMetrix 2021), respectively. Total dose for the Farm F residents during the demolition stage prior to grouting during the closure phase are summarized in Table 7.2.4-1. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 7.2.4-1: Summary of Total Dose for On-site Receptors during Demolition Stage of Closure Phase

Age Group / Exposure Scenario	Dose (mSv/a)	Percent of Dose Constraint	Percent of Public Dose Limit
Adult Harvester			
Demolition - Maximum	3.50E-03	1.40%	0.35%
Demolition - Average	9.19E-04	0.37%	0.09%
10-year old Child Harvester			
Demolition - Maximum	1.52E-03	0.61%	0.15%
Demolition - Average	4.00E-04	0.16%	0.04%
1-year old Infant Harvester			
Demolition - Maximum	6.66E-04	0.27%	0.07%
Demolition - Average	1.75E-04	0.07%	0.02%
Adult Farm F Resident			
Demolition – Maximum	3.46E-03	1.39%	0.35%
Demolition - Average	9.11E-04	0.36%	0.09%
10-year old Child Farm F Resident			
Demolition - Maximum	3.38E-03	1.35%	0.34%
Demolition - Average	8.89E-04	0.36%	0.09%
1-year old Infant – Milk fed Farm F Resident			
Demolition - Maximum	2.61E-03	1.04%	0.26%
Demolition - Average	6.86E-04	0.27%	0.07%

Table 7.2.4-1: Summary of Total Dose for On-site Receptors during Demolition Stage of Closure Phase

Age Group / Exposure Scenario	Dose (mSv/a)	Percent of Dose Constraint	Percent of Public Dose Limit
1-year old Infant – Formula fed Farm F Resident			
Demolition - Maximum	1.17E-03	0.47%	0.12%
Demolition - Average	3.07E-04	0.12%	0.03%
3-month old Infant – Nursing Farm F Resident			
Demolition - Maximum	1.23E-03	0.49%	0.12%
Demolition - Average	3.23E-04	0.13%	0.03%
3-month old Infant – Formula fed Farm F Resident			
Demolition - Maximum	3.96E-04	0.16%	0.04%
Demolition - Average	1.04E-04	0.04%	0.01%

mSv/a = microsieverts per year.

A dose of 0.0025 mSv/a is equivalent to 1% of the dose constraint and a dose of 0.01 mSv/a is equivalent to 1% of the public dose limit.

The total radiation doses to all individual human receptors assessed during closure activities are predicted to be well below the public dose limit of 1 mSv/a and the dose constraint for the Project of 0.25 mSv/a. Doses are presented for the three age groups: Adult, Child and Infant (nursing, milk fed, and formula fed). Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure to radioactive releases from the Project activities.

7.2.4.2 Grouting Activities

Maximum and average estimated radiation doses for Farm F residents are detailed in Tables 4-20 and 4-21 of the ERA (EcoMetrix 2021), respectively. Maximum and average estimated radiation dose for Farm F 3-month old Infant are detailed in EcoMetrix (2021), Tables 4-22 and 4-23, respectively. Maximum and average estimated radiation doses for Harvesters (1 year old, 10-year-old, and adult) are detailed Tables 4-18 and 4-19 of the ERA (EcoMetrix 2021), respectively. Total dose for the Farm F residents during the grouting stage in the closure phase are summarized in Table 7.2.4-2. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 7.2.4-2: Summary of Total Dose for On-site Receptors during Grouting Stage of Closure Phase

Age Group / Exposure Scenario	Dose (mSv/a)	Percent of Dose Constraint)	Percent of Public Dose Limit)
Adult Harvester			
Grouting - Maximum	1.58E-05	<0.01%	<0.01%
Grouting – Average	8.82E-06	<0.01%	<0.01%
10-year old Child Harvester			
Grouting - Maximum	1.69E-05	<0.01%	<0.01%
Grouting – Average	9.40E-06	<0.01%	<0.01%
1-year old Infant Harvester			
Grouting - Maximum	1.85E-05	<0.01%	<0.01%
Grouting – Average	1.04E-06	<0.01%	<0.01%

Table 7.2.4-2: Summary of Total Dose for On-site Receptors during Grouting Stage of Closure Phase

Age Group / Exposure Scenario	Dose (mSv/a)	Percent of Dose Constraint)	Percent of Public Dose Limit)
Adult Farm F Resident			
Grouting - Maximum	1.01E-04	0.04%	0.01%
Grouting – Average	5.69E-05	0.02%	<0.01%
10-year old Child Farm F Resident			
Grouting - Maximum	1.21E-04	0.05%	0.01%
Grouting – Average	6.79E-05	0.03%	<0.01%
1-year old Infant – Milk fed Farm F Resident			
Grouting - Maximum	1.59E-04	0.06%	0.02%
Grouting – Average	8.88E-05	0.04%	<0.01%
1-year old Infant – Formula fed Farm F Resident			
Grouting - Maximum	1.23E-04	0.05%	0.01%
Grouting – Average	6.72E-05	0.03%	<0.01%
3-month old – Nursing Farm F Resident			
Grouting - Maximum	1.09E-04	0.04%	0.01%
Grouting – Average	6.30E-05	0.03%	<0.01%
3-month old – Formula fed Farm F Resident			
Grouting - Maximum	8.91E-05	0.04%	<0.01%
Grouting – Average	4.89E-05	0.02%	<0.01%

Note:
mSv/a = microsieverts per year.
A dose of 0.0025 mSv/a is equivalent to 1% of the dose constraint and a dose of 0.01 mSv/a is equivalent to 1% of the public dose limit.

The total radiation dose to all individual human receptors assessed during closure activities are predicted to be well below the public dose limit of 1 mSv/a and the dose constraint for the Project of 0.25 mSv/a. Doses are presented for the three age groups: Adult, Child and Infant (nursing, milk fed, and formula fed). Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure radioactive releases from the Project activities.

7.2.5 Assumptions and Uncertainty

Assumptions and uncertainties regarding radiological inventories within the WR-1 Building and radiological release rates are discussed in Section 7.1.5 Assumptions and Uncertainty. As well as the modeling input assumptions, there is uncertainty in the radiological release rates to the atmosphere; however, the estimates are expected to be conservative.

There is inherent uncertainty in the air model in IMPACT that is used to estimate atmospheric dispersion factors to the critical group locations. Uncertainty in the air predictions arises from the following assumptions made in the model (COG 2013):

- the activity in the plume has a normal distribution in the vertical plane;
- the effects of building-induced turbulence on the effective release height and plume spread have been generalized, while data suggest that effects of building wakes vary substantially depending upon the geometry of the buildings and their orientation with respect to wind direction; and

- a given set of meteorological and release conditions leads to a unique air concentration, where in reality measured concentrations can vary by a factor of 2 under identical conditions.

At distances greater than 1 km, there is a two-fold uncertainty around the predictions of the sector-averaged Gaussian model used in IMPACT (COG 2013). At all distances, the Gaussian air model in IMPACT, on average, overpredicts air concentrations by approximately a factor of 1.5 (COG 2013). Considering the conservatism in the estimation of releases, and in air model, it is reasonable to conclude that doses arising from closure activities have not been underestimated.

7.3 Non-Radiological Assessment for Workers under Normal Conditions

For new and/or non-routine decommissioning tasks involving non-radiological hazards that are not well known and/or may present a risk, formal work assessments will be completed to evaluate the hazards and required mitigation.

7.3.1 Hazard Identification and Exposure Pathways

Non-radiological COPCs may be released from the WL site during grouting activities as SPM, PM_{10}, and $PM_{2.5}$. After the systematic evaluation of closure activities and taking into account the removal of excess quantities of non-radiological material and standard operating procedures, a select number of non-radiological COPCs were identified for the closure phase, and the average and maximum inventory estimates were assessed. The estimated inventory of non-radionuclides derived for the WR-1 Building is provided in Table 3-14 of the ERA (EcoMetrix 2021).

The potential effects from the ISD of the WR-1 include nuisance dust and fine particulates from grouting, building demolition, site restoration, and rehabilitation activities. This includes the generation of non-radioactive hazardous liquid or solid wastes (e.g., asbestos, lead), including dust during disturbance activities. During the closure phase, releases to surface water are not expected, because any release from the grouted reactor would occur after closure, and contaminant transport via groundwater to surface water would take additional time. Exposure pathways for on-site workers include inhalation of air on-site, and soil contact pathways for outdoor workers for non-radiological exposures (Table 7.3.1-1).

Table 7.3.1-1: Complete Exposure Pathways for Receptors Exposed to Non-Radiological COPCs during Closure Phase

Receptor	Exposure Source	Exposure Pathway	Environmental Media
On-site Worker – 40 h/wk and 50 wk/yr	Airborne Effluent	Inhalation	Air
		Dermal	Soil

h/wk = hours per week; wk/yr = weeks per year.

7.3.2 Planned Mitigation

Construction planning will identify workplace hazards associated with closure activities, specifically all non-radiological COPCs. Workplace procedures to limit worker exposures, allowable airborne exposure concentrations, compliance monitoring programs, and waste disposal plans, in accordance with applicable workplace regulatory requirements and guidance. The regulations require collection and proper disposal of materials containing designated substances (e.g., asbestos, lead, PCBs, and mercury). Planning will provide that workplace concentrations of hazardous substances are safety for workers. Accordingly, there will be very little release of these materials to the environment. The WRDF will be constructed in accordance with the design as described in the safety case and shall be constructed in a manner which preserves the safety functions that have been shown to be important for safety post-closure.

7.3.2.1 As Low As Reasonably Achievable Methods

During decommissioning, all work associated with potential non-radiological hazards will be assessed, planned and performed in accordance with the WL Decommissioning Quality Assurance Plan and CNL’s ALARA work planning and control procedures and WL job scope and safety analysis guidelines to determine that all exposures are kept ALARA. For decommissioning the work packages will be detailed into work plans prepared to document the work scopes, anticipated hazards, and waste streams and the reference approach for safely conducting work.

7.3.2.2 Workplace Characterization Confirmation

Further workplace characterization will take place prior to the start and, when deemed necessary by the responsible occupational health and safety specialist, during the execution of each work package associated with systems associated with non-radiological hazardous material. This will provide a system to determine that work is completed safely.

A specific list of industrial and non-radiological hazards will be identified for each work plan developed. Conventional hazards such as confined spaces, working at heights (i.e., WR-1 Main Hall), and potentially energized systems during utility removal, will also be considered.

7.3.2.3 Additional Controls

7.3.2.3.1 Asbestos Waste

Asbestos waste that is not filled within the ISD envelope will be managed in accordance with CNL procedures for managing asbestos waste as described in CNL’s procedure Controlling Asbestos Hazard (CNL 2017e). Provided that the asbestos is clearable and meets the classification as a non-radiological waste, it will be disposed of at an approved asbestos landfill. In the event that asbestos waste is radiologically contaminated, it will be managed at the WL WMA for radioactive waste. The regulations and procedures require that all asbestos waste is securely packaged to prevent particle release during handling and transport to the landfill site. All waste packages will be distinctly labelled indicating the contents as ACMs.

7.3.2.3.2 Lead

No immediate health concern is present if the paint remains intact, therefore, regulator monitoring performed for peeling paint and paint chips present are routinely cleaned up. Sanding and grinding activities will be avoided whenever possible, and if required will be performed with proper controls in place. Lead sheet shielding is also located in various locations within the WR-1 Building. These lead sheets will be removed where necessary to lower the lead hazard while keeping in mind exposure rates.

7.3.2.3.3 Organic Coolant

With the exception of the organic coolant, the identified hazardous substances are routinely addressed during a typical construction project. Protective measures that will be used when exposure to the organic coolant:

- use appropriate chemical resistant gloves to protect hands and skin;
- wear eye protection;
- wash hands and skin following contact; and
- confirm there is adequate ventilation.

Residual organic coolant collected during the Project will be sent for appropriate storage or disposal.

7.3.2.3.4 Hydraulic Liquid and Fuel

Heavy equipment will be used during the Project activities, spills of hydraulic liquid or fuel may occur. As part of managing these hazards:

- routine inspections of equipment are completed and repairs should be completed, if required before work begins;
- spill kits be available for each piece of heavy equipment in use on-site; and
- spill trays required for refueling.

Environmental Compliance Protection will be contacted if there is a spill.

7.3.2.3.5 Fire and Explosions

During the dismantling of system components with traces of residual flammable, combustible or reactive materials could result in fire or explosion. As part of managing this hazard:

- hazardous waste inventories will be site specific;
- adequate drainage and ventilation will be provided to accommodate firefighting efforts and confirm worker safety;
- availability of local fire extinguishers will be adequate;
- storage of extra fuel shall be labelled in CSA approved containers, sheltered from the elements, and kept in secondary containment away from storm drains;
- all refueling shall take place over a catch tray or absorbent cloth away from storm drains or watercourses;
- chemicals shall be clearly labeled and stored in appropriate chemical storage containers, incompatible materials shall not be stored together;
- compressed gases shall be secured in a ventilated and labeled compressed gas storage cage with signage indicating the flammable / explosive nature of the hazard if applicable;
- work areas such as marine containers and material lay-down areas shall be maintained in a clean state;
- waste materials shall be segregated and identified; and
- supplies and equipment shall be stored so as to minimize the risk of potential releases to the environment.

7.3.3 Airborne Non-radiological Releases

To estimate the concentration for each non-radiological COPC at the location of an on-site receptor, the daily release rate (SPM) per COPC was multiplied by the daily dispersion factor of 3.18E-05 s/m^3 determined from the IMPACT model (EcoMetrix 2021). The annual dispersion factor from IMPACT was converted to a daily value using the MOECC (2009) averaging equation to convert between averaging periods. Air concentrations were then compared against their respective air criteria in the same averaging period (i.e., 24-hour).

Applicable air quality criteria were selected for each non-radiological COPC with a preference for ambient air quality criteria from Manitoba. Where local criteria were not available, criteria from other jurisdictions such as Ontario were used. The majority of the non-radiological COPCs have air quality criteria in the 24-hour averaging period; however, where needed, predicted concentrations were converted to match the averaging period of the relevant criterion. As shown in Table 7.3.3-1, all predicted air concentrations for all non-radiological COPCs evaluated are below their relevant ambient air quality criteria; therefore, no health effects are anticipated during the closure phase due to inhalation.

Although air quality criteria have been identified for boron and potassium hydroxide, these are not considered health-based criteria. Potassium hydroxide is considered a corrosive chemical, however chronic inhalation at low concentrations is not a human health concern. Additionally, studies from the US EPA (2008) have concluded that boron inhalation has not been associated with adverse health effects in humans.

Table 7.3.3-1: Estimated Non-Radiological Airborne COPC Release Rates from WR-1 Systems during Grouting

Non-Radiological COPC	Max Air Conc. (µg/m^3)	Average Air Conc. (µg/m^3)	Averaging Period	Applicable Air Quality Guideline (µg/m^3)	Reference
Boron	1.05E-10	5.56E-12	24-hour	120	Ontario AAQC (particulate) [2]
Cadmium	1.06E-05	5.64E-07	24-hour	2	Manitoba AAQC (MAC) [1]
Chromium	1.72E-05	9.14E-07	24-hour	0.5	Ontario AAQC (health) [2]
HB-40	2.78E-03	7.36E-04	8-hour	500	OSHA[3] TWA / 10
Lead	1.43E-03	2.52E-04	24-hour	2	Manitoba AAQC (MAC) [1]
Mercury	3.84E-08	2.04E-09	24-hour	2	Ontario AAQC (health) [2]
Palladium	1.81E-06	9.57E-08	24-hour	10	Ontario AAQC (health) [2]
Potassium Hydroxide	1.16E-09	6.17E-11	24-hour	14	Ontario AAQC (corrosion)[2]
Xylene	2.21E-07	1.17E-08	24-hour	730	Ontario AAQC (health) [2]

Source: EcoMetrix 2021

COPC = constituent of potential concern; Max = maximum; Avg = Average.

Note:

1. Manitoba Conservation 2005, https://www.gov.mb.ca/sd/envprograms/airquality/pdf/criteria_table_update_july_2005.pdf.
2. MOECC 2012, http://www.airqualityontario.com/downloads/AmbientAirQualityCriteria.pdf.
3. OSHA (1989) TWA of 5 mg/m^3 for hydrogenated terphenyl cited in MSDS for HB-40 (Eastman Chemical Company 2015).

7.3.4 Assumptions and Uncertainty

With the exception of the organic coolant (HB-40), the identified hazardous substances are routinely addressed in construction projects. To address the potential for unknown or greater than expected quantities of hazardous substances being encountered within the WR-1 Building, work plans will address all non-radiological COPCs. Work plans will include procedures to limit worker exposures, allowable airborne exposure concentrations, compliance monitoring programs, and waste disposal plans. All work will be in accordance with applicable workplace safety regulation, which include proper collection and disposal of waste material. As such, there will be little release of material to the environment.

To assess a range of quantities of non-radiological substances, the maximum and average quantities estimated were assessed as a bounding assumption. The designated substances will likely not be released in significant quantities, since their collection and appropriate disposal will be encompassed by existing compliance programs at the WL site, and the activities required for removal operations will be addressed during the development of detailed work plans. Management of these risks is not “new” but are considered non-routine activities. However, it is unknown if organic coolant concentrations could approach benchmark air concentrations defined for protection of worker. Assuming COPCs are released as SPM gives the most conservative release rate for non-radionuclides.

7.4 Non-Radiological Assessment for Public under Normal Conditions

Human receptors evaluated for both the radiological are applicable to the non-radiological assessment for the closure phase and includes off-site member of the public and critical groups. Similar to the radiological assessment, these receptors are potentially exposed to airborne effluent from the WR-1 Building during the closure phase.

7.4.1 Hazard Identification and Exposure Pathways

Non-radiological COPCs are similar to the non-radiological assessment for workers and may be released from the WL site during grouting activities as SPM, PM_{10}, and $PM_{2.5}$. After the systematic evaluation of closure activities, a select number of non-radiological COPCs were identified for the closure phase and inventory estimates were assessed. The estimated inventory of non-radionuclides derived for the WR-1 Building is provided in Table 3-14 of the ERA (EcoMetrix 2021).

The potential effects from the ISD of the WR-1 include nuisance dust and fine particulates from grouting, building demolition, site restoration, and rehabilitation activities. This includes the generation of non-radioactive hazardous liquid or solid wastes (e.g., asbestos, lead), including dust during disturbance activities. During the closure phase, releases to surface water are not expected, because any release from the grouted reactor would occur after closure, and contaminant transport via groundwater to surface water would take additional time. During the closure phase, human receptors on Farm A and F will be exposed via air (inhalation, immersion), soil (incidental ingestion, ground shine), well water (ingestion, bathing), and ingestion of home-grown vegetables, fruits, and livestock, and ingestion from locally hunted deer (Table 7.4.1-1).

Table 7.4.1-1: Complete Exposure Pathways for Public Receptors Exposed to Non-Radiological COPCs during the Closure Phase

Receptor	Exposure Pathway	Environmental Media
Farm (A or F)	Inhalation	■ Air
	Ingestion	■ Water (well water) ■ Soil (incidental) ■ Terrestrial plants (homegrown) ■ Terrestrial animals (beef, pork, poultry, eggs, milk, game, honey)
	External	■ Air ■ Water (well water) ■ Soil
Harvester	Inhalation	■ Air
	Ingestion	■ Terrestrial animals (hare, deer) ■ Terrestrial plants (berries, weekay, cedar)
	External	■ Air ■ Soil

7.4.2 Planned Mitigation

All decommissioning work will be planned and executed in accordance with the Decommissioning Licence, WL Decommissioning Quality Assurance Plan, Work Planning for WL Decommissioning, Occupational Safety and Health Program, Environmental Protection Program, WL Emergency Plan, and Engineering Change Control. The operational controls described in Section 7.2.2 Planned Mitigation are also applicable.

7.4.3 Airborne Non-radiological Releases

As described in Section 7.3.3 Airborne Non-radiological Releases, airborne non-radiological releases were modelled for an on-site receptor (i.e., worker). This receptor is bounding for the public receptors (Farm A and F and harvesters) because these receptors are farther away from the source of potential airborne releases which will result in lower concentrations compared to the on-site receptor. All predicted air concentrations for all non-radiological COPCs evaluated for the on-site receptor are below their relevant ambient air quality criteria; therefore, no health effects are anticipated during the closure phase due to inhalation.

Although air quality criteria have been identified for boron and potassium hydroxide, these are not considered health-based criteria. Potassium hydroxide is considered a corrosive chemical, however chronic inhalation at low concentrations is not a human health concern. Additionally, studies from the US EPA (2008) have concluded that boron inhalation has not been associated with adverse health effects in humans.

7.4.4 Assumption and Uncertainty

With the exception of the HB-40, the identified hazardous substances are routinely addressed in construction projects. To address the potential for unknown or greater than expected quantities of hazardous substances being encountered within the WR-1 Building, work plans will address all non-radiological COPCs. All work will be in accordance with applicable workplace safety regulation, which include proper collection and disposal of waste material. As such, there will be little release of material to the environment.

To assess a range of quantities of non-radiological substances, the maximum and average quantities estimated were assessed, as a bounding assumption. The designated substances are not expected to be released in significant quantities, since their collection and appropriate disposal will be encompassed by existing compliance programs at the WL site. In addition, the activities required for removal operations will be addressed during the development of detailed work plans. Management of these risks is not "new" but are considered non-routine activities; however, it is unknown if organic coolant concentrations could approach benchmark air concentrations defined for protection of the public. Assuming COPCs are released as SPM gives the most conservative release rate for non-radionuclides.

7.5 Radiological Assessment for Non-human Biota

7.5.1 Hazard Identification and Exposure Pathways

The receptors for the radiological risk assessment were selected to be appropriate for assessment of both radiological and non-radiological stressors on ecological health. Indigenous and public input was considered when selecting Valued Components (VCs). For example, it was noted that the white-tailed deer VC is an important game species for traditional communities. For additional details on the rationale for chosen VCs refer to Section 6.1.1 Receptor Selection and Characterization of the ERA (EcoMetrix 2021).

Pathways relevant to potential exposure to liquid effluent sources are unlinked (i.e., limited by mitigation), as during the closure phase releases to surface water are not expected to occur. Therefore, exposure to atmospheric releases is the primary pathway. Exposure pathways include the routes of contamination dispersion from the source to the receptor location, as well as routes of contaminant transport thought the food chain or other media to the receptor organism (Table 7.5.1-1). For soil invertebrates and terrestrial plants, the main exposure pathway is through contact with soil and contaminant uptake from soil via bioaccumulation. The dominant exposure pathways for birds and mammals are through the uptake of contaminants via the incidental ingestion of soil and ingestion of food.

Table 7.5.1-1: Complete Exposure Pathways for Selected VC Species during Closure

VC Category	VC	Exposure Pathways	Environmental Media
Terrestrial Invertebrates	Earthworm	Direct Contact	■ In Soil
Terrestrial Plants	Grasses/Shrubs	Direct Contact	■ On Soil
	Berries	Direct Contact	■ On Soil
Terrestrial Birds	American Robin	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Earthworms ■ Fruit/Berries
	Loggerhead Shrike	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Earthworms ■ American Robin ■ Meadow Vole
Terrestrial Mammals	Meadow Vole	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Grasses ■ Fruit/Berries
	Common Shrew	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Earthworms
	Snowshoe Hare	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Grasses ■ Fruit/Berries ■ Shrubs
	White-tailed Deer	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Grasses ■ Fruit/Berries

Table 7.5.1-1: Complete Exposure Pathways for Selected VC Species during Closure

VC Category	VC	Exposure Pathways	Environmental Media
Terrestrial Mammals	Red Fox	Direct Contact	■ In and on Soil
		Ingestion	■ Soil ■ Grasses ■ Fruit/Berries ■ American Robin ■ Loggerhead Shrike ■ Meadow Vole ■ Snowshoe Hare
	Little Brown Myotis	Direct Contact	■ On Soil
		Ingestion	■ Soil ■ Earthworm

7.5.2 Planned Mitigation

All decommissioning work will be planned and executed in accordance with the Decommissioning Licence, WL Decommissioning Quality Assurance Plan, Work Planning for WL Decommissioning, Radiation Protection Program, Occupational Safety and Health Program, Environmental Protection Program, WL Emergency Plan, and Engineering Change Control. The operational controls described in Section 7.2.2 Planned Mitigation are also applicable.

7.5.3 Airborne Radiological Releases

As discussed in Section 7.1.3 Airborne Radiological Releases, the radionuclides that are considered for the ERA during the closure phase were based on operational experience at the WL Site and were primarily identified in the *Derived Release Limits for AECL's Whiteshell Laboratories* (CNL 2016c) and the *WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* (CNL 2020b). These radionuclides have been found in the WL's airborne effluent or are reasonably expected to be found in the airborne effluent.

The estimated inventory of radionuclides derived for the timeframe of 30 years following shutdown was determined to be appropriate for the closure phase. Estimated radionuclide inventories for the reactor core, biological shield, and PHT system can be found in Table 7.1.1-3.

Estimated radionuclide release rate for the dismantling of the PHT system prior to grouting and for grouting activities are presented in Table 7.1.3-1 and Table 7.1.3-2, respectively. Estimated tritium release rate for the majority of the closure activities was based on the average obtained from operational data collected from 2011 to 2019, which was 1.11E+09 Bq/wk (1.84E+03 Bq/s). During grouting, however, vibration and heating of structures and systems is expected to increase atmospheric tritium release rate. For grouting the estimated release rate used was the maximum weekly tritium release rate observed during characterization activities undertaken between 2011 and 2015, which was 1.28E+10 Bq/wk (EcoMetrix 2021).

7.5.4 Estimated Non-human Biota Dose

An environmental transport and pathways model was used to evaluate the transport and effects of contaminants on the local environment including human and ecological receptors. A more detailed description of the model is provided in the ERA (EcoMetrix 2021). Using IMPACT, radiological doses were calculated for the closure phase during the demolition prior to grouting and during grouting activities for all ecological receptors. Both the maximum and average atmospheric release scenarios were evaluated.

Maximum and average estimated radiation doses for ecological receptors during demolition activities prior to grouting are detailed in Tables 6-6 and 6-7 of the ERA (EcoMetrix 2021), respectively. For grouting activities, the maximum and average doses are detailed in Tables 6-8 and 6-9 of the ERA (EcoMetrix 2021). A summation of total dose expected exposures are provided in Table 7.5.4-1. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 7.5.4-1: Summary of Total Dose during Closure Phase for Selected Ecological Receptors

Ecological Receptor / Exposure Scenario	Dose (mGy/d)	Percent of Dose Benchmark
Demolition		
American Robin		
Demolition - Maximum	5.27E-05	<0.01%
Demolition - Average	1.39E-05	<0.01%
Loggerhead Shrike		
Demolition - Maximum	7.86E-05	<0.01%
Demolition - Average	2.07E-05	<0.01%
Meadow Vole		
Demolition - Maximum	9.71E-06	<0.01%
Demolition - Average	2.56E-06	<0.01%
Common Shrew		
Demolition - Maximum	4.84E-05	<0.01%
Demolition - Average	1.27E-05	<0.01%
Deer		
Demolition - Maximum	1.38E-04	<0.01%
Demolition - Average	3.62E-05	<0.01%
Rabbit (Snowshoe Hare)		
Demolition - Maximum	1.70E-03	0.07%
Demolition - Average	4.48E-04	0.02%
Red Fox		
Demolition - Maximum	1.28E-04	<0.01%
Demolition - Average	3.37E-05	<0.01%
Little Brown Bat		
Demolition - Maximum	1.50E-05	<0.01%
Demolition - Average	3.95E-06	<0.01%
Forage		
Demolition - Maximum	5.11E-04	0.02%
Demolition - Average	1.34E-04	<0.01%
Grass		
Demolition - Maximum	5.15E-04	0.02%
Demolition - Average	1.35E-04	<0.01%
Fruits		
Demolition - Maximum	3.95E-05	<0.01%
Demolition - Average	1.04E-05	<0.01%
Earthworm		
Demolition - Maximum	9.18E-05	<0.01%
Demolition - Average	2.41E-05	<0.01%

Table 7.5.4-1: Summary of Total Dose during Closure Phase for Selected Ecological Receptors

Ecological Receptor / Exposure Scenario	Dose (mGy/d)	Percent of Dose Benchmark
Grouting		
American Robin		
Grouting - Maximum	8.57E-06	<0.01%
Grouting – Average	4.82E-06	<0.01%
Loggerhead Shrike		
Grouting - Maximum	8.83E-06	<0.01%
Grouting – Average	4.93E-06	<0.01%
Meadow Vole		
Grouting - Maximum	8.04E-06	<0.01%
Grouting – Average	4.47E-06	<0.01%
Common Shrew		
Grouting - Maximum	8.98E-06	<0.01%
Grouting – Average	4.97E-06	<0.01%
Deer		
Grouting - Maximum	1.03E-05	<0.01%
Grouting – Average	5.71E-06	<0.01%
Rabbit (Snowshoe Hare)		
Grouting - Maximum	1.06E-05	<0.01%
Grouting – Average	5.84E-06	<0.01%
Red Fox		
Grouting - Maximum	7.31E-06	<0.01%
Grouting – Average	4.03E-06	<0.01%
Little Brown Bat		
Grouting - Maximum	4.49E-06	<0.01%
Grouting – Average	2.59E-06	<0.01%
Forage		
Grouting - Maximum	1.70E-05	<0.01%
Grouting – Average	9.46E-06	<0.01%
Grass		
Grouting - Maximum	2.54E-05	<0.01%
Grouting – Average	1.39E-05	<0.01%
Fruits		
Grouting - Maximum	5.74E-06	<0.01%
Grouting – Average	3.57E-06	<0.01%
Earthworm		
Grouting - Maximum	9.29E-06	<0.01%
Grouting – Average	5.44E-06	<0.01%

Note:

mGy/d = milligray per day; Max = maximum; Avg = Average.

There are no exceedances of the 2.4 milligray per day (mGy/d) radiation benchmark for terrestrial and riparian biota on or near the WL site. An evaluation relative to assessment criteria illustrates that all predicted doses are less than 1% of the radiation benchmark (a dose of 0.024 mGy/d is equivalent to 1% of the terrestrial and riparian ecological radiation benchmark). All predicted doses are well below this level. Therefore, it is unlikely that there would be significant adverse effects on terrestrial populations or communities as a result of radionuclide releases from closure activities. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

7.5.5 Assumptions and Uncertainty

Assumptions and uncertainties regarding radiological inventories within the WR-1 Building and radiological release rates are discussed in Section 7.1.5 Assumptions and Uncertainty. As well as the modeling input assumptions, there is uncertainty in the radiological release rates to the atmosphere; however, the estimates are expected to be conservative.

7.6 Non-Radiological Assessment for Non-human Biota

7.6.1 Hazard Identification and Exposure Pathways

The receptors for the radiological risk assessment were selected to be appropriate for assessment of both radiological and non-radiological stressors on ecological health. For details on the rationale for chosen VCs refer to Section 6.1.1 of the ERA (EcoMetrix 2021).

Pathways relevant to potential exposure to liquid effluent sources are considered to be unlinked (i.e., limited by mitigation), as during the closure phase releases to surface water are not expected to occur. Therefore, exposure to atmospheric releases is the primary pathway. Exposure pathways for non-radionuclides are the same as the radionuclides as shown in Table 7.5.1-1.

7.6.2 Planned Mitigation

All decommissioning work will be planned and executed in accordance with the Decommissioning Licence, WL Decommissioning Quality Assurance Plan, Work Planning for WL Decommissioning, Occupational Safety and Health Program, Environmental Protection Program, WL Emergency Plan, and Engineering Change Control. The operational controls described in Section 7.2.2 Planned Mitigation are also applicable.

7.6.3 Airborne Non-Radiological Releases

As described in Section 7.3.3 Airborne Non-radiological Releases, all predicted air concentrations for all non-radiological COPCs evaluated for the on-site receptor are below their relevant ambient air quality criteria; therefore, no ecological health effects are anticipated during the closure phase.

7.6.4 Assumptions and Uncertainty

Assumptions and uncertainties regarding non-radiological inventories within the WR-1 Building and non-radiological release rates are discussed in Section 7.3.4 Assumptions and Uncertainty. As well as the modeling input assumptions, there is uncertainty in the non-radiological release rates to the atmosphere; however, the estimates are expected to be conservative.

8.0 POST-CLOSURE SAFETY ASSESSMENT

This section provides the results of the safety assessment completed for the post-closure phase. Results for the Normal Evolution Scenario are provided for the radiological assessment completed for workers and the public, the non-radiological assessment completed for the public, and the radiological and non-radiological assessment completed for non-human biota during the post-closure phase of the Project. Detailed methods and results are provided in the ERA (EcoMetrix 2021).

The scope encompasses the assessment of the end-state of the WR-1 Building, and an assessment timeframe of 10,000 years was chosen based on the following (see Section 5.3 Timeframes for further detail):

- Design life of engineered barriers (i.e., the WRDF).
- Duration of institutional controls.
- Hazardous lifetime of the contaminants associated with the waste:
  - Grout mass activity comparable with natural analogue.
  - Reactor mass activity is comparable to natural analogues.
- Frequency (probability) of natural and anthropogenic changes (e.g., seismic occurrence, flood, drought, glaciation, and climate change):
  - Glaciation determined to be a probable but low risk natural event not likely to occur (i.e., initiation of glaciation) before 100,000 years have lapsed (see Section 5.4.3.2.6 Glaciation).

8.1 Radiological Assessment for Workers under Normal Conditions

Radiation monitoring for Nuclear Energy Workers and non-Nuclear Energy Workers on-site is monitored, and their doses are controlled through CNL’s Radiation Protection Program. It is recognized that worker exposures on-site are stringently controlled, documented and reported, as such the procedures and processes in place are proven to be effective. Furthermore, on-site workers are not assessed during post-closure for radiological and non-radiological exposures since there is no airborne exposure pathways (end-state of the WRDF eliminated airborne pathways) and there is no aquatic pathway to on-site workers as they will not be in contact with or consuming water from local waterbodies (i.e., accommodations and drinking water would be provided off-site [e.g., bottled water]).

8.2 Radiological Assessment for Public under Normal Evolution Scenario

The post-closure phase, including the institutional control period, is not likely to have a source of emissions for airborne contaminants, as the grouting will have been completed, the above-grade building will have been completely decommissioned and removed, and the concrete cap and engineered cover will have been installed on top of the WRDF. The focus in the post-closure phase, unlike during the closure phase, is releases from the WRDF to groundwater and subsequent migration to surface water at the Winnipeg River, as the pumping from the WR-1 Complex sumps will have ceased. Groundwater elevation after pumping cessation will recover to a new equilibrium and a significant portion of the WRDF will be below the water table. It is anticipated that the WRDF components will gradually deteriorate over time, allowing the release of solutes into the groundwater.

8.2.1 Hazard Identification and Exposure Pathways

Exposure pathways from atmospheric release are not considered relevant during the post-closure phase, as releases to air are not expected during post-closure. During the post-closure phase groundwater releases to surface water will occur. Aquatic dispersion will carry contaminants to downstream locations on the Winnipeg River. Waterborne contaminants can partition to sediment. River water and sediment will be the primary exposure media.

Pathways relevant to exposure to release of groundwater to surface water are presented for the human receptors on Figure 8.2.1-1. For assessment of non-radiological COPCs only the ingestion pathway has been considered relevant, since the dermal pathway is considered negligible for inorganics (i.e., cadmium and lead).

During post-closure, human receptors on the On-site Farm and Farm A will be exposed via use of water from the Winnipeg River for drinking, bathing, livestock watering, and irrigation (lawns and gardens), and by ingestion of home-grown vegetables, fruit, and livestock. Ingestion of terrestrial plants and animals is included for the On-site and Farm A receptors since the Winnipeg River is used for irrigation of these plants and as drinking water for the animals. Residents from the farms are also assumed to fish in the Winnipeg River. As a disruptive event (well in plume scenario), the unlikely scenario is assessed where the On-site Farm obtains drinking water from a groundwater well in the WRDF plume.

It was noted from Indigenous engagement activities that Sagkeeng First Nation members harvest wild rice and medicinal plants near the WL site. However, wild rice does not grow on the Winnipeg River downstream of WR-1. Nor are aquatic medicinal plants such as water lilies common on the river downstream of WR-1. There is uncertainty around the Harvester’s diet far into the future based on possible changes to what is available and can grow near WL; however, the diet has been developed based on the best available information at this time. Consumption of weekay has been included in the Harvester’s diet which can provide a general indication of the dose from ingestion of wild rice.

CNL (2018a) also conducted an Indigenous Food Intake Survey completed by members of the Sagkeeng First Nation to understand the types and quantities of local food consumed. The results indicate that survey participants consume animals such as wild game (e.g., moose, deer, rabbit and hare), waterfowl (e.g., duck and geese), fish, fruits and berries and medicinal plants (e.g., weekay and cedar). During the post-closure phase, the focus is on aquatic pathways since groundwater releases to surface water will occur. Since the focus is on aquatic pathways, a moose has been included instead of a deer (which was assessed during the closure phase), since a portion of the moose’s diet is from ingestion of aquatic plants. Weekay is a wetland plant and could grow along the shore of the Winnipeg River or in shallow areas. It is unlikely that weekay would be exposed to direct groundwater but could potentially be exposed to river water.

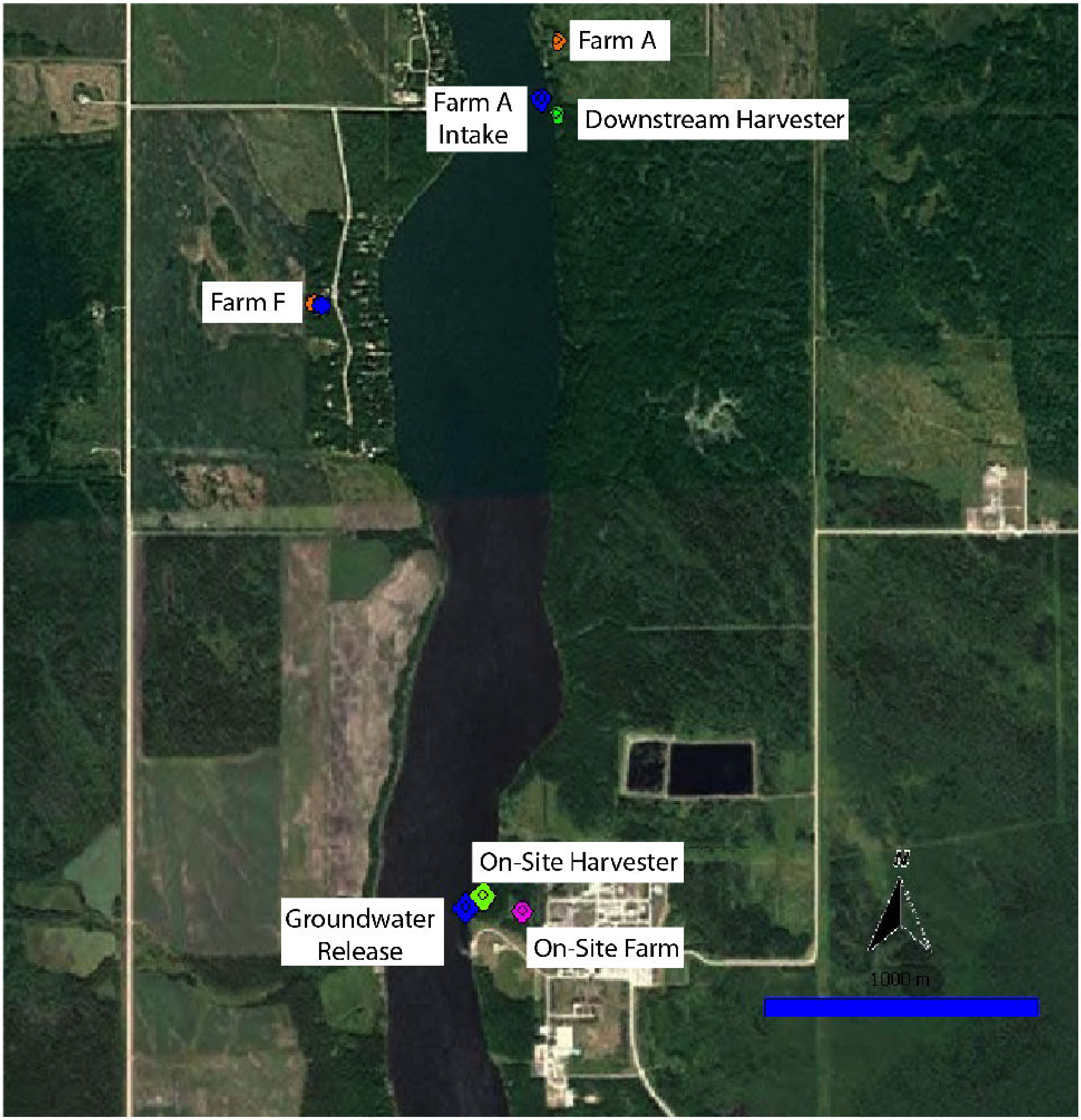
Based on these considerations, exposure via consumption of fish and waterfowl would be the important pathways for exposure of the harvester to contaminants released from WR-1 to the river. Harvesters will ingest country foods such as weekay, fish and waterfowl, as well as moose that drink from the Winnipeg River. During post-closure, aquatic release (groundwater flow to the Winnipeg River) is the relevant pathway; therefore, terrestrial pathways are not complete for the harvester in post-closure.

There is no direct release to air; however, for volatile radionuclides (tritium, carbon-14, iodine-129), receptors will be exposed via the air pathway (inhalation and immersion) through volatilization from irrigated soil. All tritium mass was conservatively assumed to migrate via the groundwater flow pathway without loss to volatilization.

Full-time residency was assumed for the residents of Farm A and On-site Farm. The harvester was assumed to be harvesting on the WL site for 2 hours per week, downstream in the vicinity of Farm A for 2 hours per week, and upstream for 2 hours per week. For Farm A and the On-site Farm, the percentage of food items in the diet that are obtained from local sources was consistent with the *Derived Release Limits for AECL’s Whiteshell*

Laboratories (CNL 2016c). For the harvester, the local percentage of food intake was assumed to be 33% (fish, duck, moose, and weekay) since the harvester only spends 33% of harvesting time at each harvest location.

For Farm A and the On-site Farm, all water for drinking, irrigation, bathing, and animal drinking is obtained from the Winnipeg River. The harvester does not consume water, but the animals that are harvested obtain their drinking water from the Winnipeg River.



REFERENCE(S)

1. ECOMETRIX, 2017

CLIENT

CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT

DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE

HUMAN RECEPTOR LOCATIONS WITHIN THE VICINITY OF WHITESHELL LABORATORIES DURING POST-CLOSURE PHASE

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
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Details pertaining to the exposure pathways to receptors in the environment are summarized in Table 8.2.1-1.

Table 8.2.1-1: Complete Exposure Pathways for Receptors Exposed to Radiological and Non-Radiological COPCs during Post-closure Phase

Receptor	Exposure Source	Exposure Pathway	Environmental Media
Farm F and Farm A – full-time residency (Adult, 10-year old Child, 1-year old, Infant (formula/milk) and 3-month old -Infant (formula/nursing)	Groundwater Release to Surface Water	Inhalation	Air
		Ingestion	Water (Winnipeg River)
			Soil (incidental)
			Terrestrial Plants (homegrown)
			Terrestrial Animals and Animal Products (livestock, game, milk, and honey)
			Aquatic Animals (fish)
		External	Air
			Water (Winnipeg River)
			Soil / Sediment
On-site Farm – full-time residency (Adult, 10-year old child; 1-year old Infant, (formula/milk) and 3-month old infant formula/nursing)		Inhalation	Air
		Ingestion	Water (Winnipeg River)
			Soil (incidental)
			Terrestrial Plants (homegrown)
			Terrestrial Animals and Animal Products (livestock, game, milk, and honey)
			Aquatic Animals (fish)
		External	Air
			Water (Winnipeg River)
			Soil / Sediment
Harvester – 2 hrs/wk on-site, downstream, and upstream from the Project activities (Adult, 10-year old Child and 1-year old, Infant)		Inhalation	Air
		Ingestion	Terrestrial Animals (waterfowl, moose)
			Aquatic Animals (fish)
			Riparian plants (weekay)
		External	Air
			Soil / Sediment

COPC = constituents of potential concern.

8.2.2 Planned Mitigation

During institutional control (i.e., 100 years after closure), the WL site will still be managed, including ongoing monitoring program. During this time, controls to restrict access to the WRDF are expected to be effective (e.g., fencing, signage, durable markings). The ISD design will consider possible events that could degrade the integrity of the WRDF. It is recognized that there are inevitable uncertainties associated with predicting the performance of the WRDF over a long-time scale. Therefore, safety is established through adequate defence-in-depth. Defence-in-depth approach, with the Project consisting not only of the WRDF (i.e., concrete foundation, grout capsule, and internal systems and components left intact), but the geosphere and biosphere as well. These defence-in-depth features were all conservatively considered in the solute transport modelling predictions (see Section 6.0 Defence-in-Depth for the In Situ Disposal System).

Solutes originating from the metal components remaining within the WR-1 Building (e.g., systems and pipes) will be gradually released overtime from these materials through corrosion. The WRDF consists of the concrete building foundation, grout capsule, and concrete cap and engineered cover, the structure has been designed to provide sufficient containment and isolation of the specific contaminated waste to be decommissioned in place. As such, the radiological decay and ingrowth over the 50-year time period since shutdown was considered. The effectiveness of the WRDF components at providing a barrier to the geosphere was cumulatively considered in the modelling, as well as the staged decrease in effectiveness of each component overtime with degradation. Components of the geosphere incorporated included the surficial geology surrounding the WRDF (including the backfill used to fill excavation required to construct the building), and the upper bedrock that represents the connection between contaminates that will start to migrate out of the WRDF and the receptors in the environment. The rate at which contaminates will reach the surface environment are substantially mediated by the bedrock, groundwater flow occurs primarily within fractures, the travel time between the bedrock and the Winnipeg River is about 100 years.

Following closure, the post-closure phase will commence and include a period of a minimum of 100 years of institutional control during which both active and passive controls will be implemented. It is recognized that institutional control will continue until the CNSC agrees it is no longer needed. During institutional control, groundwater monitoring and groundwater quality management will continue to demonstrate compliance with the safety case assumptions and illustrate that the site has reach a safe and stable state. The post-closure phase will continue indefinitely.

8.2.3 Radiological Releases

Following cessation of pumping from the sumps in the WR-1 Complex and the encapsulation of the WR-1 the groundwater elevations will be restored to an equilibrium condition and most of the grout (including the remaining components of the reactor) will be situated below the water table. As described in Section 5.5.1.2 Solute Transport Modelling, a period of 10 years was assumed for complete resaturation of the grout block and recovery of the groundwater level in the soil surrounding the WRDF.

It is anticipated that following resaturation solute release will occur as a result of gradual deterioration of the grout and the reactor components. Since the neutron activation products are generally dispersed within the bulk of the metal and concrete that once formed the WR-1, the dissolution of this bulk activity within the groundwater is anticipated to occur over a long period of time. This dissolution is a function of the material type but may occur over a period of more than 600 years to more than 10,000 years (Golder 2021). The fission products and the actinides are more concentrated as they reside within particles contained within the various pipes, tanks and pumps. Hence the rate of dissolution of the fission products and actinides is likely to increase rapidly after the encapsulating grout has deteriorated to the point where groundwater is flowing through the WR-1.

However, for the purpose of this initial screening-level assessment, it was assumed that the entire mass inventory of contaminants within the grout will be available as dissolved phase mass at the start of the release period (2035). Following the initial release, solutes will migrate through the grout via diffusion and advection to the boundary of the grout and will subsequently be transported in groundwater through the geological pathways to the downstream environment.

A groundwater model was used to predict mass loadings for radiological and non-radiological COPCs. Based on application of the groundwater model, mass loadings to the Winnipeg River were provided over a modelling timeframe of 500,000 years. Maximum mass loadings for each COPC was assessed at a single point in time, corresponding to the peak loading rate from groundwater to the Winnipeg River. For COPCs that did not reach a peak loading rate in the specified modelling time frame of 500,000 years, the simulations were extended until the peak loading rate could be determined. During post-closure no treatment of effluent released from the site (i.e., groundwater flowing through the WRDF) will be required as part of the Project design.

The radionuclides that are considered for the post-closure HHRA are those that have been identified in the *WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates* (CNL 2020b) and assessed in *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021). The radiological inventory is discussed in Section 7.1.1.1 Radionuclides. The mass inventory at 50 years following the WR-1 shutdown is provided in Table 8.2.3-1 and the simulated peak mass loading rates from the bedrock pathway are provided in Table 8.2.3-2. These radionuclides have historically been found in WL’s waterborne effluent or are reasonably expected to be found in the WRDF and have the potential to migrate from groundwater to surface water during the post-closure phase. Radionuclides are considered of public and regulatory interest, therefore, all radionuclides identified were carried forward into the HHRA.

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Table 8.2.3-1: Mass Inventories at 50 Years (2035) Following WR-1 Shutdown

Solute	Mass Inventory (g)			Solute	Mass Inventory (g)			Solute	Mass Inventory (g)			Solute	Mass Inventory (g)		
	Bioshield	Reactor	Other		Bioshield	Reactor	Other		Bioshield	Reactor	Other		Bioshield	Reactor	Other
Actinium-225	1.53E-16	0	0	Europium-154	0.0001111	0	0	Nitrogen	1.816E-06	0.08788	0	Technetium-99	0.2052	0	0
Actinium-227	3.16E-10	0	0	Europium-155	1.303E-06	0	0	Palladium	0	9.94E-06	15500	Thallium	0	0	0
Americium 243	0.002563	0	0	Gadolinium	0.003192	0	0	Plutonium -239	2.655	0	0	Thorine-227	7.30E-13	0	0
Americium-241	0.1949	0	0	Gadolinium-152	7.557E-05	0	0	Plutonium -240	1.032	0	0	Thorine-228	1.91E-19	0	0
Argon	3.45E-10	0	0	H3	0	0	2.25	Plutonium -241	0.02007	0	0	Thorine-229	4.21E-11	0	0
Barium	0.1431	0	0	HB-40	0	0	87700000	Plutonium-238	0.003336	0	0	Thorine-230	7.18E-08	0	0
Beryllium	0	0	0	Helium	0	0	0	Polonium-210	4.65E-16	0	0	Thorine-231	8.14E-11	0	0
Bismuth	4.59E-14	0	0	Iodine-129	0.04284	0	0	Potassium	1.147E-05	0	0	Thorine-232	2.50E-09	0	0
Bismuth-210	1.77E-17	0	0	Iron-55	0	0.001362	0	Protactinium-231	9.68E-07	0	0	Thorine-234	1.45E-08	0	0
Boron	0	0	0.9	KOH	0	0	10	Protactinium-233	1.56E-09	0	0	Uranium-233	5.15E-07	0	0
Cadmium	0	9.70E-07	91400	Lead	1.70E-10	0	0	Radium -224	9.82E-22	0	0	Uranium-234	0.001227	0	0
Caesium-137	0.09464	0	0	Lead-210	2.87E-14	0	0	Radium -225	2.29E-16	0	0	Uranium-235	20.02	0	0
Calcium-41	0.04421	0	0	Manganese	0	34.1	0	Radium -226	8.85E-12	0	0	Uranium-236	0.004296	0	0
Carbon-14	0.0003723	18.02	0	Mercury	0	0	330	Radium -228	6.75E-19	0	0	Uranium-237	6.24E-10	0	0
Cerium	0	0	0	Molybdenum	0	0.6031	0	Radium-223	4.44E-13	0	0	Uranium-238	997.1	0	0
Chlorine-36	3.437E-06	0	0	Neptunium-237	0.04617	0	0	Radon-222	5.68E-17	0	0	Xenon	7.71E-08	0	0
Chromium	0	0	148000	Neptunium-239	2.21E-09	0	0	Ruthenium	2.754E-05	0	0	Xylene	0	0	1900
Cobalt	0	1.025	0	Nickel	0.00128	44.21	0	Samarium	0.0001938	0	0	Yttrium-90	9.68E-06	0	0
Cobalt-60	6.827E-06	0.2358	0	Nickel-59	0	3734	0	Samarium-148	1.25E-17	0	0	Zirconium	0.0626	0	0
Copper	7.796E-05	128	0	Nickel-63	0.0002419	397.1	0	Silver-108m	3.09E-05	0.0001315	0				
Curium-244	3.176E-05	0	0	Niobium-144	1.83E-32	0	0	Strontium-90	0.03809	0	0				
Europium-152	0.0000401	0	0	Niobium-94	0	431.8	0	Sulfur	5.36E-12	0	0				

Note:

"Other" corresponds to non-radiological solute mass and tritium per CNL (2020b). “Reactor” inventory represents the mass allocated to the various reactor components.

(1) Helium produced thought radioactive decay of tritium in the period between 2015 and 2035 assumed to volatilize upon grouting during decommissioning.
(2) The inventory of mercury reflects the estimate from identified sources. Additional inventory of mercury from unidentified sources is estimated to be 0.41 kg, for a total of 0.74 kg (CNL 2020b)

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Table 8.2.3-2: Simulated Peak Mass Loading Rates from the Bedrock Pathway

Solute	Peak Mass Loading Rate (g/yr)	Peak Activity Rate (Bq/yr)	Peak Time[1] (year)	Solute	Peak Mass Loading Rate (g/yr)	Peak Activity Rate (Bq/yr)	Peak Time[1] (year)
Actinium-225	1.7E-17	3.6E+00	8.82E+04	Nitrogen	5.44E-04	N/A	1.79E+04
Actinium-227	1.4E-14	3.8E+00	1.60E+05	Palladium	2.52E-01	N/A	5.00E+05
Americium 243	N/A	0	N/A	Plutonium-238	N/A	5.49E+04	N/A
Americium-241	N/A	0	N/A	Plutonium-239	2.39E-07	1.33E+02	1.08E+05
Argon	6.9E-12	N/A	2.25E+02	Plutonium-240	1.58E-10	0.00E+00	6.70E+04
Barium	1.5E-23	N/A	5.00E+05	Plutonium-241	N/A	6.37E+01	N/A
Beryllium	N/A	N/A	N/A	Polonium-210	1.64E-10	0.00E+00	2.09E+05
Bismuth	1.1E-08	N/A	7.88E+04	Potassium	3.27E-07	N/A	1.34E+04
Bismuth-210	2.3E-14	1.1E+04	2.09E+05	Protactinium-231	2.16E-11	1.79E+01	1.61E+05
Boron	1.9E-04	N/A	4.24E+03	Protactinium-233	2.33E-16	0.00E+00	4.12E+04
Cadmium	3.7E-01	N/A	2.03E+05	Radium-223	3.36E-16	3.93E-04	1.61E+05
Caesium-137	N/A	0	N/A	Radium-224	6.66E-22	3.12E+01	4.98E+05
Calcium-41	1.6E-06	4.9E+05	2.01E+04	Radium-225	2.17E-16	6.03E+01	8.88E+04
Carbon-14	5.6E-04	3.0E+09	1.01E+03	Radium-226	1.65E-11	3.93E-04	2.09E+05
Cerium	N/A	N/A	N/A	Radium-228	3.89E-19	2.74E+06	4.98E+05
Chlorine-36	1.2E-08	1.4E+01	1.71E+02	Radon-222	4.82E-12	0.00E+00	2.09E+05
Chromium	5.0E-27	N/A	5.00E+05	Ruthenium	6.15E-09	N/A	4.32E+05
Cobalt	2.1E-02	N/A	9.38E+04	Samarium	N/A	N/A	N/A
Cobalt-60	N/A	0	N/A	Samarium-148	1.39E-24	0	2.13E+03
Copper	2.0E-04	N/A	5.00E+05	Silver-108m	8.0E-13	2.3E-01	3.66E+03
Curium-244	N/A	0	N/A	Strontium-90	N/A	1.01E+06	N/A
Europium-152	N/A	0	N/A	Sulfur	2.37E-12	N/A	1.97E+02
Europium-154	N/A	0	N/A	Technetium-99	1.60E-05	6.97E+00	1.06E+04
Europium-155	N/A	0	N/A	Thallium	N/A	N/A	N/A
Gadolinium	6.48E-06	N/A	2.15E+02	Thorium-227	6.12E-17	8.59E-05	1.62E+05
Gadolinium-152	1.71E-07	1.38E-07	2.14E+02	Thorium-228	2.83E-20	6.83E+00	4.90E+05
H3	6.37E-05	2.27E+10	6.80E+01	Thorium-229	8.69E-12	1.32E+01	8.93E+04
HB-40	4.05E+01	N/A	1.02E+04	Thorium-230	1.73E-10	9.10E+00	2.05E+05
Helium	1.07E-02	N/A	1.45E+02	Thorium-231	4.63E-18	8.59E-05	1.18E+05
Iodine-129	1.44E-04	9.42E+02	1.70E+02	Thorium-232	2.12E-10	6.29E+01	4.95E+05
Iron-55	N/A	0	N/A	Thorium-234	7.35E-16	0.00E+00	1.17E+05
KOH	4.76E-02	N/A	1.42E+02	Uranium-233	2.79E-08	3.14E+03	7.73E+04
Lead	1.05E-01	N/A	1.21E+05	Uranium-234	1.36E-07	1.28E+03	1.31E+05

Table 8.2.3-2: Simulated Peak Mass Loading Rates from the Bedrock Pathway

Solute	Peak Mass Loading Rate (g/yr)	Peak Activity Rate (Bq/yr)	Peak Time¹ (year)	Solute	Peak Mass Loading Rate (g/yr)	Peak Activity Rate (Bq/yr)	Peak Time¹ (year)
Lead-210	9.29E-13	0.00E+00	2.10E+05	Uranium-235	1.60E-04	1.73E+03	1.18E+05
Manganese	8.33E-05	N/A	3.61E+05	Uranium-236	7.21E-06	0.00E+00	1.20E+05
Mercury	1.37E-18	N/A	5.00E+05	Uranium-237	N/A	8.83E+03	N/A
Molybdenum	1.17E-02	N/A	1.79E+04	Uranium-238	7.10E-03	0.00E+00	1.17E+05
Neptunium-237	5.15E-06	0.00E+00	4.15E+04	Xenon	1.39E-09	N/A	2.04E+02
Neptunium-239	N/A	3.77E+00	N/A	**Xylene**	**4.32E+00**	**N/A**	**2.02E+02**
Nickel	2.01E-05	N/A	5.00E+05	Yttrium-90	N/A	0.00E+00	N/A
Nickel-59	5.48E-05	1.22E+07	4.92E+05	Yttrium-90	N/A	0.00E+00	N/A
Nickel-63	N/A	0	N/A	Zirconium	N/A	N/A	N/A
Niobium-144	N/A	0	N/A				

Note:

1. Peak time is expressed as years elapsed from beginning of simulation (year 2035)
2. The simulation run-time is 500,000 years. For radionuclides and non-radionuclides where the maximum loading was not reached before 500,000 years, the model runtime was extended until the maximum was reached.
3. No mass arrival occurs at the outflow of the bedrock pathway for solutes with a travel time marked as "N/A"

8.2.4 Estimated Public Dose

Exposure was assessed in the ERA (EcoMetrix 2021) for the time of maximum loadings to the river for each COPC during the post-closure phase, irrespective of when that peak occurs. This simplification is conservative as not all peaks occur at the same time point but are effectively assumed to do so for purposes of the assessment.

The total radiation dose to human receptors for the post-closure phase is summarized in Table 8.2.4-1 and in the ERA (EcoMetrix 2021). The total radiation dose to all human receptors during post-closure activities is well below the public dose limit of 1 mSv/a, and dose constraint for the Project of 0.25 mSv/a. Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure to radioactive releases from the Project activities.

Radiological doses expected for Farm A (i.e., critical group) are considered representative of those for Farm F, as radiological doses to the residents (Adult, 10-year old child, and 1-year old Infant) of Farm A would be higher than those of Farm F. Farm A residents (Adult, Child, and Infant), including a 3-month old Infant were evaluated during the time period when maximum effect is predicted to occur. This is consistent with guidance provided in CSA N288.6-12 and CSA N288.1-14, and in the CANDU Owners Group Report COG-06-3090 (2008) for the 3-month infant.

The new On-site Farm receptor has the same characteristics as Farm A; however, residents obtain water for drinking, irrigation, and bathing from the Winnipeg River directly downstream of where the groundwater that seeps from the WRDF enters into the river. This receptor would also encompass the possibility of a farm becoming established beyond the WRDF footprint, but closer to the WL site than Farm A and Farm F. The On-site Farm does not have a well in normal evolution because the well capacity would not meet all the water needs of the residential family, which would therefore be more likely to use the adjacent Winnipeg River. For a well situated in this area, the well capacity is too low and therefore, cannot be used for purposes other than drinking. This is supported by observations during routine groundwater sampling campaigns at boreholes on the WL site (i.e., inability to obtain sufficient groundwater for sampling in 2018 and 2019). This reinforces the conclusion that drinking water wells are unlikely downgradient of the WRDF due to a very low potential well capacity to support the needs of any potential future human receptor.

The Harvester represents the Indigenous traditional land user of the area who may be exposed through harvesting of country food. It is assumed that the harvesters spend part of their time on-site, part near Farm A, and part at an unexposed upstream location.

For radiological COPCs, various human age groups were assessed. For Farm receptors the age groups are: Adult; 10-year old Child; 1-year old Infant – Milk fed; 1-year old Infant – Formula fed; 3-month old Infant – Nursing; and 3-month old Infant – Formula fed. For Harvester receptors the age groups are: Adult, 10-year old Child, and 1-year old Infant. For COPCs an integrated lifetime exposure was calculated for each receptor.

Predicted radiological dose for the post-closure phase is detailed in the ERA (EcoMetrix 2021) Table 5-10 for Harvester receptors, Tables 5-11 and 5-12 for On-site Farm, and Tables 5-13 and 5-14 for Farm A.
The radiological dose during post-closure conservatively uses the maximum loadings to the Winnipeg River for each radionuclide over the modelling timeframe, assuming all maximums occur at the same point in time.
A summary of the total doses is presented in Table 8.2.4-1. The total radiological dose also includes the existing background contribution of caesium-137 in sediment; therefore, Table 8.2.4-2 presents the total dose considering the project contributions only. Background ceasium-137 sediment data (90th percentile) near the WL outfall due to historical discharge and fallout (323 Bq/kg dw) from 2010 to 2018 annual monitoring reports was used as an input to the dose calculations (CNL 2020a, 2019f, 2016d).

While the tables present the results based on the conservative assumption that maximum loadings to the river occur at the same time for all COPCs, Figure 8.2.4-1 shows a more realistic representation of predicted dose rate to human receptors over the post-closure phase from the Normal Evolution scenario. The dose increases steadily with time, generally peaking around 1,000 years after closure due to contribution from carbon-14. The exception is the 3-month-old formula-fed infant where the dose peaks at the beginning of modelling and then again after 100,000 years. This is because the dose from tritium peaks towards the beginning of the modelling timeframe, and the dose from polonium-210 peaks after 100,000 years. The peak dose due to tritium and polonium-210 to the 3-month-old formula-fed infant is lower than the peak dose due to C-14 for the other receptors; therefore, the peak dose is due to C-14 at approximately 1,000 years after closure.

All predicted exposures are well below the dose constraint of 0.25 mSv/a, and subsequently, the public dose limit of 1 mSv/a. Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure to radioactive releases from the Project activities.

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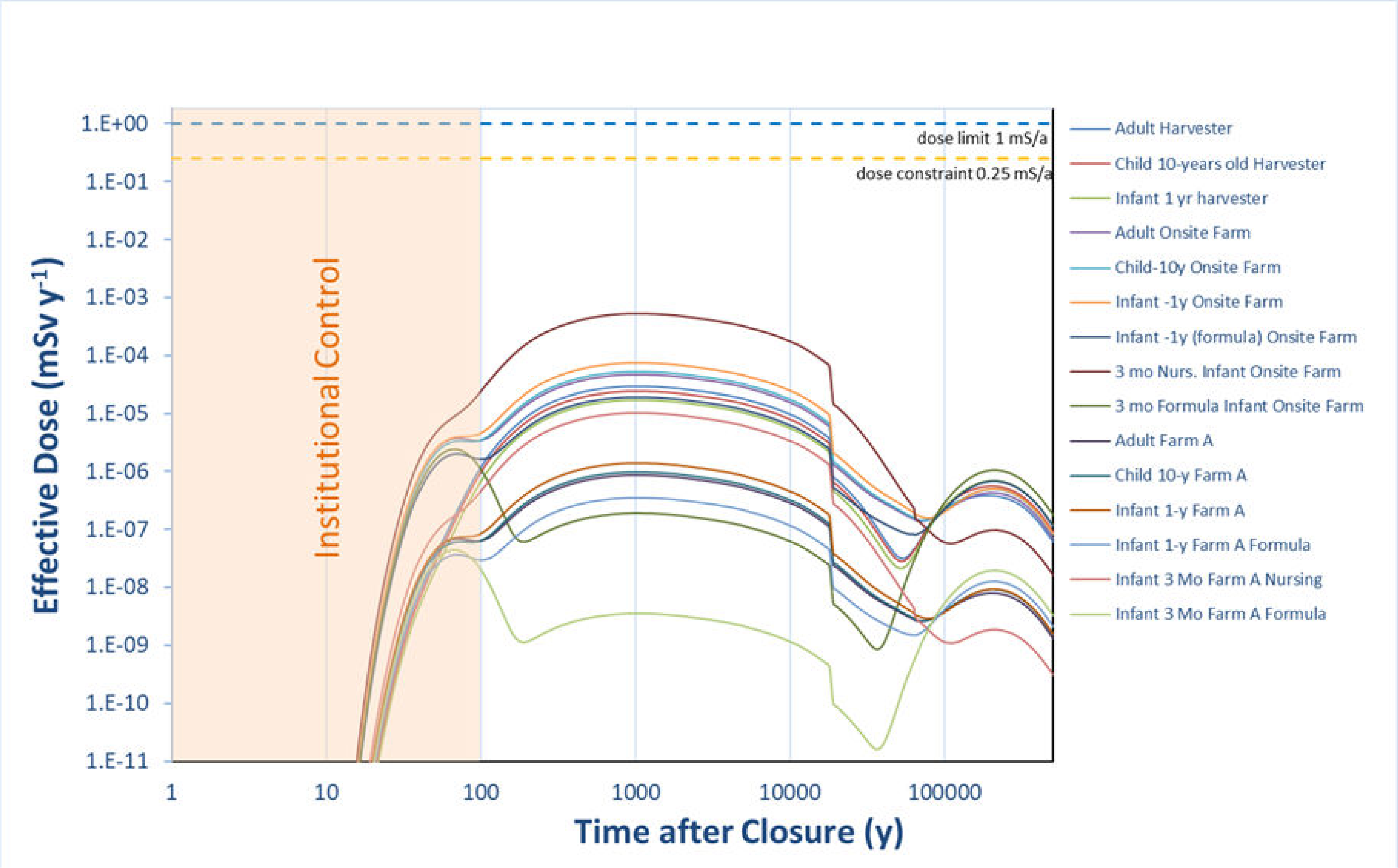

REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	SO/RRD
REVIEWED	KL
APPROVED	MM

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DOSE RATE TO HUMAN RECEPTORS FROM NORMAL EVOLUTION SCENARIO

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.2.4-1**

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Table 8.2.4-1: Summary of Total Dose for Human Health Receptors during the Post-closure Phase

Age Group	Dose (mSv/a)								
	On-site Farm			Farm A			Harvester		
	Dose (including background Cs-137)	Percent of Dose Constraint	Percent of Public Dose Limit	Dose (including background Cs-137)	Percent of Dose Constraint	Percent of Public Dose Limit	Dose (including background Cs-137)	Percent of Dose Constraint	Percent of Public Dose Limit
Adult	3.24E-03	1.30%	0.32%	3.35E-04	0.13%	0.03%	4.75E-05	0.02%	<0.01%
Child	3.25E-03	1.30%	0.33%	3.36E-04	0.13%	0.03%	3.05E-05	0.01%	<0.01%
Infant (cow's milk)	4.23E-03	1.69%	0.42%	4.37E-04	0.17%	0.04%	2.00E-05	0.01%	<0.01%
Infant (formula)	4.18E-03	1.67%	0.42%	4.36E-04	0.17%	0.04%	N/A	N/A	N/A
3-month-old (nursing)	5.48E-04	0.22%	0.05%	1.04E-05	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (formula)	3.65E-06	<0.01%	<0.01%	6.85E-08	<0.01%	<0.01%	N/A	N/A	N/A

Note:
mSv/a = millisievert per year; <= less than.

A dose of 0.0025 mSv/a is equivalent to 1% of the dose constraint and a dose of 0.01 mSv/a is equivalent to 1% of the public dose limit.

Table 8.2.4-2: Summary of Total Dose for Human Health Receptors during the Post-closure Phase – Project Contribution Only

Age Group	Dose (mSv/a)								
	On-site Farm			Farm A			Harvester		
	Dose (Project Only)	Percent of Dose Constraint	Percent of Public Dose Limit	Dose (Project Only)	Percent of Dose Constraint	Percent of Public Dose Limit	Dose (Project Only)	Percent of Dose Constraint	Percent of Public Dose Limit
Adult	5.06E-05	0.02%	0.01%	9.51E-07	<0.01%	<0.01%	3.01E-05	0.01%	<0.01%
Child	5.64E-05	0.02%	0.01%	1.06E-06	<0.01%	<0.01%	2.45E-05	0.01%	<0.01%
Infant (cow's milk)	7.90E-05	0.03%	0.01%	1.48E-06	<0.01%	<0.01%	1.76E-05	0.01%	<0.01%
Infant (formula)	2.14E-05	0.01%	<0.01%	4.03E-07	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (nursing)	5.46E-04	0.22%	0.05%	1.03E-05	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (formula)	3.65E-06	<0.01%	<0.01%	6.85E-08	<0.01%	<0.01%	N/A	N/A	N/A

Note:
mSv/a = millisievert per year; <= less than.

A dose of 0.0025 mSv/a is equivalent to 1% of the dose constraint and a dose of 0.01 mSv/a is equivalent to 1% of the public dose limit.

8.2.5 Assumptions and Uncertainty

The release rates are based on the best available information on the existing inventory to be decommissioned in place. To address the uncertainty associated with the base case and the processes affecting release, an appropriate degree of conservatism was integrated into the groundwater flow model, as well as the solute transport model (Section 5.5.1.2 Solute Transport Modelling). Further, a number of sensitivity analyses were completed (see Section 6.0 Defence-in-Depth for the In Situ Disposal System), which involved the perturbation of some of the key model input parameters to evaluate their relative influence on the base case results. Key model inputs that were examined included those related to:

- the reactor core and bioshield components;
- grout;
- internal walls and building foundation;
- local hydrology; and
- concrete cap and engineered cover.

The sensitivity analysis demonstrates the robustness of the design, as the performance does not rely on one feature but rather a set of redundant barriers and layers of passive protection (for more detail refer to Section 7.3.2 Planned Mitigation). The level of conservatism built into the modelling was illustrated to be appropriate addressing uncertainty regarding release rates. Conservative assumptions made regarding both radionuclides and non-radionuclides included:

- The groundwater flow and solute transport modelling incorporated conservative assumptions resulting in conservative mass loadings:
  - The grout was assumed to be compromised for the purposes of the solute transport modelling, which was represented by the solute mass being distributed throughout the grout block. In this configuration, the distance required for diffusive transport of solute mass is minimized, as only the building foundation separates is the source from the downgradient environment.
  - Sorption was specified for the bedrock groundwater flow path only. It was conservatively assumed that there would be no solute partitioning in the grout, foundation, or backfill materials.
  - No solute partitioning was applied to carbon-14 in the bedrock migration path, therefore, no retardation of carbon-14 has been assumed.
  - No sorption or degradation was applied to xylene.
  - For elements where no solute partitioning coefficient data were available the solute partitioning coefficient was assumed to be zero.
- For the HHRA, to assess a range of quantities for non-radionuclides, the quantity forecasted was used as an average inventory and the upper end of the uncertainty range was used to derive a maximum inventory.
- For the HHRA, maximum predicted concentration was selected for COPC screening for airborne and water borne contaminants, this is considered conservative and is not reflective of typical human exposures.
- For the HHRA, screening benchmarks for water were generally the lower of applicable provincial and federal drinking water standards and guidelines, which is a conservative approach, ensuring that the list of COPCs to be assessed is as comprehensive as possible.

- The receptors selected for the HHRA are representative of the general population and are expected to lead to conservative estimates of health risks. Receptors included an adult, 10-year-old child, 1 year old infant, and 3-month-old infant. The age groups are sufficient to capture the full range of dose effect. Each receptor represents a different life stage to capture range of intakes and dose coefficients over a lifetime.
- Atmospheric releases that occurred during closure that are dispersed and subsequently deposited on the ground or transported to a groundwater well near the WL site are not considered during the post-closure phase. The soil (internal and external) pathways and drinking water pathway (from a farm well) have small contributions to total dose during the closure phase and would have minimal impact on the total dose for the post-closure phase. Therefore, it was decided to assess the phases separately to understand the effects of each phase of the Project separately.

8.3 Non-radiological Assessment for Public under Normal Evolution Scenario

8.3.1 Hazard Identification and Exposure Pathways

Non-radiological hazardous materials will be contained within the WRDF, which will deteriorate over time. The focus in the post-closure phase is releases from the WRDF to groundwater and subsequent migration to surface water at the Winnipeg River, as the pumping from the WR-1 Complex sumps will have ceased. Groundwater elevation after pumping cessation will recover to a new equilibrium and a significant portion of the WRDF will be below the water table. It is anticipated that the WRDF components will gradually deteriorate over time, allowing the release of solutes into the groundwater (see the *WR-1 Groundwater Flow and Solute Transport Modelling* [Golder 2021] for detailed information). Details pertaining to the exposure pathways to receptors in the environment are summarized in Table 8.2.1-1.

8.3.2 Planned Mitigation

The non-radiological hazardous waste material will be further characterized, and this information will be used to informed detailed work plans, and removal of this material will be completed as necessary. Operational controls to protect workers, the public and the environment that will be in effect during the closure phase will also reduce the source term ultimately left to have an effect during the post-closure phase. Planned mitigation for the radiological assessment is also applicable to the non-radiological assessment (see Section 8.2.2 Planned Mitigation).

8.3.3 Non-radiological Releases

Maximum mass loadings for each non-radionuclide expected to be released from the WRDF over the 500,000-year modelling timeframe were derived. A number of non-radionuclides (barium, chromium, copper, mercury, nickel, lead, samarium, ruthenium, and zirconium) did not achieve maximum loadings before 500,000 years. The groundwater flow and solute transport model was run again without the 500,000 years constraint to determine the peak mass loading rates. Peak mass loading rates for radionuclides and non-radionuclides are presented in Section 8.2.3 Radionuclide Releases, in Table 8.2.3-2. Maximum mass loadings for each non-radionuclides were converted to groundwater concentrations using the anticipated flow rate through the WRDF over time. The maximum predicted concentrations in groundwater were then compared to the relevant human health guidelines, Table 8.3.3-1 summarizes the screening of non-radionuclides in groundwater. The COPCs identified with respect to transport of non-radionuclide materials through groundwater were cadmium and lead (non-carcinogenic). Mass loading rates to the Winnipeg River for non-radionuclides are presented in Table 8.3.3-2, and the exposure point concentrations are provided in Table 8.3.3-3.

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Table 8.3.3-1: Screening of Non-radionuclides in Groundwater for Normal Evolution

Non-Radionuclide	Groundwater Concentration (µg/L)	Background Winnipeg River Concentration (µg/L)	CDWS MAC (µg/L)	WQSOG MB (µg/L)	CCME WQG (µg/L)	PWQO Ontario (µg/L)	WQG BC (µg/L)	Toxicity Benchmark (µg/L)	Selected Benchmark
Argon	1.11E-10	—	—	—	—	—	—	—	Noble Gas – not applicable*
Barium	2.62E-07	1.10E+01	1,000	—	—	—	—	0.4	CDWS MAC
Bismuth	1.61E-07	<2.00E-01	—	—	—	—	—	0.25	LC_{50}/100 – Borgmann et al., 2005
Boron	2.84E-03	1.00E+01	5,000	—	1,500	200	1,200	—	CDWS MAC
Cadmium	5.35E+00	1.00E-02	5	0.137	0.08	0.1	0.114	—	CDWS MAC
Chromium	3.49E-02	1.70E+00	50	37.1	1.0 (VI)	1 (VI)	—	—	CDWS MAC
Cobalt	3.05E-01	2.00E-01	—	—	—	0.9	4	—	PWQO Ontario
Copper	7.89E-03	1.40E+01	—	4.3	2	5	2	—	CCME WQG
Gadolinium	1.04E-04	—	—	—	—	—	—	1.5	LC_{50}/100 – Borgmann et al., 2005
HB-40	5.84E+02	0.00E+00	8,800^	—	—	—	—	—	See Note^
Helium	1.73E-01	—	—	—	—	—	—	—	Noble Gas – not applicable*
Lead	7.27E+01	2.60E+00	10	0.99	1	3	4.4		CDWS MAC
Manganese	1.20E-03	1.10E+01	None - naturally occurring	—	—	—	794.2	110	WQG BC
Mercury	6.00E-04	1.00E-02	1	1	0.026	0.2	—	—	CCME WQG
Molybdenum	1.69E-01	2.00E-01	—	—	73	40	1,000	—	CCME WQG
Nickel	6.94E-04	1.78E+00		25.5	25	25		—	CCME WQG
Nitrogen	7.84E-03	—	1,000	—	—	—	3,000	—	CDWS MAC
Palladium	1.51E+00	—	—	—	—	—	—	5.7	LC_{50}/100 – Borgmann et al., 2005
Potassium	4.72E-06	9.07E+02	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Potassium hydroxide (as K)	5.37E-01	—	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Ruthenium	8.87E-08	—	—	—	—	—	—	10	LC_{50}/100 – Borgmann et al., 2005
Samarium	5.26E-11	-	-	-	-	-	-	0.74	LC_{50}/100 – Borgmann et al. 2005
Sulphur (as SO_4)	1.14E-10	—	—	—	—	—	218,000	—	WQG BC
Xenon	1.68E-08	—	—	—	—	—	—	—	Noble Gas – not applicable
Xylene	6.96E+01	—	90	—	—	2/40/30 (m/o/p)	30	—	CDWS MAC
Zirconium	2.18E-09	-	-	-	-	4	-	-	PWQO Ontario

Note:
µg/L = microgram per litre.
*Noble gases were assumed to volatilize rapidly.
^Derived drinking water limit based on a minimal effect level in mice of 250 mg/kg-day (Weeks 1974), divided by 1000, times 70 kg body weight, over 2 L/day of drinking water.
CDWS = Canadian Drinking Water Standard (Health Canada 2017)
WQSOG = Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011)
CCME WQG = Canadian Council of Ministers of the Environment Water Quality Guideline (CCME 1999)
PWQO = Ontario Provincial Water Quality Objective (MOEE 1994)
WQG BC = Water Quality Guideline British Columbia (BC MOE 2017).

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Table 8.3.3-2: Maximum Mass Loading Rates to the Winnipeg River for Non-radionuclides in Groundwater

Non-Radionuclide	Peak Mass Loading Rate (g/y)	Time of Maximum (Year)
Argon	6.88E-12	224
Barium	1.82E-08	10,740,000 (a)
Bismuth	1.12E-08	78,800
Boron	1.87E-04	4,235
Cadmium	3.71E-01	203,000
Chromium	2.42E-03	19,998,000 (a)
Cobalt	2.12E-02	93,800
Copper	5.47E-04	876,000 (a)
Gadolinium	6.48E-06	213
HB-40	4.05E+01	10,200
Helium	1.07E-02	145
Lead	5.05E+00	5,596,000 (a)
Manganese	8.33E-05	354,500
Mercury	4.16E-05	6,546,000 (a)
Molybdenum	1.17E-02	17,900
Nickel	4.81E-05	840,000 (a)
Nitrogen	5.44E-04	17,900
Palladium	1.05E-01	121,000
Potassium	3.27E-07	13,000
Potassium hydroxide	4.76E-02	142
Ruthenium	6.15E-09	374,100
Samarium	3.65E-12	20,000,000 (a)
Sulphur	2.37E-12	193
Xenon	1.39E-09	195
Xylene	4.32E+00	202
Zirconium	1.51E-10	20,000,000 (a)

a) The model was run for 500,000 years. For these non-radionuclides, the maximum loading was not reached before 500,000 years; therefore, the model runtime was extended until the maximum was reached. The peak groundwater concentrations for these non-radionuclides were all still below the screening criteria, except for lead. With respect to lead, the project contribution from lead is very small (i.e., orders of magnitude less than background levels in the river); therefore, using the higher peak would still result in a river concentration effectively at or below the background concentration. Therefore, there is no additional risk to what has been presented in this assessment based on greater than 500,000 year runs.

g/y = grams per year.

Table 8.3.3-3: Exposure Concentrations for Non-radionuclide COPCs for Human Receptors during Post-Closure

Non-radionuclide	Groundwater Concentration (µg/L)	Background Concentration (µg/L)	River Concentration at Groundwater Seep at River Bottom (µg/L)	River Concentration at Groundwater Seep – 50 m Downstream (µg/L)	River Concentration at Farm A Intake (µg/L)
Cadmium	5.35E+00	1.00E-02	1.00E-02	1.00E-02	1.00E-02
Lead	7.273E+01	2.60E+00	2.60E+00	2.60E+00	2.60E+00

µg/L = microgram per litre.

8.3.4 Estimated Public Exposure

As identified in the ERA (EcoMetrix, 2021), the relevant COPCs for the HHRA in the post-closure phase are cadmium and lead, and receptors are assumed to be exposed to surface water that has a loading contribution from the Project and a background contribution. Background water quality in the Winnipeg River is monitored at the WL intake for a variety of metals (CNL 2016d). The WL intake is located in the river at the WL site, upstream of any potential site-associated influence and represents ambient water quality for the Winnipeg River exclusive of any potential WL site related influence.

For non-radiological COPCs, various human age groups were assessed. For Farm and Harvester receptors, the age groups are Adult and Toddler. For COPCs, an integrated lifetime exposure was calculated for each receptor.

Exposures are calculated based on total concentration (background plus contribution from the Project). Exposures are provided in Table 8.3.4-1, Table 8.3.4-2 and Table 8.3.4-3 for the Harvester, Farm A and On-site Farm receptors, respectively. The HQs for incremental pathways were compared to a target value of 0.2 per medium (e.g., water, soil, food, air), HQs are provided in Table 8.3.4-4, Table 8.3.4-5, and Table 8.3.4-6 for the Harvester, Farm A and On-site Farm receptors, respectively. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 8.3.4-1: Doses to Harvester Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	Unit	Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of Weekay	Total
Total River Contribution (Dose by Pathway)							
Adult	Cadmium	mg/kg bw/d	1.40E-07	6.09E-07	1.45E-05	2.94E-06	1.82E-05
	Lead	mg/kg bw/d	6.48E-06	1.13E-05	2.63E-05	2.02E-04	2.46E-04
Toddler	Cadmium	mg/kg bw/d	1.46E-07	3.93E-07	9.35E-06	4.14E-06	1.40E-05
	Lead	mg/kg bw/d	6.79E-06	7.28E-06	1.70E-05	1.08E-04	1.39E-04
WRDF Project Contribution (Dose by Pathway)							
Adult	Cadmium	mg/kg bw/d	2.95E-11	1.29E-10	3.06E-09	6.22E-10	3.84E-09
	Lead	mg/kg bw/d	7.16E-11	1.25E-10	2.91E-10	8.45E-10	1.33E-09
Toddler	Cadmium	mg/kg bw/d	3.09E-11	8.30E-11	1.98E-09	8.75E-10	2.97E-09
	Lead	mg/kg bw/d	7.51E-11	8.05E-11	1.88E-10	1.19E-09	1.53E-09

mg/kg bw/d = milligram per kilogram of body weight per day.

Table 8.3.4-2: Doses to Farm A Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	Unit	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	7.72E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	7.74E-05
	Lead	mg/kg bw/d	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	2.01E-02
Toddler	Cadmium	mg/kg bw/d	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.72E-09	5.01E-09	2.07E-09	7.50E-09	0.00E+00	1.33E-04
	Lead	mg/kg bw/d	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	3.44E-02
WRDF Project Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	6.03E-10	9.55E-15	6.77E-15	6.02E-15	1.84E-14	8.71E-13	1.96E-14	5.76E-14	5.15E-14	3.42E-14	7.07E-15	3.44E-16	6.04E-10
	Lead	mg/kg bw/d	8.19E-09	3.07E-15	1.75E-15	1.93E-15	1.75E-15	2.12E-12	2.84E-14	4.80E-13	1.62E-14	2.38E-13	8.74E-14	4.80E-16	8.20E-09
Toddler	Cadmium	mg/kg bw/d	1.03E-09	1.64E-13	1.06E-15	1.29E-14	1.94E-14	9.14E-13	8.10E-15	5.25E-14	3.91E-14	1.62E-14	5.86E-14	0.00E+00	1.03E-09
	Lead	mg/kg bw/d	1.40E-08	5.26E-14	3.41E-15	4.14E-15	1.84E-15	2.22E-12	1.17E-14	4.38E-13	1.23E-14	1.13E-13	7.24E-13	0.00E+00	1.40E-08

mg/kg bw/d = milligram per kilogram of body weight per day.

Table 8.3.4-3: Doses to On-site Farm Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	Unit	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	7.73E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	7.74E-05
	Lead	mg/kg bw/d	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	2.01E-02
Toddler	Cadmium	mg/kg bw/d	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.73E-09	5.01E-09	2.08E-09	7.51E-09	0.00E+00	1.33E-04
	Lead	mg/kg bw/d	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	3.44E-02
WRDF Project Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	3.20E-08	5.08E-13	2.90E-14	3.20E-13	9.79E-13	4.63E-11	1.04E-12	3.06E-12	2.74E-12	1.82E-12	3.76E-13	1.83E-14	3.21E-08
	Lead	mg/kg bw/d	4.36E-07	1.63E-13	9.30E-14	1.03E-13	9.32E-14	1.13E-10	1.51E-12	2.55E-11	8.59E-13	1.26E-11	4.65E-12	2.55E-14	4.36E-07
Toddler	Cadmium	mg/kg bw/d	5.49E-08	8.71E-12	5.65E-14	6.87E-13	1.03E-12	4.86E-11	4.31E-13	2.79E-12	2.08E-12	8.61E-13	3.11E-12	0.00E+00	5.50E-08
	Lead	mg/kg bw/d	7.47E-07	2.80E-12	1.81E-13	2.21E-13	9.81E-14	1.18E-10	6.23E-13	2.33E-11	6.53E-13	5.98E-12	3.85E-11	0.00E+00	7.47E-07

mg/kg bw/d = milligram per kilogram of body weight per day.

Table 8.3.4-4: Hazard Quotients for Harvester Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	Total River Contribution (Including WRDF Project Contribution)					WRDF Project Contribution				
		Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of Weekay	Total	Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of Weekay	Total
Adult	Cadmium	1.40E-04	6.09E-04	1.45E-02	2.94E-03	1.82E-02	2.95E-08	1.29E-07	3.06E-06	6.22E-07	3.84E-06
	Lead	3.50E-03	6.10E-03	1.42E-02	1.09E-01	1.33E-01	3.87E-08	6.74E-08	1.57E-07	4.57E-07	7.20E-07
Toddler	Cadmium	1.46E-04	3.93E-04	9.35E-03	4.14E-03	1.40E-02	3.09E-08	8.30E-08	1.98E-06	8.75E-07	2.97E-06
	Lead	3.67E-03	3.93E-03	9.17E-03	5.82E-02	7.49E-02	4.06E-08	4.35E-08	1.01E-07	6.43E-07	8.28E-07

Note:

Bold values indicate HQ greater than 0.2.

Table 8.3.4-5: Hazard Quotients for Farm A Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution (Including WRDF Project Contribution)														
Adult	Cadmium	7.72E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	7.74E-02
	Lead	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	**1.09E+01**
Toddler	Cadmium	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.72E-06	5.01E-06	2.07E-06	7.50E-06	0.00E+00	1.33E-01
	Lead	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	**1.86E+01**
WRDF Project Contribution														
Adult	Cadmium	6.03E-07	9.55E-12	6.77E-12	6.02E-12	1.84E-11	8.71E-10	1.96E-11	5.76E-11	5.15E-11	3.42E-11	7.07E-12	3.44E-13	6.04E-07
	Lead	4.43E-06	1.66E-12	9.45E-13	1.05E-12	9.46E-13	1.14E-09	1.53E-11	2.60E-10	8.73E-12	1.28E-10	4.73E-11	2.59E-13	4.43E-06
Toddler	Cadmium	1.03E-06	1.64E-10	1.06E-12	1.29E-11	1.94E-11	9.14E-10	8.10E-12	5.25E-11	3.91E-11	1.62E-11	5.86E-11	0.00E+00	1.03E-06
	Lead	7.59E-06	2.84E-11	1.84E-12	2.24E-12	9.97E-13	1.20E-09	6.33E-12	2.37E-10	6.64E-12	6.08E-11	3.92E-10	0.00E+00	7.59E-06

Note:

Bold values indicate HQ greater than 0.2.

Table 8.3.4-6: Hazard Quotients for On-site Farm Receptors during Post-closure for Normal Evolution

Human Type	Non-radionuclide	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution (Including WRDF Project Contribution)														
Adult	Cadmium	7.73E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	7.74E-02
	Lead	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	**1.09E+01**
Toddler	Cadmium	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.73E-06	5.01E-06	2.08E-06	7.51E-06	0.00E+00	1.33E-01
	Lead	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	**1.86E+01**
WRDF Project Contribution														
Adult	Cadmium	3.20E-05	5.08E-10	2.90E-11	3.20E-10	9.79E-10	4.63E-08	1.04E-09	3.06E-09	2.74E-09	1.82E-09	3.76E-10	1.83E-11	3.21E-05
	Lead	2.36E-04	8.82E-11	5.03E-11	5.56E-11	5.04E-11	6.08E-08	8.16E-10	1.38E-08	4.64E-10	6.83E-09	2.51E-09	1.38E-11	2.36E-04
Toddler	Cadmium	5.49E-05	8.71E-09	5.65E-11	6.87E-10	1.03E-09	4.86E-08	4.31E-10	2.79E-09	2.08E-09	8.61E-10	3.11E-09	0.00E+00	5.50E-05
	Lead	4.04E-04	1.51E-09	9.81E-11	1.19E-10	5.30E-11	6.38E-08	3.37E-10	1.26E-08	3.53E-10	3.23E-09	2.08E-08	0.00E+00	4.04E-04

Note:

Bold values indicate HQ greater than 0.2.

The HQs for the harvester are below the acceptable risk level of 0.2 for cadmium and lead for all pathways for the toddler and adult. The HQs for the On-site Farm and Farm A are below the acceptable risk level of 0.2 for cadmium and lead for all pathways, with the exception of lead from drinking water from the Winnipeg River.

The HQs for all receptors are based on background plus project exposure. If only the project contribution is considered, the HQs to the toddler and adult for the On-site Farm and Farm A are well below the acceptable risk level of 0.2. The project contribution to the lead HQ for drinking water is 0.002%. This indicates that the project contribution to the total HQ is negligible and the exceedance is from existing background concentrations of lead in the Winnipeg River.

8.3.5 Assumptions and Uncertainties

Assumptions and uncertainties regarding radiological inventories within the WR-1 Building and radiological release rates are discussed in Section 7.3.4 Assumptions and Uncertainty and overview of modelling assumptions and uncertainty is provided in Section 8.2.5 Assumptions and Uncertainty.

8.4 Radiological Assessment for Non-Human Biota

8.4.1 Hazard Identification and Exposure Pathways

During post-closure phase, the relevant radiological release will be via groundwater transport from the WRDF to surface water at the Winnipeg River. Sessile organisms, such as benthic invertebrates may be more directly exposed to groundwater, if located at the point of discharge; therefore, it has been conservatively assumed that benthic invertebrates at the site are exposed to direct groundwater without any dilution.

In addition to routes of contaminant dispersion from the source to the receptor location, the assessment also includes routes of contaminant transport thought the food change or other media to the receptor organism. Airborne COPCs are not relevant during the post-closure assessment, as a result of the end-state established. Details pertaining to the exposure pathways to receptors in the environment are summarized in Table 8.4.1-1**,** for more information refer to Sections 7.1.4 and 7.1.5 of the ERA (EcoMetrix 2021).

Table 8.4.1-1: Complete Exposure Pathways for Receptors Exposed to Radiological and Non-Radiological COPCs during Post-closure Phase

Valued Component Category	Valued Component	Exposure Pathway	Environmental Media
Bottom Feeding Fish	Lake Sturgeon	Direct Contact	■ In Water ■ On Sediment
	Carmine Shiner	Direct Contact	■ In Water ■ On Sediment
Pelagic Fish	Walleye	Direct Contact	■ In Water
Aquatic Plants	Aquatic Plant	Direct Contact	■ In Water
Aquatic Invertebrates	Benthic Invertebrate	Direct Contact	■ In Water ■ In Sediment
Riparian Birds	Horned Grebe	Direct Contact	■ On Sediment
		Ingestion	■ Water ■ Sediment ■ Fish (forage) ■ Benthic Invertebrates

Table 8.4.1-1: Complete Exposure Pathways for Receptors Exposed to Radiological and Non-Radiological COPCs during Post-closure Phase

Valued Component Category	Valued Component	Exposure Pathway	Environmental Media
Riparian Birds (cont'd)	Trumpeter Swan	Direct Contact	On Sediment
		Ingestion	Water Sediment Aquatic Plants
	Mallard	Direct Contact	On Sediment
		Ingestion	Water Sediment Benthic Invertebrates Aquatic Plants
Riparian Mammals	Mink	Direct Contact	On Sediment
		Ingestion	Water Sediment Benthic Invertebrates Fish (forage)
Terrestrial Birds	Barn Swallow	Direct Contact	On Sediment
		Ingestion	Water Sediment Benthic Invertebrates
Terrestrial Mammals	Little Brown Myotis	Direct Contact	None
		Ingestion	Water Benthic Invertebrates
	Moose	Direct Contact	On Sediment
		Ingestion	Water Sediment Aquatic Plants Grasses

8.4.2 Planned Mitigation

Following closure, the post-closure phase will commence and include a period of a minimum of 100 years of institutional control during which both active and passive controls will be implemented. During institutional control, groundwater monitoring and groundwater quality management will continue to demonstrate compliance with the safety case assumptions and illustrate that the site has reach a safe and stable state. Planned mitigation for the radiological assessment for the HHRA is also applicable to the radiological assessment for non-human biota (see Section 8.2.2 Planned Mitigation).

8.4.3 Whiteshell Reactor Disposal Facility Radiological Releases

The radiological inventory is discussed in Section 7.1.1.1 Radionuclides, and the derived mass loadings are presented in Table 8.2.3-1. These radionuclides have historically been found in WL's waterborne effluent or are reasonably expected to be found in the WRDF and have the potential to migrate from groundwater to surface water during the post-closure phase.

8.4.4 Estimated Non-human Biota Dose

The Radiological doses derived for representative ecological receptors are presented for the post-closure phase for maximum release scenarios representing the peak release rate. Radiation exposures are detailed in Table 7-7 of the ERA (EcoMetrix 2021) and summarized in Table 8.4.4-1. The total doses are compared to dose benchmarks of 9.6 mGy/d for aquatic biota and 2.4 mGy/d for terrestrial and riparian biota. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

As a conservative assessment, benthic invertebrates have also been modelled assuming that they reside in undiluted groundwater. In reality, a concentration gradient would exist across the sediment-water interface. Radiation exposures in this scenario are shown in Table 8.4.4-2.

Table 8.4.4-1: Summary of Total Dose during Post-closure Phase for Selected Ecological Receptors

Ecological Receptor	Total Dose - including background Cs-137 (mGy/d)	Total Dose – project only (mGy/d)	Percent of Protective Benchmark (Total and Project Only)
Barn Swallow	8.13E-06	8.13E-06	<0.01%
Little Brown Bat	6.58E-06	6.58E-06	<0.01%
Carmine Shiner	1.14E-04	4.10E-06	<0.01%
Lake Sturgeon	1.14E-04	4.10E-06	<0.01%
Walleye	4.10E-06	4.10E-06	<0.01%
Freshwater plant	4.26E-06	4.26E-06	<0.01%
Benthic Invertebrate	5.77E-04	4.00E-06	<0.01%
Horned Grebe	1.55E-04	8.05E-06	<0.01%
Trumpeter Swan	1.53E-04	8.07E-06	<0.01%
Wild Waterfowl	1.52E-04	8.05E-06	<0.01%
Mink	1.54E-04	6.59E-06	<0.01%
Moose	1.01E-05	6.95E-06	<0.01%

Note:

mGy/d = milligray per day; <= less than.

Table 8.4.4-2: Summary of Total Dose for Post-Closure Benthic Invertebrate Exposed to Groundwater and Comparison to Dose Benchmarks

Ecological Receptor	Dose (mGy/d)	Percent of Protective Benchmark
Benthic Invertebrate	5.71E+00	59.48%

Note:

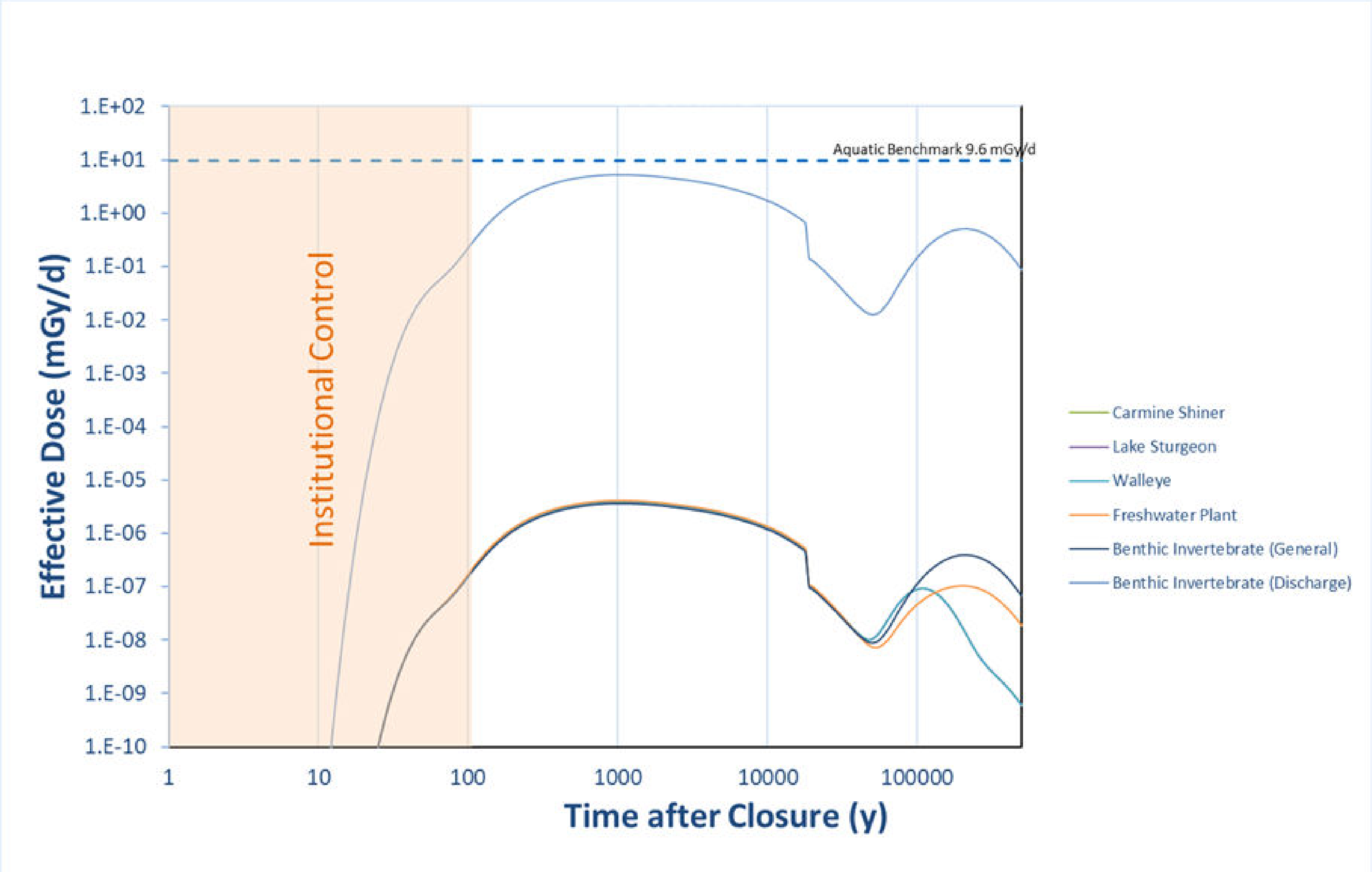
mGy/d = milligray per day.

There are no exceedances of the 9.6 mGy/d radiation benchmark for the aquatic biota in the Winnipeg River. Similarly, there are no exceedances of the 2.4 mGy/d radiation benchmark for terrestrial and riparian biota on or near the WL site. An evaluation to assessment criteria illustrates that all predicted doses are less than 0.01% of the radiation dose benchmark, with the exception of benthic invertebrates when they are exposed to maximum groundwater concentrations assuming they reside in the undiluted groundwater which is a conservative scenario.

For ecological receptors exposed to sediment (Lake Sturgeon, Carmine Shiner, Benthic Invertebrates, Horned Grebe, Trumpeter Swan, Wild Waterfowl and Mink) the dose is primarily due to existing caesium-137 in the river sediment, with carbon-14 from the Project being the next largest contributor to dose. For the remaining ecological receptors, the dominant pathway of exposure is carbon-14 from the Project through the food chain.

While the tables present the results based on the conservative assumption that maximum loadings to the river occur at the same time for all COPCs, Figure 8.4.4-1 and Figure 8.4.4-2 show a more realistic representation of predicted dose rate to ecological receptors over the post-closure phase from the Normal Evolution scenario.

A dose of 0.024 mGy/d and 0.096 mGy/d is equivalent to 1% of the terrestrial and riparian, and aquatic ecological radiation benchmark, respectively. All predicted doses are well below this level. Therefore, it is unlikely that there would be significant adverse effects on terrestrial, riparian or aquatic populations or communities as a result of radionuclide releases from the WRDF. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.



REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES

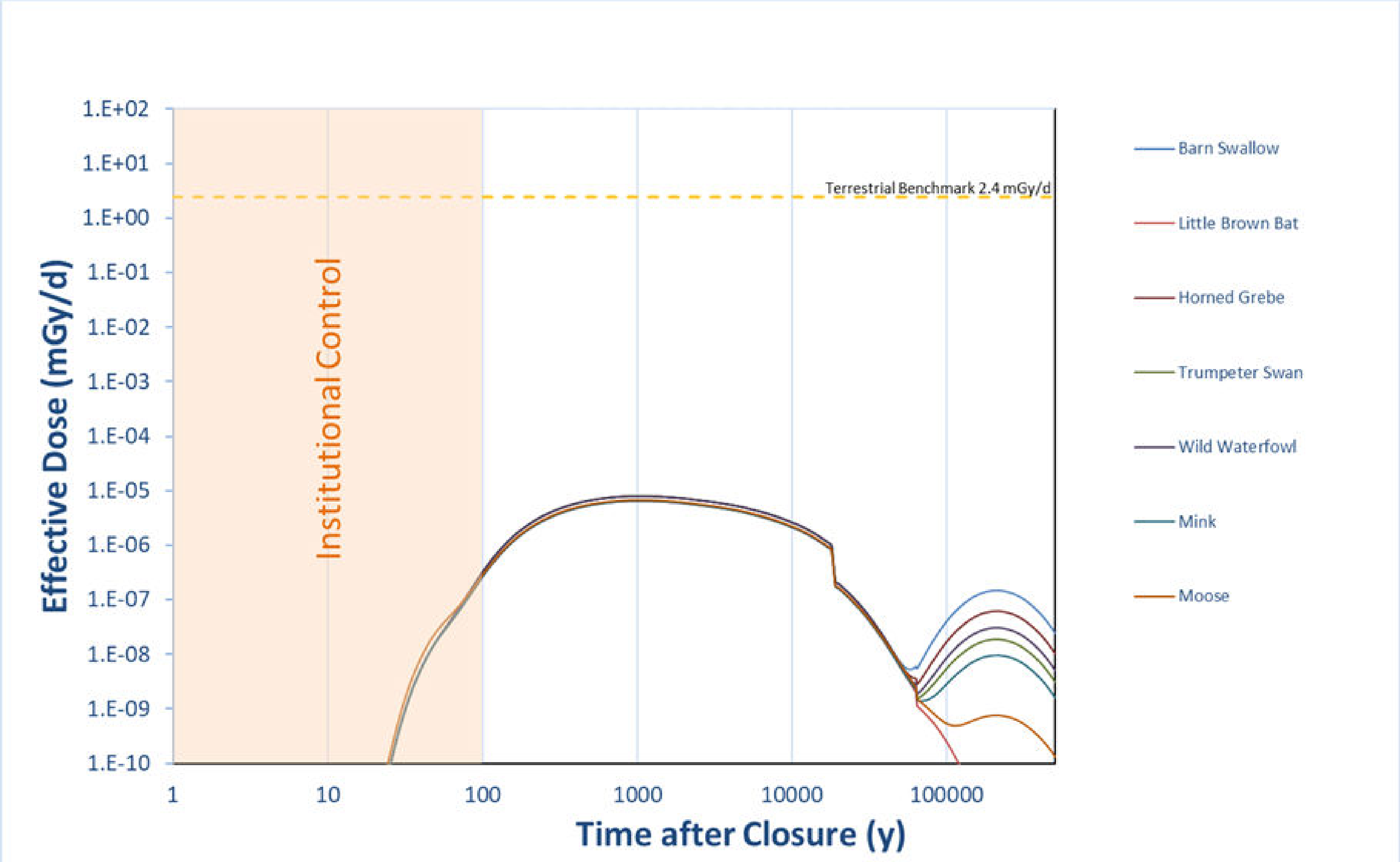
CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DOSE RATE TO ECOLOGICAL AQUATIC RECEPTORS FROM NORMAL EVOLUTION SCENARIO

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.4.4-1**



REFERENCE(S)

CLIENT	CANADIAN NUCLEAR LABORATORIES
CONSULTANT	GOLDER MEMBER OF WSP
MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	RRD
REVIEWED	KL
APPROVED	MM
PROJECT	DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT
TITLE	**DOSE RATE TO ECOLOGICAL RIPARIAN AND TERRESTRIAL RECEPTORS FROM NORMAL EVOLUTION SCENARIO**
PROJECT NO.	20145046
CONTROL	0001
REV.	4
FIGURE	**8.4.4-2**

8.4.5 Assumptions and Uncertainty

Assumptions and uncertainties regarding radiological inventories within the WR-1 Building and radiological release rates are discussed in Section 7.1.5 Assumptions and Uncertainty and overview of modelling assumptions and uncertainty is provided in Section 8.2.5 Assumptions and Uncertainty.

8.5 Non-radiological Assessment for Non-human Biota

8.5.1 Hazard Identification and Exposure Pathway

During the post-closure phase, the relevant radiological release will be via groundwater transport from the WRDF to surface water at the Winnipeg River. In addition to routes of contaminant dispersion from the source to the receptor location, the assessment also includes routes of contaminant transport through the food change or other media to the receptor organism. Airborne, COPCs are not relevant during the post-closure phase. Details pertaining to the exposure pathways to receptors in the environment are summarized in Table 7.3.1-1, for more information refer to Sections 7.1.4 and 7.1.5 of the ERA (EcoMetrix 2021).

8.5.2 Planned Mitigation

Following closure, the post-closure phase will commence and include a period of a minimum of 100 years of institutional control during which both active and passive controls will be implemented. During institutional control, groundwater monitoring and groundwater quality management will continue to demonstrate compliance with the safety case assumptions and illustrate that the site has reach a safe and stable state. Planned mitigation for the radiological assessment for the HHRA is also applicable to the radiological assessment for non-human biota (see Section 8.2.2 Planned Mitigation).

8.5.3 Non-radionuclide Releases

Modelling was extended such that it captured the maximum mass loadings for each non-radionuclide expected to be released from the WRDF. Peak mass loading rates for radiological and non-radionuclides are presented in Section 8.2.3 Radiological Releases in Table 8.2.3-2. Maximum mass loading rates for each non-radionuclide were converted to surface water concentrations using the anticipated flow rate through the WRDF over time. The maximum predicted concentrations in surface water were then compared to the relevant ecological health guidelines, Table 8.5.3-1 summarizes the screening of non-radionuclides in surface water. The more conservative and relevant of available federal and provincial guidelines and objectives were used. If there was no such guideline or objective, screening criteria were obtained from conservative toxicity benchmarks (no effect levels) in the literature. The COPCs identified with respect to transport of non-radiological materials through groundwater were cadmium, lead, HB-40 and xylene.

Table 8.5.3-1: Screening of Non-radionuclides in Surface Water for Normal Evolution

Non-radionuclide	Groundwater Concentration (µg/L)	Background Winnipeg River Concentration (µg/L)	WQSOG Manitoba (µg/L)	CCME WQG (µg/L)	PWQO Ontario (µg/L)	WQG BC (µg/L)	Toxicity Benchmark (µg/L)	Selected Benchmark
Argon	1.11E-10	—	—	—	—	—	—	Noble Gas – not applicable*
Barium	2.62E-07	1.10E+01	—	—	—	—	0.4	LCV/10 – Suter and Tsao 1996
Bismuth	1.61E-07	<2.00E-01	—	—	—	—	0.25	LC_{50}/100 – Borgmann et al. 2005
Boron	2.84E-03	1.00E+01	—	1500	200	1200	—	CCME WQG
Cadmium	5.35E+00	1.00E-02	0.137	0.08	0.1	0.114	—	CCME WQG
Chromium	3.49E-02	1.70E+00	37.1	1.0 (VI)	1 (VI)	—	—	CCME WQG
Cobalt	3.05E-01	2.00E-01	—	—	0.9	4	—	PWQO Ontario
Copper	7.89E-03	1.40E+01	4.3	2	5	2	—	CCME WQG
Gadolinium	1.04E-04	—	—	—	—	—	1.5	LC_{50}/100 – Borgmann et al. 2005
HB-40	5.84E+02	0.00E+00	—	—	—	—	2	IC_{25}/10 – EcoMetrix 2017
Helium	1.73E-01	—	—	—	—	—	—	Noble Gas – not applicable*
Lead	7.27E+01	2.60E+00	0.99	1	3	4.4	—	WQSOG Manitoba
Manganese	1.20E-03	1.10E+01	—	—	—	794.2	110	WQG BC
Mercury	6.00E-04	1.00E-02	1	0.026	0.2	—	—	CCME WQG
Molybdenum	1.69E-01	2.00E-01	—	73	40	1000	—	CCME WQG
Nickel	6.94E-04	1.78E+00	25.5	25	25		—	CCME WQG
Nitrogen	7.84E-03	—	—	—	—	3000		WQG BC
Palladium	1.51E+00	—	—	—	—	—	5.7	LC_{50}/100 – Borgmann et al. 2005
Potassium	4.72E-06	9.07E+02	—	—	—	—	5300	LCV/10 – Suter and Tsao 1996
Potassium hydroxide (as K)	5.37E-01	—	—	—	—	—	5300	LCV/10 – Suter and Tsao 1996
Ruthenium	8.87E-08	—	—	—	—	—	10	LC_{50}/100 – Borgmann et al. 2005
Samarium	5.26E-11	-	-	-	-	-	0.74	LC_{50}/100 – Borgmann et al. 2005

Table 8.5.3-1: Screening of Non-radionuclides in Surface Water for Normal Evolution

Non-radionuclide	Groundwater Concentration (µg/L)	Background Winnipeg River Concentration (µg/L)	WQSOG Manitoba (µg/L)	CCME WQG (µg/L)	PWQO Ontario (µg/L)	WQG BC (µg/L)	Toxicity Benchmark (µg/L)	Selected Benchmark
Sulphur (as SO_4)	1.14E-10	—	—	—	—	218000	—	WQG BC
Xenon	2.24E-08	—	—	—	—	—	—	Noble Gas – not applicable*
Xylene	6.96E+01	—	—	—	2/40/30 (m/o/p)	30	—	WQG BC (in preference over interim PWQO)
Zirconium	2.18E-09	-	-	-	4	-	-	PWQO Ontario

Note:

µg/L = microgram per Litre.

* Noble gases were assumed to volatilize rapidly.

** Derived drinking water limit based on a minimal effect level in mice of 250 mg/kg-day (Weeks 1974), divided by 1000, times 70 kg body weight, over 2 L/day of drinking water.

WQSOG = Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011).

CCME WQG = Canadian Council of Ministers of the Environment Water Quality Guideline (CCME 1999).

PWQO = Ontario Provincial Water Quality Objective (MOEE 1994).

WQG BC = Water Quality Guideline British Columbia (BC MOE 2017).

8.5.4 Estimated Non-human Biota Exposure

The relevant COPCs for the Ecological Risk Assessment (EcoRA) in the post-closure phase are cadmium, lead, HB-40 and xylene and receptors are assumed to be exposed to surface water that has a loading contribution from the Project and a background contribution. Exposures are calculated based on total concentration (background plus contribution from the Project). Exposures for selected aquatic ecological receptors were assumed to equal the river concentration located 50 m downstream from the groundwater seep (Table 8.5.4-1) with the exception of benthic invertebrates where the exposure was assumed to the groundwater concentration. Exposure doses derived for terrestrial ecological health receptors, are presented in Table 8.5.4-2. Exposure doses for fish, aquatic plants, and benthic invertebrates are not applicable as the HQ is calculated for those receptors based on a water concentration and not a dose. The HQs for all pathways were compared to a target value of 1. HQs are provided in Table 8.5.4-3 and Table 8.5.4-4 for aquatic receptors and in Table 8.5.4-5 for terrestrial receptors. The HQs are presented for benthic invertebrates exposed to groundwater after it is mixed with the river, and also for benthic invertebrates exposed directly to groundwater prior to mixing with the river. Refer to Section 9.0 Results Summary for a summation of expected exposures and evaluation to assessment criteria.

Table 8.5.4-1: Exposure Point Concentrations for Non-Radionuclide COPCs for Ecological Receptors during Post-Closure

Non-radionuclide	Background Concentration (µg/L)	Groundwater Seep Concentration (µg/L)	Project Contribution to River Concentrations at Groundwater Seep at River Bottom (µg/L)	River Concentration at Groundwater Seep at River Bottom (µg/L)	Project Contribution to River Concentrations at Groundwater Seep - 50 m Downstream (µg/L)	River Concentration at Groundwater Seep - 50 m Downstream (µg/L)
Cadmium	1.00E-02	5.35E+00	3.59E-05	1.00E-02	4.15E-06	1.00E-02
HB-40	0.00E+00	5.84E+02	3.92E-03	3.92E-03	4.53E-04	4.53E-04
Lead	2.60E+00	7.27E+01	4.89E-04	2.60E+00	5.64E-05	2.60E+00
Xylene	NV	6.96E+01	4.18E-04	4.18E-04	4.83E-05	4.83E-05

NV = no value available to calculate.

Table 8.5.4-2: Exposure Doses to Terrestrial Ecological Receptors during Post-closure for Normal Evolution

Non-radionuclide	Unit	Barn Swallow	Horned Grebe	Trumpeter Swan	Wild Waterfowl	Little Brown Myotis	Mink	Moose
Cadmium	mg/kg bw/d	1.07E-03	3.71E-04	1.97E-02	2.44E-02	4.97E-04	1.92E-04	1.82E-02
HB-40	mg/kg bw/d	1.07E+00	4.43E-01	2.72E-02	1.21E-01	5.09E-01	2.61E-01	2.50E-02
Lead	mg/kg bw/d	7.76E-02	2.42E-02	5.13E-01	6.38E-01	2.86E-02	1.14E-02	4.72E-01
Xylene	mg/kg bw/d	1.94E-04	6.79E-05	1.15E-05	3.00E-05	9.19E-05	3.56E-05	1.06E-05

mg/kg bw/d = milligrams per kilogram of body weight per day.

Table 8.5.4-3: Hazard Quotients for Aquatic Receptors during Post-closure for Normal Evolution

Non-radionuclide	Benthic Invertebrates	Fish	Aquatic Plants
	unitless	unitless	unitless
Cadmium	6.69E-02	5.90E-03	5.02E-03
HB-40	1.96E-04	4.56E-06	8.35E-06
Lead	2.12E-01	1.38E-01	5.20E-03
Xylene	4.18E-06	1.56E-07	1.07E-07

Note:

NV = no value available to calculate.

Table 8.5.4-4: Hazard Quotients for Benthic Invertebrates Exposed to Groundwater during Post-closure for Normal Evolution

Non-radionuclide	Benthic Invertebrates (Groundwater)
	unitless
Cadmium	**3.57E+01**
HB-40	**2.92E+01**
Lead	**5.93E+00**
Xylene	6.96E-01

Note:

Bolded and shaded cells indicate exceedance of greater than 1% or more (value of 1.01 represents an exceedance of 1%).

NV = no value available to calculate.

Table 8.5.4-5: Hazard Quotients for Bird and Mammal Receptors during Post-closure for Normal Evolution

Non-radionuclide	Barn Swallow	Horned Grebe	Trumpeter Swan	Wild Waterfowl	Little Brown Myotis	Mink	Moose
	unitless	unitless	unitless	unitless	unitless	unitless	Unitless
Cadmium	5.36E-05	1.85E-05	9.87E-04	1.22E-03	4.97E-05	1.92E-05	1.82E-03
HB-40	NV	NV	NV	NV	2.04E-03	1.04E-03	1.00E-04
Lead	6.87E-03	2.15E-03	4.54E-02	5.65E-02	3.58E-04	1.42E-04	5.90E-03
Xylene	3.46E-06	1.21E-06	2.06E-07	5.37E-07	2.57E-07	9.96E-08	2.98E-08

Note:

NV = no value available to calculate.

There are no exceedances of the HQs identified for exposure of the ecological receptors to cadmium, lead, HB-40 and xylene. Therefore, it is unlikely that there would be significant adverse effects on either aquatic or terrestrial populations or communities as a result of these chemical releases.

The HQs for cadmium, lead and HB-40 exceeded 1 for the conservative scenario where benthic invertebrates are exposed directly to groundwater. The assumption of direct exposure to undiluted groundwater is conservative because a diffusion gradient will exist across the sediment-water interface, resulting in some degree of dilution into the top layer of sediment where most benthic organisms reside. Moreover, the seepage area represents a small part of the benthic community habitat, and the maximum groundwater concentration assumed represents the worst-case time period. As such, benthic invertebrates may not be at risk due to cadmium, lead and HB-40 exposure, and any adverse effects will be spatially and temporally limited.

8.5.5 Assumptions and Uncertainty

Assumptions and uncertainties regarding radiological inventories within the WR-1 Building and radiological release rates are discussed in Section 7.3.4 Assumptions and Uncertainty and overview of modelling assumptions and uncertainty is provided in Section 8.2.5 Assumptions and Uncertainty.

There were no data to determine HB-40 benchmarks for birds. As such, there is uncertainty around the potential health risks to birds due to HB-40 exposure. Considering the major constituent is closely related to the aromatic hydrocarbon group and that petroleum hydrocarbons are metabolized by vertebrates (CCME 2008), it can be suggested that exposure of birds to HB-40 is limited.

8.6 Bounding Scenarios

Taking into consideration the descriptions of disruptive events, three bounding scenarios were identified as “worst case”, with consequences greater than the other disruptive events considered. These were:

- Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes (Base Case Scenario in the solute transport model);
- WRDF Barrier Failure Bounding Scenario: a WRDF barrier failure (Scenario 3 in the sensitivity cases evaluated in the solute transport model); and
- Well in Plume Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF (Scenario 16 in the sensitivity cases evaluated in the solute transport model).

These bounding scenarios are described in detail in Section 5.4.3.3, and the results of the assessment are provided below.

8.6.1 Human Intrusion

For this bounding scenario, it was assumed that immediately following the 100 years of institutional control, an exploration borehole was drilled through the concrete cap and engineered cover, grout, concrete structure, and ISD waste from ground surface to bedrock. The material encountered was brought to surface, handled by the driller, and dumped on the ground. Once the driller had left, trespassers would spend time at the drill location. The driller and trespasser would be exposed to the waste material through incidental soil ingestion, dermal contact with soil and groundshine. For the trespasser, there may be inhalation of dust from resuspension of dried waste material, which is not the case for the driller since the material would be considered wet when it was initially brought to surface. This is an unlikely scenario but is considered as a conservative assessment for the disruptive events. For details refer to Section 5.4.3.2.2 Human Intrusion and Section 5.4.3.3.1 Human Intrusion Bounding Scenario.

8.6.1.1 Radiological Releases

Concentrations of radionuclides projected in the WRDF post-closure are presented in Table 8.6.1-1 and in Appendix D of the ERA (EcoMetrix 2021; Golder 2021). Radionuclides and non-radionuclides associated with WR-1 and released from WR-1 to the environment were simulated with an analytical (GoldSim®, Version 11.1) model as described in the *WR-1 Groundwater Flow and Solute Transport Modelling* (Golder 2021). This model was used to estimate the dissolved mass and the solid mass remaining in WR-1, which were combined to estimate a total mass concentration in the waste material. Solute concentrations (g/m^3) were calculated based on the total solute mass divided by the volume of the grout (i.e., the source area, as it was assumed that the solute mass would be distributed throughout the grout). For radionuclides, the solute concentrations (g/m^3) were converted to activity concentrations (Bq/kg) using the equation in Section 3.2.1 of the ERA (EcoMetrix 2021) and the density of the waste (2,100 kg/m^3). For non-radionuclides the received concentration was divided by the density of the waste to convert the concentrations to mg/kg. The density of the remainder of WR-1 Building foundation and grout filling used in the solute transport model was 2,100 kg/m^3 and was used as the density of material in the borehole for this assessment (EcoMetrix 2021). The concentration was adjusted by a factor of 0.947 to account for mixing of the clean “soil” in the cover with the waste material.

All radionuclides were carried forward for further analysis.

Table 8.6.1-1: Concentrations of Radionuclides Projected in the WRDF Post-closure

Radionuclide	Concentration (g/m^3)	Atomic mass (g/mol)	Half-life (s)	Concentration (Bq/kg)
Actinium-225	4.38E-18	225.02	8.64E+05	4.24E-06
Actinium-227	3.01E-13	227.03	6.87E+08	3.63E-04
Americium 243	2.58E-07	243.06	2.32E+11	8.59E-01
Americium-241	1.86E-05	241.06	1.36E+10	1.07E+03
Bismuth-210	4.66E-18	209.98	4.33E+05	9.66E-06
Caesium-137	6.66E-07	137.91	9.51E+08	9.55E+02
Calcium-41	4.21E-06	40.96	3.22E+12	6.01E+00
Carbon-14	2.63E-03	14.00	1.81E+11	1.96E+05
Chlorine-36	2.45E-10	35.97	9.46E+12	1.36E-04
Cobalt-60	7.37E-11	59.93	1.67E+08	1.38E+00
Curium-244	6.88E-11	244.06	5.71E+08	9.30E-02
Europium 152	2.40E-11	151.92	4.26E+08	6.98E-02
Europium 154	3.56E-12	153.92	2.71E+08	1.61E-02
Europium 155	7.13E-17	154.92	1.50E+08	5.77E-07
Gadolinium-152	8.80E-09	151.92	3.41E+21	3.20E-12
Iodine-129	3.04E-06	128.90	4.95E+14	8.98E-03
Iron-55	3.41E-18	54.94	8.51E+07	1.37E-07
Lead-210	7.55E-15	209.98	7.03E+08	9.63E-06
Neodymium-144	2.69E-34	143.91	7.22E+22	4.87E-39
Neptunium-237	8.77E-06	237.05	6.75E+13	1.03E-01

Table 8.6.1-1: Concentrations of Radionuclides Projected in the WRDF Post-closure

Radionuclide	Concentration (g/m^3)	Atomic mass (g/mol)	Half-life (s)	Concentration (Bq/kg)
Neptunium-239	2.22E-13	239.05	2.04E+05	8.59E-01
Nickel-59	5.52E-01	58.93	3.19E+12	5.53E+05
Nickel-63	2.92E-02	62.93	3.15E+09	2.77E+07
Niobium-94	6.36E-02	93.91	6.40E+11	1.99E+05
Plutonium 238	1.63E-07	238.05	2.77E+09	4.65E+01
Plutonium-239	2.86E-04	239.05	7.60E+11	2.97E+02
Plutonium-240	1.10E-04	240.05	2.07E+11	4.19E+02
Plutonium-241	1.70E-08	241.06	4.42E+08	3.01E+01
Polonium-210	1.27E-16	209.98	1.19E+07	9.59E-06
Protactinium-231	4.81E-10	231.04	1.03E+12	3.80E-04
Protactinium-233	2.97E-13	233.04	2.33E+06	1.03E-01
Radium-223	4.24E-16	223.02	9.88E+05	3.63E-04
Radium-224	1.87E-23	224.02	3.14E+05	5.01E-11
Radium-225	6.53E-18	225.02	1.29E+06	4.24E-06
Radium-226	5.33E-13	226.03	5.05E+10	8.79E-06
Radium-228	1.10E-20	228.03	1.80E+08	5.04E-11
Radon-222	3.43E-18	222.02	3.30E+05	8.80E-06
Samarium-148	1.02E-20	147.91	2.21E+23	5.89E-26
Silver-108m	1.84E-08	107.91	1.38E+10	2.33E+00
Strontium 90	3.22E-07	89.91	9.08E+08	7.42E+02
Technetium-99	1.88E-05	98.91	6.65E+12	5.37E+00
Thorium-227	6.96E-16	227.03	1.61E+06	3.58E-04
Thorium-228	3.63E-21	228.03	6.03E+07	4.97E-11
Thorium-229	1.20E-12	229.03	2.30E+11	4.27E-06
Thorium-230	2.92E-10	230.03	2.38E+12	1.00E-04
Thorium-231	8.80E-15	231.04	9.19E+04	7.81E-02
Thorium-232	2.82E-11	232.04	4.45E+17	5.15E-11
Thorium-234	1.57E-12	234.04	2.08E+06	6.05E-01
Tritium	3.68E-07	3.02	3.89E+08	5.91E+04
Uranium-233	7.68E-10	233.04	5.02E+12	1.24E-04
Uranium-234	3.43E-07	234.04	7.74E+12	3.56E-02
Uranium-235	2.17E-03	235.04	2.22E+16	7.82E-02
Uranium-236	2.56E-06	236.05	7.40E+14	2.76E-03
Uranium-237	5.29E-16	237.05	5.83E+05	7.21E-04
Uranium-238	1.08E-01	238.05	1.41E+17	6.05E-01
Yttrium-90	8.18E-11	89.91	2.31E+05	7.43E+02

g/m^3 = grams per cubic metre; g/mol = grams per mole; Bq/kg = becquerels per kilogram.

8.6.1.2 Non-Radiological Releases

Concentrations of non-radionuclides projected in the WRDF post-closure are presented in Table 8.6.1-2. Non-radionuclides were screened against CCME Soil Quality Guidelines for human health. For non-radionuclides without soil guidelines, background soil concentrations were taken from the Ontario Ministry of Environment and Energy 98th percentile, as well as background values reported for soils. For HB-40 and palladium, the mammalian Lowest Observable Adverse Effect Level was used. For HB-40 250 mg/kg bw/day and for palladium 1.2 mg/kg bw/day (EcoMetrix 2021). This was divided by 10 to estimate a No Observable Adverse Effect Level and by further factor of 100 to allow for uncertainty in animal to human extrapolation and to be protective of the public. This value was then converted to a screening benchmark (EcoMetrix 2021). Screening of the non-radionuclides is summarized in Table 8.6.1-3 (EcoMetrix 2021). Lead, HB-40 and palladium exceeded the screening criteria; therefore, they were carried forward for further analysis.

Table 8.6.1-2: Concentrations of Non-Radionuclides in WRDF Post-closure

Non-Radionuclide	Concentration (g/m^3)	Concentration (mg/kg)
Argon	8.72E-14	3.94E-14
Barium	2.06E-05	9.31E-06
Bismuth	2.09E-14	9.42E-15
Boron	7.67E-05	3.46E-05
Cadmium	8.91E+00	4.02E+00
Chromium	1.50E+01	6.78E+00
Cobalt	3.02E-02	1.36E-02
Copper	7.66E-02	3.45E-02
Gadolinium	3.36E-07	1.51E-07
HB-40	1.30E+04	5.85E+03
Helium	1.85E-04	8.34E-05
Lead	6.04E+03	2.72E+03
Manganese	5.04E-03	2.27E-03
Mercury	3.03E-02	1.37E-02
Molybdenum	1.32E-02	5.95E-03
Nickel	6.57E-03	2.96E-03
Nitrogen	1.25E-03	5.65E-04
Palladium	1.68E+00	7.56E-01
Potassium	3.59E-09	1.62E-09
Potassium hydroxide	4.05E-04	1.83E-04
Ruthenium	1.07E-08	4.83E-09
Samarium	2.25E-08	1.01E-08
Sulphur	1.86E-15	8.40E-16
Xenon	2.58E-11	1.16E-11
Xylene	1.80E-01	8.14E-02
Zirconium	1.00E-05	4.52E-06

g/m^3 = grams per cubic metre; mg/kg = milligrams per kilogram.

Table 8.6.1-3: Screening Non-radionuclides in Soil

Non-Radionuclide	Concentration (mg/kg)	CCME SGQ$_{HH}$ (mg/kg)	OTR98 (mg/kg)	Dragun and Chiasson, 1991 (mg/kg)	Shackette and Boerngen, 1984 (mg/kg)	Other
Argon	3.94E-14	Noble gas – Not Applicable				
Barium	9.31E-06	500	—	—	—	—
Bismuth	9.42E-15	—	—	10	—	—
Boron	3.46E-05	2	—	—	—	—
Cadmium	4.02E+00	14	—	—	—	—
Chromium	6.78E+00	220	—	—	—	—
Cobalt	1.36E-02	50	—	—	—	—
Copper	3.45E-02	1100	—	—	—	—
Gadolinium	1.51E-07	—	—	2.8	—	—
HB-40	5.85E+03	—	—	—	—	15
Helium	8.34E-05	Noble gas – Not Applicable				
Lead	2.72E+03	140	—	—	—	—
Manganese	2.27E-03	—	1,300	—	—	—
Mercury	1.37E-02	6.6	—	—	—	—
Molybdenum	5.95E-03	10	—	—	—	—
Nickel	2.96E-03	200	—	—	—	—
Nitrogen	5.65E-04	—	5,700	—	—	—
Palladium	7.56E-01	—	—	—	1	—
Potassium	1.62E-09	—	6,500	—	—	—
Potassium hydroxide	1.83E-04	—	6,500	—	—	—
Ruthenium	4.83E-09	—	—	72	—	—
Samarium	1.01E-08	—	—	4.4	—	—
Sulphur	8.40E-16	500	—		—	—
Xenon	1.16E-11	Noble gas – Not Applicable				
Xylene	8.14E-02	2.4	—	—	—	—
Zirconium	4.52E-06	—	—	159	—	—

mg/kg = milligrams per kilogram.

8.6.1.3 Estimated Public Dose and Risk Characterization

For all bounding scenarios, the radionuclides total doses were compared to the IAEA reference level ranging from 1 to 20 mSv/a for disruptive events. The total dose to a drill crew member (adult exposed during drilling the borehole) was below both the upper and lower IAEA reference level for disruptive events. It was also considered that trespassers could interact with the site following a human intrusion (assumed to be spending 1 hour a day on-site). The dose predictions for this trespasser receptor were below both the upper (20 mSv/a) and lower (1 mSv/a) IAEA reference level for disruptive events (refer to Table 8.6.1-4).

The dominant contributor to the total dose is niobium-94 through groundshine (i.e., from the drilled material improperly disposed of on-site).

Table 8.6.1-4: Summary of Total Dose for Trespassers and Driller under Human Intrusion Conditions

Age Group	Dose (mSv/a)	Percent of IAEA Lower Reference Level for Disruptive Events	Percent of IAEA Upper Reference Level for Disruptive Events
Trespasser - Adult	1.98E-01	20%	1%
Trespasser - Child	2.01E-01	20%	1%
Trespasser - 1-year old Infant	3.19E-01	32%	2%
Trespasser - 3-month old Infant	2.86E-01	29%	1%
Driller	6.35E-03	1%	0%

Note:
mSv/a = millisievert per year.
1 millisievert per year (mSv/a) IAEA Lower Reference Level 20 mSv/a IAEA Upper Reference Level for Disruptive Events.

The doses to human receptors that could be exposed to the non-radionuclides identified as potentially occurring at elevated concentrations (i.e., HB-40, lead) as a result of a human intrusion event occurring during post-closure were calculated based on total concentration (background plus Project contribution). The HQ for HB-40 exceeded the target values for both the adult and the toddler for soil ingestion and soil dermal contact (i.e., exposure as a result of material being improperly disposed of on-site). The HQ for lead exceeded the target values for the adult and the toddler through soil ingestion, soil dermal contact and dust inhalation (i.e., exposure as a result of material being improperly disposed of on-site) and was exceeded for the driller through soil ingestion and soil dermal contact. The estimated non-radiological exposures for receptors near the borehole are presented in Table 8.6.1-5.

In Table 8.6.1-6, HQs were presented and compared to a target value of 0.2 per medium. HQs greater than 0.2 (per pathway) are not statistical probabilities of harm occurring. Instead, they are a simple statement of whether (and by how much) an exposure dose exceeds the reference dose.

The TRVs for HB-40 and lead incorporate safety factors to account for uncertainty, making the results conservative. The HB-40 TRV incorporates a safety factor of 1,000 and the lead TRV incorporates a safety factor of 2 (EcoMetrix 2021). In this scenario, it was assumed that an exploration borehole was drilled through the engineered cover, grout, cement, and WRDF from ground surface to bedrock. The waste emplacement strategy and design of the engineered cover is more robust than a typical hazardous waste landfill (Figure 8.6.1-1 and Figure 8.6.1-2); therefore, the likelihood of installing an exploration borehole directly into the WRDF as well as extruding the highest concentrations is low.

Table 8.6.1-5: Exposures to Human Receptors Immediate to the Borehole

Human Type	Non-radionuclide	Unit	Dose by Pathway			Total Dose
			Soil Ingestion	Soil Dermal Contact	Dust Inhalation	
Adult	HB-40	mg/kg bw/d	8.80E-01	1.14E-01	2.89E-02	1.02E+00
	Lead	mg/kg bw/d	4.09E-01	2.65E-02	1.34E-02	4.49E-01
Toddler	HB-40	mg/kg bw/d	5.13E-02	5.85E-02	1.35E-02	1.23E-01
	Lead	mg/kg bw/d	2.39E-02	1.36E-02	6.28E-03	4.38E-02
Driller	HB-40	mg/kg bw/d	1.66E-03	1.89E-02	NA	2.05E-02
	Lead	mg/kg bw/d	7.70E-04	4.39E-03	NA	5.16E-03

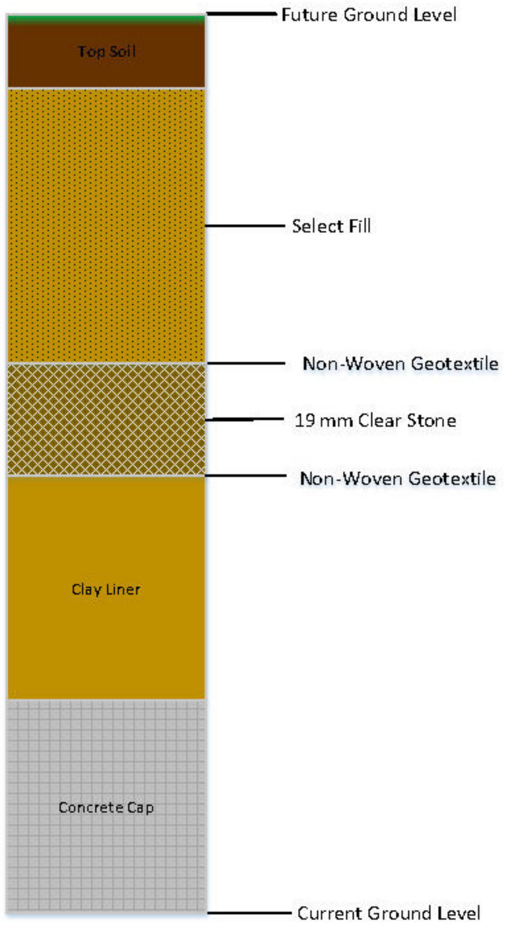
mg/kg bw/d = milligrams per kilogram per body weight per day.

Table 8.6.1-6: Hazard Quotients for Human Receptors Immediate to the Borehole

Human Type	Non-radionuclide	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Total
Adult	HB-40	**3.52**	**4.57E-01**	1.16E-01	**4.09**
	Lead	**3.41E+02**	**2.21E+01**	**1.12E+01**	**3.74E+02**
Toddler	HB-40	**2.05E-01**	**2.34E-01**	5.40E-02	**4.93E-01**
	Lead	**1.99E+01**	**1.13E+01**	**5.23**	**3.65E+01**
Driller	HB-40	6.62E-03	7.55E-02	NA	8.21E-02
	Lead	**6.42E-01**	**3.66**	NA	**4.30**

Note:

Bold and shaded cells indicate exceedance of HQ benchmark of 0.2.



Layers	Height (mm)
Top Soil	300
Select Fill	1100
19 mm Clear Stone	450
Clay Liner	900
Concrete Cap	850

🟩	Top Soil
▒	Select Fill
▩	19 mm Clear Stone
🟫	Clay
▦	Reinforced Concrete

Scaling: 50/1000

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DETAILS OF THE CONCRETE CAP AND ENGINEERED COVER

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.6.1-1**

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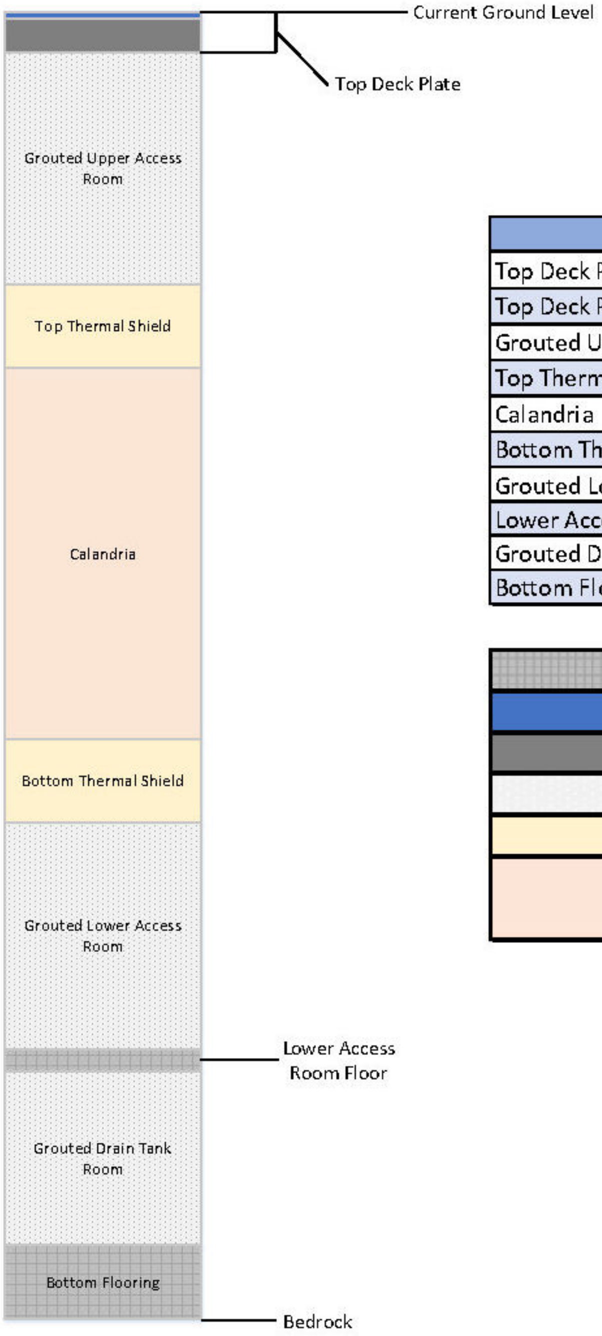
IF THIS MEASUREMENT DOES NOT MATCH WHAT IS SHOWN, THE SHEET SIZE HAS BEEN MODIFIED FROM:
25mm
0



Scaling: 15/1000

Layers	Height (m)
Top Deck Plate: Masonite	0.08
Top Deck Plate: Carbon Steel	0.46
Grouted Upper Access Room	3.15
Top Thermal Shield	1.12
Calandria	5.03
Bottom Thermal Shield	1.12
Grouted Lower Access Room	3.07
Lower Access Room Floor	0.30
Grouted Drain Tank Room	2.36
Bottom Flooring	0.99

▦	Reinforced Concrete
🟦	Masonite Benelex 70
■	Carbon Steel A27-58-Gr 65
░	Grout
🟨	Carbon Steel A353 Gr B
🟧	SS 304L, Al B210-61 Alloy 5052, Zr-2.5%Nb, Ozhennite

CLIENT
CANADIAN NUCLEAR LABORATORIES LTD.

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DETAILS OF THE END-STATE OF BELOW-GRADE STRUCTURES

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	--
PREPARED	PR/RRD
REVIEWED	KL
APPROVED	MM

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.6.1-2**

S:\Clients\Canadian_Nuclear_Laboratories\Manitoba\99_PROJ\20145046\40_PROD\0001_DSAR\20145046-0001-G-0031.mxd

IF THIS MEASUREMENT DOES NOT MATCH WHAT IS SHOWN, THE SHEET SIZE HAS BEEN MODIFIED FROM:
25mm
0

The assessment demonstrates that human intrusion into the WRDF could result in exposures of human receptors to HB-40 and lead in waste material brought to the surface at levels where risks cannot be ruled out. As such, while this is a very unlikely worst-case scenario, reasonable effort is warranted to reduce the probability of these unplanned events from occurring. During the post-institutional control period, passive controls will still be in place including the limited footprint, the WRDF composition being relatively imperviousness and made of material of no economic value, and the land use restriction acting to reduce the likelihood of a human intrusion event.

8.6.2 Whiteshell Reactor Disposal Facility Barrier Failure

For this bounding scenario, an open fracture was modelled in the foundation of the WR-1 Building that will remain in place as a component of the WRDF. The foundation floor and walls for the WR-1 Building were specified as a 1-m thick continuous barrier with a uniform hydraulic conductivity (grout failure was included in the Normal Evolution Scenario). To examine potential effect of a failure in this barrier, a 2 m-wide zone of enhanced hydraulic conductivity was simulated. Exposure pathway characterization are the same as was modelled in the Normal Evolution Scenario, except that groundwater loadings to the surface water are based on an open fracture. For details on this scenario refer to Section 5.4.3.2.4 WRDF Barrier Failure, as well as Section 5.4.3.3.2 WRDF Barrier Failure Bounding Scenario.

8.6.2.1 Radiological Releases

In Table 8.6.2-1, the mass loading rate and peak times (i.e., time of maximum [year]) are presented for radionuclides in groundwater. Peak times are presented in years after closure. All radionuclides were carried forward for assessment.

Table 8.6.2-1: Mass and Activity Loadings to the Winnipeg River for Radionuclides in Groundwater for WRDF Barrier Failure

Radionuclide	Maximum Loading Rate (g/yr)	Time of Maximum (yr)	Atomic mass, (g/mol)	Half-life (s)	Release Rate (Bq/s)
Actinium-225	1.70E-17	88,400	225.02	8.64E+05	1.16E-09
Actinium-227	1.41E-14	159,800	227.03	6.87E+08	1.20E-09
Americium-241	0	N/A	241.06	1.36E+10	0.00E+00
Americium-243	0	N/A	243.06	2.32E+11	0.00E+00
Bismuth-210	2.29E-14	208,100	209.98	4.33E+05	3.33E-06
Caesium-137	0	N/A	137.91	9.51E+08	0.00E+00
Calcium-41	1.56E-06	20,300	40.96	3.22E+12	1.56E-04
Carbon-14	5.65E-04	937	14.00	1.81E+11	2.96E+00
Chlorine-36	1.20E-08	168	35.97	9.46E+12	4.67E-07
Cobalt-60	0	N/A	59.93	1.67E+08	0.00E+00
Curium-244	0	N/A	244.06	5.71E+08	0.00E+00
Europium 152	0	N/A	151.92	4.26E+08	0.00E+00
Europium 154	0	N/A	153.92	2.71E+08	0.00E+00
Europium 155	0	N/A	154.92	1.50E+08	0.00E+00
Gadolinium-152	1.84E-07	208	151.92	3.41E+21	4.72E-15
Iodine-129	1.50E-04	168	128.90	4.95E+14	3.11E-05
Iron-55	0	N/A	54.94	8.51E+07	0.00E+00

Table 8.6.2-1: Mass and Activity Loadings to the Winnipeg River for Radionuclides in Groundwater for WRDF Barrier Failure

Radionuclide	Maximum Loading Rate (g/yr)	Time of Maximum (yr)	Atomic mass, (g/mol)	Half-life (s)	Release Rate (Bq/s)
Lead-210	9.29E-13	209,700	209.98	7.03E+08	8.33E-08
Neodymium-144	0	N/A	143.91	7.22E+22	0.00E+00
Neptunium-237	5.15E-06	41,400	237.05	6.75E+13	4.26E-06
Neptunium-239	0	1 N/A	239.05	2.04E+05	0.00E+00
Nickel-59	5.48E-05	491,200	58.93	3.19E+12	3.86E-03
Nickel-63	0	N/A	62.93	3.15E+09	0.00E+00
Niobium-94	6.62E-21	626,000*	93.91	6.40E+11	1.46E-18
Plutonium-238	0	N/A	238.05	2.77E+09	0.00E+00
Plutonium-239	2.39E-07	108,400	239.05	7.60E+11	1.74E-05
Plutonium-240	1.59E-10	67,100	240.05	2.07E+11	4.23E-08
Plutonium-241	0	N/A	241.06	4.42E+08	0.00E+00
Polonium-210	1.64E-10	206,900	209.98	1.19E+07	8.69E-04
Protactinium-231	2.16E-11	161,700	231.04	1.03E+12	1.20E-09
Protactinium-233	2.33E-16	41,600	233.04	2.33E+06	5.68E-09
Radium-223	3.36E-16	161,200	223.02	9.88E+05	2.02E-08
Radium-224	6.67E-22	1,894,000*	224.02	3.14E+05	1.26E-13
Radium-225	2.17E-16	89,500	225.02	1.29E+06	9.91E-09
Radium-226	1.65E-11	207,500	226.03	5.05E+10	1.91E-08
Radium-228	3.9E-19	930,000*	228.03	1.80E+08	1.26E-13
Radon-222	4.82E-12	208,600	222.02	3.30E+05	8.70E-04
Samarium-148	1.40E-24	2,066	147.91	2.21E+23	5.65E-34
Silver-108m	8.10E-13	3,645	107.906	1.38E+10	7.19E-09
Strontium-90	0	N/A	89.91	9.08E+08	0.00E+00
Technetium-99	1.60E-05	10,500	98.91	6.65E+12	3.21E-04
Thorium-227	6.12E-17	161,000	227.03	1.61E+06	2.21E-09
Thorium-228	2.84E-20	1,082,000*	228.03	6.03E+07	2.73E-14
Thorium-229	8.68E-12	89,000	229.03	2.30E+11	2.18E-09
Thorium-230	1.73E-10	206,200	230.03	2.38E+12	4.19E-09
Thorium-231	4.36E-18	118,000	231.04	9.19E+04	2.89E-09
Thorium-232	2.12E-10	632,000*	232.04	4.45E+17	2.72E-14
Thorium-234	7.35E-16	116,400	234.04	2.08E+06	2.00E-08
Tritium	6.73E-05	68	3.02	3.89E+08	7.61E+02
Uranium-233	2.79E-08	76,800	233.04	5.02E+12	3.16E-07
Uranium-234	1.36E-07	130,300	234.04	7.74E+12	9.94E-07
Uranium-235	1.60E-04	118,200	235.04	2.22E+16	4.05E-07
Uranium-236	7.21E-06	120,100	236.05	7.40E+14	5.46E-07
Uranium-237	0	N/A	237.05	5.83E+05	0.00E+00

Table 8.6.2-1: Mass and Activity Loadings to the Winnipeg River for Radionuclides in Groundwater for WRDF Barrier Failure

Radionuclide	Maximum Loading Rate (g/yr)	Time of Maximum (yr)	Atomic mass, (g/mol)	Half-life (s)	Release Rate (Bq/s)
Uranium-238	7.10E-03	116,300	238.05	1.41E+17	2.80E-0
Yttrium-90	0	N/A	89.91	2.31E+05	0.00E+00

g/yr = grams per year; g/mol = grams per mol.
* The model was run for 500,000 years. For these radionuclide contaminants, the maximum loading was not reached before 500,000 years; therefore, the model runtime was extended until the maximum was reached.

8.6.2.2 Non-Radiological Releases

Mass loading rates to the Winnipeg River for non-radionuclides are presented in Table 8.6.2-2, non-radionuclides were screened against effects criteria, for the human health screening refer to Table 8.6.2-3 and for the ecological health screening refer to Table 8.6.2-4. As with the Normal Evolution Scenario, lead exceeded the screening criteria, and was carried forward for further analysis. Cadmium did not exceed human health screening criteria but was carried forward for comparison against the Normal Evolution Scenario. As with the Normal Evolution Scenario, cadmium, lead, HB-40, and xylene exceeded ecological health screening criteria and were carried forward for further analysis. In Table 8.6.2-5 the exposure point concentrations for non-radionuclide COPCs are presented.

Table 8.6.2-2: Maximum Mass Loading Rates to the Winnipeg River for Non-radionuclides in Groundwater with WRDF Barrier Failure

Non-Radionuclide	Maximum Loading Rate (g/yr)	Time of Maximum (Year)
Argon	7.07E-12	222
Barium	1.49E-23	499,900
Bismuth	1.11E-08	78,200
Boron	1.87E-04	4,171
Cadmium	3.71E-01	203,300
Chromium	4.94E-27	500,000
Cobalt	2.12E-02	93,900
Copper	5.47E-04	876,000*
Gadolinium	6.98E-06	212
HB-40	4.13E+01	10,300
Helium	1.12E-02	146
Lead	5.14E+00	4,430,000*
Manganese	8.33E-05	359,100
Mercury	4.16E-05	6,576,000*
Molybdenum	1.17E-02	17,900
Nickel	4.81E-05	842,000*
Nitrogen	5.44E-04	17,900
Palladium	1.05E-01	121,100
Potassium	3.27E-07	13,500

Table 8.6.2-2: Maximum Mass Loading Rates to the Winnipeg River for Non-radionuclides in Groundwater with WRDF Barrier Failure

Non-Radionuclide	Maximum Loading Rate (g/yr)	Time of Maximum (Year)
Potassium hydroxide	4.94E-02	140
Ruthenium	6.15E-09	438,500
Sulphur	2.46E-12	195
Xenon	1.45E-09	203
Xylene	4.59E+00	200

g/yr = grams per year.
* The model was run for 500,000 years. For these non-radionuclide contaminants, the maximum loading was not reached before 500,000 years; therefore, the model runtime was extended until the maximum was reached.

Table 8.6.2-3: Human Health Screening of Non-radionuclides in Surface Water for WRDF Barrier Failure

Non-radionuclide	Concentration (µg/L)	Background Concentration (µg/L)	CDWS MAC	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Argon	1.09E-10	Noble Gas – not applicable*							
Barium	1.95E-22	1.10E+01	1,000	—	—	—	—	0.4	CDWS MAC
Bismuth	1.46E-07	<2.00E-01	—	—	—	—	—	0.25	LC$_{50}$/100 – Borgmann et al., 2005
Boron	2.30E-03	1.00E+01	5,000	—	1,500	200	1,200	—	CDWS MAC
Cadmium	4.86E+00	1.00E-02	5	0.137	0.08	0.1	0.114	—	CDWS MAC
Chromium	6.47E-26	1.70E+00	50	37.1	1.0 (VI)	1 (VI)	—	—	CDWS MAC
Cobalt	2.77E-01	2.00E-01	—	—	—	0.9	4	—	PWQO Ontario
Copper	7.17E-03	1.40E+01	—	4.3	2	5	2	—	CCME WQG
Gadolinium	1.07E-04	—	—	—	—	—	—	1.5	LC$_{50}$/100 – Borgmann et al., 2005
HB-40	5.41E+02	0	8,800^	—	—	—	—	—	See Note^
Helium	1.72E-01	—	—	—	—	—	—	—	Noble Gas – not applicable*
Lead	**6.74E+01**	**2.60E+00**	**10**	**0.99**	**1**	**3**	**4.4**		**CDWS MAC**
Manganese	1.09E-03	1.10E+01	None - naturally occurring	—	—	—	794.2	110	WQG BC
Mercury	5.45E-04	1.00E-02	1	1	0.026	0.2	—	—	CCME WQG
Molybdenum	1.54E-01	2.00E-01	—	—	73	40	1,000	—	CCME WQG
Nickel	6.31E-04	1.78E+00	—	25.5	25	25	—	—	CCME WQG
Nitrogen	7.13E-03	—	1,000	—	—	—	3,000	—	CDWS MAC
Palladium	1.37E+00	—	—	—	—	—	—	5.7	LC$_{50}$/100 – Borgmann et al., 2005
Potassium	4.29E-06	9.07E+02	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Potassium hydroxide (as K)	5.32E-01	—	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Ruthenium	8.06E-08	—	—	—	—	—	—	10	LC$_{50}$/100 – Borgmann et al., 2005
Sulphur (as SO$_4$)	1.14E-10	—	—	—	—	—	218,000	—	WQG BC

Table 8.6.2-3: Human Health Screening of Non-radionuclides in Surface Water for WRDF Barrier Failure

Non-radionuclide	Concentration (µg/L)	Background Concentration (µg/L)	CDWS MAC	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Xenon	2.24E-08	—	—	—	—	—	—	—	Noble Gas – not applicable
Xylene	7.06E+01	—	90	—	—	2/40/30 (m/o/p)	30	—	CDWS MAC

Note:

Shaded and bolded indicates exceedance of human health screening criteria.

µg/L = microgram per Litre.

*Noble gases were assumed to volatilize rapidly.

^Derived drinking water limit based on a minimal effect level in mice of 250 mg/kg-day (Weeks 1974), divided by 1000, times 70 kg body weight, over 2 L/day of drinking water.

CDWS = Canadian Drinking Water Standard (Health Canada 2017).

WQSOG = Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011).

CCME WQG = Canadian Council of Ministers of the Environment Water Quality Guideline (CCME 1999).

PWQO = Ontario Provincial Water Quality Objective (MOEE 1994).

WQG BC = Water Quality Guideline British Columbia (BC MOE 2017).

Table 8.6.2-4: Ecological Health Screening of Non-radionuclides in Surface Water for WRDF Barrier Failure

Non-radionuclide	Groundwater Concentration (µg/L)	Background Winnipeg River Concentration (µg/L)	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Argon	1.09E-10	Noble Gas – not applicable*						
Barium	1.95E-22	1.10E+01	—	—	—	—	0.4	LCV/10 – Suter and Tsao 1996
Bismuth	1.46E-07	<2.00E-01	—	—	—	—	0.25	LC_{50}/100 – Borgmann et al., 2005
Boron	2.30E-03	1.00E+01	—	1,500	200	1,200	—	CCME WQG
Cadmium	**4.86E+00**	**1.00E-02**	**0.137**	**0.08**	**0.1**	**0.114**	**—**	**CCME WQG**
Chromium	6.47E-26	1.70E+00	37.1	1.0 (VI)	1 (VI)	—	—	CCME WQG
Cobalt	2.77E-01	2.00E-01			0.9	4	—	PWQO Ontario
Copper	7.17E-03	1.40E+01	4.3	2	5	2	—	CCME WQG
Gadolinium	1.07E-04	—	—	—	—	—	1.5	LC_{50}/100 – Borgmann et al., 2005
HB-40	**5.41E+02**	**0.00E+00**	—	—	—	—	**2**	**IC_{25}/10-EcoMetrix 2021**
Helium	1.72E-01	Noble Gas – not applicable*						
Lead	**6.74E+01**	**2.60E+00**	**0.99**	**1**	**3**	**4.4**	**—**	**WQSOG Manitoba**
Manganese	1.09E-03	1.10E+01	—	—	—	794.2	110	WQG BC

Table 8.6.2-4: Ecological Health Screening of Non-radionuclides in Surface Water for WRDF Barrier Failure

Non-radionuclide	Groundwater Concentration (µg/L)	Background Winnipeg River Concentration (µg/L)	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Mercury	5.45E-04	1.00E-02	1	0.026	0.2	—	—	CCME WQG
Molybdenum	1.54E-01	2.00E-01	—	73	40	1,000	—	CCME WQG
Nickel	6.31E-04	1.78E+00	25.5	25	25	—	—	CCME WQG
Nitrogen	7.13E-03	—	—	—	—	3,000	—	WQG BC
Palladium	1.37E+00	—	—	—	—	—	5.7	LC_{50}/100 – Borgmann et al., 2005
Potassium	4.29E-06	9.07E+02	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Potassium hydroxide (as K)	5.32E-01	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Ruthenium	8.06E-08	—	—	—	—	—	10	LC_{50}/100 – Borgmann et al., 2005
Sulphur (as SO_4)	1.14E-10	—	—	—	—	218,000	—	WQG BC
Xenon	2.24E-08	Noble Gas – not applicable						
Xylene	**7.06E+01**	—	—	—	**2/40/30 (m/o/p)**	**30**	—	**WQG BC (in preference over interim PWQO)**

Note:

µg/L = microgram per Litre.

Shaded and bolded cells indicate exceedance above the selected ecological screening criteria.

*Noble gases were assumed to volatilize rapidly.

WQSOG = Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011)

CCME WQG = Canadian Council of Ministers of the Environment Water Quality Guideline (CCME 1999)

PWQO = Ontario Provincial Water Quality Objective (MOEE 1994)

WQG BC = Water Quality Guideline British Columbia (BC MOE 2017).

Table 8.6.2-5: Exposure Point Concentrations for Non-radionuclide COPCs during Post-closure with WRDF Barrier Failure

Non-radionuclide	Groundwater Concentration (µg/L)	Background Concentration (µg/L)	Project Contribution to River Concentrations at Groundwater Seep at River Bottom (µg/L)	River Concentration at Groundwater Seep at River Bottom (µg/L)	Project Contribution to River Concentrations at Groundwater Seep - 50 m Downstream (µg/L)	River Concentration at Groundwater Seep - 50 m Downstream (µg/L)	Project Contribution to River Concentration at Farm A Intake (µg/L)	River Concentration at Farm A Intake (µg/L)
Cadmium	4.86E+00	1.00E-02	3.59E-05	1.00E-02	4.15E-06	1.00E-02	7.80E-08	1.00E-02
HB-40	5.41E+02	0.00E+00	4.00E-03	4.00E-03	4.61E-04	4.61E-04	8.68E-06	8.68E-06
Lead	6.74E+01	2.60E+00	4.98E-04	2.60E+00	5.75E-05	2.60E+00	1.08E-06	2.60E+00
Xylene	7.06E+01	NV	4.44E-04	4.44E-04	5.13E-05	5.13E-05	9.65E-07	9.65E-07

µg/L = microgram per Litre.

NV = no value available to calculate.

8.6.2.3 Estimated Public Dose and Risk Characterization

Detailed radiological doses for Harvester, Farm A, and On-site Farm receptors are detailed in Tables D-20 to D24 of the ERA (EcoMetrix 2021), and doses for non-radionuclides in Tables D-29 to D-30. Total radiological doses for human heath receptors are presented Table 8.6.2-6 (total dose including contribution from background caesium-137) and Table 8.6.2-7 (project contribution only). The total radiological dose for ecological health receptors is presented in Table 8.6.2-8. For the HHRA, all radionuclide doses were below the IAEA reference level (lower and upper level) ranging from 1 to 20 mSv/a. For the EcoRA, all predicted radionuclide doses to ecological receptors were well below benchmarks.

The dominant pathway for the On-Site Farm as well as for Farm A is caesium-137 through external exposure to sediment. This constitutes the measured background and is not associated with the disruptive scenario for the project. Omitting the dose due to caesium-137 in sediment, the highest contributor to the total dose is carbon-14 mainly through the consumption of terrestrial animals. For 3-month-old infants in both farm locations, the dominant uptake pathway is carbon-14 through breast milk for the nursing infant, and tritium from water for the formula consuming infant. In the case of the harvester the majority of the dose is accounted for by carbon-14 through uptake of terrestrial and aquatic animals.

While the tables present the results based on the conservative assumption that maximum loadings to the river occur at the same time for all COPCs, Figure 8.6.2-1 shows a more realistic representation of predicted dose rate to human receptors after a hypothetical WRDF barrier failure. After a hypothetical barrier failure, the dose steadily increases steadily with time, generally peaking around 1,000 years after closure due to contribution from carbon-14. The exception is the 3-month-old formula-fed infant where the dose peaks at the beginning of modelling and then again after 100,000 years. This is because the dose from tritium peaks towards the beginning of the modelling timeframe, and the dose from polonium-210 peaks after 100,000 years.

The dose to receptors in the WRDF Barrier Failure Bounding Scenario is similar to the Normal Evolution Scenario since the source inventory is the same, and in both situations the end point is the Winnipeg River. In the WRDF barrier failure scenario the groundwater pathway flowrate through the fracture is faster that in normal evolution; therefore, groundwater concentrations are lower than in normal evolution.

Table 8.6.2-6: Summary of Total Dose for Human Health Receptors for WRDF Barrier Failure Bounding Scenario

Age Group	Dose (mSv/a)								
	On-site Farm			Farm A			Harvester		
	Dose (including background Cs-137)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level for Disruptive Events	Dose (including background Cs-137)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level	Dose (including background Cs-137)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level
Adult	3.24E-03	0.32%	0.02%	3.35E-04	0.03%	<0.01%	4.81E-05	<0.01%	<0.01%
Child	3.25E-03	0.33%	0.02%	3.36E-04	0.03%	<0.01%	3.09E-05	<0.01%	<0.01%
Infant (cow's milk)	4.23E-03	0.42%	0.02%	4.37E-04	0.04%	<0.01%	2.03E-05	<0.01%	<0.01%
Infant (formula)	4.18E-03	0.42%	0.02%	4.36E-04	0.04%	<0.01%	N/A	N/A	N/A
3-month-old (nursing)	5.58E-04	0.06%	<0.01%	1.06E-05	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (formula)	3.79E-06	<0.01%	<0.01%	7.12E-08	<0.01%	<0.01%	N/A	N/A	N/A

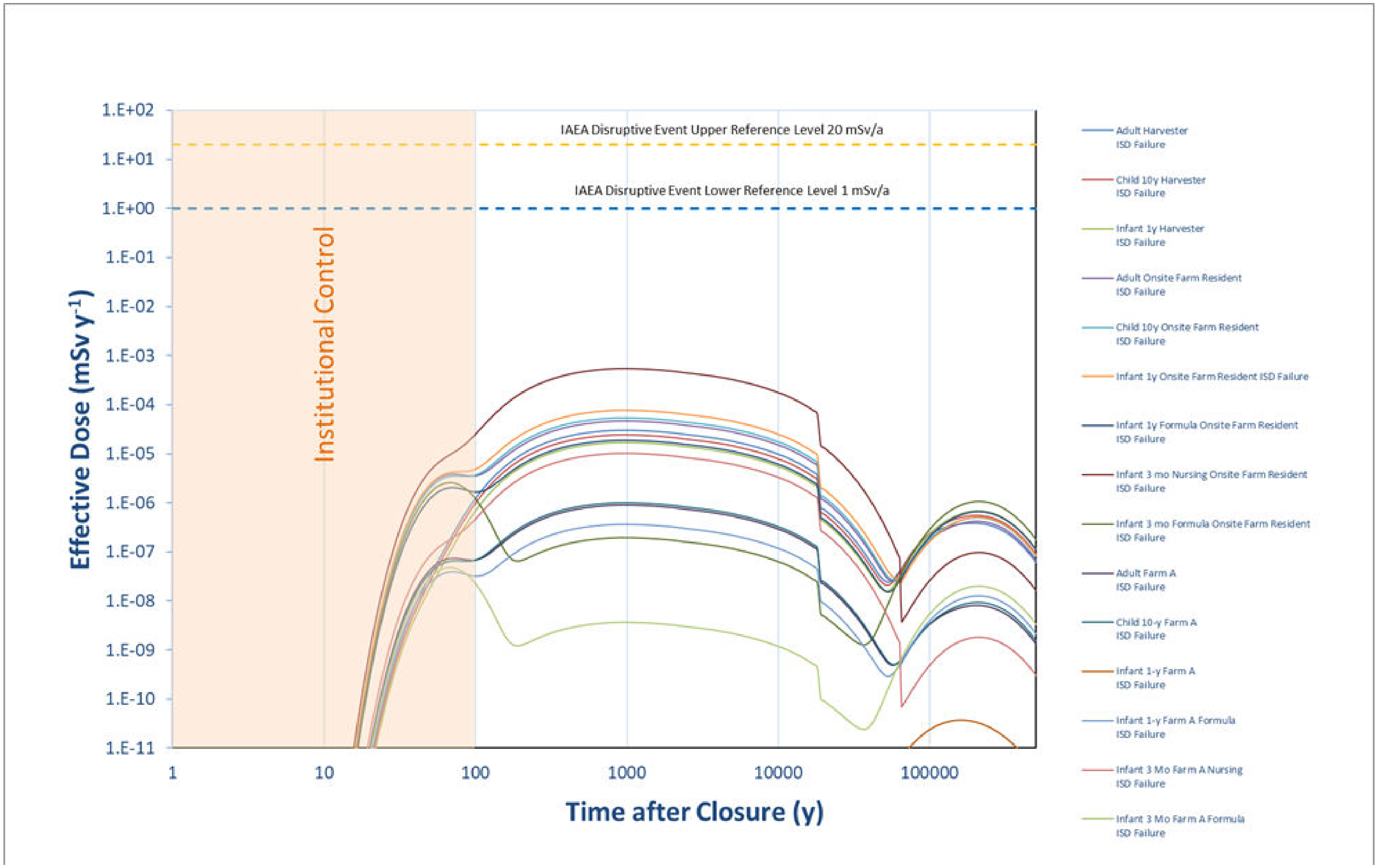
Note:
mSv/a = millisievert per year; <= less than.

1 millisievert per year (mSv/a) IAEA Lower Reference Level; 20 mSv/a IAEA Upper Reference Level for Disruptive Events.

Table 8.6.2-7: Summary of Total Dose for Human Health Receptors for WRDF Barrier Failure Bounding Scenario – Project Contribution Only

Age Group	Dose (mSv/a)								
	On-site Farm			Farm A			Harvester		
	Dose (Project Only)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level for Disruptive Events	Dose (Project Only)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level	Dose (Project Only)	Percent of IAEA Lower Reference Level	Percent of IAEA Upper Reference Level
Adult	5.17E-05	0.01%	<0.01%	9.71E-07	<0.01%	<0.01%	3.07E-05	<0.01%	<0.01%
Child	5.76E-05	0.01%	<0.01%	1.08E-06	<0.01%	<0.01%	2.49E-05	<0.01%	<0.01%
Infant (cow's milk)	8.05E-05	0.01%	<0.01%	1.51E-06	<0.01%	<0.01%	1.79E-05	<0.01%	<0.01%
Infant (formula)	2.19E-05	<0.01%	<0.01%	4.12E-07	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (nursing)	5.56E-04	0.06%	<0.01%	1.05E-05	<0.01%	<0.01%	N/A	N/A	N/A
3-month-old (formula)	3.79E-06	<0.01%	<0.01%	7.12E-08	<0.01%	<0.01%	N/A	N/A	N/A

Note:
mSv/a = millisievert per year; <= less than.
1 millisievert per year (mSv/a) IAEA Lower Reference Level; 20 mSv/a IAEA Upper Reference Level for Disruptive Events.



REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES

CONSULTANT
GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DOSE RATE TO HUMAN RECEPTORS - WRDF BARRIER FAILURE BOUNDING SCENARIO

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.6.2-1**

Table 8.6.2-8: Summary of Total Dose during WRDF Barrier Failure for Selected Ecological Receptors

Ecological Receptor	Total (including background Cs-137) (mGy/day)	Project only (mGy/day)	Percent of Dose Benchmark
Barn Swallow	8.28E-06	8.28E-06	<0.01%
Little Brown Bat	6.70E-06	6.70E-06	<0.01%
Carmine Shiner	1.14E-04	4.17E-06	<0.01%
Lake Sturgeon	1.14E-04	4.17E-06	<0.01%
Walleye	4.17E-06	4.17E-06	<0.01%
Freshwater plant	4.34E-06	4.34E-06	<0.01%
Benthic Invertebrate	5.77E-04	4.07E-06	<0.01%
Horned Grebe	1.55E-04	8.19E-06	<0.01%
Trumpeter Swan	1.53E-04	8.21E-06	<0.01%
Wild Waterfowl	1.52E-04	8.20E-06	<0.01%
Mink	1.54E-04	6.71E-06	<0.01%
Moose	1.02E-05	7.08E-06	<0.01%

Note:
mGy/d = milligray per day; <= less than.

All predicted radiological doses were well below benchmark values for aquatic and terrestrial biota. There are no exceedances of the 9.6 mGy/d radiation benchmark for the aquatic biota in the Winnipeg River. Similarly, there are no exceedances of the 2.4 mGy/d radiation benchmark for terrestrial and riparian biota on or near the WL site. An evaluation to assessment criteria illustrates that all predicted doses are less than 1% of the radiation benchmark. A dose of 0.024 mGy/d and 0.096 mGy/d is equivalent to 1% of the terrestrial and riparian, and aquatic ecological radiation benchmark, respectively. All predicted doses are well below this level. Therefore, it is unlikely that there would be significant adverse effects on human or ecological populations or communities as a result of radionuclide releases from the WRDF.

For non-radionuclides, human health doses are presented in Table 8.6.2-9, Table 8.6.2-10 and Table 8.6.2-11, for Harvester, On-site Farm, and Farm A receptors, respectively. Subsequent HQ are presented in Table 8.6.2-12 to Table 8.6.2-14. Exposures derived for selected terrestrial ecological health receptors, are presented in Table 8.6.2-15. Exposures for selected aquatic ecological receptors were assumed to equal the river concentration located 50 m downstream from the groundwater seep (Table 8.6.2-16). Non-radiological HQs are presented and compared to benchmark in Table 8.6.2-17 (aquatic) and Table 8.6.2-18 (terrestrial).

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Table 8.6.2-9: Doses for Non-Radionuclides to Harvester during Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	Unit	Dose by Pathway				Total Dose
			Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of WeeKay	
Adult	Cadmium	mg/kg bw/d	1.40E-07	6.09E-07	1.45E-05	2.94E-06	1.82E-05
	Lead	mg/kg bw/d	6.48E-06	1.13E-05	2.63E-05	2.02E-04	2.46E-04
Toddler	Cadmium	mg/kg bw/d	1.46E-07	3.93E-07	9.35E-06	4.14E-06	1.40E-05
	Lead	mg/kg bw/d	6.79E-06	7.28E-06	1.70E-05	1.08E-04	1.39E-04

mg/kg bw/d = milligram per kilogram body weight per day.

Table 8.6.2-10: Doses for Non-Radionuclides to On-site Farm during Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	Unit	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total Dose
Total River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	7.73E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	7.74E-05
	Lead	mg/kg bw/d	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	2.01E-02
Toddler	Cadmium	mg/kg bw/d	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.73E-09	5.01E-09	2.08E-09	7.51E-09	0.00E+00	1.33E-04
	Lead	mg/kg bw/d	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	3.44E-02
WRDF River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	3.20E-08	5.08E-13	2.90E-14	3.20E-13	9.79E-13	4.63E-11	1.04E-12	3.06E-12	2.74E-12	1.82E-12	3.76E-13	1.83E-14	3.21E-08
	Lead	mg/kg bw/d	4.44E-07	1.63E-13	9.30E-14	1.03E-13	9.32E-14	1.15E-10	1.54E-12	2.60E-11	8.75E-13	1.29E-11	4.74E-12	2.60E-14	4.44E-07
Toddler	Cadmium	mg/kg bw/d	5.49E-08	8.71E-12	5.65E-14	6.87E-13	1.03E-12	4.86E-11	4.31E-13	2.79E-12	2.08E-12	8.61E-13	3.11E-12	0.00E+00	5.50E-08
	Lead	mg/kg bw/d	7.61E-07	2.80E-12	1.81E-13	2.21E-13	9.81E-14	1.20E-10	6.34E-13	2.37E-11	6.65E-13	6.10E-12	3.92E-11	0.00E+00	7.61E-07

mg/kg bw/d = milligram per kilogram body weight per day.

Table 8.6.2-11: Doses for Non-Radionuclides to Farm A during Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	Unit	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total Dose
Total River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	7.72E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	7.74E-05
	Lead	mg/kg bw/d	8.35E-09	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	4.85E-07
Toddler	Cadmium	mg/kg bw/d	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.72E-09	5.01E-09	2.07E-09	7.50E-09	0.00E+00	1.33E-04
	Lead	mg/kg bw/d	1.43E-08	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	3.14E-06
WRDF River Contribution (Dose by Pathway)															
Adult	Cadmium	mg/kg bw/d	6.03E-10	9.55E-15	6.77E-15	6.02E-15	1.84E-14	8.71E-13	1.96E-14	5.76E-14	5.15E-14	3.42E-14	7.07E-15	3.44E-16	6.04E-10
	Lead	mg/kg bw/d	8.35E-09	3.07E-15	1.75E-15	1.93E-15	1.75E-15	2.16E-12	2.89E-14	4.89E-13	1.65E-14	2.42E-13	8.91E-14	4.89E-16	8.35E-09
Toddler	Cadmium	mg/kg bw/d	1.03E-09	1.64E-13	1.06E-15	1.29E-14	1.94E-14	9.14E-13	8.10E-15	5.25E-14	3.91E-14	1.62E-14	5.86E-14	0.00E+00	1.03E-09
	Lead	mg/kg bw/d	1.43E-08	5.26E-14	3.41E-15	4.14E-15	1.84E-15	2.26E-12	1.19E-14	4.46E-13	1.25E-14	1.15E-13	7.38E-13	0.00E+00	1.43E-08

mg/kg bw/d = milligram per kilogram body weight per day.

Table 8.6.2-12: Hazard Quotients for Harvester During Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	Total River Contribution (Including WRDF Project Contribution)					WRDF Project Contribution				
		Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of WeeKay	Total	Ingestion of Fish	Ingestion of Wild Waterfowl	Ingestion of Moose	Ingestion of WeeKay	Total
Adult	Cadmium	1.40E-04	6.09E-04	1.45E-02	2.94E-03	1.82E-02	2.95E-08	1.29E-07	3.06E-06	6.22E-07	3.84E-06
	Lead	3.50E-03	6.10E-03	1.42E-02	1.09E-01	1.33E-01	3.94E-08	6.87E-08	1.60E-07	4.65E-07	7.34E-07
Toddler	Cadmium	1.46E-04	3.93E-04	9.35E-03	4.14E-03	1.40E-02	3.09E-08	8.30E-08	1.98E-06	8.75E-07	2.97E-06
	Lead	3.67E-03	3.93E-03	9.17E-03	5.82E-02	7.49E-02	4.13E-08	4.43E-08	1.03E-07	6.55E-07	8.44E-07

Table 8.6.2-13: Hazard Quotients for On-Site Farm During Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution														
Adult	Cadmium	7.73E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	7.74E-02
	Lead	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	**1.09E+01**
Toddler	Cadmium	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.73E-06	5.01E-06	2.08E-06	7.51E-06	0.00E+00	1.33E-01
	Lead	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	**1.86E+01**
WRDF Project Contributions														
Adult	Cadmium	3.20E-05	5.08E-10	2.90E-11	3.20E-10	9.79E-10	4.63E-08	1.04E-09	3.06E-09	2.74E-09	1.82E-09	3.76E-10	1.83E-11	3.21E-05
	Lead	2.40E-04	8.82E-11	5.03E-11	5.56E-11	5.04E-11	6.20E-08	8.31E-10	1.41E-08	4.73E-10	6.96E-09	2.56E-09	1.40E-11	2.40E-04
Toddler	Cadmium	5.49E-05	8.71E-09	5.65E-11	6.87E-10	1.03E-09	4.86E-08	4.31E-10	2.79E-09	2.08E-09	8.61E-10	3.11E-09	0.00E+00	5.50E-05
	Lead	4.11E-04	1.51E-09	9.81E-11	1.19E-10	5.30E-11	6.50E-08	3.43E-10	1.28E-08	3.60E-10	3.30E-09	2.12E-08	0.00E+00	4.11E-04

Table 8.6.2-14: Hazard Quotients for Farm A During Post-Closure with WRDF Barrier Failure

Human Type	Non-radionuclide	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Total River Contribution														
Adult	Cadmium	7.72E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	7.74E-02
	Lead	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	**1.09E+01**
Toddler	Cadmium	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.72E-06	5.01E-06	2.07E-06	7.50E-06	0.00E+00	1.33E-01
	Lead	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	**1.86E+01**
WRDF Project Contributions														
Adult	Cadmium	6.03E-07	9.55E-12	6.77E-12	6.02E-12	1.84E-11	8.71E-10	1.96E-11	5.76E-11	5.15E-11	3.42E-11	7.07E-12	3.44E-13	6.04E-07
	Lead	4.51E-06	1.66E-12	9.45E-13	1.05E-12	9.46E-13	1.17E-09	1.56E-11	2.64E-10	8.90E-12	1.31E-10	4.82E-11	2.64E-13	4.51E-06
Toddler	Cadmium	1.03E-06	1.64E-10	1.06E-12	1.29E-11	1.94E-11	9.14E-10	8.10E-12	5.25E-11	3.91E-11	1.62E-11	5.86E-11	0.00E+00	1.03E-06
	Lead	7.73E-06	2.84E-11	1.84E-12	2.24E-12	9.97E-13	1.22E-09	6.45E-12	2.41E-10	6.76E-12	6.20E-11	3.99E-10	0.00E+00	7.74E-06

Table 8.6.2-15: Doses for Non-Radionuclides for Birds and Mammals in Post-Closure with WRDF Barrier Failure

Non-radionuclide	Unit	Barn Swallow	Horned Grebe	Trumpeter Swan	Wild Waterfowl	Little Brown Myotis	Mink	Moose
Cadmium	mg/kg bw/d	1.07E-03	3.71E-04	1.97E-02	2.44E-02	4.97E-04	1.92E-04	1.82E-02
HB-40	mg/kg bw/d	1.09E+00	4.52E-01	2.77E-02	1.23E-01	5.19E-01	2.66E-01	2.55E-02
Lead	mg/kg bw/d	7.76E-02	2.42E-02	5.13E-01	6.38E-01	2.86E-02	1.14E-02	4.72E-01
Xylene	mg/kg bw/d	2.06E-04	7.22E-05	1.23E-05	3.19E-05	9.76E-05	3.78E-05	1.13E-05

mg/kg bw/d = milligram per kilogram body weight per day.

Table 8.6.2-16: Exposure Point Concentrations for Non-Radionuclide COPCs for during Post-Closure with WRDF Barrier Failure

Non-radionuclide	Groundwater Concentration (µg/L)	Background Concentration (µg/L)	Project Contribution to River Concentrations at Groundwater Seep at River Bottom (µg/L)	River Concentration at Groundwater Seep at River Bottom (µg/L)	Project Contribution to River Concentrations at Groundwater Seep - 50 m Downstream (µg/L)	River Concentration at Groundwater Seep - 50 m Downstream (µg/L)
Cadmium	4.86E+00	1.00E-02	3.59E-05	1.00E-02	4.15E-06	1.00E-02
HB-40	5.41E+02	0.00E+00	4.00E-03	4.00E-03	4.61E-04	4.61E-04
Lead	6.74E+01	2.60E+00	4.98E-04	2.60E+00	5.75E-05	2.60E+00
Xylene	7.06E+01	NV	4.44E-04	4.44E-04	5.13E-05	5.13E-05

NV = no value available to calculate.

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Table 8.6.2-17: Non-Radionuclide Hazard Quotients for Aquatic Receptors in Post-Closure with WRDF Barrier Failure

Non-radionuclide	Benthic Invertebrates	Fish	Aquatic Plants
Cadmium	6.69E-02	5.90E-03	5.02E-03
HB-40	2.00E-04	4.65E-06	8.50E-06
Lead	2.12E-01	1.38E-01	5.20E-03
Xylene	4.44E-06	1.66E-07	1.14E-07

Table 8.6.2-18: Non-radionuclide Hazard Quotients for Birds and Mammals in Post-Closure with WRDF Barrier Failure

Non-radionuclide	Barn Swallow	Horned Grebe	Trumpeter Swan	Wild Waterfowl	Little Brown Myotis	Mink	Moose
Cadmium	5.36E-05	1.85E-05	9.87E-04	1.22E-03	4.97E-05	1.92E-05	1.82E-03
HB-40	NV	NV	NV	NV	2.07E-03	1.06E-03	1.02E-04
Lead	6.87E-03	2.15E-03	4.54E-02	5.65E-02	3.58E-04	1.42E-04	5.90E-03
Xylene	3.68E-06	1.29E-06	2.19E-07	5.71E-07	2.73E-07	1.06E-07	3.16E-08

Note:

NV = no value available to calculate HQ.

All radionuclides were below the public dose limit and the project dose constraint.

The HQs for the harvester, On-Site Farm, and Farm A are below the acceptable risk level of 0.2 for cadmium and lead for all ingestion pathways for the toddler and adult. The HQs for the On-site Farm and Farm A are below the acceptable risk level of 0.2 for cadmium and lead for all pathways, with the exception of lead from drinking water from the Winnipeg River. The HQs for all receptors are based on background plus project exposure. If only the project contribution is considered, the HQs for the toddler and adult for the On-site Farm and Farm A are well below the acceptable risk level of 0.2. This indicates that the project contribution to the total HQ is negligible and the exceedance is from existing background concentrations of lead in the Winnipeg River. Therefore, adverse effects to human or ecological receptors are not anticipated from WRDF failure.

The HQs for exposure of the ecological receptors to cadmium, lead, HB-40 and xylene are all below 1. Therefore, it is unlikely that there would be significant adverse effects on either aquatic or terrestrial populations or communities as a result of WRDF failure.

8.6.3 Well in Plume

This bounding scenario is identical to the Normal Evolution Scenario, except that the on-site farm receptors have a well located half-way between the WRDF and the Winnipeg River and use it for drinking water. The on-site farmer in the Normal Evolution Scenario receives drinking water from the Winnipeg River. In this bounding scenario the same farmer receives drinking water instead from a groundwater well in the groundwater plume from the WRDF.

Due to limitations on groundwater well capacity (i.e., inability to obtain sufficient groundwater for sampling in 2018 and 2019), water for other purposes, including bathing and irrigation of garden crops, is taken from the Winnipeg

River near the site. Calculations of well capacity were completed based on the methods in Driscoll (1995) for an overburden well (0.051 m radius) located in the basal till unit (i.e., the overburden unit with the greatest capacity for water production). For a well situated in this unit pumping at its maximum capacity it is reasonable to assume that the flow to that well would be governed by the average aquifer properties due to its radius of influence. Under these conditions the estimated well capacity is 0.02 m^3/d. This is adequate for use as drinking water but not for all water needs of the farmer receptor. A typical Canadian household uses approximately 0.3 m^3/day for all purposes. Therefore, all other water uses are still from the Winnipeg River directly adjacent to the site.

Further, a well in the groundwater plume was considered not feasible until after institutional control ends (i.e., 100 years after closure). During institutional control, long-term performance monitoring and maintenance activities will occur to demonstrate compliance with the safety case assumptions; therefore, a well in the groundwater plume is unlikely. Groundwater concentrations used to calculate radiological and non-radiological dose are based on mass loadings beyond the 100 years of institutional control.

8.6.3.1 Radiological Releases

In Table 8.6.3-1, the mass loading rate and peak times (i.e., time of maximum [year]) are presented for radionuclides in the groundwater well located half-way between the WRDF and the Winnipeg River. Peak times are presented in years after closure. All radionuclides were carried forward for assessment.

Table 8.6.3-1: Mass, Activity Loadings and Groundwater Concentrations – Well in Plume Bounding Scenario

Radionuclide	Maximum Loading Rate (g/y)	Time of Maximum (yr)	Release Rate (Bq/d)	Groundwater Concentration (Bq/L)
Actinium-225	1.75E-17	52,200	1.03E-04	5.41E-07
Actinium-227	2.13E-14	93,200	1.56E-04	8.23E-07
Bismuth-210	2.04E-14	120,400	2.57E-01	1.35E-03
Calcium-41	3.32E-06	10,500	2.88E+01	1.52E-01
Carbon-14	5.59E-04	951	2.53E+05	1.48E+03
Chlorine-36	1.36E-08	107	4.57E-02	2.69E-04
Gadolinium-152	1.88E-07	140	4.16E-10	2.44E-12
Iodine-129	1.70E-04	106	3.05E+00	1.79E-02
Lead-210	8.29E-13	121,200	6.42E-03	3.38E-05
Neptunium-237	1.04E-05	21,100	7.41E-01	3.90E-03
Nickel-59	6.43E-04	307,400	3.92E+03	2.06E+01
Niobium-94	2.88E-15	444,700	5.47E-08	2.88E-10
Plutonium-239	2.63E-06	65,600	1.65E+01	8.71E-02
Plutonium-240	1.74E-08	44,100	4.01E-01	2.11E-03
Polonium-210	1.46E-10	119,200	6.68E+01	3.51E-01
Protactinium-231	3.26E-11	93,400	1.56E-04	8.23E-07
Protactinium-233	4.68E-16	21,200	9.86E-04	5.19E-06
Radium-223	5.09E-16	93,200	2.64E-03	1.39E-05
Radium-224	6.68E-22	310,200	1.09E-08	5.72E-11
Radium-225	2.23E-16	52,000	8.81E-04	4.64E-06
Radium-226	1.47E-11	121,300	1.47E-03	7.75E-06

Table 8.6.3-1: Mass, Activity Loadings and Groundwater Concentrations – Well in Plume Bounding Scenario

Radionuclide	Maximum Loading Rate (g/y)	Time of Maximum (yr)	Release Rate (Bq/d)	Groundwater Concentration (Bq/L)
Radium-228	3.91E-19	304,800	1.09E-08	5.73E-11
Radon-222	4.3E-12	121,800	6.70E+01	3.53E-01
Samarium-148	1.27E-23	14,822,000	4.45E-28	2.34E-30
Silver-108m	3.95E-11	2,429	3.03E-02	1.68E-04
Technetium-99	3.23E-05	5,419	5.61E+01	2.95E-01
Thorium-227	9.26E-17	93,400	2.89E-04	1.52E-06
Thorium-228	2.84E-20	354,400	2.36E-09	1.24E-11
Thorium-229	8.94E-12	52,300	1.94E-04	1.02E-06
Thorium-230	1.54E-10	117,600	3.23E-04	1.70E-06
Thorium-231	9.07E-18	59,900	4.89E-04	2.57E-06
Thorium-232	2.12E-10	297,900	2.35E-09	1.24E-11
Thorium-234	1.47E-15	58,500	3.45E-03	1.82E-05
Tritium	4.69E-05	101	4.58E+07	2.69E+05
Uranium-233	3.09E-08	40,700	3.02E-02	1.59E-04
Uranium-234	1.77E-07	65,000	1.12E-01	5.89E-04
Uranium-235	3.13E-04	59,900	6.86E-02	3.61E-04
Uranium-236	1.43E-05	61,700	9.38E-02	4.93E-04
Uranium-238	1.42E-02	58,400	4.84E-01	2.55E-03

g/y = grams per year.

8.6.3.2 Non-Radiological Releases

Groundwater concentrations for non-radionuclides are presented in Table D-10 of the ERA (Ecometrix, 2021). Non-radionuclides were selected based on the human health screening criteria provided in Table 5-2 of the main ERA report. The non-radionuclides that exceed human health screening criteria are cadmium and lead.

Mass loading rates and groundwater concentrations for non-radionuclides in the groundwater well located half-way between the WRDF and the Winnipeg River are presented in Table 8.6.3-2.

Non-radionuclides were screened against effects criteria, for the human health screening. As with the Normal Evolution Scenario, cadmium and lead exceeded the screening criteria, and were carried forward for further analysis.

Table 8.6.3-2: Human Health Screening of Non-radionuclides – Well in Plume Bounding Scenario

Non-Radionuclide	Maximum Loading (g/a)	Time of Maximum (yr)	Concentration (µg/L)	Background Concentration (µg/L)	CDWS MAC	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Argon	5.81E-12	162	9.37E-11	Noble Gas – not applicable*							
Barium	3.65E-08	5,382,000	5.26E-07	1.10E+01	1,000	—	—	—	—	0.4	CDWS MAC
Bismuth	5.51E-09	46,400	7.95E-08	<2.00E-01	—	—	—	—	—	0.25	LC$_{50}$/100 – Borgmann et al., 2005
Boron	3.63E-04	2,260	5.52E-03	1.00E+01	5,000	—	1,500	200	1,200	—	CDWS MAC
Cadmium	7.42E-01	102,000	**1.07E+01**	1.00E-02	5	0.137	0.08	0.1	0.114	—	CDWS MAC
Chromium	8.01E-03	15,276,000	1.16E-01	1.70E+00	50	37.1	1.0 (VI)	1 (VI)	—	—	CDWS MAC
Cobalt	2.49E-02	74,700	3.59E-01	2.00E-01	—	—	—	0.9	4	—	PWQO Ontario
Copper	1.02E-03	444,600	1.47E-02	1.40E+01	—	4.3	2	5	2	—	CCME WQG
Gadolinium	7.09E-06	143	1.14E-04	—	—	—	—	—	—	1.5	LC$_{50}$/100 – Borgmann et al., 2005
HB-40	4.05E+01	10,200	5.84E+02	0	8,800^	—	—	—	—	—	See Note^
Helium	1.29E-02	101	2.07E-01	—	—	—	—	—	—	—	Noble Gas – not applicable*
Lead	5.05E+00	2,796,000	**7.28E+01**	2.60E+00	10	0.99	1	3	4.4		CDWS MAC
Manganese	1.62E-04	190,900	2.34E-03	1.10E+01	None - naturally occurring	—	—	—	794.2	110	WQG BC
Mercury	8.32E-05	3,282,000	1.20E-03	1.00E-02	1	1	0.026	0.2	—	—	CCME WQG
Molybdenum	1.17E-02	17,900	1.69E-01	2.00E-01	—	—	73	40	1,000	—	CCME WQG
Nickel	8.99E-05	430,900	1.30E-03	1.78E+00	—	25.5	25	25	—	—	CCME WQG
Nitrogen	5.44E-04	17,900	7.84E-03	—	1,000	—	—	—	3,000	—	CDWS MAC
Palladium	2.10E-01	61,600	3.02E+00	—	—	—	—	—	—	5.7	LC$_{50}$/100 – Borgmann et al., 2005
Potassium	3.39E-07	6,872	4.89E-06	9.07E+02	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996
Potassium hydroxide (as K)	5.76E-02	101	6.50E-01	—	—	—	—	—	—	5,300	LCV/10 – Suter and Tsao, 1996

Table 8.6.3-2: Human Health Screening of Non-radionuclides – Well in Plume Bounding Scenario

Non-Radionuclide	Maximum Loading (g/a)	Time of Maximum (yr)	Concentration (µg/L)	Background Concentration (µg/L)	CDWS MAC	WQSOG MB	CCME WQG	PWQO Ontario	WQG BC	Toxicity Benchmark	Selected Benchmark
Ruthenium	6.27E-09	232,600	9.04E-08	—	—	—	—	—	—	10	LC_{50}/100 – Borgmann et al., 2005
Samarium	1.21E-11	15,180,000	1.74E-10	-	-	-	-	-	-	0.74	LC50/100 – Borgmann et al. 2005
Sulphur (as SO_4)	1.45E-12	120	7.01E-11	—	—	—	—	—	218,000	—	WQG BC
Xenon	1.15E-09	116	1.86E-08	—	—	—	—	—	—	—	Noble Gas – not applicable
Xylene	4.81E+00	132	7.75E+01	—	90	—	—	2/40/30 (m/o/p)	30	—	CDWS MAC
Zirconium	3.43E-09	24,398,000	4.95E-08	-	-	-	-	4	-	-	PWQO Ontario

Note:

Bolded value indicates exceedance of human health screening criteria.

µg/L = microgram per Litre.

*Noble gases were assumed to volatilize rapidly.

^Derived drinking water limit based on a minimal effect level in mice of 250 mg/kg-day (Weeks 1974), divided by 1000, times 70 kg body weight, over 2 L/day of drinking water.

CDWS = Canadian Drinking Water Standard (Health Canada 2017).

WQSOG = Manitoba Water Quality Standards, Objectives and Guidelines (MWS 2011).

CCME WQG = Canadian Council of Ministers of the Environment Water Quality Guideline (CCME 1999).

PWQO = Ontario Provincial Water Quality Objective (MOEE 1994).

WQG BC = Water Quality Guideline British Columbia (BC MOE 2017).

8.6.3.3 Estimated Public Dose and Risk Characterization

The radiological dose for the On-site Farm residents who drink groundwater are detailed in Tables D-11 and D-12 of the ERA (EcoMetrix 2021). A summary of the total dose is provided in Table 8.6.3-3. For non-radionuclides, doses are provided in Table 8.6.3-4, and the hazard quotients are presented and compared to benchmarks in Table 8.6.3-5.

Table 8.6.3-3: Radionuclide Total Dose for On-Site Farm – Well in Plume Bounding Scenario

Age Group	Dose (mSv/a)	Percent of IAEA Lower Reference Level for Disruptive Events	Percent of IAEA Upper Reference Level for Disruptive Events
Adult	**2.54E+00**	**254.00%**	12.70%
10-year old Child	**1.34E+00**	**134.00%**	6.70%
1-year old Infant – Milk fed	4.23E-03	0.42%	0.02%
1-year old Infant – Formula fed	**1.96E+00**	**196.00%**	9.80%
3-month old – Nursing	**3.29E+00**	**329.00%**	16.45%
3-month old – Formula fed	**4.02E+00**	**402.00%**	20.10%

Note:
mSv/a = millisievert per year; <= less than.
Bolded cells indicate exceedance of the lower IAEA reference level for Disruptive Events of 1 mSv/a

For the residents of an on-site farm who drink groundwater, the total radiation dose from a disruptive event of a well in the plume from WRDF does not exceed the upper IAEA reference level for all receptors, but does exceed the lower IAEA reference level for all receptors, except the infant who drinks cow's milk. The dominant contributor to the total dose is from tritium through ingestion of water (i.e., drinking the groundwater). While Table 8.6.3-3 presents the results based on the conservative assumption that the on-site residents are exposed to the maximum groundwater concentrations for all COPCs in the plume at the same time, Figure 8.6.3-1.

Table 8.6.3-4 shows a more realistic representation of predicted dose rate to the on-site residents over the post-closure phase from the disruptive event. Around the 1,000-year time period the dose is dominated by carbon-14, and around the 100,000 year mark the dose is dominated by polonium-210. The dose to most receptors peak prior to the end of institutional control due to tritium and is estimated to potentially exceed the upper reference level for disruptive events; however, the well in plume scenario is only considered credible after the end of the 100-year institutional control stage which is the assessed peak dose for tritium in Table 8.6.3-1. Combined with the conservatism within the assessment model, there is no reasonable expectation that a member of the public would ever receive the peak dose rate depicted in Figure 8.6.3-1.

For non-radionuclides, the assessment demonstrates that human habitation with groundwater use for drinking water could result in exposures to cadmium and lead at levels where risks cannot be ruled out. The TRV for lead incorporates a safety factor of 2 to account for uncertainty, making the results conservative (EcoMetrix 2021). While the TRV for cadmium does not incorporate a safety factor, the assessment is considered conservative as it assumes that the maximum concentrations of COPCs occur at the same time, where in reality maximum concentrations occur at different timeframes during post-closure. Additionally, the likelihood of installing a groundwater well directly in the path of the potential plume is low.

Overall, the failure of passive controls to prevent human intrusion into the WRDF and/or a well in the groundwater plume used for drinking water during the post-institutional control period (i.e., 100 years after closure) is conceivable. The assessment was conservatively conducted to predict exposures to human receptors under these future unlikely conditions. However, it should be noted that for a well situated in this area, the well capacity is too low and therefore, cannot be used for purposes other than drinking. This is supported by observations during routine groundwater sampling campaigns at boreholes on the WL site (i.e., inability to obtain sufficient groundwater for sampling in 2018 and 2019)

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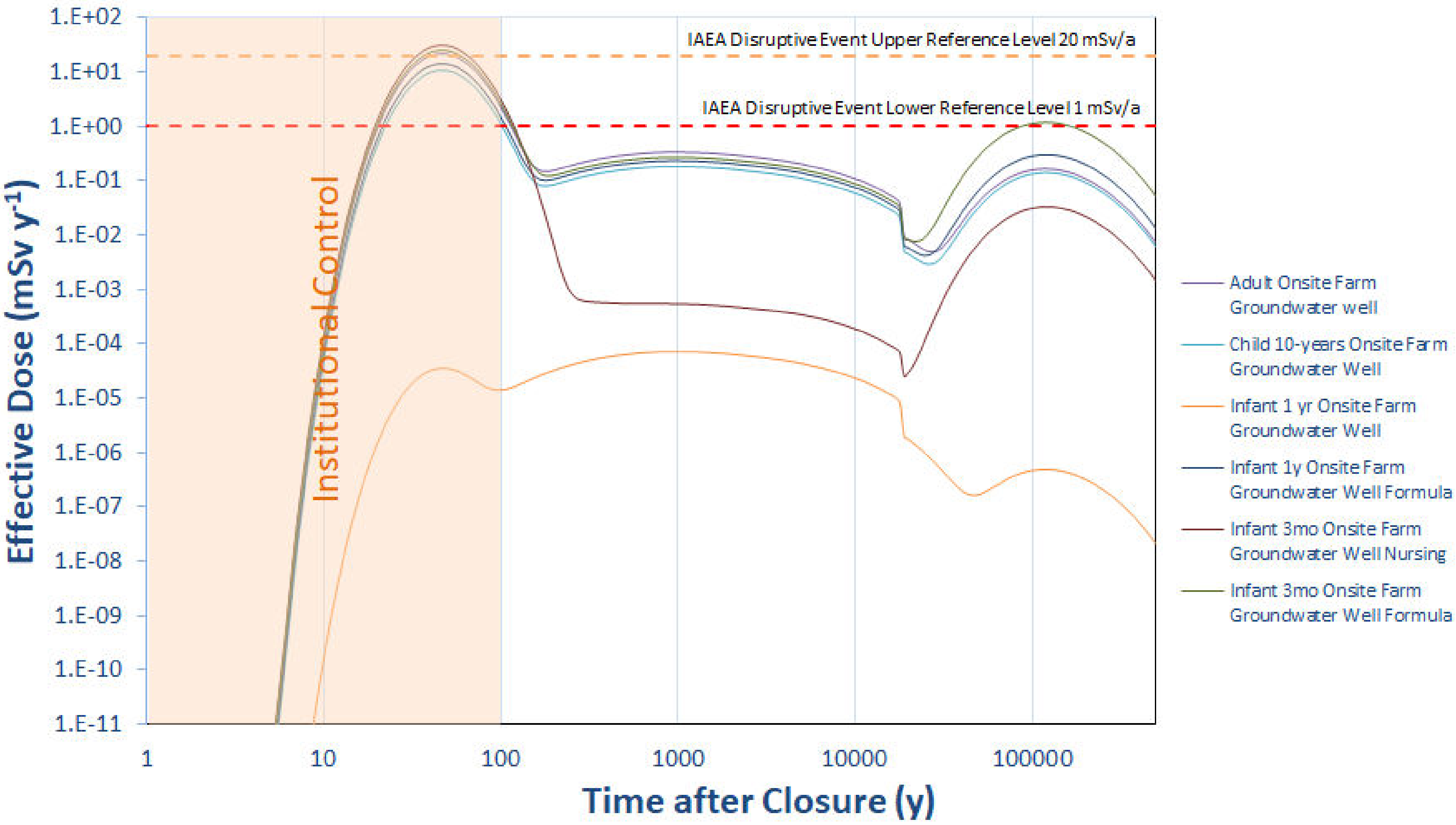

REFERENCE(S)

CLIENT
CANADIAN NUCLEAR LABORATORIES

CONSULTANT

GOLDER
MEMBER OF WSP

MM-YYYY	DECEMBER 2021
DESIGNED	PR
PREPARED	RRD
REVIEWED	KL
APPROVED	MM

PROJECT
DECOMMISSIONING SAFETY ASSESSMENT REPORT - IN SITU DECOMMISSIONING OF WR-1 PROJECT

TITLE
DOSE RATE TO HUMAN RECEPTORS - WELL IN PLUME BOUNDING SCENARIO

PROJECT NO.	CONTROL	REV.	FIGURE
20145046	0001	4	**8.6.3-1**

Table 8.6.3-4: Doses for Non-Radionuclides to On-site Farm with a Groundwater Well during Post-Closure

Human Type	Non-radionuclide	Unit	Dose by Pathway												Total Dose
			Groundwater Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	
Adult	Cadmium	mg/kg bw/d	8.26E-02	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	8.26E-02
	Lead	mg/kg bw/d	5.62E-01	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	5.62E-01
Toddler	Cadmium	mg/kg bw/d	1.42E-01	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.73E-09	5.01E-09	2.08E-09	7.51E-09	0.00E+00	1.42E-01
	Lead	mg/kg bw/d	9.64E-01	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	9.64E-01

Table 8.6.3-5: Hazard Quotients for On-site Farm with Groundwater Well during Post-Closure

Human Type	Non-radionuclide	Groundwater Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer	Total
Adult	Cadmium	**8.26E+01**	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	**8.26E+01**
	Lead	**3.04E+02**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	**3.04E+02**
Toddler	Cadmium	**1.42E+02**	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.73E-06	5.01E-06	2.08E-06	7.51E-06	0.00E+00	**1.42E+02**
	Lead	**5.21E+02**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	**5.21E+02**

Note:

Bold and shaded values indicate HQ greater than 0.2.

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9.0 RESULTS SUMMARY

9.1 Closure Phase

The HAZOP, FEPs Analysis and ERA completed for the closure phase illustrate that the operational controls, management practices, and standard operating procedures that will remain in place and govern closure activities, are adequate to provide the safety of workers, the public and the environment. The closure activities to be completed were not all evaluated under the CSR (AECL 2001a) nor are they all considered routine to the WL site (e.g., grouting of the PHT system). However, the occupational health and safety risk associated with closure activities, both conventional and contaminant exposure risks, are not ‘new’ to the site and have been successfully managed in recent history.

The HHRA confirms that the total radiation dose to all human receptors during closure activities (demolition prior to grouting and grouting) is well below the public dose limit of 1 mSv/a and dose constraint for the Project of 0.25 mSv/a for both maximum and average release rates. The EcoRA confirms there are no exceedances of the 2.4 mGy/d radiation benchmark for terrestrial and riparian biota on or near the WL site for both maximum and average release rates. Table 9.1-1 summarizes the ERA results for the closure phase.

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Table 9.1-1: Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Closure Phase

Receptor	Exposure Pathway	Environmental Media	Project Phase	Project Stage	Receptor Age Classification	Radiological Dose Exposure Calculation (mSv/a)		Health Assessment Benchmark	Comparison to Benchmarks (dose percent of benchmark)		Non-radiological Maximum Exposure Calculation
						Maximum	Average		Maximum	Average	
On-site Receptor	Inhalation	■ Air	Closure Phase	Demolition	Adult	6.04E-03	1.59E-03	0.25 mSv/a	2.42%	0.63%	No Non-Radiological COPCs Identified for Closure
	External	■ Soil (incidental)		Grouting	Adult	1.80E-04	9.76E-05		0.07%	0.04%	
Farm F and Farm A – full-time residency	Inhalation	■ Air		Demolition	Adult	3.46E-03	9.11E-04		1.39%	0.36%	
	Ingestion	■ Water (well water) ■ Soil (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products			10-year old	3.38E-03	8.89E-04		1.35%	0.36%	
					1-year old Milk-fed	2.61E-03	6.86E-04		1.04%	0.27%	
					1-year old Formula-fed	1.17E-03	3.07E-04		0.47%	0.12%	
					3-month old – Nursing	1.23E-03	3.23E-04		0.49%	0.13%	
					3-month old – Formula-fed	3.96E-04	1.04E-04		0.16%	0.04%	
	External	■ Air ■ Water (Winnipeg River) ■ Soil / Sediment		Grouting	Adult	1.01E-04	5.69E-05		0.04%	0.02%	
					10-year old	1.21E-04	6.79E-05		0.05%	0.03%	
					1-year old – Milk-fed	1.59E-04	8.88E-05		0.06%	0.04%	
					1-year old = Formula-fed	1.23E-04	6.72E-05		0.05%	0.03%	
					3-month old – Nursing	1.09E-04	6.30E-05		0.04%	0.03%	
					3-month old - Formula-fed	8.91E-05	4.89E-05		0.04%	0.02%	
Harvester - 2 hrs/wk on-site, downwind, and upwind from the Project activities	Inhalation	■ Air		Demolition	Adult	3.50E-03	9.19E-04		1.40%	0.37%	
	Ingestion	■ Terrestrial Animals (waterfowl, deer)			10-year old	1.52E-03	4.00E-04		0.61%	0.16%	
	External	■ Air			1-year old	6.66E-04	1.75E-04		0.27%	0.07%	
		■ Soil		Grouting	Adult	1.58E-05	8.82E-06		<0.01%	<0.01%	
					10-year old	1.69E-05	9.40E-06		<0.01%	<0.01%	
					1-year old	1.85E-05	1.04E-06		<0.01%	<0.01%	
Earthworm	Direct Contact	■ In Soil		Demolition	—	9.18E-05	2.41E-05	2.4 mGy/d	<0.01%	<0.01%	
				Grouting		9.29E-06	5.44E-06		<0.01%	<0.01%	
Grasses / Shrubs	Direct Contact	■ On Soil		Demolition	—	5.15E-04	1.35E-04		0.02%	<0.01%	
				Grouting		2.54E-05	1.39E-05		<0.01%	<0.01%	
Fruits	Direct Contact	■ On Soil		Demolition	—	3.95E-05	1.04E-05		<0.01%	<0.01%	
				Grouting		5.74E-06	3.57E-06		<0.01%	<0.01%	
American Robin	Direct Contact	■ On Soil		Demolition	—	5.27E-05	1.39E-05	2.4 mGy/d	<0.01%	<0.01%	
	Ingestion	■ Soil									
		■ Earthworms		Grouting		8.57E-06	4.82E-06		<0.01%	<0.01%	
		■ Fruit / Berries									
Loggerhead Shrike	Direct Contact	■ On Soil		Demolition	—	7.86E-05	2.07E-05		<0.01%	<0.01%	
	Ingestion	■ Soil									
		■ Earthworms									
		■ American Robin		Grouting		8.83E-06	4.93E-06		<0.01%	<0.01%	
		■ Meadow Vole									

Table 9.1-1: Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Closure Phase

Receptor	Exposure Pathway	Environmental Media	Project Phase	Project Stage	Receptor Age Classification	Radiological Dose Exposure Calculation (mSv/a)		Health Assessment Benchmark	Comparison to Benchmarks (dose percent of benchmark)		Non-radiological Maximum Exposure Calculation
						Maximum	Average		Maximum	Average	
Meadow Vole	Direct Contact	🟩 On Soil	Closure Phase	Demolition	—	9.71E-06	2.56E-06	2.4 mGy/d	<0.01%	<0.01%	No Non-Radiological COPCs Identified for Closure
	Ingestion	🟩 Soil									
		🟩 Grasses									
		🟩 Fruit / Berries		Grouting		8.04E-06	4.47E-06		<0.01%	<0.01%	
Common Shrew	Direct Contact	🟩 On Soil		Demolition	—	4.84E-05	1.27E-05		<0.01%	<0.01%	
	Ingestion	🟩 Soil									
		🟩 Earthworms		Grouting		8.98E-06	4.97E-06		<0.01%	<0.01%	
Snowshoe Hare	Direct Contact	🟩 On Soil		Demolition	—	1.70E-03	4.48E-04		0.07%	0.02%	
	Ingestion	🟩 Soil									
		🟩 Grasses									
		🟩 Fruit / Berries		Grouting		1.06E-05	5.84E-06		<0.01%	<0.01%	
		🟩 Shrubs									
White-tailed Deer	Direct Contact	🟩 On Soil		Demolition	—	1.38E-04	3.62E-05		<0.01%	<0.01%	
	Ingestion	🟩 Soil									
		🟩 Grasses									
		🟩 Fruit / Berries		Grouting		1.03E-05	5.71E-06		<0.01%	<0.01%	
Red Fox	Direct Contact	🟩 On Soil		Demolition	—	1.28E-04	3.37E-05		<0.01%	<0.01%	
	Ingestion	🟩 Soil									
		🟩 Grasses									
		🟩 Fruit / Berries									
		🟩 American Robin		Grouting		7.31E-06	4.03E-06		<0.01%	<0.01%	
		🟩 Loggerhead Shrike									
		🟩 Meadow Vole									
		🟩 Snowshoe Hare									
Little Brown Myotis	Direct Contact	🟩 On Soil		Demolition	—	1.50E-05	3.95E-06	2.4 mGy/d	<0.01%	<0.01%	
	Ingestion	🟩 Soil									
		🟩 Earthworms		Grouting		4.49E-06	2.59E-06		<0.01%	<0.01%	

Note:
Maximum and average dose during demolition activities prior to grouting (based on maximum and average airborne emission calculations [EIS Section 6.2]).
Maximum and average dose during grouting activities (based on maximum and average airborne emission calculations [EIS Section 6.2]).
Hrs/wk = hours per week; mSv/a = millisievert per year; COPC = constituents of potential concern; mGy/d = milligray per day.

9.2 Post-closure Phase

The FEPs Analysis and ERA completed for the post-closure phase, illustrate that the Project design is adequate to confirm the end-state protects the safety of workers, the public and the environment, demonstrating safety over the long-term. Table 9.2-1, Table 9.2-2 and Table 9.2-3 summarize the radiological and non-radiological ERA results for the post-closure phase – Normal Evolution Scenario, respectively. Table 9.2-4, Table 9.2-5 and Table 9.2-6 summarize the radiological and non-radiological ERA results for the bounding scenarios for the post-closure phase.

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Table 9.2-1: Radiological Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Post-closure Phase - Normal Evolution Scenario

Receptor	Exposure Pathway	Environmental Media	Receptor Type	Receptor Age Classification	Radiological Dose Exposure Calculation	Unit	Health Assessment Benchmark	Comparison to Benchmarks (dose percent of benchmark)
Farm F and Farm A - full-time residency	Inhalation	■ Air	Human receptors	Adult	3.35E-04	mSv/a	0.25 mSv/a	0.13%
	Ingestion	■ Water (well water) ■ Soil/Sediment (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products ■ Aquatic Animals (fish)		10-year old	3.36E-04			0.13%
				1-year old – Milk Fed	4.37E-04			0.17%
				1-year old – Formula	4.36E-04			0.17%
				3-month old – Nursing	1.04E-05			<0.01%
	External	■ Air ■ Water (Winnipeg River) ■ Soil / Sediment		3-month old – Formula-fed	6.85E-08			<0.01%
On-site Farm - full-time residency	Inhalation	■ Air		Adult	3.24E-03			1.30%
	Ingestion	■ Water (Winnipeg River) ■ Soil/Sediment (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products ■ Aquatic Animals (fish)		10-year old	3.25E-03			1.30%
				1-year old – Milk Fed	4.23E-03			1.69%
				1-year old – Formula-fed	4.18E-03			1.67%
				3-month old – Nursing	5.48E-04			0.22%
	External	■ Air ■ Water (Winnipeg River) ■ Soil / Sediment		3-month old – Formula-fed	3.65E-06			<0.01%
Harvester - 2 hrs/wk on-site, downwind, and upwind from the Project activities	Inhalation	■ Air		Adult	4.75E-05			0.02%
	Ingestion	■ Terrestrial Animals (waterfowl, moose) ■ Aquatic Animals (fish)		10-year old	3.05E-05			0.01%
	External	■ Air ■ Soil		1-year old	2.00E-05			0.01%
Lake Sturgeon	Direct Contact	■ In Water ■ On Sediment	Aquatic ecological receptor	—	1.14E-04	mGy/d	9.6 mGy/d	<0.01%
Carmine Shiner	Direct Contact	■ In Water ■ On Sediment		—	1.14E-04			<0.01%
Walleye	Direct Contact	■ In Water		—	4.10E-06			<0.01%
Benthic Invertebrate	Direct Contact	■ In Water		—	5.77E-04	—	—	<0.01%

Table 9.2-1: Radiological Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Post-closure Phase - Normal Evolution Scenario

Receptor	Exposure Pathway	Environmental Media	Receptor Type	Receptor Age Classification	Radiological Dose Exposure Calculation	Unit	Health Assessment Benchmark	Comparison to Benchmarks (dose percent of benchmark)
Horned Grebe	Direct Contact	■ On Sediment	Terrestrial ecological receptor	—	1.55E-04	mGy/d	2.4 mGy/d	<0.01%
	Ingestion	■ Water ■ Sediment ■ Fish (forage) ■ Benthic Invertebrates						
Trumpeter Swan	Direct Contact	■ On Sediment		—	1.53E-04	mGy/d	2.4 mGy/d	<0.01%
	Ingestion	■ Water ■ Sediment ■ Aquatic Plants						
Mallard	Direct Contact	■ On Sediment		—	1.52E-04			<0.01%
	Ingestion	■ Water ■ Sediment ■ Aquatic Plants ■ Benthic Invertebrates						
Mink	Direct Contact	■ On Sediment		—	1.54E-04			<0.01%
	Ingestion	■ Water ■ Sediment ■ Benthic Invertebrates ■ Fish (forage)						
Barn Swallow	Direct Contact	■ On Sediment		—	8.13E-06			<0.01%
	Ingestion	■ Water ■ Sediment ■ Benthic Invertebrates						
Little Brown Myotis	Ingestion	■ Water ■ Benthic Invertebrates		—	8.13E-06			<0.01%
Moose	Direct Contact	■ On Sediment		—	1.01E-05	—	—	<0.01%
	Ingestion	■ Water ■ Sediment ■ Aquatic Plants ■ Grasses		—		mGy/d	—	

Note:

Maximum and average dose during demolition activities prior to grouting (based on maximum and average airborne emission calculations [EIS Section 6.2])

Maximum and average dose during grouting activities (based on maximum and average airborne emission calculations [EIS Section 6.2])

— indicates not applicable.

Table 9.2-2: Non-Radionuclide Dose and Risk by Exposure Pathways Evaluation for Human Receptors during Post-closure Phase - Normal Evolution Scenario

Receptor	Non-radionuclide	Variable	Receptor Age Classification	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer/Moose	Ingestion of Waterfowl	Ingestion of Weekay	Total	Health Assessment Exposure Protective Benchmark	Hazard Quotients	Health Assessment Hazard Quotient Protective Benchmark
Farm F and Farm A - full-time residency	Cadmium	Exposure Dose (mg/kg bw/d)	Adult	7.72E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	—	—	7.74E-05	1.00E-03	Presented Within Body of the Table	—
			Toddler	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.73E-09	5.01E-09	2.08E-09	7.51E-09	0.00E+00	—	—	1.33E-04			
On-site Farm – full-time residency			Adult	7.73E-05	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	—	—	7.74E-05			
			Toddler	1.32E-04	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.72E-09	5.01E-09	2.08E-09	7.50E-09	0.00E+00	—	—	1.33E-04			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	1.40E-07	—	—	—	—	—	1.45E-05	6.09E-07	2.94E-06	1.82E-05			
			Toddler	—	—	—	—	—	1.46E-07	—	—	—	—	—	9.35E-06	3.93E-07	4.14E-06	1.40E-05			
Farm F and Farm A - full-time residency		Contribution from the Project Hazard Quotients (unitless)	Adult	6.03E-07	9.55E-12	6.77E-12	6.02E-12	1.84E-11	8.72E-10	1.96E-11	5.76E-11	5.15E-11	3.42E-11	7.07E-12	3.44E-13	—	—	6.04E-07	—		2.00E-01
			Toddler	1.03E-06	1.64E-10	1.06E-12	1.29E-11	1.94E-11	9.14E-10	8.11E-12	5.25E-11	3.91E-11	1.62E-11	5.86E-11	0.00E+00	—	—	1.03E-06			
On-site Farm – full-time residency			Adult	3.20E-05	5.08E-10	2.90E-11	3.20E-10	9.79E-10	4.63E-08	1.04E-09	3.06E-09	2.74E-09	1.82E-09	3.76E-10	1.83E-11	—	—	3.21E-05			
			Toddler	5.49E-05	8.71E-09	5.65E-11	6.87E-10	1.03E-09	4.86E-08	4.31E-10	2.79E-09	2.08E-09	8.61E-10	3.11E-09	0.00E+00	—	—	5.50E-05			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	2.95E-08	—	—	—	—	—	3.06E-06	1.29E-07	6.22E-07	3.84E-06			
			Toddler	—	—	—	—	—	3.09E-08	—	—	—	—	—	1.98E-06	8.30E-08	8.75E-07	2.97E-06			
Farm F and Farm A - full-time residency		Total River + Contribution from the Project Hazard Quotients (unitless)	Adult	7.72E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	—	—	7.74E-02	—		2.00E-01
			Toddler	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.72E-06	5.01E-06	2.07E-06	7.50E-06	0.00E+00	—	—	1.33E-01			
On-site Farm – full-time residency			Adult	7.73E-02	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	—	—	7.74E-02			
			Toddler	1.32E-01	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.73E-06	5.01E-06	2.08E-06	7.51E-06	0.00E+00	—	—	1.33E-01			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	1.40E-04	—	—	—	—	—	1.45E-02	6.09E-04	2.94E-03	1.82E-02			
			Toddler	—	—	—	—	—	1.46E-04	—	—	—	—	—	9.35E-03	3.93E-04	4.14E-03	1.40E-02			
Farm F and Farm A - full-time residency	Lead	Exposure Dose (mg/kg bw/d)	Adult	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	—	—	2.01E-02	1.85E-03		—
			Toddler	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	—	—	3.44E-02			
On-site Farm – full-time residency			Adult	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	—	—	2.01E-02			
			Toddler	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	—	—	3.44E-02			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	6.48E-06	—	—	—	—	—	2.63E-05	1.13E-05	2.02E-04	2.46E-04			
			Toddler	—	—	—	—	—	6.79E-06	—	—	—	—	—	1.70E-05	7.28E-06	1.08E-04	1.39E-04			
Farm F and Farm A - full-time residency		Contribution from the Project Hazard Quotients (unitless)	Adult	4.43E-06	1.66E-12	9.45E-13	1.05E-12	9.46E-13	5.71E-11	7.90E-13	1.33E-11	4.44E-13	6.60E-12	2.41E-12	1.32E-14	—	—	4.43E-06	—		2.00E-01
			Toddler	7.59E-06	2.84E-11	1.84E-12	2.24E-12	9.97E-13	5.99E-11	3.26E-13	1.21E-11	3.37E-13	3.12E-12	2.00E-11	0.00E+00	—	—	7.59E-06			
On-site Farm – full-time residency			Adult	2.36E-04	8.82E-11	5.03E-11	5.56E-11	5.04E-11	3.04E-09	4.20E-11	7.08E-10	2.36E-11	3.51E-10	1.28E-10	7.04E-13	—	—	2.36E-04			
			Toddler	4.04E-04	1.51E-09	9.81E-11	1.19E-10	5.30E-11	3.18E-09	1.73E-11	6.46E-10	1.79E-11	1.66E-10	1.06E-09	0.00E+00	—	—	4.04E-04			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	3.87E-08	—	—	—	—	—	1.57E-07	6.74E-08	4.57E-07	7.20E-07			
			Toddler	—	—	—	—	—	4.06E-08	—	—	—	—	—	1.01E-07	4.35E-08	6.43E-07	8.28E-07			

Table 9.2-2: Non-Radionuclide Dose and Risk by Exposure Pathways Evaluation for Human Receptors during Post-closure Phase - Normal Evolution Scenario

Receptor	Non-radionuclide	Variable	Receptor Age Classification	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of Deer/Moose	Ingestion of Waterfowl	Ingestion of Weekay	Total	Health Assessment Exposure Protective Benchmark	Hazard Quotients	Health Assessment Hazard Quotient Protective Benchmark
Farm F and Farm A - full-time residency	Lead	Total River + Contribution from the Project Hazard Quotients (unitless)	Adult	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	—	—	**1.09E+01**	—	Presented Within Body of the Table	2.00E-01
			Toddler	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	—	—	**1.86E+01**			
On-site Farm – full-time residency			Adult	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	—	—	**1.09E+01**			
			Toddler	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	—	—	**1.86E+01**			
Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities			Adult	—	—	—	—	—	3.50E-03	—	—	—	—	—	1.42E-02	6.10E-03	1.09E-01	1.33E-01			
			Toddler	—	—	—	—	—	3.67E-03	—	—	—	—	—	9.17E-03	3.93E-03	5.82E-02	7.49E-02			

Note:

hrs/wk = hours per week; mSv/a = millisieverts per year; mGy/d = milligray per day; mg/kg bw/d = milligram per kilogram of body weight per day.

Bolded font and shaded cell = calculated HQ exceeds the protective benchmark.

— indicates not applicable.

Table 9.2-3: Non-Radionuclide Dose and Risk by Exposure Pathways Evaluation for Ecological Receptors during Post-closure Phase - Normal Evolution Scenario

Receptor	Non-radionuclide	Concentration (mg/L)	Total Dose (mg/kg bw d)	Health Assessment Exposure Protective Benchmark (mg/L or mg/kg bw d)	Hazard Quotient	Health Assessment Hazard Quotient Protective Benchmark
Benthic Invertebrates (seep)	Cadmium	5.35E-03	—	1.50E-04	**3.57E+01**	1.00E+00
Fish		1.00E-05	—	1.70E-03	5.90E-03	
Aquatic Plants		1.00E-05	—	2.00E-03	5.02E-03	
Barn Swallow		—	1.07E-03	2.00E+01	5.36E-05	
Horned Grebe		—	3.71E-04	2.00E+01	1.85E-05	
Trumpeter Swan		—	1.97E-02	2.00E+01	9.87E-04	
Wild Waterfowl		—	2.44E-02	2.00E+01	1.22E-03	
Little Brown Myotis		—	4.97E-04	1.00E+01	4.97E-05	
Mink		—	1.92E-04	1.00E+01	1.92E-05	
Moose		—	1.82E-02	1.00E+01	1.82E-03	
Benthic Invertebrates (seep)	HB-40	5.84E-01	—	2.00E-02	**2.92E+01**	
Fish		3.92E-06	—	8.60E-01	4.56E-06	
Aquatic Plants		3.92E-06	—	4.70E-01	8.35E-06	
Barn Swallow		—	1.07E+00	NV	NV	NV
Horned Grebe		—	4.43E-01	NV	NV	
Trumpeter Swan		—	2.72E-02	NV	NV	
Wild Waterfowl		—	1.21E-01	NV	NV	
Little Brown Myotis		—	5.09E-01	2.50E+02	2.04E-03	1.00E+00
Mink		—	2.61E-01	2.50E+02	1.04E-03	
Moose		—	2.50E-02	2.50E+02	1.00E-04	
Benthic Invertebrates (seep)	Lead	7.27E-02	—	1.23E-02	**5.93E+00**	
Fish		2.60E-03	—	1.89E-02	1.38E-01	
Aquatic Plants		2.60E-03	—	5.00E-01	5.20E-03	
Barn Swallow		—	7.76E-02	1.13E+01	6.87E-03	
Horned Grebe		—	2.42E-02	1.13E+01	2.15E-03	
Trumpeter Swan		—	5.13E-01	1.13E+01	4.54E-02	
Wild Waterfowl		—	6.38E-01	1.13E+01	5.65E-02	
Little Brown Myotis		—	2.86E-02	8.00E+01	3.58E-04	
Mink		—	1.14E-02	8.00E+01	1.42E-04	
Moose		—	4.72E-01	8.00E+01	5.90E-03	
Benthic Invertebrates (seep)	Xylene	6.96E-02	—	1.00E-01	6.96E-01	
Fish		4.18E-07	—	2.68E+00	1.56E-07	
Aquatic Plants		4.18E-07	—	3.90E+00	1.07E-07	
Barn Swallow		—	1.94E-04	5.59E+01	3.46E-06	
Horned Grebe		—	6.79E-05	5.59E+01	1.21E-06	
Trumpeter Swan		—	1.15E-05	5.59E+01	2.06E-07	
Wild Waterfowl		—	3.00E-05	5.59E+01	5.37E-07	
Little Brown Myotis		—	9.19E-05	3.57E+02	2.57E-07	
Mink		—	3.56E-05	3.57E+02	9.96E-08	
Moose		—	1.06E-05	3.57E+02	2.98E-08	

Bolded font and shaded cell = calculated HQ exceeds the protective benchmark.

— indicates not applicable; NV = no value available to calculate.

Table 9.2-4: Radiological Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Post-closure Phase - Bounding Scenarios

Scenario	Receptor	Exposure Pathway	Environmental Media	Receptor Type	Receptor Age Classification	Exposure (mSv/a)	Protective Benchmark	Percent of IAEA Lower Reference Level for Disruptive Events	Percent of IAEA Upper Reference Level for Disruptive Events
Unsealed Borehole / Human Intrusion	On-site Worker (Driller) - 1 hrs	Inhalation	Air	Human Receptors	Adult	6.35E-03	1 mSv/a (IAEA Lower Reference Level for Disruptive Events) 20 mSv/a (IAEA Upper Reference Level for Disruptive Events)	1%	0%
		Dermal Contact	Soil						
		Groundshine							
	Trespasser - exposed 1 hr daily	Inhalation	Air		Adult	1.98E-01		20%	1%
		Dermal Contact	Soil		10-year old	2.01E-01		20%	1%
		Groundshine			1-year old	3.19E-01		32%	2%
					3-month old	2.86E-01		29%	1%
WRDF Barrier Failure	On-site Farm – full-time residency	Inhalation	Air		Adult	3.24E-03		0.32%	0.02%
		Ingestion	■ Water (Winnipeg River) ■ Soil (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products		10-year old	3.25E-03		0.33%	0.02%
					1-year old – Milk-fed	4.23E-03		0.42%	0.02%
					1-year old – Formula-fed	4.18E-03		0.42%	0.02%
					3-month old – Nursing	5.58E-04		0.06%	<0.01%
					3-month old – Formula-fed	3.79E-06		<0.01%	<0.01%
WRDF Barrier Failure	Farm F and Farm A - full-time residency	Inhalation	Air		Adult	3.35E-04		0.03%	<0.01%
		Ingestion	■ Water (well water) ■ Soil (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products		10-year old	3.36E-04		0.03%	<0.01%
					1-year old – Milk-fed	4.37E-04		0.04%	<0.01%
					1-year old – Formula-fed	4.36E-04		0.04%	<0.01%
					3-month old – Nursing	1.06E-05		<0.01%	<0.01%
					3-month old – Formula-fed	7.12E-08		<0.01%	<0.01%
WRDF Barrier Failure	Harvester - 2 hrs/wk on-site, downwind, and upwind from the Project activities	Inhalation	Air		Adult	4.81E-05		<0.01%	<0.01%
		Ingestion	Terrestrial Animals (waterfowl, moose)		10-year old	3.09E-05		<0.01%	<0.01%
		External	Air		1-year old	2.03E-05		<0.01%	<0.01%
			Soil						
Well in Plume – Groundwater Consumption	On-site Farm – full-time residency	Inhalation	Air		Adult	**2.54E+00**		**254.00%**	12.70%
		Ingestion	■ Water (well water) ■ Soil (incidental) ■ Terrestrial Plants (homegrown) ■ Terrestrial Animals and Animal Products ■ Soil		10-year old	**1.34E+00**		**134.00%**	6.70%
					1-year old – Milk-fed	4.23E-03		0.42%	0.02%
					1-year old – Formula-fed	**1.96E+00**		**196.00%**	9.80%
					3-month old – Nursing	**3.29E+00**		**329.00%**	16.45%
					3-month old – Formula-fed	**4.02E+00**		**402.00%**	20.10%
		Dermal	Soil						

Table 9.2-4: Radiological Exposure Pathways Evaluation for Receptors (Human and Ecological Health) during Post-closure Phase - Bounding Scenarios

Scenario	Receptor	Exposure Pathway	Environmental Media	Receptor Type	Receptor Age Classification	Exposure (mSv/a)	Protective Benchmark	Percent of IAEA Lower Reference Level for Disruptive Events	Percent of IAEA Upper Reference Level for Disruptive Events
WRDF Barrier Failure	Lake Sturgeon	Direct Contact	■ In Water ■ On Sediment	Aquatic ecological receptors	—	1.14E-04	9.6 mGy/d	<0.01%	<0.01%
	Carmine Shiner	Direct Contact	■ In Water ■ On Sediment		—	1.14E-04	9.6 mGy/d	<0.01%	<0.01%
	Walleye	Direct Contact	In Water		—	4.17E-06		<0.01%	<0.01%
	Benthic Invertebrate	Direct Contact	■ In Water ■ In Sediment		—	5.77E-04		<0.01%	<0.01%
	Horned Grebe	Direct Contact	On Sediment	Terrestrial ecological receptors	—	1.55E-04	2.4 mGy/d	<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Fish (forage) ■ Benthic Invertebrates						
	Trumpeter Swan	Direct Contact	On Sediment		—	1.53E-04		<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Aquatic Plants						
	Mallard	Direct Contact	On Sediment		—	1.52E-04		<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Aquatic Plants ■ Benthic Invertebrates						
	Mink	Direct Contact	On Sediment		—	1.54E-04		<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Benthic Invertebrates ■ Fish (forage)						
	Barn Swallow	Direct Contact	On Sediment		—	8.28E-06		<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Benthic Invertebrates						
	Little Brown Myotis	Ingestion	■ Water ■ Benthic Invertebrates		—	6.70E-06		<0.01%	<0.01%
	Moose	Direct Contact	On Sediment		—	1.02E-05		<0.01%	<0.01%
		Ingestion	■ Water ■ Sediment ■ Aquatic Plants ■ Grasses						

Note

hr = hour; hrs/wk = hours per week; mSv/a= millisieverts per year; mGy/d = milligray per day.

Bolded font = calculated value exceeds the lower protective benchmark.

Shaded cell = calculated value exceeds upper protective benchmark.

— indicates not applicable.

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Table 9.2-5: Non-radionuclide Dose and Risk by Exposure Pathways for Human Receptors during Post-closure Phase – Bounding Scenarios

Scenario	Non-radionuclide	Variable	Receptor	Receptor Age Classification	Groundwater Ingestion	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of /Moose	Ingestion of Waterfowl	Ingestion of WeeKay	Total	Health Assessment Exposure Protective Benchmark	Hazard Quotients	Health Assessment Hazard Benchmark Protective Benchmark
WRDF Barrier Failure	Lead	Exposure Dose (mg/kg bw/d)	Farm F and Farm A – full time residency	Adult	—	8.35E-09	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	—	—	4.85E-07	1.85E 03	Presented Within Body of the Table	2.00E 01
			Farm F and Farm A – full time residency	Toddler	—	1.43E-08	2.58E-06	1.67E-07	2.03E-07	9.04E-08	5.44E-06	2.96E-08	1.10E-06	3.06E-08	2.83E-07	1.82E-06	0.00E+00	—	—	3.14E-06			
			On-site Farm – full time residency	Adult	—	2.01E-02	1.50E-07	8.58E-08	9.49E-08	8.59E 08	5.19E 06	7.17E 08	1.21E-06	4.03E-08	5.99E 07	2.19E-07	1.20E 09	—	—	2.01E-02			
			On-site Farm – full time residency	Toddler	—	3.44E-02	2.58E-06	1.67E-07	2.03E-07	9.04E 08	5.44E 06	2.96E 08	1.10E-06	3.06E-08	2.83E 07	1.82E-06	0.00E+00	—	—	3.44E-02			
			Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities	Adult	—	—	—	—	—	—	6.48E-06	—	—	—	—	—	2.63E-05	1.13E-05	2.02E-04	2.46E-04			
			Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities	Toddler	—	—	—	—	—	—	6.79E-06	—	—	—	—	—	1.70E-05	7.28E-06	1.08E-04	1.39E-04			
		Hazard Quotient (unitless)	Farm F and Farm A – full time residency	Adult	—	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	—	—	**1.09E+01**	—	Presented Within Body of the Table	2.00E 01
			Farm F and Farm A – full time residency	Toddler	—	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	—	—	**1.86E+01**			
			On-site Farm – full time residency	Adult	—	**1.09E+01**	8.13E-05	4.64E-05	5.13E-05	4.64E-05	2.80E-03	3.87E-05	6.54E-04	2.18E-05	3.24E-04	1.18E-04	6.49E-07	—	—	**1.09E+01**			
			On-site Farm – full time residency	Toddler	—	**1.86E+01**	1.39E-03	9.04E-05	1.10E-04	4.89E-05	2.94E-03	1.60E-05	5.96E-04	1.66E-05	1.53E-04	9.81E-04	0.00E+00	—	—	**1.86E+01**			
			Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities	Adult	—	—	—	—	—	—	3.50E-03	—	—	—	—	—	1.42E-02	6.10E-03	1.09E-01	1.33E-01			
			Harvester – 2 hrs/wk on-site, downwind, and upwind from the Project activities	Toddler	—	—	—	—	—	—	3.67E-03	—	—	—	—	—	9.17E-03	3.93E-03	5.82E-02	7.49E-02			

Note:

hrs/wk = hours per week; mg/kg bw/d = milligram per kilogram of body weight per day.

Bolded font and shaded cell = calculated HQ exceeds the protective benchmark.

— indicates not applicable.

Table 9.2-5: Non-radionuclide Dose and Risk by Exposure Pathways for Human Receptors during Post-closure Phase – Bounding Scenarios (cont'd)

Scenario	Non-radionuclide	Variable	Receptor	Receptor Age Classification	Groundwater Ingestion	River Water Ingestion	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of /Moose	Ingestion of Waterfowl	Ingestion of WeeKay	Total	Health Assessment Exposure Protective Benchmark	Hazard Quotients	Health Assessment Hazard Benchmark Protective Benchmark
Human Intrusion	HB-40	Exposure Dose (mg/kg bw/d)	On-site Receptor	Adult	—	—	8.80E-01	1.14E-01	2.89E-02	—	—	—	—	—	—	—	—	—	—	1.02E+00	2.50E+02	Presented Within Body of the Table	2.00E-01
				Toddler	—	—	5.13E-02	5.85E-02	1.35E-02	—	—	—	—	—	—	—	—	—	—	1.23E-01			
				Driller	—	—	1.66E-03	1.89E-02	NA	—	—	—	—	—	—	—	—	—	—	2.05E-02			
		Hazard Quotient (unitless)		Adult	—	—	**3.52E+00**	**4.57E-01**	1.16E-01	—	—	—	—	—	—	—	—	—	—	**4.09E+00**	—		
				Toddler	—	—	**2.05E-01**	**2.34E-01**	5.40E-02	—	—	—	—	—	—	—	—	—	—	**4.93E-01**			
				Driller	—	—	6.62E-03	7.55E-02	NA	—	—	—	—	—	—	—	—	—	—	8.21E-02			
	Lead	Exposure Dose (mg/kg bw/d)	On-site Receptor	Adult	—	—	4.09E-01	2.65E-02	1.34E-02	—	—	—	—	—	—	—	—	—	—	4.49E-01	1.85E-03		2.00E-01
				Toddler	—	—	2.39E-02	1.36E-02	6.28E-03	—	—	—	—	—	—	—	—	—	—	4.38E-02			
				Driller	—	—	7.70E-04	4.39E-03	NA	—	—	—	—	—	—	—	—	—	—	5.16E-03			
		Hazard Quotient (unitless)		Adult	—	—	**3.41E+02**	**2.21E+01**	**1.12E+01**	—	—	—	—	—	—	—	—	—	—	**3.74E+02**	—		
				Toddler	—	—	**1.99E+01**	**1.13E+01**	**5.23E+00**	—	—	—	—	—	—	—	—	—	—	**3.65E+01**			
				Driller	—	—	**6.42E-01**	**3.66E+00**	NA	—	—	—	—	—	—	—	—	—	—	**4.30E+00**			

Note:

hrs/wk = hours per week; mg/kg bw/d = milligram per kilogram of body weight per day.

Bolded font and shaded cell = calculated HQ exceeds the protective benchmark.

— indicates not applicable.

Table 9.2-5: Non-radionuclide Dose and Risk by Exposure Pathways for Human Receptors during Post-closure Phase – Bounding Scenarios (cont'd)

Scenario	Non-radionuclide	Variable	Receptor	Receptor Age Classification	Groundwater Ingestion	River Water	Soil Ingestion	Soil Dermal Contact	Dust Inhalation	Ingestion of Plants	Ingestion of Fish	Ingestion of Beef	Ingestion of Poultry	Ingestion of Pork	Ingestion of Eggs	Ingestion of Milk	Ingestion of /Moose	Ingestion of	Ingestion of WeeKay	Total	Health Assessment Exposure Protective Benchmark	Hazard Quotients	Health Assessment Hazard Benchmark Protective Benchmark
Well in Plume	Cadmium	Exposure Dose (mg/kg bw/d)	On-site Receptor	Adult	8.26E-02	—	1.22E-09	6.98E-11	7.72E-10	2.36E-09	1.12E-07	2.52E-09	7.38E-09	6.60E-09	4.38E-09	9.06E-10	4.41E-11	—	—	8.26E-02	1.00E-03	Presented Within Body of the Table	2.00E-01
				Toddler	1.42E-01	—	2.10E-08	1.36E-10	1.65E-09	2.48E-09	1.17E-07	1.04E-09	6.73E-09	5.01E-09	2.08E-09	7.51E-09	0.00E+00	—	—	1.42E-01			
		Hazard Quotient (unitless)		Adult	**8.26E+01**	—	1.22E-06	6.98E-08	7.72E-07	2.36E-06	1.12E-04	2.52E-06	7.38E-06	6.60E-06	4.38E-06	9.06E-07	4.41E-08	—	—	**8.26E+01**	—		
				Toddler	**1.42E+02**	—	2.10E-05	1.36E-07	1.65E-06	2.48E-06	1.17E-04	1.04E-06	6.73E-06	5.01E-06	2.08E-06	7.51E-06	0.00E+00	—	—	**1.42E+02**			
	Lead	Exposure Dose (mg/kg bw/d)	On-site Receptor	Adult	5.62E-01	—	1.50E-07	8.58E-08	9.49E-08	8.59E-08	5.19E-06	7.17E-08	1.21E-06	4.03E-08	5.99E-07	2.19E-07	1.20E-09	—	—	5.62E-01	1.85E-03	Presented Within Body of the Table	2.00E-01

Table 9.2-6: Non-radionuclide Dose and Risk by Exposure Pathways for Ecological Receptors during Post-closure Phase – Bounding Scenarios

Scenario	Receptor	Non-radionuclide	Concentration (mg/L)	Total Dose (mg/kg bw d)	Health Assessment Exposure Protective Benchmark (mg/L or mg/kg bw d)	Hazard Quotient	Health Assessment Hazard Quotient Protective Benchmark
WRDF Barrier Failure	Benthic Invertebrates	Cadmium	1.00E-05	—	1.50E-04	6.69E-02	1.00E+00
	Fish		1.00E-05	—	1.70E-03	5.90E-03	
	Aquatic Plants		1.00E-05	—	2.00E-03	5.02E-03	
	Barn Swallow		—	1.07E-03	2.00E+01	5.90E-03	
	Horned Grebe		—	3.71E-04	2.00E+01	5.90E-03	
	Trumpeter Swan		—	1.97E-02	2.00E+01	5.90E-03	
	Wild Waterfowl		—	2.44E-02	2.00E+01	5.90E-03	
	Little Brown Myotis		—	4.97E-04	1.00E+01	5.90E-03	
	Mink		—	1.92E-04	1.00E+01	5.90E-03	
	Moose		—	1.82E-02	1.00E+01	5.90E-03	
	Benthic Invertebrates	HB-40	4.00E-06	—	2.00E-02	2.00E-04	
	Fish		4.00E-06	—	8.60E-01	4.65E-06	
	Aquatic Plants		4.00E-06	—	4.70E-01	8.50E-06	
	Barn Swallow		—	1.09E+00	NV	NV	NV
	Horned Grebe		—	4.52E-01	NV	NV	
	Trumpeter Swan		—	2.77E-02	NV	NV	
	Wild Waterfowl		—	1.23E-01	NV	NV	
	Little Brown Myotis		—	5.19E-01	2.50E+02	2.07E-03	1.00E+00
	Mink		—	2.66E-01	2.50E+02	1.06E-03	
	Moose		—	2.55E-02	2.50E+02	1.02E-04	
	Benthic Invertebrates	Lead	2.60E-03	—	1.23E-02	2.12E-01	
	Fish		2.60E-03	—	1.89E-02	1.38E-01	
	Aquatic Plants		2.60E-03	—	5.00E-01	5.20E-03	
	Barn Swallow		—	7.76E-02	1.13E+01	6.87E-03	
	Horned Grebe		—	2.42E-02	1.13E+01	2.15E-03	
	Trumpeter Swan		—	5.13E-01	1.13E+01	4.54E-02	
	Wild Waterfowl		—	6.38E-01	1.13E+01	5.65E-02	
	Little Brown Myotis		—	2.86E-02	8.00E+01	3.58E-04	
	Mink		—	1.14E-02	8.00E+01	1.42E-04	
	Moose		—	4.72E-01	8.00E+01	5.90E-03	
	Benthic Invertebrates	Xylene	4.44E-07	—	1.00E-01	4.44E-06	
	Fish		4.44E-07	—	2.68E+00	1.66E-07	
	Aquatic Plants		4.44E-07	—	3.90E+00	1.14E-07	
	Barn Swallow		—	2.06E-04	5.59E+01	3.68E-06	
	Horned Grebe		—	7.22E-05	5.59E+01	1.29E-06	
	Trumpeter Swan		—	1.23E-05	5.59E+01	2.19E-07	
	Wild Waterfowl		—	3.19E-05	5.59E+01	5.71E-07	
	Little Brown Myotis		—	9.76E-05	3.57E+02	2.73E-07	
	Mink		—	3.78E-05	3.57E+02	1.06E-07	
	Moose		—	1.13E-05	3.57E+02	3.16E-08	

Bolded font and shaded cell = calculated HQ exceeds the protective benchmark.

— indicates not applicable; NV = no value available to calculate.

The total radiation dose to all individual human receptors assessed during the closure and post-closure phases for the Normal Evolution Scenario are predicted to be well below the public dose limit of 1 mSv/a and the dose constraint for the Project of 0.25 mSv/a. Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure radioactive releases from the Project activities. These radiation doses are comparable to typical annual exposures from natural background radiation. The average annual radiation dose to members of the public in Canadian cities is presented in Table 9.2-6.

9.3 Integration of Safety Arguments

The safety assessment demonstrates that the data, assumptions and models have been tested and that a systematic analysis has been performed. Areas of uncertainty have been identified and assessed so that the limitations of the data and hence the models are fully understood. The quality of the assessment is reliant on the development of the scenarios and the arguments associated with the scenarios are realistic and based on specific evidence related to the WR-1 Building. Where there is uncertainty in the data or the models, conservative assumptions have been made. Further confidence in the models is provided by the use of commercially available software that has been tested and validated in a wider range of international projects. The analyses performed has shown that the WRDF is able to meet the acceptance criteria specified in Section 5.2 Assessment Acceptance Criteria and Endpoints. The key safety arguments that demonstrate that the WRDF provides long-term containment and isolation are described below.

The disposal system design uses the addition of grout to limit the mobility of radionuclides and lead. The grout acts as an engineered barrier to isolate short-lived radionuclides, which allows for substantial decay of these wastes. This isolation of the radioactive waste in the WRDF by the grout provides that the unacceptable release of wastes cannot occur as the short-lived radionuclides have decayed before they are able to migrate from the facility. Other design features of the disposal facility that provide isolation and containment and wastes are:

- The disposal facility is located below-grade and is enclosed by thick concrete walls and floor, with the outer walls being surrounded by clay till overburden and the ISD is covered with a concrete cap and earth mound.
- The chemical and hydrogeological conditions in the disposal facility provided by the grout limit contaminant mobility.
- The facility is designed and constructed using well known techniques and sound engineering practices. This provides confidence in their effectiveness and overall performance.

The safety assessment has demonstrated dose values that are below dose constraints in a wide range of scenarios. For the few scenarios where the dose constraint was exceeded, they still met international dose criteria for unlikely events (e.g., human intrusion). For each scenario, appropriate conservatism has been used in numerical models where uncertainties were present. Where assumptions have been made, they have undergone a review process to confirm that they are valid and appropriate for the scenario being studied. The safety analyses have demonstrated the following conclusions.

- WR-1 Building was a relatively low power reactor that has undergone 32 years of decay and has little radioactive inventory in the process equipment. The low inventory means that the potential doses from the Normal Evolution Scenario to humans and non-human biota are less than the dose acceptance criteria.
- Due to the characteristics of the waste, the long-lived radionuclides are bound to the stainless steel, zirconium alloy and Ozhennite components, which corrode relatively slowly, allowing the inventory of radionuclides diminish prior to release through corrosion. The grout, although not credited as a barrier under

the current assessment, is anticipated to impede the migration of solutes within the WRDF. The foundation provides structural support to the WRDF and due to its low hydraulic conductivity (relative to the surrounding environment) serves to limit the migration of contaminants through the structure. Similarly, the concrete cap and engineered cover limits vertical infiltration through the structure. Because of the effective containment offered by the engineered barriers, which function independently from one another, the reduction is manly a result of radioactive decay. This demonstrates that the engineered barriers are able to meet the defence-in-depth principles of containment and isolation and are effective.

- The low inventory coupled with the effective containment and isolation limit the risks to humans and non-human biota to acceptance levels, even when Disruptive Events are considered.

The assessments incorporate cumulative effects from other potential exposure sources at the WL site, including contaminated river sediments. While not included in the assessment explicitly, there are no significant cumulative effects expected from the in situ disposal of low level waste trenches in the WMA. The final safety assessments for these wastes have not been completed. However, both the Project and the WMA are subject to the same regulatory requirements, which require that no member of the public receive a dose greater than 1 mSv/a. REGDOC-2.11.1, *Waste Management, Volume III: Assessing the Long-Term Safety of Radioactive Waste Management* (CNSC 2018), Section 6.2.1, specifies a 1 mSv/a limit, and also specifies that a lower dose rate target should be set to account for emissions from other sources. Therefore, a value of 0.3 mSv/a dose limit to public has been established for the WL site. The Project specified a 0.25 mSv/a target dose rate limit for the public, and results of the assessment for the normal evolution indicate that the Project is well below this target. The expectation is that the final assessment of the WMA will demonstrate negligible dose rates compared to these targets. Consequently, there are negligible cumulative effects of the Project.

The WR-1 is located on the WL site which is comprised of other buildings and facilities that are currently undergoing various decommissioning activities according to the overall decommissioning plan for the WL site. The WL site has been divided into four post-closure land-use categories: agricultural, residential, industrial and recreational, and associated clearance levels and cleanup criteria for radionuclides and non-radionuclides have been developed. Each facility will be decommissioned and remediated to meet the cleanup criteria for the designated future land-use (CNL 2019a), which are protective of the anticipated future receptors and pathways on those lands.

Since cleanup criteria for the WL site will be met, and since risks to human and ecological receptors from radionuclides and non-radionuclides anticipated to be released during the closure and post-closure phases of the WR-1 Project are considered acceptable, cumulative effects from the WL site are not anticipated.

Additionally, a site-wide ERA is currently underway to assess the impacts of current operations and decommissioning activities at the WL site in terms of dose and risk to human and ecological receptors. Results of this site-wide ERA can be used to provide additional context to the cumulative effects of the WL site and the Project.

10.0 INSTITUTIONAL CONTROL

Reasonable effort is warranted to reduce the probability of human intrusion into the WRDF and institutional controls are required to confirm the long-term safe performance of the WRDF. After closure, the WL site will be under institutional control for the duration of the Decommissioning Licence. Approval will be required to release the site from regulatory control. The conditions required to grant this approval may change during the post-closure phase; however, it is understood that the site will not be released from regulatory control until it can be demonstrated that the hazard posed by the Project is acceptably low (i.e., the site is in a safe and stable state). An institutional control timeframe of a minimum of 100 years is anticipated. The 100-year timeframe is not related to the design life of the WRDF barriers; it is the time required to confirm the long-term safe performance of the WRDF. Additionally, beyond 100 years, there is less confidence that institutional controls, like access restrictions, can be relied upon.

During institutional control (i.e., 100 years after closure), the WRDF is anticipated to be under active management, surveillance, and monitoring to demonstrate the site conditions evolve as predicted and the WRDF performs as expected. The WRDF will be inspected for damage, any potential effects on long-term performance after closure will be detected during monitoring and deficiencies will be mitigated. Institutional control also includes passive controls, specifically government controls (e.g., zoning designation, land use restrictions, or orders) and societal memory (i.e., long-term maintenance of records and site recognition), as well as physical barriers such as fencing and signage. Ultimately, institutional control will continue until the CNSC agrees it is no longer needed. Institutional control can be extended beyond 100 years if the institutions involved have the resources and desire to do so. However, it is conservatively assumed that institutional control is lost beyond 100 years from closure so that the potential bounding effects of the WRDF can be assessed. This is a conservative assumption. For institutional controls to be lost, implies government control of land titles and land use restrictions are lost on a local, provincial and federal level.

The solute transport modelling indicates that a failure of the WRDF would be detectable at very low contamination levels via groundwater monitoring within the first 100 years. Monitoring will be required to continue as long as is necessary to demonstrate that the concrete cap and engineered cover and other containment features are performing sufficiently to meet design, safety and environmental requirements, and provide the protection of the public and environment in the long-term.

After closure, the passive engineered barriers will provide protection to the public and the environment by independently limiting the release of contaminants to the environment. The hydraulic conductivities of the grout and foundation were assumed to increase over time to account for degradation of these WRDF components. As expected, the groundwater flow rates through these components increases proportionally to the increase in hydraulic conductivity. The performance of these passive engineered barriers was accounted for in the HHRA and EcoRA. Passive controls, including the limited footprint, the WRDF composition being relatively impervious and made of material of no economic value, as well as any remaining land use restrictions, are expected to last beyond the 100 years of institutional control, but are not accounted for in the ERA.

11.0 MONITORING AND SURVEILLANCE

CNL’s Environmental Protection Program is designed to provide protection of the environment and the public with respect to environmental aspects that result from operation of CNL’s facilities. Decommissioning work will be conducted in accordance with the requirements of the Decommissioning Licence. Operations and activities conducted at CNL sites in Canada are bound by environmental requirements specified in the *Nuclear Safety and Control Act*, *Canadian Environmental Protection Act*, *Canadian Environmental Assessment Act, 2012*, *Fisheries Act*, *Transportation of Dangerous Goods Act*, and *Species at Risk Act*. Program requirements are outlined in CNL's document titled *Environmental Protection* (CNL 2018d) which documents the framework, roles and responsibilities, processes and procedures for the program. The program's requirements are implemented company-wide. The Environmental Monitoring Program will be maintained throughout the Project to monitor the effects of disposal activities and to verify that the requirements and objectives of the Environmental Protection Program are met.

During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions. CNL operates an extensive Environmental Monitoring Program that will govern monitoring throughout the closure phase. CNL has revised the EAFP for the WL site to incorporate the proposed monitoring and reporting specific to the Project. For further information see work package #10 in Table 3 of the EAFP (CNL 2018e).

The requirements outlined in the EAFP for the WL site have been integrated into this Environmental Monitoring Program. CNL will implement an EAFP for the Project to verify the accuracy of environmental effects and determine the effectiveness of mitigation that has been implemented. Follow-up programs will be carefully integrated with the existing EAFP for the WL site, as well as ongoing monitoring and management plans currently part of WL's Integrated Environmental Monitoring Program. The EAFP will be prepared consistent with the Canadian Standards Association's Standards N288.4-10 (Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills [CSA Group 2010]), N288.5-11 (Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills [CSA Group 2011]) and N288.7-15 (Groundwater Protection Programs At Class I Nuclear Facilities and Uranium Mines and Mills [CSA Group 2015]), as applicable.

Towards the end of the institutional control period, a Licence to Abandon will be sought and as a prerequisite for this it will need to be demonstrated that the facility is in a long-term, passive, safe state. If abandonment of the facility is allowed, monitoring and surveillance will no longer be required as, at this time, the facility will have been demonstrated to no longer pose a hazard to humans or the environment.

12.0 LIMITS, CONTROLS, AND CONDITIONS

The limits, controls and conditions for the Project have been determined by way of the safety analyses performed, the DDP, and the groundwater flow and solute transport modelling and the ERA. The limits, controls, and conditions are a set of rules that set limits, functional capability and performance levels of components and personnel for the safe decommissioning of the WR-1 Building. The limiting conditions, including safety related systems, are aimed at reducing the exposure of personnel, the public and the environment to radionuclide and non-radionuclide materials at all stages of the Project.

During the closure phase, the controls applied will be subject to CNL waste management systems and CNSC regulations. During post-closure, the controls are primarily related to inspection and monitoring of accessible areas of the WR-1 during the institutional control period. During the transition of the WR-1 Building to the WRDF, some of the inspection and monitoring activities already completed at the WL site will be continued until the systems are formally deactivated during decommissioning. All work will be controlled using existing programs and policies, as well as work control documents in accordance with the WL site management system.

Limiting factors include implementation of institutional controls that will limit the residual risks at the site after it has been decommissioned. Institutional controls will include active measures (e.g., monitoring, surveillance, and maintenance) and passive measures (e.g., land use restrictions). Institutional controls can also be administrative, legal, or land use controls. Administrative and legal controls are used to limit the potential for exposure to contamination or protect the integrity of the WRDF. Land use controls will limit the use of the site and will consist of engineering and physical barriers, such as fences or security guards. A combination of the two sets of controls will be used to prevent unwarranted access to the WRDF during the institutional control period.

The prevention of human intrusion is a key requirement to prevent accidental exposure to the wastes contains in the WRDF. When the facility enters the institutional control period, it will be fenced and remain under CNL control. During this period access restrictions to the site will be in place together with maintenance activities. After 100 years, passive controls will continue to provide controlled mitigation. These controls in the long-term will provide safety, reduce the probability of intrusion, and provide public confidence in the safety of the WRDF.

CNL will implement a long-term monitoring and surveillance program for the Project to verify the conditions of the site. The monitoring program will be developed as part of the EAFP and will be completed to verify that the condition of the site remains safe in the long-term for humans and the environment. The monitoring program will cover the period from the time decommissioning commences until the end of institutional control (i.e., a minimum of 100 years). After the end of institutional control, CNL will need to demonstrate that the WRDF is in a long-term passive state. If abandonment of the facility is allowed, monitoring and surveillance will no longer be required as the facility will have been demonstrated to no longer pose a hazard to humans and the environment.

13.0 CONCLUSIONS

The purpose of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public, and the releases of contaminants to the surrounding environment. The scope of the assessment considers the closure phase (which includes decommissioning and reclamation) and long-term performance during the post-closure phase (which includes institutional control and post-institutional control).

The DSAR has been prepared in accordance with CNSC's *REGDOC-2.11.1 Waste Management, Volume III: Assessing the Long-Term Safety of Radioactive Waste Management* (*REGDOC-2.11.1, Volume III* [CNSC 2018a]) and incorporates guidance outlined by the International Atomic Energy Agency (IAEA), specifically *SSG-23 The Safety Case and Safety Assessment for the Disposal of Radioactive Waste* (IAEA 2012) and *SSR-5 Disposal of Radioactive Waste* (IAEA 2011). As per CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), demonstrating long-term safety consists of providing reasonable assurance that waste management will be completed in a manner that protects human health and the environment.

The DSAR provides a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy. The DSAR demonstrates the level of protection provided to people and the environment by the Project and provides assurance that regulatory safety requirements will be met.

The ISD approach provides a permanent, passive decommissioning end state, and incorporates proven technologies and best industry practices, including documented experience from the IAEA and other similar international facilities. The decommissioning approach for the WR-1 draws upon experience and lessons learned from the decommissioning of many other similar facilities.

The ISD approach was selected as:

- the safety assessment demonstrates it is safe for the environment and the public;
- it does not rely on undefined future disposal options or technologies;
- it reduces risk for the exposure of radiological and industrial hazards to workers, meeting the ALARA principle, taking into account risk, cost, and goals pertaining to economic and social factors; and
- it reduces waste transport/handling risk to workers, the public and the environment.

The Project encompasses closure and post-closure (institutional control, including verification of end-state and post-institutional control) activities. Closure activities were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. Expansion of the CNL's Management System is not necessary to encompass the Project activities, as no new hazards were identified for the WL site. Radiological doses to workers, the public and the environment during closure will meet the ALARA principle in accordance with the procedures and practices in effect at the WL site. Through environmental monitoring (including the existing Environmental Assessment Follow-up Program), it is illustrated that the controls in place at the WL site are sufficient to ensure, with a degree of caution, that airborne and liquid effluent released from site are protective of workers, the public and the environment.

The safety of the Project post-closure is provided by means of passive features that slow the dispersion of contaminants and remove the need for active management, which is in alignment with IAEA Requirement 5 of SSR-5 (IAEA 2011). The performance of the combined natural and engineered barriers assure safety following the decommissioning of WR-1. In addition, institutional controls, including restrictions on land use, and a program for monitoring will be completed in the post-closure phase to help ensure the long-term safety of the public and the environment.

Institutional controls are required to confirm the long-term safe performance of the WRDF. Institutional control is estimated to last 100 years during which long-term performance monitoring and maintenance activities will continue, to demonstrate compliance with the safety case assumptions. The 100-year timeframe is not a design life for the barriers of the WRDF; rather it is an assumed duration of institutional control selected based on a reasonable assumption of the reliability of institutional controls. Beyond 100 years, there is less confidence that institutional controls can be relied upon as a barrier. As such, 100 years was selected as the limit beyond which the WRDF must be safe without reliance upon human intervention.

The safety functions of the Project components are containment and isolation. The ISD approach is designed to control the rate of release of nuclear and hazardous substances from the WRDF and retain the waste away from people and the environment. The design considers possible events that could degrade the integrity of the WRDF. It is recognized that there are inevitable uncertainties associated with predicting the performance of the WRDF over a long-time scale (i.e., thousands of years); therefore, safety is established through adequate defence-in-depth and verified through long-term environmental monitoring during institutional control.

The key aspect of the defence-in-depth principle is the provision of multiple layers of protection against abnormal events. In other words, the safety performance of the WRDF Project is not dependent on any single safety function. System robustness is demonstrated through a combination of design (multiple barriers) and analysis. Conceptual models were developed using qualified software packages, which have been validated appropriate for the intended use and compliant with appropriate standards. The models were developed taking into account uncertainty in input parameters and assumptions. The results have been compared with the appropriate safety criteria, considering uncertainty.

Central to the safety assessment, a wide range of scenarios are considered to develop an understanding of the system and provide a thorough safety case for a project. Scenario development does not try to predict the future, rather it aims to demonstrate the importance of sources of uncertainty, providing meaningful illustrations of future conditions to assist decision makers. The scenarios selected for detailed assessment are those most likely to occur (i.e., the Normal Evolution Scenario) and various unlikely disruptive events that could result in substantially higher exposure doses to the public and the environment (i.e., bounding scenarios).

The Normal Evolution Scenario is the expected long-term evolution of the WL site after closure has been completed and is a reasonable extrapolation of present-day site features and receptor lifestyles. The WRDF is expected to degrade over time due to mechanical stresses and chemical reactions. Contaminants will be released from the WRDF in the future due to corrosion of the reactor components and the degradation of the ISD components. Therefore, over time contaminants will migrate into the geosphere and discharge into shallow groundwater and ultimately be realized in surface water.

The total radiation dose to all individual human receptors assessed during the closure and post-closure phases for the Normal Evolution Scenario are predicted to be well below the public dose limit of 1 mSv/a and the dose constraint for the Project of 0.25 mSv/a. Since the dose estimates are a small fraction of the public dose limit, no discernable health effects are anticipated due to exposure radioactive releases from the Project activities. Further, for the two non-radiological COPCs with anticipated concentration requiring assessment (lead and cadmium), the HQs for all receptors are below the acceptable risk level for all pathways, with one exception, which is driven by background concentrations.

Maximum radiation doses predicted for non-human biota during post-closure were also well below UNSCEAR (2008) radiation benchmarks and all HQs were below the protective target value. Therefore, it is unlikely that there would be significant health effects on non-human biota as a result of radiological and non-radiological releases from the Project. Uncertainty and sensitivity analyses illustrated that there is a high confidence level in the predicted exposure doses, and the assumptions made for the assessment ensure that the forecast is conservative. The analyses also demonstrate the robustness of the design, as the performance does not rely on one feature but rather a set of redundant barriers and layers of passive protection.

Disruptive events are variants on the Normal Evolution Scenario, designed to address uncertainties that have arisen during the definition of scenarios and conceptual models. Each disruptive event is described with scenario-specific assumptions. Bounding scenarios are then identified out of the Disruptive Events to represent the “worst case”, with consequences greater than the other disruptive events considered. In the evaluation of bounding scenario results, it is recognized that deterministic effects will be prevented if effective whole-body annual dose exposures are limited to 20 mSv/a. Bounding scenarios considered in the safety assessment included Human Intrusion, WRDF Barrier Failure, and Well in Plume.

For the Human Intrusion Bounding Scenario, the total dose to a drill crew member (adult exposed during drilling the borehole) was below both the upper and lower IAEA reference level for Disruptive Events. The Human Intrusion Bounding Scenario also assumed that a trespasser receptor group would spend time daily on the site where core material obtained from within the WR-1 would be improperly disposed on on-site. The dose predictions for the trespasser receptors were below both the upper (20 mSv/a) and lower (1 mSv/a) IAEA reference level for disruptive events. Non-radiological hazardous material doses were calculated based on total concentration (background plus project contribution). The HQ for HB-40 exceeded the target value for the Adult and Toddler receptors for soil ingestion and soil dermal contact. The HQ for lead exceeded the target values for the adult and toddler trespasser receptors through soil ingestion, soil dermal contact, dust inhalation (exposure as a result of material being improperly disposed of on-site) and was exceeded for the driller receptor through soil ingestion and soil dermal contact.

The assessment demonstrates that human intrusion into the WRDF could result in exposures of human receptors to HB 40 and lead in waste material brought to the surface at levels where risks cannot be ruled out. As such, while this is a very unlikely worst case scenario, reasonable effort is warranted to reduce the probability of these unplanned events from occurring. During the post-institutional control period, passive controls will still be in place including the limited footprint, the WRDF composition being relatively imperviousness and made of material of no economic value, and the land use restriction acting to reduce the likelihood of a human intrusion event.

For the WRDF Barrier Failure Bounding Scenario, all radionuclide doses to human receptors were below the IAEA reference level (lower and upper level) ranging from 1 to 20 mSv/y and all radionuclide doses to non-human biota receptors were well below benchmarks.

The HQs for the Harvester receptors are below the acceptable risk level of 0.2 for cadmium and lead for all ingestion pathways. The HQs for the On Site Farm and Farm A are below the acceptable risk level of 0.2 for cadmium and lead for all pathways, with the exception of lead from drinking water from the Winnipeg River. The HQs for all receptors are based on background plus project exposure. If only the project contribution is considered, the HQs for the On Site Farm and Farm A are well below the acceptable risk level of 0.2. This indicates that the Project contribution to the total HQ is negligible and the exceedance is from existing background concentrations of lead in the Winnipeg River. Therefore, adverse effects to human receptors are not anticipated from WRDF barrier failure.

The HQs for exposure of the non-human biota receptors to cadmium, lead, HB 40, and xylene are all below the acceptable risk level of 1. Therefore, it is unlikely that there would be significant adverse effects on either aquatic or terrestrial populations or communities as a result of WRDF barrier failure.

For the Well in Plume Bounding Scenario, the total radiation dose does not exceed the upper IAEA reference level for any receptor, but does exceed the lower IAEA reference level for all receptors except the infant who drinks cow's milk. For non-radionuclides, the assessment demonstrates that human habitation with groundwater use for drinking water could result in exposures to cadmium and lead at levels above those that are known to prevent any adverse effects from occurring. The TRV for lead incorporates a safety factor of 2 to account for uncertainty, making the results conservative (EcoMetrix 2021). While the TRV for cadmium does not incorporate a safety factor, the assessment is considered conservative as it assumes that the maximum concentrations of COPCs occur at the same time, where in reality maximum concentrations occur at varying timeframes. Additionally, the likelihood of installing a groundwater well directly in the path of the potential plume is low.

Overall, the failure of passive controls to prevent human intrusion into the WRDF and/or a well in the groundwater plume used for drinking water during the post-institutional control period (100 years after closure) is conceivable. However, it should be noted that drinking water wells are unlikely downgradient of the WRDF due to a very low potential well capacity and the close proximity of the Winnipeg River.

To provide confidence in the long-term safety evaluation of the WRDF, a glaciation disruptive event was compared to CNSC Unrestricted Clearance Levels. The worst-case scenario is assumed to include the glacial advance having completely removed the concrete cap and engineered cover and excised the WRDF (i.e., glacial erosion), and glacial retreat having dispersed the ISD waste within the surface environment. The total remaining activity concentration is 95% of the CNSC Unconditional Clearance Level. This corresponds to a probable annual dose of 9.5 μSv for people living near or on the wastes 140,000 years from now, which is much less than the public dose limit of 1 mSv/y. Hence, under this scenario, the dispersal of the wastes in the WRDF after the next glaciation cycle is anticipated to lead to an acceptable radiation exposure to future inhabitants of the area.

The specific radioactivity of the WRDF was compared to natural analogues to enhance confidence in the safety features and provide a greater understanding of the disposal system. The standard method for directly comparing the environmental hazards from the artificial radionuclides within the WR-1 to the naturally occurring radionuclides present everywhere in the Earth's crust, is to consider the dose to members of the critical group.

Radiological consequences to a hypothetical exposure group settling in the vicinity of the WL site area after the glacial retreat will be bound by the current levels of exposure to members of the public living in the vicinity of surficial uranium deposits. In 2007, the CCME concluded that environmental levels of radionuclides at several locations containing subsurface uranium deposits, including Prairie Flats, met regulatory guidelines for the protection of the health of human and non-human biota, and that "no adverse effects are expected". Experience has shown that a sound knowledge of the potential radiological effects associated with the presence of these natural deposits has generally resulted in no measurable effect on human health (CCME 2007b).

The safety assessment illustrates that the Project, based on conceptual design details for the WRDF components and the characteristic of the environmental setting (i.e., the rate at which contaminants will reach the surface environment are substantially mediated by the bedrock), will provide long-term protection for the public and the environment. The design meets the criteria of providing long-term safety by passive means and minimizing the need for active controls and systems (active management of the site during which monitoring, and surveillance activities are completed). The long-term safety of the end-state has been demonstrated, and the Project meets the safety strategy for decommissioning.

The DSAR is a "living document" (i.e., continued iterative use as needed) and will be periodically reviewed and updated (as required), approximately every 5 years, over the lifetime of the facility. The safety envelope delineated by the current safety assessment is based on preliminary design information that was conservatively developed based on experience from similar long-term waste management and decommissioning projects. As outlined in IAEA SSR-5, safety assessments are updated as necessary to reflect actual experience and increasing knowledge. Anticipated future detailed design work includes:

- developing detailed work planning to control the potential spread of contamination and control worker exposure;
- reviewing detailed grout formulation and grout emplacement plan;
- reviewing detailed concrete cap and engineered cover plan; and
- delineating any required excavation around the WR-1 Building to facilitate decommissioning.

Finally, this new information will be evaluated to confirm the bounds of the safety assessment are not exceeded. If required, the safety assessment will be updated to reflect the evolution of the Project. Currently, it is anticipated that adequate conservatism has been integrated into the assessment and assumptions to accommodate future detailed design decisions and outcomes.

14.0 THIRD-PARTY REVIEWS AND FINDINGS

The analysis was subject to an internal review and verification in accordance with CNL's Quality Assurance Program, as well as an independent review by CNL's SRC.

Furthermore, the DSAR was subjected to third party review by several experienced industry recognized experts, active and retired, with various affiliations including the US DOE, Ontario Power Generation (OPG) and Nuclear Waste Management Organization (NWMO), and AECL.

In general, the reviews were positive, that the analysis presented a defensible case for ISD, though they highlighted several areas where the justification for conclusions could be better presented, and information could be made clearer (US DOE 2020; Melnyk 2020; Garisto 2020). The feedback from these reviews has been incorporated as appropriate to both clarify and bolster the justifications and conclusions provided throughout this assessment.

Signature Page

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APPENDIX A

Detailed Concordance Table

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
REGDOC 2.11.1 WASTE MANAGEMENT, VOLUME III: ASSESSING THE LONG-TERM SAFETY OF RADIOACTIVE WASTE MANAGEMENT (CNSC 2018)		
5.0	Developing a Long-term Safety Case	

5.1	Safety Assessment
	• Performance of the facility
		Section 7.0 WR-1 Building Closure Safety Assessment
		This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the closure phase of the WR-1 Project.
		
		The safety assessments completed for the closure phase included:
		• Radiological assessment for workers under normal conditions;
		• Radiological assessment for public under normal conditions;
		• Non-radiological assessment for workers under normal conditions; and
		• Radiological assessment for non-human biota.
		
		Section 8.0 Post-Closure Safety Assessment
		This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the post-closure phase of the WR-1 Project.
		
		The safety assessments completed for the post-closure phase included:
		• Radiological assessment for workers under normal conditions;
		• Radiological assessment for public under normal conditions;
		• Non-radiological assessment for workers under normal conditions;
		• Radiological assessment for non-human biota;
		• Non-radiological assessment for non-human biota;
		• Bounding Scenarios:
		  ○ Inadvertent Human Intrusion
		  ○ IDF Structure Failure
		  ○ Unplanned Human Habitation
		
		Section 1.1 Scope and Purpose
		The DSAR is a “living document” (i.e., continued iterative use as needed) and as necessary it is updated as the design transitions from pre feasibility to feasibility to detailed design and finally to as-built. The safety assessment will be updated to take into account the availability of new information gathered from experience, monitoring results, decommissioning modifications, and improvement in knowledge.

	Safety Assessment
	• Pathways analysis to predict:
	  1. Contaminant release;
	  2. Contaminant transport;
	  3. Receptor exposure; and
	  4. Potential effects resulting from the exposure.
		Groundwater Flow and Solute Transport Modelling Report
		A three-dimensional numerical groundwater model was constructed and calibrated to represent the best estimate of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. Predictive simulations were completed using the model to evaluate the post-closure groundwater conditions in the vicinity of the project area. A sensitivity analysis was completed to address uncertainty.
		
		Solute transport modelling was also completed for the WR-1 Project. The solute transport model was configured so that the solute mass is tracked from the source area, through the subsurface pathways to its ultimate discharge location at downgradient receptors.
		
		Section 7.2.1 Hazard Identification and Exposure Pathways
		The receptors for the HHRA were selected to be appropriate for assessment of both radiological and non-radiological stressors on human health. An on-site receptor (e.g., personnel leasing office/business space on the WL site) was evaluated for the closure phase, assessed as being present during demolition activities prior to grouting activities (i.e., during dismantling and relocating of above-grade portions of the PHT system).
		
		Off-site members of the public are potentially exposed to low levels of airborne contaminants. The most affected off-site member of the public is identified as the “critical group”. The critical group identified for the WR-1 Project is Farm A and Farm F (year-round occupants), as well as harvesters (i.e., traditional users of the area who may be exposed through harvesting country foods).
		
		Section 7.4.1 Hazard Identification and Exposure Pathways
		The receptors for the radiological risk assessment were selected to be appropriate for assessment of both radiological and non-radiological stressors on ecological health. For details on the rationale for chosen Valued Components refer to the ERA, Section 6.1.1.
		
		WR-1 at the Whiteshell Laboratories Site: Environmental Risk Assessment
		An environmental risk assessment was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota. The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
5.2	Use of different assessment strategies	
5.2.1	Scoping and bounding assessments	**Section 1.1 Scope and Purpose** The scope of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public and the releases of contaminants to the surrounding environment. The Purpose of the DSAR is to provide a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy. **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as “worst case” with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.
5.2.2	Realistic best estimates vs. conservative overestimation	**Section 5.7.1 Conservatism and Realism** The safety analyses were performed using a conservative approach to take uncertainties in data into account, and models were used providing bounding assessment using conservative assumptions. In those cases where scientifically informed knowledge and data is available, realistic assumptions are made. Additionally, where measurement data is available, it is used as input to calculations, for performing a comparison between results, or to limiting the range of variants for a scenario.
5.2.3	Deterministic and probabilistic calculations	**Section 5.5 Conceptual and Deterministic Models** Conceptual models were used to illustrate the performance of facilities under varying conditions and provide an analytical, quantitative analysis of performance. The conceptual model developed for the WR-1 Project represents the environmental setting and the conceptual design of the WRDF (including mitigation to protect against radiological and non-radiological hazards associated with the equipment and infrastructure to remain in place). A deterministic model, which uses single-valued input data to calculate a single-valued result, was compared with the assessment acceptance criteria. Deterministic calculations, with sensitivity assessment, relied on numerical software models, specifically for groundwater flow modelling, solute transport modelling and environmental risk assessment. These are further described in this section. Probabilistic analysis is used for a portion of the analyses of the long-term performance of the WRDF by way of computer modelling code IMPACT. This code was used to develop the human health risk assessment. According to CNSC guidance for applying probabilistic safety assessments (Regulatory Document-2.4.2 for existing facilities), the requirements do not apply unless they have been included in whole or in part, in the licence or the licencing basis. For WR-1, such analysis is not required according to the licencing basis.
5.3	Robustness and natural analogues	**Section 4.1.3 Robustness** The robust nature of the WRDF is demonstrated by evaluating a range of conservative bounding scenarios, using a suite of different calculation variations, and by using conservative models and data inputs. This section provides an overview of how conservatism was applied, and confidence increased for the safety assessments. **Section 5.4.3.2.6.2 Natural Analogue** Two natural analogues were considered: • The Maqarin Site; and • Natural radioactivity in soils and rocks. **Section 9.3 Integration of Safety Arguments** The safety assessment demonstrates that the data, assumptions and models have been tested and that a systematic analysis has been performed. Areas of uncertainty have been identified and assessed so that the limitations of the data and hence the models are fully understood. The quality of the assessment is reliant on the development of the scenarios and the arguments associated with the scenarios are realistic and based on specific evidence related to the WR-1 Building. The key safety arguments that demonstrate that the WRDF provides long-term containment and isolation are described.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
5.4	Use of complementary indicators of safety	**Section 13.6 Complementary Safety Indicators and Performance Indicators** The WR-1 Project is to decommission the WR-1 in a manner that meets end-state criteria, and will align with current international best practices, including the protection of present and future generations and the avoidance of imposing undue burden on future generations (IAEA 2006). The decommissioning strategy for the WR-1 draws upon experience and lessons learned from the decommissioning of many other similar facilities (CNL 2017a, 2015b). **Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection. The safety assessment approach uses a combination of complementary assessments at various levels of detail. **Section 6.0 Defence-in-Depth for the In Situ Disposal System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, components of the barrier system, including the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Groundwater Flow and Solute Transport Modelling Report** A three-dimensional numerical groundwater model was constructed and calibrated to represent the best estimate of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. Predictive simulations were completed using the model to evaluate the post-closure groundwater conditions in the vicinity of the project area. A sensitivity analysis was completed to address uncertainty. Solute transport modelling was also completed for the WR-1 Project. The solute transport model was configured so that the solute mass is tracked from the source area, through the subsurface pathways to its ultimate discharge location at downgradient receptors.
6.0	Defining Acceptance Criteria	
6.1	Overview	**Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
6.2	Criteria for protections of persons and the environment	
6.2.1	Radiological protection of persons	**Section 5.2.1.1 Radiological** Whiteshell Laboratories maintains a Radiation Protection Program, which provides a management framework and processes that are designed to confirm that radiation exposures arising are maintained below regulatory dose limits and are kept ALARA. The operational limits for the site are specified in the Licence Conditions Handbook for WL and encompass decommissioning activities.
6.2.2	Protection of persons from hazardous substances	**Section 5.2.1.2 Non-Radiological** Public exposure criteria adopted for airborne non-radiological hazardous substances are the United States Department of Energy Protective Action Criteria, which are a comprehensive set of short-term public exposure guidelines based on the United States Environmental Protection Agency Acute Exposure Guideline Levels or the American Industrial Hygiene Association one-hour Emergency Response Guidelines, as available.
6.2.3	Radiological protection of the environment	**Section 5.2.2.1 Radiological** For the protection of non-human biota from radiation exposure, the primary concern is the total radiation dose to the organisms resulting in deterministic effects. Benchmark values for mean radiation doses to non-human biota have been derived for various types of organisms.
6.2.4	Protection of the environment from hazardous substances	**Section 5.2.2.2 Non-Radiological** For non-radiological substances, COPCs were identified by comparing the maximum concentration of each contaminant in each medium measured at the site to appropriate guidelines for the protection of ecological receptors. Where appropriate guidelines were not available, upper background concentrations were used as the screening criteria.
7.0	Performing long-term assessments	
7.1	Selection of appropriate methodology	**Section 5.0 Assessment Approach** The assessment approach was selected to provide reasonable assurance that the WR-1 Project and the management of radioactive waste that arise are consistent with all applicable requirements. This section includes a description of the general approach used to demonstrate safety over the long-term, confidence in the results, and how the approach addresses the principles of radioactive waste management put forward in CNSC Regulatory Document 2.11.1.
7.2	Assessment context	**Section 5.0 Assessment Approach** The assessment approach was selected to provide reasonable assurance that the WR-1 Project and the management of radioactive waste that arise are consistent with all applicable requirements. This section includes a description of the general approach used to demonstrate safety over the long-term, confidence in the results, and how the approach addresses the principles of radioactive waste management put forward in CNSC Regulatory Document 2.11.1.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.2.1	Terms of Reference	**Section 1.1 Scope and Purpose** The scope of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public and the releases of contaminants to the surrounding environment. The Purpose of the DSAR is to provide a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy.
7.2.2	Regulatory requirements to be met	**Section 1.2.1 Regulatory Requirements** This section provides an overview of the federal and provincial requirements applicable to the DSAR. **Section 5.1 Scientific and Engineering Principles** The appropriate Canadian and international standards have been identified to confirm that robust scientific and engineering principles are applied rigorously to the design and construction of the WRDF. Specifically, the requirements of the following standards and guides have been identified for the design: • IAEA Safety Standard SSG-29 Near Surface Disposal of Radioactive Waste; • IAEA Safety Standard SSG-31 Monitoring and Surveillance of Radioactive Waste Disposal Facilities; • IAEA Safety Standard SSR-5 Disposal of Radioactive Waste; • CSA N292.0-14 Management of Low and Intermediate Level Radioactive Waste; • CNSC REGDOC-2.11.1 Assessment the Long-term Safety of Radioactive Waste Management; • CSA A23.3-12 Design of Concrete Structures; and • CSA N294-09 (R2014) Decommissioning of Facilities Containing Nuclear Substances.
7.2.3	Criteria to be met	**Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
7.2.4	Approach used to demonstrate safety	**Section 1.2.2 Guidance Documents and Safety Standards** Guidance documents and Safety Standards used in the development of the DSAR include: • Decommissioning Planning for Licensed Activities, Regulatory Guide G-219; and • CNSC's Regulatory Document 2.11.1 Waste Management, Volume III: Assessing the Long-Term Safety of Radioactive Waste Management. CSA guidance documents also provided meaningful guidance on how to meet regulatory requirements. The DSAR development considered international recommendations relating to the safe management of radioactive waste, including: • IAEA SF-1 Fundamental Safety Principles; • SSR-5 Disposal of Radioactive Waste; • IAEA SSG-23 Safety Case and Safety Assessment for Disposal of Radioactive Waste; • IAEA WS-G-5.2 Safety Assessment for the Decommissioning of Facilities Using Radioactive Material; and • IAEA Near Surface Disposal Facilities for Radioactive Waste Specific Safety Guide SSG-29. **Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture. **Section 9.3 Integration of Safety Arguments** The safety assessment demonstrates that the data, assumptions and models have been tested and that a systematic analysis has been performed. Areas of uncertainty have been identified and assessed so that the limitations of the data and hence the models are fully understood. The quality of the assessment is reliant on the development of the scenarios and the arguments associated with the scenarios are realistic and based on specific evidence related to the WR-1 Building. The key safety arguments that demonstrate that the WRDF provides long-term containment and isolation are described.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.3	System Description	**Section 3.1 WR-1 Decommissioning Plan** A high-level description of the In Situ Disposal (ISD) approach to the decommissioning of WR-1 is provided in this section. The below-grade reactor systems, components and structures will be permanently disposed in situ. The above-grade structures will be demolished and removed using traditional demolition methods. Details of the system are described further in Section 3.0 with Project Activities in Section 3.1.2. **Section 3.1.2 Project Activities** A description of the decommissioning activities and components proposed as part of the WR-1 Project is provided in this section and include: • Preparation for In Situ Decommissioning • Grouting of Below Grade Structures and Systems; • Removal of Above-grade WR-1 Building Structures; • Installation of Engineered Cover; • Final Site Restoration; • Preparation for institutional control; • Temporary Supporting Infrastructure; • Waste Generation and Management; and • End-State and Post-Closure Activities. Assumptions upon which the design is based are also discussed. **Section 3.1.3 Waste Classification, Inventory and Characterization** This section discusses the radiological hazards, non-radiological hazards, and other hazardous chemicals contained within the WR-1 Building. **Section 3.1.3.1.3 WR-1 Building** Table 3.1.3-1 summarizes building rooms with radiological concerns is provided, as well as a prediction of the total radionuclide inventory on the room surface. **Section 3.1.3.2.6 Other Hazardous Chemicals** Table 3.1.3-2 provides the location, description and quantity of hazardous chemicals within the WR-1 Building. **Section 3.1.3.3 Waste Generation and Management** Waste material resulting from decommissioning of the WR-1 building and their associated management are described in this section. Waste expected to be generated during the decommissioning of WR-1 includes: • Radiological waste; • Hazardous non-radiological waste; and • Clean wastes and likely clean waste. **Section 4.1.1 Containment** This section describes the containment and isolation features of the WR-1 Project. Specifically, containment is achieved through the combined natural and engineered barriers to assure safety following the decommissioning of WR-1. Systematically, these natural and engineered barriers consist of the reactor care and bioshield, grout surround, building foundation (i.e., internal walls and concrete surround), engineered cover, and the geological and hydrogeological characteristics, and post-closure monitoring. These barriers are described in this section, and an evaluation of Adequacy of Containment is provided. **Section 4.2.1 Defence-in-Depth Principle** This section provides a discussion of the defence-in-depth principle and how it has been applied to the WR-1 Project. **Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as Features, Events and Processes (FEPs). A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases. **Section 5.6.1 Alternative Options and Facility Design Selection** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option. **Section 5.6.2 Iteration and Design Optimization** The documentation supporting this DSAR (see Section 5.9 Technical Supporting Studies) has followed an iterative process, with the results used to refine the assessment of the WR-1 Project. These technical studies have been undertaken to builds on the outcomes of the previous work to adopt more realistic assumptions, progressively reduce those uncertainties and increase confidence in the projected outcomes. Overall, the technical support studies completed and subsequent refinements to the models were used to optimize the design of the WR-1 Project.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.3 (cont'd)	System Description	**Section 6.0 Defence-In-Depth for the In Situ Disposal System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, components of the barrier system, including the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report** This technical study report presents a summary of existing characterization information of the WR-1 Reactor that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided. **Non-Radiological Inventory Estimates for WR-1 ISD** Further details on the WR-1 non-radiological inventory estimates are provided in this document. **Detailed Decommissioning Plan** The Detailed Decommissioning Plan provides a list of the potential contamination present, and an estimate of the contamination level and general dose rates expected in each area. **Appendix D CNL WR-1 Decommissioning Project Features, Events and Processes Analysis** The purpose of this report was to establish and document the FEP Analysis that was completed as part of the safety case establishment for the Project. The development of the FEPs list is performed through a comprehensively and systematically examination of Project activities and components to identify all factors that may be relevant to the safety of the Project, during both the closure and post-closure phases. A screening analysis was completed to determine the applicability of each potential FEP on the safety of the Project and the relevant factors were encompassed in the development of bounding scenarios.
7.3.1	Site Characterization: • Subsurface characterization	**Section 2.1 Environmental Setting** This section provides a description of the biosphere including the geologic and hydrogeologic environment. **Geosynthesis for WR-1 Environmental Impact Statement** Integration and presentation of site geological, hydrogeological and geomechanical characterization data within the broader context of regional geoscientific data based on research completed on the Lac du Bonnet Batholith and in support of WL site operations and the safe storage of nuclear wastes held at the WL site. The Geosynthesis places the detailed geoscientific description of the site within the broader geoscientific understanding. **Hydrogeological Study Report** Detailed site-specific information on the hydrogeology can be found in this Technical Support Document. The work in the Hydrogeological Study Report provides baseline data for refining the conceptual hydrogeological model for the site, which is used in developing and calibrating the three-dimensional groundwater flow and solute transport model.The understanding of the hydrostratigraphy of the site has been enhanced by the recent field investigations at the main campus of the WL site and the ongoing monitoring programs at the WMA, lagoons, and landfill. The work herein provides baseline data for refining the conceptual hydrogeological model for the site, which will be used in developing and calibrating a 3-D groundwater flow and transport model **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report** This technical study report presents a summary of existing characterization information of the WR-1 Reactor that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided. **Building Condition Assessment Report** The Building Condition Assessment is part of the ISD program to evaluate the integrity of the existing subsurface concrete foundation and bottom slab of the WR-1 prior to grouting the existing structure of the WR-1 Building. The scope of the work was a condition assessment of the exposed and accessible concrete elements of the substructure, supplemented with laboratory testing of recovered cores to establish the condition and properties of the concrete.
	Site Characterizations: • Surface characterization	**Section 2.1 Environmental Setting** This section provides a description of the biosphere including the: • atmospheric environment; • surface water environment; • aquatic environment; • terrestrial environment; • land and resource use; and socio-economic environment.
	Site Characterization: • Monitoring	**Section 11.0 Monitoring and Surveillance** During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions. CNL operates an extensive Environmental Monitoring Program that will govern monitoring throughout the closure phase. The requirements outlined in the EA Follow-up Program for the WL site have been integrated into this Environmental Monitoring Program. The monitoring program will focus on groundwater quality and the functioning of the containment.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.3.1 (cont'd)	Site Characterization: • Current and foreseeable land use	**Section 3.1.2.9 Post-closure Activities** Future use of the WL site will depend on the ability of AECL to release parts of the site for unrestricted use upon completion of the WR-1 Project. CNL is developing the WL Closure Land-use and End-state Plan, along with appropriate criteria for site remediation and clean-up activities.
7.3.2	Waste management systems	**Section 3.1.2 Project Activities** A description of the decommissioning activities and components proposed as part of the WR-1 Project is provided in this section and include: • Preparation for In Situ Decommissioning • Grouting of Below Grade Structures and Systems; • Removal of Above-grade WR-1 Building Structures; • Installation of Engineered Cover; • Final Site Restoration; • Preparation for institutional control; • Temporary Supporting Infrastructure; • Waste Generation and Management; and • End-State and Post-Closure Activities. Assumptions upon which the design is based are also discussed. **Section 3.1.3 Waste Classification, Inventory and Characterization** This section discusses the radiological hazards, non-radiological hazards, and other hazardous chemicals contained within the WR-1 Building. **Section 3.1.3.1.3 WR-1 Building** Table 3.1.3-1 summarizes building rooms with radiological concerns is provided, as well as a prediction of the total radionuclide inventory on the room surface. **Section 3.1.3.2.6 Other Hazardous Chemicals** Table 3.1.3-2 provides the location, description and quantity of hazardous chemicals within the WR-1 Building. **Section 3.1.3.3 Waste Generation and Management** Waste material resulting from decommissioning of the WR-1 building and their associated management are described in this section. Waste expected to be generated during the decommissioning of WR-1 includes: • Radiological waste; • Hazardous non-radiological waste; and • Clean wastes and likely clean waste. **Section 4.1.1 Containment** This section describes the containment and isolation features of the WR-1 Project. Specifically, containment is achieved through the combined natural and engineered barriers to assure safety following the decommissioning of WR-1. Systematically, these natural and engineered barriers consist of the reactor care and bioshield, grout surround, building foundation (i.e., internal walls and concrete surround), engineered cover, and the geological and hydrogeological characteristics, and post-closure monitoring. These barriers are described in this section, and an evaluation of Adequacy of Containment is provided. **Section 4.2.1 Defence-in-Depth Principle** This section provides a discussion of the defence-in-depth principle and how it has been applied to the WR-1 Project. **Section 6.0 Defence-In-Depth for the In Situ Disposal System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, components of the barrier system, including the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report** This technical study report presents a summary of existing characterization information of the WR-1 Reactor that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided. **Non-Radiological Inventory Estimates for WR-1 ISD** Further details on the WR-1 non-radiological inventory estimates are provided in this document. **Detailed Decommissioning Plan** The Detailed Decommissioning Plan provides a list of the potential contamination present, and an estimate of the contamination level and general dose rates expected in each area.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.3.2 (cont'd)	Waste management systems	**Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as Features, Events and Processes (FEPs). A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases. **Appendix D CNL WR-1 Decommissioning Project Features, Events and Processes Analysis** The purpose of this report was to establish and document the FEP Analysis that was completed as part of the safety case establishment for the Project. The development of the FEPs list is performed through a comprehensively and systematically examination of Project activities and components to identify all factors that may be relevant to the safety of the Project, during both the closure and post-closure phases. A screening analysis was completed to determine the applicability of each potential FEP on the safety of the Project and the relevant factors were encompassed in the development of bounding scenarios.
7.4	Assessment time frames	**Section 5.3 Timeframes** The assessment timeframe is established consistent with the CNSC Regulatory Document-2.11.1 and is described in this section. The temporal boundaries associated with the WR-1 Project are divided into two main phases: the closure phase and the post-closure phase. The post-closure phase will continue indefinitely; however, the timeframe defined for the assessment of potential effects as part of the normal evolution of the WR-1 Project is 10,000 years. This time period encompasses the phase in which peak effects are anticipated. Natural analogues have been considered in the assessment timeframe.
7.5	Assessment scenarios	**Section 5.4.3 Assessment Scenarios** Table 5.4.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario (described in Section 5.4.3.1) is a reasonable extrapolation of present-day site features and receptor lifestyles and it includes the expected evolution of the site post-closure and degradation of engineered controls. Disruptive events (described in Section 5.4.3.2) postulate the occurrence of very unlikely events that could lead to high risk conditions
7.5.1	Normal evolution scenario	**Section 5.4.3.1 Normal Evolution Scenario** The normal evolution scenario is the expected long-term evolution of the WL site after closure has been completed. The normal evolution scenario includes extreme conditions such as climate shift other extreme events identified as very rare (e.g., glaciation) were analyzed separately. The details of the normal evolution scenario are further described in this section.
7.5.2	Disruptive event scenarios, including human intrusion	**Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic
7.5.3	Institutional controls	**Section 10.0 Institutional Control** Institutional controls are required to provide the long-term safety from residual contamination. Institutional control is estimated to last 100 years during which long-term performance monitoring and maintenance activities will continue, to demonstrate compliance with the safety case assumptions. Passive controls such as access restrictions (e.g., physical barriers/fencing, signage, and land title instruments/deed restrictions) will remain in place at the end of the institutional control period. Passive controls will continue to provide controlled mitigation following the release from regulatory control. These passive controls in the long-term will provide safety, reduce the probability of intrusion, reduce the consequence of intrusion, and provide public confidence in the safety of the WRDF. **Section 11.0 Monitoring and Surveillance** During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions. CNL operates an extensive Environmental Monitoring Program that will govern monitoring throughout the closure phase. The requirements outlined in the EA Follow-up Program for the WL site have been integrated into this Environmental Monitoring Program. The monitoring program will focus on groundwater quality and the functioning of the containment.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.5.4	Identification of critical groups and environmental receptors	**Section 7.1 Radiological Assessment for Workers Under Normal Conditions** According to CSA N288.6-12, Nuclear Energy Workers who participate in a Radiation Protection Program do not require radiological assessment in the ERA because their radiation exposure is monitored, and their doses are controlled. Workers on the WL site will participate in CNL's Radiation Protection Program. However, on-site WL workers have been assessed in the ERA for radiological exposures. **Section 7.2.1 Hazard Identification and Exposure Pathways** The receptors for the HHRA were selected to be appropriate for assessment of both radiological and non-radiological stressors on human health. An on-site receptor (e.g., personnel leasing office/business space on the WL site) was evaluated for the closure phase, assessed as being present during demolition activities prior to grouting activities (i.e., during dismantling and relocating of above-grade portions of the PHT system). Off-site members of the public are potentially exposed to low levels of airborne contaminants. The most affected off-site member of the public is identified as the "critical group". The critical group identified for the WR-1 Project is Farm A and Farm F (year-round occupants), as well as harvesters (i.e., traditional users of the area who may be exposed through harvesting country foods). **Section 7.4.1 Hazard Identification and Exposure Pathways** The receptors for the radiological risk assessment were selected to be appropriate for assessment of both radiological and non-radiological stressors on ecological health. For details on the rationale for chosen Valued Components refer to the ERA, Section 6.1.1. **WR-1 at the Whiteshell Laboratories Site: Environmental Risk Assessment** An environmental risk assessment was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota. The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern.
7.6	Developing and using assessment models	
7.6.1	Developing assessment models	**Section 5.5 Conceptual and Deterministic Models** Conceptual models were used to illustrate the performance of facilities under varying conditions and provide an analytical, quantitative analysis of performance. The conceptual model developed for the WR-1 Project represents the environmental setting and the conceptual design of the WRDF (including mitigation to protect against radiological and non-radiological hazards associated with the equipment and infrastructure to remain in place). A deterministic model, which uses single-valued input data to calculate a single-valued result, was compared with the assessment acceptance criteria. Deterministic calculations, with sensitivity assessment, relied on numerical software models, specifically for groundwater flow modelling, solute transport modelling and environmental risk assessment. These are further described in this section. Probabilistic analysis is used for a portion of the analyses of the long-term performance of the WRDF by way of computer modelling code IMPACT. This code was used to develop the human health risk assessment. According to CNSC guidance for applying probabilistic safety assessments (Regulatory Document-2.4.2 for existing facilities, the requirement do not apply unless they have been included in hole or in part, in the licence or the licencing basis. For WR-1, such analysis is not required according to the licencing basis. **Section 5.9 Technical Support Studies** **WR-1 Groundwater Flow and Solute Transport Modelling** A three-dimensional numerical groundwater model was constructed and calibrated using site-specific data to represent the best estimate of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. **Environmental Risk Assessment** An environmental risk assessment was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota. The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern. **Hydrogeological Study Report** Detailed site-specific information on the hydrogeology can be found in this Technical Support Document. The work in the Hydrogeological Study Report provides baseline data for refining the conceptual hydrogeological model for the site, which is used in developing and calibrating the three-dimensional groundwater flow and solute transport model.The understanding of the hydrostratigraphy of the site has been enhanced by the recent field investigations at the main campus of the WL site and the ongoing monitoring programs at the WMA, lagoons, and landfill. The work herein provides baseline data for refining the conceptual hydrogeological model for the site, which will be used in developing and calibrating a 3-D groundwater flow and transport model

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
7.6.2	Confidence in computing tools	**Section 5.8 Confidence in Numerical Models** The majority of the safety analysis relies on the use of commercially available software to develop numerical models to quantity the various scenarios. A brief overview of the following software is provided in this section: • MODFLOW-2005 • Visual MODFLOW • MODPATH • GoldSim • IMPACT • ERICA • ONEDANT • ORIGEN-S **WR-1 Groundwater Flow and Solute Transport Modelling** A three-dimensional numerical groundwater model was constructed and calibrated using site-specific data to represent the best estimate of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. **Environmental Risk Assessment** An environmental risk assessment was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota. The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern. **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report** This technical study report presents a summary of existing characterization information of the WR-1 Reactor that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided.
7.6.3	Confidence in assessment models	**Section 5.7 Management of Uncertainty** The safety assessment is performed using a conservative approach to take uncertainties in data into account, and models were used to provide bounding assessments using conservative assumptions. The general approach to management of uncertainty is provided in this section. **Section 5.8 Confidence in Numerical Models** The majority of the safety analysis relies on the use of commercially available software to develop numerical models to quantity the various scenarios. A brief overview of the following software is provided in this section: • MODFLOW-2005 • Visual MODFLOW • MODPATH • GoldSim • IMPACT • ERICA • ONEDANT • ORIGEN-S **Section 7.1.5 Assumptions and Uncertainty (Closure Assessment)** Uncertainty and the conservative assumptions used in the closure assessment are discussed. **Section 8.2.5 Assumptions and Uncertainty (Post-closure Assessment)** Uncertainty and the conservative assumptions used in the post-closure assessment are discussed. **WR-1 Groundwater Flow and Solute Transport Modelling** A three-dimensional numerical groundwater model was constructed and calibrated using site-specific data to represent the best estimate of groundwater flow conditions based on the site conceptual model, which incorporates the primary hydrostatigraphic units and groundwater flow boundaries. **Environmental Risk Assessment** An environmental risk assessment was completed to assess the radiological and non-radiological stressors of off-site members of the public and on-site workers, as well as non-human biota. The assessment integrates the conceptual hydrogeological model and the site-specific sources of constituents of potential concern. **Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates Report** This technical study report presents a summary of existing characterization information of the WR-1 Reactor that may be of relevance in assessing the decommissioning strategy of ISD. Radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms workplace radiological hazards are provided.

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8.0	Interpretation of Results	**Section 5.1 Scientific and Engineering Principles** The appropriate Canadian and international standards have been identified to confirm that robust scientific and engineering principles are applied rigorously to the design and construction of the WRDF. Specifically, the requirements of the following standards and guides have been identified for the design: • IAEA Safety Standard SSG-29 Near Surface Disposal of Radioactive Waste; • IAEA Safety Standard SSG-31 Monitoring and Surveillance of Radioactive Waste Disposal Facilities; • IAEA Safety Standard SSR-5 Disposal of Radioactive Waste; • CSA N292.0-14 Management of Low and Intermediate Level Radioactive Waste; • CNSC REGDOC-2.11.1 Assessment the Long-term Safety of Radioactive Waste Management; • CSA A23.3-12 Design of Concrete Structures; and • CSA N294-09 (R2014) Decommissioning of Facilities Containing Nuclear Substances.
8.1	Comparing assessment results with acceptance criteria	**Section 9.0 Results Summary** The adequacy of the site and engineering for the WR-1 Project is demonstrated through the results of the closure and post-closure safety assessments. The HAZOP, FEPs Analysis and ERA completed for the closure phase illustrate that the operational controls, management practices, and standard operating procedures that will remain in place and govern closure activities, are adequate to provide the safety of workers, the public and the environment. The key safety arguments that demonstrate that the WRDF provides long-term containment and isolation are described.
8.2	Analyzing uncertainties	**Section 7.1.5 Assumptions and Uncertainty (Closure Assessment)** Uncertainty and the conservative assumptions used in the closure assessment are discussed. **Section 8.2.5 Assumptions and Uncertainty (Post-closure Assessment)** Uncertainty and the conservative assumptions used in the post-closure assessment are discussed.
SSR-5 DISPOSAL OF RADIOACTIVE WASTE (IAEA 2011)		
1.19	Consideration of a range of options for the project	**Section 5.6.1 Facility Design Selection and Alternative Options** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option.
1.23	Concerned with providing for the protection of people and the environmental against hazards associated with waste management activities. Assurance of this protection will be provided by application of legal and regulatory requirements for closure and post-closure periods	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
1.26	The safety case for a disposal facility will be developed together with the development of the facility	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture.

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2.70	The radiation safety requirements and the related safety criteria for Closure are established in the International Basic Safety Standards	**Section 1.2.2 Guidance Documents and Safety Standards** Regulating nuclear safety in Canada is the responsibility of the CNSC. Therefore, the Project has been designed to be compliant with the CNSC guidance documents. The IAEA is a valuable resource to provide guidance for decisions concerning safety related to CNL's plans to decommission WR-1. The Project has been designed to be in alignment with the IAEA safety standards**.** This section describes the applicable guidance documents and safety standards. **Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture.
2.80	The primary goal is to ensure that radiation doses are as low as reasonably achievable (economic and social factors taken into account) and within the applicable system of dose limitation	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture. **Section 4.2.2 As Low As Reasonably Achievable Principle** The ALARA principle is that the exposures to persons shall be as low as reasonably achievable, social and economic factors being taken into account. CNL's ALARA program includes: • demonstrated management commitment to the ALARA principle; • implementation of ALARA through design, organization and management, selection and training of personnel, oversight of the Radiation Protection Program, resources, and documentation; • establishment of nuclear safety culture; • planning and control of all work; • application of task-specific dose and dose-rate radiological control hold points; and • performance of regular operational reviews.
2.11	During closure, even in the event of an accident, radiological releases are unlikely to have any radiological consequences outside the facility	**Section 7.0 Closure Safety Assessment** Provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the closure phase of the WR-1 Project. **Section 13.4 Closure Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions.

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2.13	An operational radiation protection programme, commensurate with the radiological hazards, is required to be put in place to ensure that doses to workers during Closure are controlled and that the requirements for the limitation of radiation doses are met	**Section 5.2.1 Human Health** Whiteshell Laboratories maintains a Radiation Protection Program, which provides a management framework and processes that are designed to confirm that radiation exposures arising are maintained below regulatory dose limits and are kept ALARA.
	Emergency plans are required to be put in place for dealing with accidents and other incidents, and for ensuring that any consequent radiation doses are controlled to the extent possible, with due regard for the relevant emergency action levels	**Section 3.1.4 CNL Management System and Quality Assurance** The compliance programs currently in place, and that will be applied to the WR-1 Project are listed in this section. Reference to the WL Decommissioning Quality Assurance Plan is also made.
2.14	The doses and risks associated with the transport of radioactive waste through public areas to a disposal facility are required to be managed and in accordance with IAEA's Regulations for the Safety Transport of Radioactive Material	**Section 3.1.3.4 Transporting Radioactive Waste Off-site** The transport of radioactive waste is regulated under the *Packaging and Transport of Nuclear Substances Regulations.*
Requirement 3	Shall carry out safety assessment and develop and maintain a safety case, and shall carry out all the necessary activities for site selection and evaluation, design, construction, operation, closure and after closure, in accordance with national strategy, in compliance with the regulatory requirements and within the legal and regulatory infrastructure	**Section 1.1 Scope and Purpose** The scope of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public and the releases of contaminants to the surrounding environment. The Purpose of the DSAR is to provide a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy. **Section 1.3 Documentation Framework and Structure** This section describes some of documents available which provide supporting analyses and content for the WR-1 Project and include: • the Decommissioning Safety Assessment Report; • the Environmental Impact Statement; • the Safety Case Report; and • the Detailed Decommissioning Plan. **Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture.
3.12	Develop a Safety case on the basis of which decisions on the development, operation and closure have to be made	**Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste.
3.14	Establish technical specification that are justified by safety assessment, to ensure development in accordance with the safety case	**Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture.
Requirement 4	An understanding of the relevance and the implications for safety of the available options for the facility shall be developed	**Section 5.6.1 Facility Design Selection and Alternative Options** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option.

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3.19	If more than one option is capable of providing the required level of safety, then other factors also have to be considered. These factors could include public acceptability, cost, site ownership, existing infrastructure and transport routes	**Section 5.6.1 Facility Design Selection and Alternative Options** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option.
3.20; 3.45	Consideration has to be given to locating the facility away from significant known mineral resources, geothermal water and other valuable subsurface resources	**Section 2.1 Environmental Setting** This section provides a description of the biosphere including the: • atmospheric environment; • geologic and hydrogeologic environment; • surface water environment; • aquatic environment; • terrestrial environment; • land and resource use; and • socio-economic environment.
Requirement 5	Shall evaluate the site and shall design, construct, operate and close the disposal facility in such a way that safety is ensured by passive means to the fullest extent possible and the need for action to be taken after closure of the facility is minimized	**Section 3.1.4 Multilayered Barrier System** The In Situ Disposal approach relies on a number of barriers, which passively resist release of contaminants. This section describes the multilayered barrier system that has been designed to be protective of the environment. **Section 4.1.1 Containment** This section describes the containment and isolation features of the WR-1 Project. Specifically, containment is achieved through the combined natural and engineered barriers to assure safety following the decommissioning of WR-1. Systematically, these natural and engineered barriers consist of the reactor care and bioshield, grout surround, building foundation (i.e., internal walls and concrete surround), engineered cover, and the geological and hydrogeological characteristics, and post-closure monitoring. These barriers are described in this section, and an evaluation of Adequacy of Containment is provided. **Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section. **Section 13.6 Complementary Safety Indicators and Performance Indicators** The WR-1 Project is to decommission the WR-1 in a manner that meets end-state criteria, and will align with current international best practices, including the protection of present and future generations and the avoidance of imposing undue burden on future generations (IAEA 2006). The decommissioning strategy for the WR-1 draws upon experience and lessons learned from the decommissioning of many other similar facilities (CNL 2017a, 2015b).
Requirement 6	Shall develop an understanding of the features of the facility and its host environment and of the factors that influence its safety after closure over suitably long time periods, so that a sufficient level of confidence in safety can be achieved	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 6.0 Defence-in-Depth for the Multilayered In Situ Decommissioning System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Section 4.2.1 Defence-in-Depth Principle** This section provides a discussion of the defence-in-depth principle and how it has been applied to the WR-1 Project. **Section 13.1 Site** The WRDF will be located on the east bank of the Winnipeg River within the WL site and therefore, the site characterization is based on decades of environmental monitoring. Site selection was considered in the establishment of the WL site, and the aspects of the site that made it an appropriate choice then still apply today.
3.26	Demonstrate that these features and factors are sufficiently well characterized and understood. Any uncertainties have to be taken into consideration in the assessment of safety	**Section 13.1 Site** The WRDF will be located on the east bank of the Winnipeg River within the WL site and therefore, the site characterization is based on decades of environmental monitoring. Site selection was considered in the establishment of the WL site, and the aspects of the site that made it an appropriate choice then still apply today. **Section 6.0 Defence-in-Depth for the Multilayered In Situ Decommissioning System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described.
3.28	Demonstrate dependability of certain design features; provide evidence feasibility and effectiveness before construction activities are commenced	**Section 6.0 Defence-in-Depth for the Multilayered In Situ Decommissioning System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described.

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3.29	Subject to agreement by regulatory body, shall consider the range of possible events and processes causing disturbances that it is reasonable to include; develop an understanding of whether or not such events and processes cause disturbances that could lead to the widespread loss of safety functions	**Section 5.4.1 Hazard and Operability Study** From the combination of planning, design and mitigation risks are addressed thereby avoiding or limiting the potential for environmental effects, worker injury (harm) and/or costly operational disruptions. In addition, the HAZOP is used to identify potential hazards and associated conditions that could arise during the Phase 2 decommissioning activities of the WR-1 Building. **Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as FEPs. A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases.
3.30	The level of understanding has to be sufficient to support the safety case fulfilling regulatory requirements applicable for the particular stage of the project.	**Section 5.4 Assessment Scenario Development** A wide range of scenarios are considered in order to develop an understanding of the system and provide a thorough safety case for the project. In accordance with IAEA 2014, the selection of scenarios for detailed assessment for the WR-1 Project are justified and, where appropriate, supporting evidence is provided. **Section 13.4 Closure Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 13.5 Post-closure Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation.
3.31	Has to be recognized that there are various types and components of uncertainty inherent in modelling complex environmental systems.	**Assumption and Uncertainty sections are provided for each component of the assessment, Sections 7.1.5, 7.2.5, 7.3.4, 7.4.5, 8.2.5, 8.3.5, 8.4.5 and 8.5.5** There is uncertainty in the estimated inventory of radioactive material that could be encountered during the cutting and perforation of components, during grouting preparation activities, or dismantling of the WR-1 Building. As needed, system characterization and radiological surveys will be completed to inform the development of detailed work plans.
Requirement 7	The host environment shall be selected, the engineered barrier of the disposal facility shall be designed and the facility shall be operated to ensure that safety is provided by means of multiple safety functions; the capability of the individual barriers and controls together with that of the overall disposal system to perform as assumed in the safety case shall be demonstrated	**Section 3.1.4 Multilayered Barrier System** The In Situ Disposal approach relies on a number of barriers, which passively resist release of contaminants. This section describes the multilayered barrier system that has been designed to be protective of the environment. **Section 7.0 WR-1 Building Closure Safety Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 8.0 Post-closure Safety Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation.
3.38	The safety case has to explain and justify the functions performed by each physical element and other features, identify the time periods over which physical components and other features are expected to perform, and identify the additional safety functions that are available if a physical element does not fully perform	**Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste.

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Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Requirement 8	Containment shall be provided until radioactive decay has significantly reduced the hazards posed by the waste	**Section 7.0 WR-1 Building Closure Safety Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 8.0 Post-closure Safety Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation.
3.40	Containment over a defined period has to ensure that the majority of shorter-lived radionuclides decay in situ	**Section 7.0 WR-1 Building Closure Safety Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 8.0 Post-closure Safety Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation.
3.41	If the waste has activity level for which the dose and/or risk criteria for human intrusion might be exceeded, alternative disposal options, will have to be considered	**Section 7.0 WR-1 Building Closure Safety Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 8.0 Post-closure Safety Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation. **Section 8.6.1 Inadvertent Human Intrusion** This Bounding Scenario encompasses uncertainty in the geological pathways that exist between the WR-1 Project area and the Winnipeg River (e.g., unsealed borehole), as well as potential disruption (i.e., inadvertent human intrusion), was assessed through the inclusion of a preferential pathway in the solute transport model.
2.15	Criteria (d) If human intrusion were expected to lead to a possible annual dose of more than 20 mSv to those living around the site, then alternative options for waste disposal are to be considered, for example	**Section 8.6.1 Inadvertent Human Intrusion** This Bounding Scenario encompasses uncertainty in the geological pathways that exist between the WR-1 Project area and the Winnipeg River (e.g., unsealed borehole), as well as potential disruption (i.e., inadvertent human intrusion), was assessed through the inclusion of a preferential pathway in the solute transport model.
Requirement 9	Consideration shall be given to both the natural evolution of the disposal system and events causing disturbance of the facility	**Section 5.4.3.1 Normal Evolution Scenario** The normal evolution scenario is the expected long-term evolution of the WL site after closure has been completed. The normal evolution scenario includes extreme conditions such as climate shift other extreme events identified as very rare (e.g., glaciation) were analyzed separately. The details of the normal evolution scenario are further described in this section. **Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as "worst case" with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
3.44	Access to waste has to be made difficult to gain without, for example, violation of institutional controls for near surface disposal	**Section 8.6.1 Inadvertent Human Intrusion** This Bounding Scenario encompasses uncertainty in the geological pathways that exist between the WR-1 Project area and the Winnipeg River (e.g., unsealed borehole), as well as potential disruption (i.e., inadvertent human intrusion), was assessed through the inclusion of a preferential pathway in the solute transport model.
3.47	The safety criteria for assessing releases over time periods of several thousand years or more are set out in paragraph 2.15	**Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
Requirement 10	An appropriate level of surveillance and control shall be applied to protect and preserve the passive safety features, so that they can fulfil the functions that are assigned in the safety case for safety after closure	**Section 3.1.2.10 Post-closure Activities** Future use of the WL site will depend on the ability of AECL to release parts of the site for unrestricted use upon completion of the WR-1 Project. CNL is developing the WL Closure Land-use and End-state Plan, along with appropriate criteria for site remediation and clean-up activities. **Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section. **Section 11.0 Monitoring and Surveillance** During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions. CNL operates an extensive Environmental Monitoring Program that will govern monitoring throughout the closure phase. The requirements outlined in the EA Follow-up Program for the WL site have been integrated into this Environmental Monitoring Program. The monitoring program will focus on groundwater quality and the functioning of the containment.
3.48	The passive safety features (barriers) have to be sufficiently robust so as not to require repair or upgrading; surveillance and monitoring as a method of checking whether performance is as specified ensuring the continuing fulfilment of safety functions	**Section 3.1.2.10 Post-closure Activities** Future use of the WL site will depend on the ability of AECL to release parts of the site for unrestricted use upon completion of the WR-1 Project. CNL is developing the WL Closure Land-use and End-state Plan, along with appropriate criteria for site remediation and clean-up activities. **Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section. **Section 11.0 Monitoring and Surveillance** During institutional control, long-term performance monitoring and maintenance activities will continue to demonstrate compliance with the safety case assumptions. CNL operates an extensive Environmental Monitoring Program that will govern monitoring throughout the closure phase. The requirements outlined in the EA Follow-up Program for the WL site have been integrated into this Environmental Monitoring Program. The monitoring program will focus on groundwater quality and the functioning of the containment.
Requirement 11	Step by step development and evaluation of disposal facilities; each step supported by iterative evaluation of the site, of the options for design, construction, operation and management, and of the performance and safety	**Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 5.4 Assessment Scenario Development** A wide range of scenarios are considered in order to develop an understanding of the system and provide a thorough safety case for the project. In accordance with IAEA 2014, the selection of scenarios for detailed assessment for the WR-1 Project are justified and, where appropriate, supporting evidence is provided. **Section 5.6.1 Facility Design Selection and Alternative Options** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
1.18	Step by step approach imposed by the regulatory body and by political decision-making processes	**Section 1.2 Regulatory Requirements and Guidance Documents** The WR-1 Project is required to comply with applicable federal and provincial legislation. This section provides an overview of the federal and provincial requirements applicable to the DSAR. **Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture. **Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste.
4.7	Safety case also has to identify and acknowledge the unresolved uncertainties that exist at that stage and their significance, and approaches for their management	**Assumption and Uncertainty sections are provided for each component of the assessment, Sections 7.1.5, 7.2.5, 7.3.4, 7.4.5, 8.2.5, 8.3.5, 8.4.5 and 8.5.5** There is uncertainty in the estimated inventory of radioactive material that could be encountered during the cutting and perforation of components, during grouting preparation activities, or dismantling of the WR-1 Building. As needed, system characterization and radiological surveys will be completed to inform the development of detailed work plans.
Requirement 12	The safety case and the supporting safety assessment shall be sufficiently detailed and comprehensive to provide the necessary technical input for informing the regulatory body and for informing the decisions necessary at each step	**Section 1.1 Scope and Purpose** The scope of the DSAR is to demonstrate that proposed activities can be safely completed in compliance with the prescribed protective limits, including radiological doses to workers and members of the public and the releases of contaminants to the surrounding environment. The Purpose of the DSAR is to provide a clear and transparent safety assessment and documents the rationale supporting the preferred decommissioning strategy. **Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste.
Requirement 13	The safety case for a disposal facility shall describe all safety relevant aspects of the site, the design of the facility and the managerial control measures and regulatory controls; The safety case and supporting safety assessment shall demonstrate the level of protection of people and the environment provided and shall provide assurance to the regulatory body and other interested parties that safety requirements will be met	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
4.15	All aspects of operation relevant to safety are considered, including surface and underground excavation, construction and mining work, waste emplacement, and backfilling, sealing and closing operations; Consideration has to be given to both occupational exposure and public exposure resulting from conditions of normal operation and anticipated operational occurrences over the operating lifetime	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 7.0 WR-1 Building Closure Safety Assessment** This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the closure phase of the WR-1 Project. The safety assessments completed for the closure phase included: • Radiological assessment for workers under normal conditions; • Radiological assessment for public under normal conditions; • Non-radiological assessment for workers under normal conditions; and • Radiological assessment for non-human biota. **Section 8.0 Post-closure Safety Assessment** This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the post-closure phase of the WR-1 Project. The safety assessments completed for the post-closure phase included: • Radiological assessment for workers under normal conditions; • Radiological assessment for public under normal conditions; • Non-radiological assessment for workers under normal conditions; • Radiological assessment for non-human biota; • Non-radiological assessment for non-human biota; • Bounding Scenarios: o Inadvertent Human Intrusion o IDF Structure Failure o Unplanned Human Habitation
4.16	Accidents of a lesser frequency, but with significant radiological consequences have to be considered with regard to both their likelihood of occurrence and the magnitude of possible radiation doses.	**Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as “worst case” with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.
4.19	If necessary, sensitivity analyses and uncertainty analyses would be undertaken to gain an understanding of the performance of the disposal system and its components under a range of evolutions and events.	**Section 6.0 Defence-in-Depth for the Multilayered In Situ Decommissioning System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Section 4.2.1 Defence-in-Depth Principle** This section provides a discussion of the defence-in-depth principle and how it has been applied to the WR-1 Project.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
4.20	The resilience of the disposal system has to be assessed; Quantitative analyses have to be undertaken, at least over the tie period for which regulatory requirements apply	**Section 7.0 WR-1 Building Closure Safety Assessment** This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the closure phase of the WR-1 Project. The safety assessments completed for the closure phase included: • Radiological assessment for workers under normal conditions; • Radiological assessment for public under normal conditions; • Non-radiological assessment for workers under normal conditions; and • Radiological assessment for non-human biota. **Section 8.0 Post-closure Safety Assessment** This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the post-closure phase of the WR-1 Project. The safety assessments completed for the post-closure phase included: • Radiological assessment for workers under normal conditions; • Radiological assessment for public under normal conditions; • Non-radiological assessment for workers under normal conditions; • Radiological assessment for non-human biota; • Non-radiological assessment for non-human biota; • Bounding Scenarios: ○ Inadvertent Human Intrusion ○ IDF Structure Failure ○ Unplanned Human Habitation
4.22	The management systems established to provide assurance of quality have to be addressed in the safety case	**Section 3.1.4 CNL Management System and Quality Assurance** The compliance programs currently in place, and that will be applied to the WR-1 Project are listed in this section. Reference to the WL Decommissioning Quality Assurance Plan is also made.
Requirement 14	The safety case and supporting safety assessment shall be documented to a level of detail and quality sufficient to inform and support the decision to be made at each step and to allow for independent review of the safety case and supporting safety assessment	**Section 1.3 Documentation Framework and Structure** This section describes some of documents available which provide supporting analyses and content for the WR-1 Project and include: • the Decommissioning Safety Assessment Report; • the Environmental Impact Statement; • the Safety Case Report; and • the Detailed Decommissioning Plan.
3.15	Retain all the information relevant to the safety case and the supporting safety assessment	**Section 1.3 Documentation Framework and Structure** This section describes some of documents available which provide supporting analyses and content for the WR-1 Project and include: • the Decommissioning Safety Assessment Report; • the Environmental Impact Statement; • the Safety Case Report; and • the Detailed Decommissioning Plan.
Requirement 15	The site shall be characterized at a level of detail sufficient to support a general understanding the characteristics of the site and how the site will evolve over time	**Section 2.1 Environmental Setting** This section provides a description of the biosphere including the: • atmospheric environment; • geologic and hydrogeologic environment; • surface water environment; • aquatic environment; • terrestrial environment; • land and resource use; and • socio-economic environment.
4.26	Focus on features, events and processes relating to the site that could have an impact on safety and that are addressed in the safety case and supporting safety assessment	**Section 5.4 Assessment Scenario Development** A wide range of scenarios are considered in order to develop an understanding of the system and provide a thorough safety case for the project. In accordance with IAEA 2014, the selection of scenarios for detailed assessment for the WR-1 Project are justified and, where appropriate, supporting evidence is provided. **Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as FEPs. A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
4.29	Characterization of the surface environmental features has to include natural aspects, such as hydrological and meteorological aspects and flora and fauna; It also has to cover human activities in the vicinity of the site relating to normal residential settlement patterns and industrial and agricultural activities.	**Section 2.1 Environmental Setting** This section provides a description of the biosphere including the: • atmospheric environment; • geologic and hydrogeologic environment; • surface water environment; • aquatic environment; • terrestrial environment; • land and resource use; and • socio-economic environment.
Requirement 16	Shall be designed to contain the waste with its associated hazards, to be physically and chemically compatible with the host geological formation and/or surface environment	**Section 3.1.4 Multilayered Barrier System** The In Situ Disposal approach relies on a number of barriers, which passively resist release of contaminants. This section describes the multilayered barrier system that has been designed to be protective of the environment.
Requirement 17	Shall be constructed in such a way as to preserve the safety functions of the host environment that have been shown by the safety case to be important for safety after closure; Construction activities shall be carried out in such a way as to ensure safety	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 3.1.4 CNL Management System and Quality Assurance** The compliance programs currently in place, and that will be applied to the WR-1 Project are listed in this section. Reference to the WL Decommissioning Quality Assurance Plan is also made. **Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste.
Requirement 18	Shall be operated with the conditions of the licence and the relevant regulatory requirements	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
Requirement 19	Provides for those safety functions that have been shown by the safety case to be important after closure; Plans for closure, including the transition from active management of the facility, shall be well defined and practicable, so that closure can be carried out safely at an appropriate time	**Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section.
Requirement 20	Waste shall conform to criteria consistent with the safety case for the disposal facility in after closure.	**Section 4.0 Safety Strategy** The WR-1 Project in accordance with 4.26 of IAEA SSG-23, has a safety strategy which provides overall management strategy for the various activities required in planning, operation and closure of the facility, including siting and design, site characterization, waste characterization, and development of the safety case. **Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.
5.2	Modelling and/or testing of the behaviour of waste forms has to be undertaken to ensure the physical and chemical stability under the conditions expected	**Section 6.0 Defence-in-Depth for the Multilayered In Situ Decommissioning System** This section describes the multilayered In Situ Decommissioning System of the WR-1 Project. Specifically, the reactor care and bioshield components, grout, internal walls, concrete surround, local geosphere, engineered cover, and post-closure monitoring are described. **Section 4.2.1 Defence-in-Depth Principle** This section provides a discussion of the defence-in-depth principle and how it has been applied to the WR-1 Project.
5.3	Waste intended for disposal has to be characterized to provide sufficient information to ensure compliance	**Section 3.1.3 Waste Classification, Inventory and Characterization** This section discusses the radiological hazards, non-radiological hazards, and other hazardous chemicals contained within the WR-1 Building.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Requirement 21	A programme of monitoring shall be carried out prior to, and during closure and after closure; this programme shall be designed to collect and update information necessary for the purposes of protection and safety	**Section 3.1.2.10 Post-closure Activities** Future use of the WL site will depend on the ability of AECL to release parts of the site for unrestricted use upon completion of the WR-1 Project. CNL is developing the WL Closure Land-use and End-state Plan, along with appropriate criteria for site remediation and clean-up activities. **Section 11.0 Maintenance and Surveillance** The Environmental Protection Program at WL is designed to provide protection of the environment and the public with respect to environmental aspects that result from operation of CNL's facilities. The WL Environmental Monitoring Program will be maintained throughout the WR-1 Project to monitor the effects of decommissioning activities and verify that the requirements and objectives of the Environmental Protection Program are met.
Requirement 22	Plans shall be prepared for the period after closure to address institutional control and the arrangements of maintaining the availability of information on the disposal facility	**Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section.
5.14	While the facility remains licensed, the operator has to provide institutional controls	**Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section.
Requirement 23	Consideration of the State system for accounting for, and control of, nuclear material	**Section 10.0 Institutional Control** Institutional controls are also established to protect humans and the environment and are described further in this section.
Requirement 24	Measures shall be implemented to ensure an integrated approach to safety measures and nuclear security measures in the disposal of radioactive waste	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. For the WR-1 Project, the intended safety objectives are the containment and isolation of the waste. **Section 4.2 Safety and design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture.
Requirement 25	Management systems to provide for the assurance of quality shall be applied to all safety related activities, systems and components throughout all the steps	**Section 3.1.4 CNL Management System and Quality Assurance** The compliance programs currently in place, and that will be applied to the WR-1 Project are listed in this section. Reference to the WL Decommissioning Quality Assurance Plan is also made.
Requirement 26	In the event that any requirements set down in SSR-5 are not met, measures shall be put in place to upgrade the safety of the facility, economic and social factors being taken into account	**Section 11.0 Maintenance and Surveillance** The Environmental Protection Program at WL is designed to provide protection of the environment and the public with respect to environmental aspects that result from operation of CNL's facilities. The WL Environmental Monitoring Program will be maintained throughout the WR-1 Project to monitor the effects of decommissioning activities and verify that the requirements and objectives of the Environmental Protection Program are met.
	It includes an analysis of the operational experience acquired and possible improvements that could be made, with account taken of the existing situation and of whatever new technological developments or changes in regulatory control there might be.	**Section 11.0 Maintenance and Surveillance** The Environmental Protection Program at WL is designed to provide protection of the environment and the public with respect to environmental aspects that result from operation of CNL's facilities. The WL Environmental Monitoring Program will be maintained throughout the WR-1 Project to monitor the effects of decommissioning activities and verify that the requirements and objectives of the Environmental Protection Program are met. **Section 13.6 Complementary Safety Indicators and Performance Indicators** The WR-1 Project is to decommission the WR-1 in a manner that meets end-state criteria, and will align with current international best practices, including the protection of present and future generations and the avoidance of imposing undue burden on future generations (IAEA 2006). The decommissioning strategy for the WR-1 draws upon experience and lessons learned from the decommissioning of many other similar facilities (CNL 2017a, 2015b).

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Appendix – A.3	The optimization of protection and safety for a disposal facility for radioactive waste is a judgemental process that is applied to the decisions made in the development of the facility's design. Most important is that sound engineering design and technical features are adopted and sound principles of management are applied throughout the development, operation and closure of the disposal facility. Given these considerations, protection and safety can then be considered optimized, provided that: a) Due attention has been paid to the implications for long term safety of various design options at each step in the development and in the operation of the disposal facility;	**Section 3.0 WR-1 Project Description** The objective of the WR-1 Project is to safely decommission the WR-1 Building, while maintaining protection to the environment (i.e., human and ecological), and reducing risk to workers during the decommissioning phase. Table 3.1.1-1 shows the proposed overall schedule for the WR-1 Project. **Section 5.6.1 Facility Design Selection and Alternative Options** Alternative options in the facility design were evaluated and the preferred option selected. This section provides a description of the alternative options considered and rationale for the selection of the preferred option.
	b) There is reasonable assurance that the assessed doses and/or risks arising from the generally expected range over the natural evolution of the disposal system do not exceed the relevant constraint, over timescales for which the uncertainties are not so large as to prevent meaningful interpretation of the results;	**Section 7.0 WR-1 Building Closure Safety Assessment** The closure phase was assessed through the completion of a HAZOP, Accidents and Malfunctions Analysis, and FEPs Analysis. The closure activities, even the non-routine activities, were determined to be well encompassed by existing engineering and administrative controls in place at the WL site. The Predicted effects to the environment as a result of WR-1 Project activities (Section 7.5 Radiological Assessment for Non-human Biota), illustrate that the effectiveness of operational controls will also sufficiently protect the environment from WR-1 Project emissions. **Section 8.0 Post-closure Safety Assessment** The WR-1 Project will result in the WR-1 being decommissioned in a manner that meet all of the assessment criteria, including being in accordance with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a). The end-state of the WR-1 Project provides the long-term safety of present and future generations, while avoiding undue burden on future generation.
Appendix – A.3	c) The likelihood of events that might affect the performance of the disposal facility in such a way as to give rise to higher doses or greater risks has been reduced as far as reasonably possible by site selection and evaluation and/or design.	**Section 5.4.1 Hazard and Operability Study** From the combination of planning, design and mitigation risks are addressed thereby avoiding or limiting the potential for environmental effects, worker injury (harm) and/or costly operational disruptions. In addition, the HAZOP is used to identify potential hazards and associated conditions that could arise during the Phase 2 decommissioning activities of the WR-1 Building. **Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as FEPs. A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases. **Section 5.4.3 Assessment Scenario** Table 5.4.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario is a reasonable extrapolation of present-day site features and receptor lifestyles, and it includes the expected evolution of the site post-closure and degradation of engineered controls. **Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as "worst case" with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Appendix - A.4	It is recognized that calculated possible radiation doses to individuals in the future due to a disposal facility are only estimates and that the uncertainties associated with the estimates will increase for timescales extending farther into the future. Nevertheless, estimates of possible doses and risks for long time periods can be made and can be used as indicators for comparison with the safety criteria.	**Section 5.2.2.1 Radiological** For the protection of non-human biota from radiation exposure, the primary concern is the total radiation dose to the organisms resulting in deterministic effects. Benchmark values for mean radiation doses to non-human biota have been derived for various types of organisms. **Section 7.1 Radiological Assessment for Workers Under Normal Conditions** According to CSA N288.6-12, Nuclear Energy Workers who participate in a Radiation Protection Program do not require radiological assessment in the ERA because their radiation exposure is monitored, and their doses are controlled. Workers on the WL site will participate in CNL's Radiation Protection Program. However, on-site WL workers have been assessed in the ERA for radiological exposures. In addition, the maximum and average estimated radiation doses for the on-site receptor during dismantling activities prior to grouting are presented in EcoMetrix 2019, Tables 4.9 and 4.10, respectively. **Section 7.2 Radiological Assessment for Public Under Normal Conditions** Human receptors evaluated for both the radiological and non-radiological assessment for the closure phase includes off-site member of the public and those critical groups. The maximum and average estimated radiation doses for Farm F residents are detailed in EcoMetrix (2019), Tables 4.13 and 4.14, respectively. **Estimated doses for the worker, public and nonhuman biota during WR-1 building closure and post-closure are provided in sections 7.1.4, 7.2.4, 7.4.4, 8.2.4, 8.3.4, 8.4.4.** Adherence to the CNL's Radiation Protection and Occupational Safety and Health programs will provide that exposures are controlled and kept within regulated limits.
Appendix - A.5	In estimating doses to individuals in the future due to a disposal facility, the assumption is made that people will be present locally, and that they will make some use of local resources that may contain radionuclides originating from the waste in the disposal facility. It is not possible to predict the behaviour of people in the future with any certainty, and its representation in assessment models is necessarily stylized13. The rationale and possible approaches to the modelling of the biosphere and the estimation of doses arising from waste disposal facilities have been considered in the IAEA BIOMASS Project[26].	**Section 5.4.3 Assessment Scenario** Table 5.5.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario is a reasonable extrapolation of present-day site features and receptor lifestyles, and it includes the expected evolution of the site post-closure and degradation of engineered controls. **Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as "worst case" with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.
Appendix - A.6	The possibility exists that in the future, an activity or activities undertaken by people could cause some type of intrusion into a disposal facility for radioactive waste. It is not possible to say definitively what form such an intrusion will take or what the likelihood of the intrusion event will be, owing to the unpredictability of the behaviour of people in the future. Nevertheless, the impact of certain generic intrusion events, such as construction work, mining or drilling, can be evaluated as reference scenarios.	**Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.4.3.2 Disruptive Events, three Bounding Scenarios were identified as "worst case" with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Appendix - A.7	Generic intrusion events such as construction work, mining or drilling could possibly occur, but will not necessarily occur. On this basis, an approach to evaluating the implications for safety of such events has been proposed by the ICRP, which makes use of the type of criteria set down in para. 2.15. An agreement would have to be reached with the regulatory body as to when such an approach was appropriate and exactly how the criteria would be used. Arbitrary decisions may have to be made as to what would be considered a normal activity that would be expected to occur and what would be considered intrusion events.	**Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.3.3.2 Disruptive Events, three Bounding Scenarios were identified as “worst case” with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF.
Appendix - A.8	In the event of inadvertent human intrusion into a disposal facility, a small number of individuals involved in activities such as drilling into the facility or mining could receive high radiation doses and exposures of other persons could also arise as a result of the intrusion. The doses and risks involved for any individuals authorized to take part in activities that deliberately disturb the disposal facility or its waste need not be taken into consideration in this context, as such activities would constitute planned exposure situations.	**Section 5.4.3 Assessment Scenarios** Table 5.4.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario (described in Section 5.4.3.1) is a reasonable extrapolation of present-day site features and receptor lifestyles and it includes the expected evolution of the site post-closure and degradation of engineered controls. Disruptive events (described in Section 5.4.3.2) postulate the occurrence of very unlikely events that could lead to high risk conditions.
Appendix - A.9	In general, the likelihood of inadvertent human intrusion into the waste will be low as a consequence of the chosen depth for a geological disposal facility. The likelihood will be low owing to institutional controls in the case of a near surface disposal facility, and because of the decision to site the facility away from known significant mineral resources or other valuable resources. The possible doses that would be received from such an inadvertent intrusion could be high. However, since the likelihood of inadvertent intrusion is low, the associated risk is likely to be outweighed by the higher level of protection and safety afforded by the disposal of waste in comparison with other strategies.	**Section 5.4.3 Assessment Scenarios** Table 5.5.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario (described in Section 5.4.3.1) is a reasonable extrapolation of present-day site features and receptor lifestyles and it includes the expected evolution of the site post-closure and degradation of engineered controls. Disruptive events (described in Section 5.4.3.2) postulate the occurrence of very unlikely events that could lead to high risk conditions. **Section 8.0 Post-closure Safety Assessment** This section provides a summary of the radiological and non-radiological safety assessment completed for workers, and the risk assessment completed for human and non-human biota for the post-closure phase of the WR-1 Project. The safety assessments completed for the post-closure phase included: • Radiological assessment for workers under normal conditions; • Radiological assessment for public under normal conditions; • Non-radiological assessment for workers under normal conditions; • Radiological assessment for non-human biota; • Non-radiological assessment for non-human biota; • Bounding Scenarios: o Inadvertent Human Intrusion o IDF Structure Failure o Unplanned Human Habitation

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Appendix - A.10	A disposal facility may be affected by a range of possible evolutions and events. Some evolutions and events may be judged to be relatively likely to occur over the period of assessment and some may be rather unlikely or very unlikely to occur. With a view to optimizing protection and safety, the design process will focus on ensuring that the disposal system provides for safety (i.e. through compliance with dose constraints and/or risk constraints). Such provision will be made in consideration of the expected evolution of the disposal system. Account will also be taken of uncertainties concerning that evolution and the natural events that are likely to occur over the period of assessment.	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. **Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture. **Section 5.7 Management of Uncertainty** The safety assessment is performed using a conservative approach to take uncertainties in data into account, and models were used to provide bounding assessments using conservative assumptions. The general approach to management of uncertainty is provided in this section. **Section 7.1.5 Assumptions and Uncertainty (Closure Assessment)** Uncertainty and the conservative assumptions used in the closure assessment are discussed. **Section 8.2.5 Assumptions and Uncertainty (Post-closure Assessment)** Uncertainty and the conservative assumptions used in the post-closure assessment are discussed. **Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as FEPs. A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases.
Appendix - A.11	The achievement of a level of protection and safety such that calculated doses are less than the dose constraint is not in itself sufficient for the acceptance of a safety case for a disposal facility, since protection is also required to be optimized[3]. Conversely, an indication that calculated doses could exceed the dose constraint in some unlikely circumstances need not necessarily result in the rejection of a safety case. Over very long timescales, radioactive decay of the waste will reduce the hazard associated with a disposal facility. However, uncertainties could become much larger and calculated estimates of doses might exceed the dose constraint.	**Section 8.2.2 Planned Mitigation** The ISD design will consider possible events that could degrade the integrity of the WRDF while recognizing the inevitable uncertainties associated with predicting the performance of the WRDF over a long-time scale. Therefore, safety is established through adequate defence-in-depth approach.
Appendix - A.12	Comparison of doses with doses due to radionuclides of natural origin may provide a useful indication of the significance of such cases. Caution needs to be exercised in applying criteria for periods far into the future. Beyond such timescales, the uncertainties associated with dose estimates become so large that the criteria might no longer serve as a reasonable basis for decision making (see the criteria in para. 2.15).	**Section 5.2 Assessment Acceptance Criteria and Endpoints** The radiological and hazardous non-radiological criteria for the decommissioning work are specified in the Licence Conditions Handbook for WL and are presented in this section. Acceptance criteria are provided for Human Health and Non-Human Biota Protection.

Table 1: Concordance to Regulatory Guidance

Document Section	Legislated Requirement	Decommissioning Safety Assessment Report Section
Appendix - A.13	The evaluation of whether or not the design of a disposal facility will provide an optimized level of protection and safety could require a judgement in which several factors would be considered. These factors might include, for example, the quality of the design of the facility and of the safety assessment, and any significant qualitative or quantitative uncertainties in the calculation of exposures in the long term	**Section 4.1 Safety Objectives** The Safety Objectives for the In Situ Disposal approach. The In Situ Disposal approach includes three main objectives: • Apply international best practises to safely decommission the WR-1 Building while protecting the human and ecological environment; • Apply CNL and international safety design principles to limit radiation doses to the public and workers; • Limit risk to workers during the decommissioning phase by avoiding and reducing industrial hazards. **Section 4.2 Safety and Design Principles** The Safety and Design Principles applied to the design and implementation of the WR-1 Project to confirm that the Safety Objectives can be met, include: • Defence-in-depth principle; • ALARA principle; and • Nuclear safety culture. **Section 5.1 Scientific and Engineering Principles** The appropriate Canadian and international standards have been identified to confirm that robust scientific and engineering principles are applied rigorously to the design and construction of the WRDF. Specifically, the requirements of the following standards and guides have been identified for the design: • IAEA Safety Standard SSG-29 Near Surface Disposal of Radioactive Waste; • IAEA Safety Standard SSG-31 Monitoring and Surveillance of Radioactive Waste Disposal Facilities; • IAEA Safety Standard SSR-5 Disposal of Radioactive Waste; • CSA N292.0-14 Management of Low and Intermediate Level Radioactive Waste; • CNSC REGDOC-2.11.1 Assessment the Long-term Safety of Radioactive Waste Management; • CSA A23.3-12 Design of Concrete Structures; and • CSA N294-09 (R2014) Decommissioning of Facilities Containing Nuclear Substances. **Section 5.7 Management of Uncertainty** The safety assessment is performed using a conservative approach to take uncertainties in data into account, and models were used to provide bounding assessments using conservative assumptions. The general approach to management of uncertainty is provided in this section.
Appendix - A.14	In general, when irreducible uncertainties make the results of calculations for safety assessment purposes less reliable, then comparisons with dose constraints or risk constraints need to be treated with caution. For a disposal facility, the uncertainties mean that caution is necessary in considering possible human intrusion events and very low frequency natural events. Caution is also necessary in considering calculated doses for timescales extending into the far future. The robustness of the disposal system can be demonstrated, however, by making an assessment of reference events that are typical of very low frequency natural events.	**Section 5.4.2 Features, Events and Processes** To define the range of potential future condition and scenarios, the analysis considers numerous factors that could affect performance referred to as FEPs. A detailed FEPs list has been developed for the ISD of WR-1, followed by a screening process that determined which FEPs to be included in the assessment. These FEPs were used to develop scenarios that are classified into Normal Evolution Scenario and Bounding Scenario (i.e., upset conditions). The FEPs have been developed for both the closure and post-closure phases. **Section 5.4.3 Assessment Scenario** Table 5.5.3-1 presents the scenarios identified as part of the DSAR. The Normal Evolution Scenario is a reasonable extrapolation of present-day site features and receptor lifestyles, and it includes the expected evolution of the site post-closure and degradation of engineered controls. **Section 5.4.3.2 Disruptive Events** Disruptive events include the analysis of the following scenarios: • Unsealed Borehole; • Human Habitation; • Localized failure of ISD Structure; • Substantial failure of ISD Structure; • Inadvertent Human Intrusion; • Glaciation; and • Seismic **Section 5.4.3.3 Bounding Scenarios** Taking into consideration the descriptions of disruptive events provided in Section 5.3.3.2 Disruptive Events, three Bounding Scenarios were identified as “worst case” with consequences greater than the other events considered. These were: • Inadvertent Human Intrusion Bounding Scenario: an exploration borehole drilled into the WRDF and exposure of wastes; • WRDF Failure Bounding Scenario: an ISD barrier failure; and • Unplanned Human Habitation Bounding Scenario: an on-site resident drinking groundwater from a well capturing the plume from the WRDF. **Section 5.7 Management of Uncertainty** The safety assessment is performed using a conservative approach to take uncertainties in data into account, and models were used to provide bounding assessments using conservative assumptions. The general approach to management of uncertainty is provided in this section.

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APPENDIX B

CNL WR-1 In Situ Disposal Activities Hazard Identification

CNL WR-1 IN SITU DECOMMISSIONING ACTIVITIES HAZARD IDENTIFICATION

ISR Report 3014-01-01
Version 2.0
28 August 2017

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QUALITY ASSURANCE AND VERSION TRACKING

Authorization

Title	CNL WR-1 In Situ Decommissioning Activities Hazard Identification	
Report number	3014-01-01	
Version	2.0	Signature
Prepared by	R. Corrigan	Robin Corrigan
Reviewed by	K. Potter	Potter
Approved by	T. Mahilrajan	Mahilrajan
Approved for Corporate Release by	F. Lemay	Lemay

Version Tracking

Ver.	Action	By	Date
1.0	Release to Client	M. McCall	18-Nov-16
2.0	Release to Client	F. Lemay	28-Jul-17

TABLE OF CONTENTS

LIST OF TABLES

1. INTRODUCTION

1.1 Background

Decommissioning of Whiteshell Laboratories is being completed in two phases. Phase 1 activities for the WR-1 reactor included removal of the organic coolant, heavy water moderator, irradiated fuel, and some decontamination and dismantling activities and are complete. CNL has now proposed In Situ Decommissioning (ISD) as the strategy for the Phase 2 decommissioning of the WR-1 Reactor. This is a change to the original Phase 2 strategy of dismantlement and removal [1] and requires an EA under the Canadian Environmental Assessment Act, 2012. To support the EA, an Environmental Impacts Statement (EIS) is being developed, which includes a section on accidents and malfunctions.

1.2 Objective

The objective of this hazard identification report is to identify potential hazards and associated malfunctions and accidents which could arise during the Phase 2 decommissioning activities of WR-1, causing harm to people or the environment. The findings from this report support development of the *Accidents and Malfunctions* section of the EIS for the in-situ decommissioning of WR-1.

1.3 Scope

This hazard identification assesses Phase 2 activities, which encompass all remaining activities to place the WR-1 in a final decommissioned state. The Phase 2 activities include:

- Emplacement of radiological wastes from the above grade structure into the below grade structure.
- Crawlspace remediation into below grade structure.
- Establishment of a supply of mixed grout.
- Grouting of WR-1 reactor system components where void spaces cannot be effectively filled during final grouting.
- Grouting of WR-1 below grade building structure/systems including all remaining penetrations and void spaces.
- Installation of an engineered cover and environmental controls.

Potential radiological and non-radiological hazards associated with accidents and malfunctions are to be identified for the WR-1 ISD activities of Phase 2. The associated consequence and likelihood are also to be estimated for each hazard, as well as any mitigating factors.

Hazards which result in consequences only over long timeframes (i.e., after the engineered cover is complete and the WR-1 site closed) will be identified as part of the Decommissioning Safety Analysis Report and are not in the scope of this assessment.

2. METHODOLOGY

This hazard ID report was developed through the following activities:

- Document reviews
- What-If workshop

Document Reviews

Numerous documents from CNL and other sources were reviewed to identify the activities which will be carried out for decommissioning and to determine the types of hazards present. This review was also used to develop a checklist to guide the What-If Workshop.

What-If Workshop

This hazard identification was executed as a structured *What-If* workshop, which is a *What-If* workshop guided by a checklist. In this case, the checklist (shown in Appendix A) was populated with a set of hazards relevant to the decommissioning activities, identified from various sources. Team brainstorming of potential malfunctions and accidents was not constrained by the checklist and participants were encouraged to present all ideas which may result in the identification of hazards associated with malfunctions or accidents. An initial estimate of the consequence, likelihood, and mitigating factors was also attempted for each hazard or *What-If* question posed by the team. Likelihoods were roughly estimated based on the frequency indices shown in Table 1**Error! Reference source not found.**.

Table 1: Frequency Indices

Index	Events/Year
Occasional	$3x10^{-2}$ to $3x10^{-1}$
Rare	10^{-4} to $3x10^{-2}$
Extremely Rare	10^{-6} to 10^{-4}

The process is organized according to the Phase 2 Decommissioning Work Packages identified in [1]. These include the following:

1. Building Systems
2. Prepare Reactor Systems
3. Grout Installation
4. Building Demolition and Site Remediation

What-If tables (see Appendix B) were used to record results of the workshop discussions. A separate table was used for each Decommissioning Work Package.

3. RESULTS

3.1 Type of Hazards

The literature review identified the following radiological and non-radiological high level hazards.

3.1.1 Radiological Hazards

Radiological hazards associated with WR-1 decommissioning include potential exposure to beta, gamma, and alpha-emitting radionuclides produced during the operational period of WR-1. These consist primarily of the following:

1. Fission products
2. Activation products
3. Tritium
4. Corrosion products

Fission products are normally contained within the fuel, although fuel defects or failures result in the release of fission products in small quantities, which are then circulated throughout the reactor core and heat transport system during the operational life of the reactor.

Activation products are produced by irradiation of neutron absorbing materials, which become radioactive.

Tritium is produced when heavy water captures neutrons.

Corrosion products are small, mobile activation products, produced when small debris from corrosion is transported through the core (e.g., in the moderator or coolant) and are irradiated.

3.1.2 Non-Radiological Hazards

Non-radiological hazards present during the ISD of WR-1 include conventional construction and demolition hazards, chemical hazards, biological hazards, fire hazards, and hazards from external events.

Conventional industrial hazards associated with the decommissioning activities include working at heights, in confined spaces, and with energized systems, hoisting and rigging, grouting, and falling objects. Dismantling and demolition of B100, will involve the use of cranes and movement of heavy loads.

Hazardous materials that are present include asbestos, lead, PCBs, mercury, mould, and various chemicals.

Potential external events include fires, floods, aircraft strikes, earthquakes, extreme weather events (e.g., tornadoes, heavy precipitation).

3.2 Potential Accident Scenarios

The main results from the What-If workshop discussions are recorded directly in the What-If tables in Appendix B. These results include the What-If questions, consequences, likelihood, and mitigating factors.

4. REFERENCES

[1] AECL, "WLDP-03702-041-000, Rev. 2, - Whitehsll Laboratories Decommissioning Project Comprehensive Study Report, Vol. 1," 2001.

[2] CNL, "WLDP26400-DDP-001, Rev. 4D1, Detailed Decommissioning Plan: Volume 6 - Whiteshell Readtor #1: Building 100," 2016.

[3] NRC, "NUREG-0586, Supplement 1, Vol. 1 - Generic Environmental Impact Statement on Decommissioning of Nuclear Facilitie," 2002.

[4] IAEA, "TSS No. 439 - Decommissioning of Underground Structures, Systems and Components," Vienna, 2006.

[5] SRNL, "SAVANNAH RIVER SITE R-REACTOR DISASSEMBLY BASIN IN-SITU DECOMMISSIONING -10499," 2010.

[6] SRNL, "SRNL-STI-2010-00361 - R & P Reacdtor Building In-Situ Decommissioning Visualization," Aiken, SC, 2010.

[7] DOE, "Draft EIS - Decommissioning of Eight Surplus Reactors at the Hanfort Site, Richland, Washington," 1989.

[8] AMEC, "NWMO DGR-TR-2011-07, Malfunctions, Accidents and Malevolent Acts Technical Support Document," 2011.

[9] DOE, "In Situ Decommissioning Lessons Learned - 14042," 2014.

APPENDIX A. WHAT-IF CHECKLIST/GUIDANCE SHEET

Table 2: What-If Guidance Sheet

Work Package	Item #	Hazards	Source
1. Building Systems Air gap building services, Remove Auxiliary Equipment, Remediate Crawlspace, Remove Building Support Systems	1	No unique hazards expected	WR-1 DDP [2]
	2	Some work will be in confined space – safety issues to consider	
	3	Some work will be in contaminated areas (e.g., crawlspace remediation).	
	4	Accidental cutting of contaminated piping	NUREG 0586 [3]
	5	Heavy equipment operation	IAEA, Decommissioning of Underground SSCs [4]
	6	Lifting and rigging	
2. Prepare Reactor Systems Prepare Heat Transport System, Reactor Vessel, Heavy Water & Helium Systems, Secondary Cooling and Auxiliary Systems and Active Ventilation Systems to receive grout; Remove, package and ship Fuel Transfer and Fuel Channel Flasks to CRL for long term management	7	Tritium removal procedures will need to be implemented to ensure worker safety.	WR-1 DDP [2]
	8	Radioactive water, organic coolant, particles and sludge may be present in some equipment.	
	9	Impact on ventilation patterns when active ventilation systems are shutdown, removed or if an opening needs to be cut.	
	10	High radiation fields exist in the reactor vault, upper and lower access rooms.	
	11	Damaged asbestos may be present on piping or tanks that require penetration or other preparatory work.	
	12	The need to remove some shielding to allow vessel preparation/grouting – use of temporary shielding to ensure worker safety.	
3. Grout Installation	13	Airborne particulate from grout constituents may be hazardous to respiratory health and precautions will	WR-1 DDP [2]

Work Package	Item #	Hazards	Source
Apply grout to selected areas, in and around special materials; Flood grouting of below grade structure		be taken to minimize dust generation and limit or prevent inhalation using respirators as appropriate.	
	14	Heavy truck traffic may occur at times during delivery of grout materials.	
	15	Impact on ventilation patterns when ventilation pathways are filled with grout.	
	16	Structural reinforcements may be necessary to ensure grout pressure does not compromise the structure and blow outs do not occur.	
	17	Deflagration/explosion: Aluminum scrap and metal alloys corrode when exposed to high pH (e.g., grout used at SRNL was 12+), producing hydrogen gas.	SRNL R-Reactor ISD [5]
	18	Structural failure: An unknown void space was discovered at the SRNL R-Reactor when a 3D CAD model was created.	[6]
4. Building Demolition and Site Remediation Demolition of B100 control room and office complex, reactor hall and ventilation stack; Installation of engineered cover; Site remediation and final grading	19	Work execution from heights – multiple floors above and below grade.	WR-1 DDP [2]
	20	Methods to backfill individual rooms in the lower levels need to be evaluated.	
	21	Radiological clearance needs to be completed well below ground prior to backfilling the excavation.	
	22	Heavy equipment operation	IAEA, Decommissioning of Underground SSCs [4]
	23	Lifting and rigging	
	24	Noise	
	25	Falling objects	
	26	Eye hazards	
	27	Explosion during demolition	WL CSR Vol. 1 [1]
	28	Weather event during dismantling activities.	DOE, Draft EIS [7]
General	29	Vehicle accident	DGR Malfunctions,

Work Package	Item #	Hazards	Source
			Accidents, Malevolent Acts [8]
	30	Fire: • At facility or involving equipment • During vehicle accident • During cutting	DGR Malfunctions, Accidents, Malevolent Acts [8]
	31	Electrical accidents • Misuse or poor maintenance of electrical equipment • Damage to electrical equipment • Access to live electrical equipment • Severe weather conditions, such as lightning	DGR Malfunctions, Accidents, Malevolent Acts [8]
	32	Spill of fuel, chemicals, lubricants and oils	DGR Malfunctions, Accidents, Malevolent Acts [8]
	33	Occupational accidents • Falls of workers • Injury during cutting • Injury during material handling • Heat exhaustion/stroke • Frostbite • Accidents related to moving/rotating machinery or other equipment • Machinery-related accidents during operation of drill, dozer or other equipment • Injury due to falling objects	DGR Malfunctions, Accidents, Malevolent Acts [8]
	34	Entrapment	

APPENDIX B. WHAT-IF WORKSHOP

Table 3: Workshop Form - 1 Building Systems

What-If	Potential Consequence	Likelihood	Mitigating Factors
1-1 What if the contamination levels of crawl spaces are higher than expected or take greater effort to remediate.	Higher worker exposure.	Rare	Previous characterization. RP surveys, PPE&C, contingency planning.
1-2 What if contaminated soil has spread (cross-contamination)	May require more cleanup/decontamination or higher worker exposures.	Occasional	Work planning to minimize potential spread of contamination. RP surveys, PPE&C.
1-3 What if there is more friable or loose asbestos in the soil or crawl spaces	Worker exposure to asbestos.	Occasional	Previous characterization. PPE&C
1-4 What if there is contamination released during cutting of tubing/conduit during air gapping. (e.g., cross contamination from other systems)	Potential spill or leak and higher worker exposure to radioactivity.	Occasional	Use of remote cutting techniques, PPE&C, radiation surveys, RP protection
1-5 What if there are some systems that are missed for isolation and air gapping (it's a large building)?	Electrical shocks, high pressure relief, industrial hazards, fire hazard, explosion hazard, etc.	Rare	Field verification, verification of design and design changes. Fire Protection, Emergency Preparedness plans.
1-6 What if the required systems (e.g., low level liquid waste collection, ventilation) are inadvertently isolated?	Loss of a safety-significant system. Higher radiological hazards. Potential loss of confinement and release of contamination (airborne and waterborne) to the environment. Potential worker injury.	Rare	Fail-safe modes of systems, backup power supplies. Detailed work plans. Verification of tie-ins.
1-7 What if failure of some remaining	Hazards may exist due to repair or maintenance being	Rare	Risks can be mitigated through careful assessment and work

What-If	Potential Consequence	Likelihood	Mitigating Factors
systems (e.g., low level liquid waste system) requires repair or more services than currently available to the building?	performed under conditions with limited building services.		planning for repair or remediation efforts.
1-8 What if organic coolant contains higher than expected levels of contaminated debris?	Higher exposure from coolant contaminated with radioactive debris.	Rare	RP surveys, PPE&C, potentially visible color change in coolant.

Table 4: Workshop Form - 2 Prepare Reactor Systems

What-If	Potential Consequence	Likelihood	Mitigating Factors
2-1 What if heavy equipment from the PHT or AVS is dropped or impacts the structure?	Structure could be damaged. Clean up for repair requires workers to spend more time in reactor incurring various hazards (e.g., industrial, radiological). Potential for airborne releases.	Rare	Procedural adherence. Verification of equipment. Personnel training, confinement by ventilation, emergency procedures.
2-2 What if negative pressure is not maintained in the reactor building?	Increased spread of airborne contamination, exposure to workers, and potential release to environment.	Occasional	Continuous air monitoring, PPE&C, work planning.
2-3 What if high levels of contamination exist in systems/components being prepared?	Potential for leaks and spills, worker exposure.	Occasional	Radiation Protection surveys, work permit, Personal Protective Equipment & Clothing (PPE&C), remote technology (e.g., automated guillotine cutters), temporary shielding, ventilation enclosure, conservative decision-making, work planning.
2-4 What if cutting or perforating causes	Potential for explosion.	Rare	Removal of combustibles. Use of cold cutting techniques, remote

What-If	Potential Consequence	Likelihood	Mitigating Factors
heating and pressurization of a sealed component during preparation?			technology. Fire protection. Emergency preparedness.
2-5 What if the systems state is unknown or different from expected? (more contamination than expected, or different in nature)	Various types of hazards (e.g., asbestos). The component may not be safe to prepare.	Occasional	Work planning, PPE&C, automated guillotine cutters.
2-6 What if there is an accident or incident resulting in a worker spending prolonged time in the reactor?	Higher worker exposure.	Rare	RP program, work plans, pre-job briefs, contingency plans, safety culture.
2-7 What if there is an accident during removal handling of heavy equipment (during hoisting, transferring)?	Heavy lifting, hoisting, transferring, transportation.	Rare	OSH procedures, PPE&C. Emergency procedures for rescue.
2-8 What if there is extreme weather during grouting (e.g., extreme cold)	See What-If **4-2.**		

Table 5: Workshop Form - 3 Grout Installation

What-If	Potential Consequence	Likelihood	Mitigating Factors
3-1 Can additional tritium be released during this phase (e.g., wet decontamination by, cutting of the ventilation system)?	Higher potential exposure to tritium.	Occasional	PPE&C, work plans, permit, decontamination techniques that do not create liquid waste. Temporary ventilation system during grouting.
3-2 What if there are unknown voids?	Structural integrity, chemical reactions? Higher likelihood of cracks near voids and potential release pathways.	Rare	Careful grout planning, sequencing. Use of visual tools (e.g., 3 D model, graphical method of display, video cameras during grouting).
3-3 What if there are reactions with metals, gases, or other materials (e.g., what if the grout is the wrong specification)?	Potential hydrogen explosion. Grout may not fill voids adequately. May affect post-closure environmental effects.	Rare	Design of grout plan. Knowledgeable grout specialist. Quality Control and Surveillance during grouting. Testing to determine performance of grout with HB-40.
3-4 What if it's not practical to reach certain areas with the grout (e.g., piping)?	Greater worker exposure to hazards such as radiation, confined spaces. Higher likelihood of void spaces.	Occasional	Identification of alternative methods for grouting difficult areas.
3-5 What if the contamination levels of crawl spaces are higher than expected or take greater effort to remediate.	Higher worker exposure.	Rare	Previous characterization.
3-6 What if heavy objects are dropped on the curing grout during this phase?	Potential damage to grout.	Rare /Occasional	Procedural adherence
3-7 What if the grout does not cure as expected to meet end-state objectives (e.g., thermal expansion during curing).	Failure modes of grout may be different or more likely.	Rare	Field testing of grout (including testing with HB-40) before ISD grouting. Grouting plan, and adherence to plan. Quality control

What-If	Potential Consequence	Likelihood	Mitigating Factors
			and surveillance in phases throughout grouting. Limit the grout lift volumes to prevent damage to the curing grout.
3-8 What if contaminated air is displaced by grout?	Spread of airborne contamination.	Rare	Provisions for controlling spread of contamination can be made (e.g., placement of filtered vents), temporary ventilation system during grouting.

Table 6: Workshop Form - 4 Building Demolition and Site Remediation

What-If	Potential Consequence	Likelihood	Mitigating Factors
4-1 What if heavy objects are dropped onto the grouted monolith?	Potential for generation of airborne contamination. Damage or breaks to contaminated systems or components below.	Rare	Ventilation system can confine airborne releases. Emergency and clean up procedures.
4-2 What if the grout behaves unexpectedly due to the organic coolant?	Failure modes of grout may be different or more likely. Migration of contaminants may be more likely.	Unknown	Study of HB-40.

Table 7: Workshop Form – 5 External Events

What-If	Potential Consequence	Likelihood	Mitigating Factors
5-1 What if there is extreme cold weather.	Electronic equipment, filters, respirators may not function, resulting in potential for dose.	Occasional	Work planning, contingency planning, stop work. Heat building.
5-2 What if fire fighting is necessary (also applies to internal fires)?	Large spread of contamination and release to environment. Higher worker doses during	Rare	Fire protection, PPE&C

What-If	Potential Consequence	Likelihood	Mitigating Factors
	cleanup/decontamination of reactor hall outside of ISD envelope. Higher worker doses due to contamination spread during reactor systems preparation (if not complete).		
5-3 What if there is an earthquake during ISD activities?	Worker hazards include falling objects, falling workers, potential collapse. Some spread of contamination is expected. Grout performance may be compromised if during or after grout installation (see What-If **4-2**).	Extremely Rare	WR-1 design basis. Grout design.
5-4 What if there is a flood during ISD activities?	Flooded areas could result in spread of contamination and greater worker doses. Electrical hazards.	Rare	[1] shows extremely low risk from flooding.
5-5 What if there are high winds or a tornado during ISD activities?	Most severe result is collapse of reactor hall, which could also cause a fire. This could result in spread of contamination and higher worker doses due to extensive clean-up activities.	Rare	Stop work during extreme weather events. WR-1 design basis.

APPENDIX C

WL Whiteshell Reactor 1 Decommissioning Features, Events and Process Analysis

CANADIAN NUCLEAR LABORATORIES WHITESHELL REACTOR DECOMMISSIONING PROJECT

Feature, Events and Processes Analysis

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Table of Contents

ATTACHMENTS

ATTACHMENT A
Comprehensive Features, Events and Processes List

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1.0 INTRODUCTION

1.1 Background

The scope of the Canadian Nuclear Laboratories In Situ Decommissioning of WR-1 at the Whiteshell Laboratories Site (the WR-1 Project) is to decommission the WR-1 in a manner that meets end-state criteria, and will align with current international best practices, including the protection of present and future generations and the voidance of imposing undue burden on future generation (IAEA 2006). The decommissioning strategy for the WR-1 draws upon experience and lessons learned from the decommissioning of many other similar facilities (CNL 2017).

The strategy chosen for the Project is complete in situ disposal (ISD), where below-grade WR-1 systems, components and structures, and associated radiological and non-radiological hazards will be permanently encased with grout within the building's foundation. The general approach to ISD for the WR-1 Building involves preparing systems and structures for grouting, then the below-grade structure will be filled with grout to encapsulate and immobilize radiological sources and hazardous materials for a defined period of institutional control. The decommissioning activities proposed as part of the Project include:

- preparation for ISD, including placement of portions of the Primary Heat Transport system (PHT) that is contaminated below-grade;
- grouting of below-grade structures and systems;
- removal of above-grade WR-1 Building;
- installation of a concrete cap and engineered cover over the Whiteshell Reactor Disposal Facility (WRDF);
- final site restoration; and
- institutional control.

Activities will include the transport and disposal of Low Level Radioactive Waste off-site; however, the volume of waste will be considerably lower compared to complete decommissioning following deferment, the original decommissioning strategy assessed in the *Whiteshell Laboratories Decommissioning WR-1 Project Comprehensive Study Report* (AECL 2001).

1.2 Purpose

The purpose of a Features, Events and Processes (FEPs) analysis is to define the range of potential future conditions and scenarios that could affect performance. These factors are referred to as FEPs. A FEP is a feature, event, process or other factor that could directly or indirectly influence the long-term safety and performance of the disposal system. A Feature is a prominent or distinctive part or characteristic of the Project or its environment (e.g., engineered cap), an Event is a change or complex of changes located in a restricted portion of time and space (e.g., rainfall), and a Process is a phenomenon marked by gradual changes that lead towards a particular result (e.g., climate change; IAEA 2004).

Consistent with CNSC's *REGDOC-2.11.1, Volume III* (CNSC 2018a), the FEPs are used to develop scenarios that are classified into those considered in the expected evolution of the Project (i.e., the Normal Evolution Scenario) and those low-probability events that could occur leading to upset conditions (i.e., bounding scenarios for disruptive events. The Normal Evolution Scenario and bounding scenarios to be evaluated and documented in the Decommissioning Safety Assessment Report (DSAR; CNL 2021).

1.3 Organization of Report

The structure of this report is as follows:

- Section 2: Methods - Describes the methodology used for establishing the initial FEPs list and for performing the screening analysis to identify FEPs to be carried forward into the DSAR.
- Section 3: Results – Presents the outcome of the screening analysis.
- Section 4: References
- Attachment A: Comprehensive Features, Events and Processes List

2.0 METHODS

The development of the FEPs list is performed through a comprehensive and systematic examination of the Project activities and components to identify all factors that may be relevant to the safety of the Project. A comprehensive list of FEPs to be considered was developed for the Project safety case based on international guidance including the Nuclear Energy Agency's Features, Events, and Processes for the Disposal of Radioactive Waste: An International Database (NEA 2000) and the International Atomic Energy Agency's Safety Assessment Methodologies for Near Surface Disposal Facilities: Review and Enhancement of Safety Assessment Approaches and Tools (IAEA 2004). As well, previous Chalk River Laboratories and Canadian Nuclear Laboratories (CNL) waste disposal projects with similarities to the Project were reviewed (AECL 2014, AECL 2013).

The FEPs Analysis was completed in two phases. The first phase encompassed FEPs relevant to the closure phase as part of a Hazard and Operability Study (ISR 2017). The second phase encompassed the FEPs relevant to the post-closure phase as part of a FEPs Analysis. The FEPs are identified based on the following categories:

- Assessment Basis Factors;
- External Factors;
- Internal Disposal System Domain Environmental Factors; and
- Radionuclide and Contaminant Factors.

Assessment Basis Factors include factors that considered in determining the scope of the analysis. These may include factors related to regulatory requirements, definition of desired calculation end-points and requirements in a particular phase of the assessment (AECL 2014, 2013).

External Factors are FEPs with causes or origins outside the disposal system domain (i.e., natural or human factors of a more global nature) and their immediate effects (AECL 2014, 2013). External FEPs are those beyond the control of project execution, originating outside the Project. External FEPs include geological processes and events, climatic processes and events, and future human action.

Internal Disposal System Domain Environmental Factors are FEPs occurring with the spatial and temporal (post-closure) domain. The primary purpose is to determine the evolution of the physical, chemical, biological, and human conditions relevant to estimating the release and migration of radionuclides and consequent exposure to humans and the environment (AECL 2014, 2013). Internal FEPs include engineered control features, subgeological surround, surface environment, human behaviour, source-term characteristics, solute transport factors, and exposure pathway factors.

Radionuclide and Contaminant Factors are FEPs that take place in the disposal system domain that directly affect the release and migration of radionuclides and other contaminants, or directly affect the dose to humans and the environment from given concentrations of radiological and non-radiological contaminants in environmental media (AECL 2014, 2013).

Following the development of the FEPs list, a screening analysis was completed to determine the applicability of each potential FEP on the safety of the Project. Specific FEPs were screened out if:

- FEP is not applicable to the waste types to be encountered, project design, project activities or environmental setting;
- There is an extremely low likelihood the FEP would occur (i.e., non-credible event) without the implementation of controls/mitigation or inconsideration of establishment of proven controls/mitigation; and/or
- The FEP would have low consequence or negligible impact if it did occur (i.e., non-consequential event) without the implementation of controls/mitigation or inconsideration of establishment of proven controls/ mitigation.

The FEPs that were not screened out (i.e., included for further assessment) were carried forward into the safety assessment and the relevant factors were encompassed in the development of scenarios to be considered.

The FEPs developed for the closure phase were influenced by the “What-if” questions raised during a HAZOP exercise (ISR 2017). Most of these FEPs were addressed through existing procedures that would be in place during closure work that would mitigate the risks or uncertainty introduced by a specific FEP. Therefore, these FEPs were excluded from the safety assessment. The FEPs included in the safety assessment for the closure phase were related to exposure to airborne emissions from closure activities.

The FEPs that were carried forward into the post-closure phase assessment of the WRDF, guided the development of the groundwater flow and solute transport model and the ERA. They helped determine the parameters to be included in the models, as well as key events and processes to be modelled through the sensitivity analyses.

The “Features” portion of the FEPs informed the key parameters and features of the existing structures, waste forms and inventory, the surrounding environmental setting, climatic setting, hydrogeological setting, geological setting, and human habitation and land use patterns that needed to be included in the overall model. The “Events” portion of the FEPs provided specific types of occurrences that are likely or unlikely to occur, that can affect the long-term safety and performance of the WRDF. The “Processes” portion of the FEPs provided guidance on modelling the long-term safety and performance of the WRDF. These events and processes are evaluated as part of the expected evolution (Normal Evolution Scenario) or through Disruptive Events or Bounding Scenarios.

The FEPs determine the parameters and equations of the models; uncertainty in the parameters and equations determine the scenarios to be assessed. The Normal Evolution Scenario include site-specific values, or reasonably conservative values for FEPs that occur or are likely to occur during the assessment timeframe. Disruptive Events and Bounding Scenarios use deliberately extreme values for FEPs that are shown to influence the model outcomes through the sensitivity cases. Sensitivity cases use a range of values for parameters to understand the relative significance of a particular FEP to the overall system performance.

3.0 RESULTS

The comprehensive list of FEPs that could potentially be relevant to the safety case of the Project for the post-closure phase is provided in Attachment A. The FEPs were defined as being excluded (not relevant) to the safety assessment or were included. For those FEPs included in the assessment, the applicability of a FEP to the assessment scenarios, or if it was captured as part of the sensitivity analysis, was identified. The following categories were used to describe how a FEP was considered in the Normal Evolution Scenario, Disruptive Events/Bounding Scenario, and Sensitivity Analysis (Table 3.0-1).

Table 3.0-1: FEP Categories Used in the Safety Assessment

Category	Description
1 – Normal Evolution Scenario	The analysis related to the FEP was completed using available data with reasonable ranges.
2 – Disruptive Events/Bounding Scenarios	The analysis related to the FEP was completed using available data with extreme value ranges.
3 – Sensitivity Analysis	The analysis related to the FEP was completed with variability on input parameters.

4.0 REFERENCES

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ATTACHMENT A

Comprehensive Features, Events and Processes List

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
0			**Assessment Basis Factors**						
0	0	1	Impacts of concern	The impacts of concern are the long-term human health and environmental effects or risks that may arise from the Whiteshell Reactor Disposal Facility (WRDF).	Demonstrating the long-term safety consists of providing reasonable assurance that the WRDF will be constructed in a manner that protects the human health and the environment. The impacts of concern will be assessed in the decommissioning safety assessment report (DSAR) through the completion of an Environmental Risk Assessment (ERA).	Include	Y	N	Y
0	0	2	Timescales of concern	Timescales of concern are the time periods over which the WRDF may present some significant human health or environmental hazards.	The assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompasses the period when the maximum impact is predicted to occur.	Include	Y	N	Y
0	0	3	Spatial domain of concern	The spatial domain of concern is the domain over which the WRDF may present some significant human health or environmental hazard.	The spatial domain of concern is the area over which the performance/safety of the WRDF is estimated, or the area where the movement of contaminants and exposure may occur.	Include	Y	N	Y
0	0	4	Repository assumptions	The assumptions that are made in the safety assessment regarding the closure and post-closure of the WRDF.	The safety analysis includes justified assumptions and key assumptions and rationales clearly identified.	Include	Y	Y	Y
0	0	5	Future human action assumptions	The assumptions made in the safety assessment concerning boundary conditions pertaining to assessing future human action.	Assumptions pertaining to predicting human behaviour are based on reasonable conservative and plausible assumptions that consider current lifestyles and available site-specific or region-specific information.	Include	Y	Y	N
0	0	6	Future human behaviour (target group) assumptions	Future human behaviour assumptions made concerning potentially exposed individuals or population groups considered in the safety assessment.	The habits and characteristics assumed for humans are based on reasonable conservative and plausible assumptions that consider current lifestyles and available site-specific or region-specific information.	Include	Y	Y	N
0	0	7	Dose response assumptions	Dose response assumptions made in a safety assessment to convert received dose to a measure of risk to an individual or population group.	Safety analysis uses justified assumptions and key assumptions, and rationales are clearly identified.	Include	Y	Y	Y
0	0	8	Assessment purpose	The purpose for which the safety assessment is being undertaken.	The purpose of the DSAR is to provide a safety case for the Project, demonstrating the long-term safety by being protective of workers, the public and the environment.	Include	Y	Y	Y
0	0	9	Regulatory requirements and exclusions	Regulatory requirements and exclusions are the specific terms or conditions in the national regulations or guidance relating to the post-closure safety assessment.	Regulatory requirements and guidance are central to defining the DSAR assessment criteria (CNSC 2004, CNSC 2018, IAEA 2011, IAEA 2014).	Include	Y	Y	Y
0	0	10	Model and data issues	Model and data issues are general issues affecting the safety assessment modelling process and use of data.	The modelling assumptions and uncertainty regarding modelling inputs are documented as part of the DSAR, including the appropriate conservatism incorporated. Uncertainty and sensitivity analyses are completed to illustrate the robustness of the modelling and the ISD design.	Include	Y	Y	Y

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1			**External Factors**						
1	1		**ISD Issues**						
1	1	1	Site investigation	Factors related to the investigations carried out to characterize the site both during closure and post-closure.	In addition to existing baseline and operational data, geological and hydrogeological investigations were completed for the Project site. Site specific data available are used, and an appropriate level of conservatism is built into the modelling.	Include	Y	N	Y
1	1	2	Excavation / construction	Factors related to closure activities at the site.	Proven operational controls/mitigation will be in place during closure activities to effectively manage the associated industrial hazards, including conventional, radiological, and non-radiological hazards. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
1	1	3	Emplacement of wastes and backfilling	The methods employed for the isolation and containment of waste in the WRDF.	The concept for long-term waste management is based on the passive containment and isolation of the waste within the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	Y
1	1	4	Closure	Factors related to the decommissioning of the WR-1 Complex (e.g., grout formulation, grouting plan, and engineered cover).	The long-term performance and safety of the WRDF is assessed.	Include	Y	Y	Y
1	1	6	Waste allocation	Factors related to the placement of wastes into the WRDF, including waste type(s) and amount(s).	The remaining hazards in the WR-1 Building are largely radiological, with asbestos and organic coolant remaining in some reactor components. A comprehensive characterization campaign was performed to address data gaps and to provide quantitative estimates of residual radionuclide content remaining within WR-1 systems. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	Y
1	1	7	ISD design	Factors related to the design of the WRDF including both the safety concept (i.e., the general features of design and how they are expected to lead to a satisfactory performance), and the more detailed engineering specification for closure.	The DSAR encompasses the FEPs relevant to the design of the WRDF (e.g., grout formulation, grout application plan, and engineered cover design), including an appropriate level of conservatism used in modelling. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	Y
1	1	8	Quality assurance / Quality control	Factors related to quality assurance and control procedures during closure and post-closure.	Closure activities will be completed in accordance with CNL procedures and detailed work plans, including quality control and assurance. Long-term performance monitoring will be implemented to validate assumptions and effects predictions. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
1	1	9	Schedule and planning	Factors related to the sequence of events and activities occurring during closure.	Closure activities will be completed in accordance with CNL procedures and detailed work plans that were systematically developed.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1	1	10	Administrative control	Factors related to measures to control events at or around the WRDF both during the closure and post-closure periods.	Active site management will persist for an extended period during Institutional control. Administrative controls (e.g., records, programs, and procedures relevant to WRDF) are assumed to cease immediately following Institutional control. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
1	1	11	Monitoring of ISD Waste	Factors related to monitoring carried out during closure and post-closure periods. This includes monitoring for closure safety and monitoring of parameters related to the long-term safety and performance.	The design of the monitoring program provides a safety indicator and performance indicator. The purpose will be to confirm predicted performance of the WRDF.	Include	Y	N	N
1	1	12	Accidents and unplanned events	Factors related to accidents and unplanned events during closure and post-closure which might have an impact on long term safety and performance.	An accident or unplanned event could occur during closure activities; however, it is assumed that this event will not negatively impact the WRDF performance. This FEP is related to the long-term performance and safety of the WRDF.	Include	N	Y	Y
1	1	13	Retrievability	Factors related to any special design, emplacement, operational or administrative measure that might be applied or considered to enable or ease retrieval of wastes.	ISD waste is decommissioned in place with no intention of retrieval. Therefore, no measures are considered to facilitate retrieval of waste in the future.	Excluded	N	N	N
1	2		**Geological Processes and Effects**						
1	2	1	Tectonic movements and orogeny	Tectonic movements are movements of rock masses as a result of movements of the Earth's crustal plates; regionally the surface rocks respond to the underlying movements of plates. Orogeny is the process or period of crust folding and deforming by lateral compression to form a mountain range, often occurring over periods of hundreds of millions of years.	The process of tectonic plate movement and the process of orogeny are considered to occur over millions of years, or up to several tens of millions of years. The DSAR timescale of concern is expected to be 10,000 years; therefore, this FEP is not relevant.	Exclude	N	N	N
1	2	2	Deformation: elastic, plastic, brittle or ductile	Factors related to the physical deformation of geological structures in response to geological forces. This includes faulting, fracturing, extrusion, and compression of rocks.	Elastic and brittle are possible near the earth's surface. Brittle results in fracture, when it occurs, without displacement it results in a joint, with displacement it results in a fault. Deformation via geological forces is not expected to occur during the DSAR timescale.	Exclude	N	N	N
1	2	3	Seismicity	Factors related to seismic events and also the potential for seismic events. A seismic event is caused by rapid relative movements within the Earth's crust usually along existing faults or geological interfaces. The accompanying release of energy may result in ground movement and/or rupture (e.g., earthquakes).	The NBCC placed the WL site (and all of Manitoba) within a Seismic Zone 0. This FEP is related to the long-term performance and safety of the WRDF.	Include	N	Y	N
1	2	4	Volcanic and magmatic activity	Magma is molten, or frequently partly molten, mobile rock material, generated within or below the Earth's crust, which gives rise to igneous rocks when solidified. Magmatic activity occurs when there is movement of magma in the crust. A volcano is a vent or fissure in the Earth's surface through which molten or part-molten materials (lava) may flow, and ash and hot gases be expelled.	The Project site is within a region recognized as void of volcanic activity. This FEP is not relevant for the WRDF geographical setting.	Exclude	N	N	N
1	2	5	Metamorphism	Metamorphism is the processes by which rocks are changed by the action of heat (temperature >200 C) and pressure at great depths (usually several kilometres) beneath the Earth's surface or in the vicinity of magmatic activity.	The depth of the WRDF (near surface) excludes the potential for impacts from metamorphic processes.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1	2	6	Hydrothermal activity	Factors associated with high temperature groundwaters, including processes such as density-driven groundwater flow and hydrothermal alteration of minerals in the rocks through which the high temperature groundwater flows.	The depth of the WRDF (near surface) excludes the potential for impacts from hydrothermal activity.	Exclude	N	N	N
1	2	7	Erosion and sedimentation	Factors related to the large scale (geological) removal and accumulation of rocks and sediments, with associated changes in topography and geological/hydrogeological conditions of the host rock.	Large scale geological erosion and sedimentation is not relevant for the WRDF long-term performance and safety, and the timescale of concern. Surface erosion is addressed in FEP #2.3.12, which is concerned with more local processes over shorter periods of time.	Exclude	N	N	N
1	2	8	Diagenesis	The processes by which deposited sediments at or near the Earth's surface are formed in rocks by compaction, cementation, and crystallization; that is, under conditions of temperature and pressure normal to the upper few kilometres of the Earth's crust.	The depth of the WRDF (near surface) excludes the potential for impacts from diagenesis, and the process is not expected to occur during the DSAR timescale.	Exclude	N	N	N
1	2	9	Salt diapirism and dissolution	Salt diapirism and dissolution involves the large-scale evolution of salt formations. Diapirism is the lateral or vertical intrusion or upwelling of either buoyant or non-buoyant rock into overlying strata (the overburden) from deeper levels. Dissolution of the salt may occur where the evolving salt formation is in contact with groundwaters with salt content below saturation.	There are no salt formations present within the Project site.	Exclude	N	N	N
1	2	10	Hydrological/hydrogeological response to geological changes	Factors related to groundwater flow and pressures arising from large-scale geological changes. These could include changes of hydrological boundary conditions due to glaciation, effects of erosion on topography, and changes of hydraulic properties of geological units due to changes in rock stress or fault movements.	Large scale geological changes associated with glaciation are expected to occur 140,000 after present following glacial retreat. The process is not expected to occur within the DSAR timescale of 10,000 years. However, it is of public and regulatory interest, therefore, it is included.	Include	N	Y	Y
1	3		**Climatic Processes and Effects**						
1	3	1	Climate change, global	Factors related to the possible future, and evidence for past, long term change of global climate. This is distinct from resulting changes that may occur at specific locations according to their regional setting and also climate fluctuations (FEP #1.3.2).	Climate change is the variation on global or regional climates over time. Global climate change will lead to local climate changes around the WRDF. Current trends towards global warming are likely to lead to changes in global climate. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
1	3	2	Climate change, regional and local	Factors related to the possible future changes, and evidence for past changes, of local climate. This is likely to occur in response to global climate change, but the changes will be specific to situation, and may include shorter term fluctuations included in FEP #1.3.1.	Climate change may affect regional precipitation and evapotranspiration rates, groundwater flow, occurrence and extent of surface waterbodies, and receptor occurrence and characteristics. Global warming is likely to cause temperature and precipitation changes that could impact the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
1	3	4	Periglacial effects	Factors related to the physical processes and associated landforms in cold but ice-sheet-free environments.	Frost penetration in soils may result in ice lens formation and heaving. Climate change would result in warmer and wetter, not cold enough for permafrost conditions to develop. Permafrost would only develop at the onset of a glaciation period, which is not expected within the 10,000-year assessment time period.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1	3	5	Glacial and ice sheet effects, local	Factors related to effects of glaciers and ice sheets within the region (e.g., changes in the geomorphology, erosion, melt water, and hydraulic effects). This is distinct from the effect of large ice masses on global and regional climate included in FEP # 1.3.1 and 1.3.2.	Glaciation is expected to occur after 100,000 years, with retreat occurring 140,000 years from now or later. Exposure of the ISD waste to the surface environment 140,000 years from present is not expected to result in adverse effects to human and non-human biota. The process is not expected to occur within the DSAR timescale of 10,000 years. However, it is of public and regulatory interest and was included.	Include	N	Y	N
1	3	6	Warm climate effects (tropical and desert)	Factors related to warm tropical and desert climates, including seasonal effects, and meteorological and geomorphological effects special to these climates.	The Project site is expected to get warmer and wetter; however, tropical and desert climatic conditions are not expected to develop within the DSAR timescale.	Exclude	N	N	N
1	3	7	Hydrological/ hydrogeological response to climate changes	Factors related to changes in hydrology and hydrogeology (e.g., recharge, sediment load, and seasonality), in response to climate change in a region.	Climate change may alter hydrological conditions at the site, including precipitation and evapotranspiration and groundwater recharge. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
1	3	8	Ecological response to climate changes	Factors related to changes in ecology (e.g., vegetation, plant, and animal populations), in response to climate change in a region.	Climate change may influence the ecological receptors in the vicinity of the Project site (occurrence and characteristics). The habits and characteristics for the receptors are based on conservative and plausible assumptions.	Exclude	N	N	N
1	3	9	Human response to climate change	Factors related to human behaviour (e.g., habits, diet, size of communities), in response to climate change in a region.	Climate change may influence the human habits and characteristics of receptors in the vicinity of the Project. The habits and characteristics for the receptors are based on conservative and plausible assumptions.	Exclude	N	N	N
1	4		**Future Human Actions**						
1	4	1	Human influences on climate	Factors related to human activity that could affect change of climate either globally or in a region (i.e., greenhouse effect, deforestation).	Man-made emissions of greenhouse gases have been implicated as factors in global warming. Changes to the current climate (e.g., increased temperature and precipitation) may affect the WRDF long-term performance and safety. Human activities that affect climate change may also influence the long-term safety case for the WRDF.	Exclude	N	N	N
1	4	2	Motivation and knowledge issues (inadvertent/deliberate human actions)	Factors related to the degree of knowledge of the existence, location and/or natural of the WRDF. In addition, reasoning for deliberate interference with or intrusion into WRDF after closure, with complete or incomplete knowledge.	Deliberate interference with or intrusion into the WRDF is not encompassed by the DSAR as it is assumed the associated risks are understood with such activities. The design of the WRDF itself (i.e., value or use of the structure components) aims at preventing deliberate intrusion. Inadvertent intrusion (i.e., exploratory drilling) is taken without knowledge or awareness of the WRDF and assumes that long-term controls have failed following Institutional control. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	N
1	4	3	Un-intrusive site investigation	Factors related to airborne, geophysical or other surface-based investigation of an WRDF after closure.	Future deliberate un-intrusive site investigations of the WRDF may be undertaken after consideration of the safety aspects and associated hazards. Un-intrusive site investigations will result in negligible human exposure and will not affect the long-term performance and safety of the WRDF.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1	4	4	Drilling activities (human intrusion)	Factors related to any type of drilling or tunnelling activity in the vicinity of the WRDF. These may be conducted with or without knowledge of the structure (see FEP #1.4.2).	Inadvertent intrusion (i.e., exploratory drilling) is taken without knowledge or awareness of the WRDF and assumes that long-term controls have failed following Institutional control. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	N
1	4	6	Surface environment, human activities	Factors related to any type of human activities that may be carried out on the surface environment that can potentially affect the performance of the engineered and/or natural (geology) barriers, or the exposure pathways.	Surface activities during the closure and institutional control periods would be taken with full knowledge of the WRDF existence, location and associated hazards. Surface activities occurring after the post-institutional control period could occur with partial or without knowledge of the WRDF. Surface activities (inadvertent human intrusion) could potentially impact the performance and safety of the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	N
1	4	7	Waste management (wells, reservoirs, dams)	Factors related to groundwater and surface water management including water extraction, reservoirs, dams, and river management.	After the post-institutional control period, it is assumed that surface water will be extracted for consumption, domestic, and irrigation needs. However, there is potential that the groundwater in the vicinity of the WRDF could be used as a source of water for human use and/or industrial purposes. This water could introduce pathways for contaminant movement and may impact the WRDF long-term performance and safety.	Include	Y	Y	N
1	4	8	Social and institutional developments	Factors related to changes in social patterns, local government, and regulations.	After the post-institutional control period, changes in land use, regulatory requirements, loss of archives of the WRDF, loss of societal memory, changes in planning controls and environmental legislation, and demographic change and urban development. The decisions made in the future concerning social and institutional development may impact the WRDF long-term performance and safety.	Include	Y	Y	N
1	4	9	Technological developments	Factors related to future developments in human technology and changes in the capacity and motivation to implement technologies. This may include retrograde developments (e.g., loss of capacity to implement a technology).	The technological developments of interest are those that might change the capacity of humans to intrude deliberately or otherwise into the repository, to cause changes that would affect the movement of contaminants and affect the exposure pathways or health implications. Technical developments in the future are likely but are unpredictable over the timescale of concern. Due to the uncertainty of predictions made far into the future, the habits and characteristics of humans in the future should be based on reasonably conservative and plausible assumptions that consider current lifestyles. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
1	4	10	Remedial actions	Factors related to actions that might be taken following WRDF closure to remedy problems with the structure that, either, was not performing to the standards required, had been disrupted by some natural event or process, or had been inadvertently or deliberately damaged by human action.	During institutional control, environmental monitoring will be performed. The purpose will be to confirm predicted performance of the WRDF. If remedial actions are required, they will not impact the WRDF performance and safety, and are assumed to improve the WRDF performance and safety. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
1	4	11	Explosions and crashes	Factors related to deliberate or accidental explosions and crashes such as might have some impact on the WRDF post-closure.	The WL site is about 13 km from the nearest airfield (Lac du Bonnet airport), and over 100 km away from the nearest international airport (Winnipeg Jameson Armstrong Richardson). Given the small size of the WR-1 Building and the small footprint of the WRDF that will be present throughout post-closure, the likelihood of an aircraft crash directly affecting the Project is beyond extremely rare (<1/1,000,000 likelihood events/year). This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
1	4	12	Site development	Factors related to any type of human activities during site development that can potentially affect the performance of the engineered and/or natural (geological) barriers, or the exposure pathways.	The potential for future site development may result in inadvertent human intrusion and release of contaminants. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	N
1	4	13	Deliberate Human Intrusion	Factors related to any deliberate human intrusion within the WRDF.	Future deliberate intrusive actions within the WRDF (if required) will be carried out after consideration of the safety aspects. These future deliberate intrusive actions will be undertaken with knowledge of the WRDF and associated hazards. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
1	4	14	Pollution	Factors related to any type of human activities associated with pollution that can potentially affect the performance of the engineered and/or natural (geological) barriers, or the exposure pathways.	Pollution in the vicinity of the WRDF could occur as acid rain, soil pollution, groundwater pollution, and air pollution. The pollution could be generated dure to agricultural, industrial, or urban activities.	Exclude	N	N	N
1	4	15	Archaeology	Factors related to any type of human activities associated with archaeology that can potentially affect the performance of the engineered and/or natural (geological) barrier, or the exposure pathways (e.g., inadvertent intrusion).	Archaeological inadvertent human intrusion is taken without knowledge or awareness of the WRDF and could occur after the post-institutional control period. This type of inadvertent human intrusion (archaeology) is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
1	**5**		**Other**						
1	5	1	Meteorite impact	Factors related to the possibility of a large meteorite impact occurring at or close to the WRDF and related consequences.	Given the small size of the WR-1 Building and the small footprint of the WRDF that will be present throughout post-closure, the likelihood of an meteorite impact directly affecting the Project is beyond extremely rare (<1/1,000,000 likelihood events/year). This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
1	5	2	Species evolution	Factors related to the possibility of biological evolution of receptors by both natural selection and selective breeding/culturing occurring at or close to the WRDF and related consequences.	Species evolution relevant to the long-term safety of the Project (e.g., contaminant tolerance adaptation) are highly unpredictable. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2			**Internal Disposal System Domain Environmental Factors**						
2	**1**		**Wastes and Engineered Features**						
2	1	1	Inventory, radionuclide and other material	Factors related to the total content of the WRDF of a given type of material, substance, element, individual radionuclides, total radioactivity or inventory of hazardous substances.	The long-term safety of the Project is dependent on the waste inventory to be decommissioned in place. The available information on the radiological status is primarily based on post-operation surveys, the end-state survey completed after the preliminary decommissioning work, and measurements in the reactor core taken in 2019. Furthermore, a comprehensive characterization campaign was performed during 2017 and 2018 to address data gaps and to provide quantitative estimates of residual radionuclide content remaining within WR-1 systems. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	2	Waste form materials and characteristics	Factors related to the waste form materials and characteristics. The waste form will usually be conditioned prior to disposal (e.g., by solidification and inclusion of grout materials). The waste characteristics will evolve due to various processes that will be affected by the physical and chemical conditions of the environment. Processes that are relevant specifically as waste degradation processes, as compared to general evolution of the near field.	The physical and chemical degradation of the waste material to be decommissioned in place will be encompassed by the DSAR (e.g., hazardous lifetime). This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	3	Containment materials and characteristics	Factors related to the physical, chemical, biological characteristics of the containment at the time of disposal and also as they may evolve in the WRDF, including FEPS that are relevant specifically as degradation/failure processes.	In the long-term, engineered barriers (e.g., systems and components of the WR-1 and associated facilities, WR-1 Building foundation, grout encapsule, and engineered cover) that will comprise the WRDF are assumed to degrade in the long-term. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	4	Backfill materials and characteristics	Factors related to the physical, chemical, biological characteristics of the backfill/buffer material at the time of disposal and also as these materials evolve in the WRDF, including FEPs which are relevant specifically to buffer/backfill degradation processes.	The movement of groundwater and contaminants is affected by the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	5	Engineered barriers system (EBS), characteristics and degradation process	Factors related to the design, physical, chemical, hydraulic characteristics of the engineered barriers system at the time of closure, and also as they may evolve in the WRDF, including FEPs which are relevant specifically as EBS degradation processes.	Design life of the engineered barriers (e.g., systems and components of the WR-1 and associated facilities, WR-1 Building foundation, grout encapsule, and engineered cover) that will comprise the WRDF are assumed to degrade in the long-term. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	6	Other engineered features materials and characteristics	Factors related to the physical, chemical, biological characteristics of the engineered features (other than containers, buffer/backfill, and cover) at the time of disposal and also as they may evolve in the WRDF, including FEPs which are relevant specifically as degradation processes acting on the engineered features.	There are no other engineered features, materials and characteristics for the WRDF that are not covered in FEP #2.1.5.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	1	7	Mechanical processes and conditions (in wastes and EBS)	Factors related to the mechanical processes that affect the wastes, containment, cover, EBS, and other engineered features, and the overall mechanical evolution of near field with time. This includes the effects of hydraulic and mechanical loads imposed on wastes, containment and WRDF components by the surrounding geology.	Mechanical loads during closure activities will be addressed during detailed work planning, including the development of a systematic grout application plan. During the post-closure phase unplanned mechanical processes could lead to failure of the WRDF (i.e., concrete cap and engineered cover). This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	1	8	Hydraulic/hydrogeological processes and conditions (in wastes and EBS)	Factors related to the hydraulic/hydrogeological processes that affect the wastes, containment, cover, and other engineered features, and the overall hydraulic/hydrogeological evolution of near field with time. This includes the effects of hydraulic/hydrogeological influences on wastes, containers and WRDF components by the surrounding geology.	Factors such as precipitation infiltration and hydraulic conductivity rates and surface runoff and groundwater flow could affect the long-term safety of the WRDF. These factors may impact the performance of the WRDF and/or the migration of contaminants into the surrounding environment. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	9	Chemical/geochemical processes and conditions (in wastes and EBS)	Factors related to the biological/biochemical processes that affect the wastes, containment, cover, EBS, and other engineered features, and the overall chemical/geochemical evolution of near field with time. This includes the effects of chemical/geochemical influences on wastes, containment and WRDF components by the surrounding geology.	Chemical processes will impact the contaminant solubility, mobility and transport pathways of contaminants into the surrounding environment. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	1	10	Biological/biochemical processes and condition (in wastes and EBS)	Factors related to the biological/biochemical processes that affect the wastes, containment, cover, EBS, and other engineered features, and the overall biological/biochemical evolution of near field with time. This includes the effects of biological/biochemical influences on wastes, containment and WRDF components by the surrounding geology.	Biological/biochemical processes can play an important role in the behaviour and transport of contaminants in the environment. Biological activity (micro-organisms, bacteria) could change the physical and chemical environment around the WRDF (e.g., mobility of contaminants, selective release of specific radionuclides, and generation of gases). This FEP is related to the long-term performance and safety of the WRDF.	Include	N	N	Y
2	1	11	Thermal processes and conditions (in wastes and EBS)	Factors related to the thermal processes that affect the wastes, containment, cover, EBS, and other engineered features, and the overall thermal evolution of the near field with time. This includes the effects of heat on wastes, containment and WRDF components from the surrounding geology.	No thermal processes are expected within the WRDF, as the waste to be decommissioned in place will produce minimal heat. Some thermal fracturing of the WRDF is expected during curing, for modelling appropriate conservatism will be included. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	1	12	Gas sources and effects (in wastes and EBS)	Factors within and around the wastes, containment and engineered features (i.e., EBS) resulting in the generation of gases and their subsequent effects on the repository system.	Gases may be generated in the WRDF as a result of chemical interactions (e.g., grout pH and system alloy interaction or decay of radionuclides), or decomposition and degradation. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	1	13	Radiation effects (in wastes and EBS)	Factors related to the effects that result from the radiation emitted from the wastes that affect the wastes, containment, cover, EBS, and other engineered features, and the overall radiogenic evolution of the near field with time.	Radiation effects will not impact the WRDF performance since the waste to be decommissioned in place includes Low Level Radioactive Waste. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	1	14	Nuclear criticality	Factors related to the possibility and effects of a spontaneous nuclear fission chain reactions within the WRDF.	The irradiated fuel , heavy water and bulk organic coolant have been removed, therefore, there is no risk of criticality occurring. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	1	15	Extraneous Materials	Factors related to the effect of extraneous materials introduced into the WRDF during closure.	The available information on the radiological status is primarily based on post-operation surveys, the end-state survey completed after the preliminary decommissioning work, and measurements in the reactor core taken in 2019. Furthermore, a comprehensive characterization campaign was performed during 2017 and 2018 to address data gaps and to provide quantitative estimates of residual radionuclide content remaining within WR-1 systems. Procedures will be in place to prevent introduction of extraneous materials into the WRDF. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	**2**		**Geological Environment**						
2	2	1	Disturbed host rock	Factors related to the zone of rock around the WRDF that may be mechanically disturbed during closure, and the properties and characteristics as they may evolve both before and after closure.	The depth of the WRDF (near surface) excludes the potential for impacts from host rock disruption. performance and safety. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	2	2	Host rock	Factors related to the zone of rock around the WRDF that may be mechanically disturbed during closure, and the properties and characteristics as they may evolve both before and after closure.	The depth of the WRDF (near surface) excludes the potential for impacts from host rock disruption. performance and safety. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	2	3	Geological units, other	Factors related to the properties and characteristics of rocks other than the host rock as they may evolve both before and after closure.	The depth of the WRDF (near surface) excludes the potential for impacts from substrate disruption. performance and safety. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	2	4	Discontinuities, large scale (in geosphere)	Factors related to the properties and characteristics of discontinuities in and between the host rock and geological units, including faults, shear zones, intrusive dykes, and interfaces between different rock types.	Groundwater flow is relevant to the long-term safety of the Project and will be encompassed by the DSAR. For modelling, appropriate conservatism will be included.	Include	Y	N	Y
2	2	5	Contaminant transport path characteristics (in geosphere)	Factors related to the properties and characteristics of smaller discontinuities and features within the host rock and other geological units that are expected to be the main paths for contaminant transport through the geosphere, as they may evolve both before and after closure.	The contaminant transport pathway is influenced by the groundwater flow, release of contaminants, and contaminant mobility. The contaminant transport path will influence the WRDF long-term performance and safety.	Include	Y	N	Y
2	2	6	Mechanical processes and conditions (in geosphere)	Factors related to the mechanical processes that affect the host rock and other rock units, and the overall evolution of conditions with time. This includes the effects of changes in condition (e.g., rock stress), due to the closure and long-term presence of the in situ decommissioning structure.	The depth of the WRDF (near surface) excludes the potential for impacts from mechanical changes in host rock and substrate condition.	Exclude	N	N	N
2	2	7	Hydraulic/hydrogeological processes and conditions (in geosphere)	Factors related to the hydraulic and hydrogeological processes that affect the host rock and other rock units, and the overall evolution of conditions with time. This includes the effects of changes in condition (e.g., hydraulic head), due to the closure and long-term presence of the WRDF.	The hydraulic properties and hydrogeological processes of the host rock and other rock units after the migration of water and contaminants. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	Y	Y

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	2	8	Chemical/geochemical processes and conditions (in geosphere)	Factors related to the chemical and geochemical processes that affect the host rock and other rock units, and the overall evolution of conditions with time. This includes the effects of changes in condition (e.g., pH), due to closure and long-term presence of the WRDF.	Chemical conditions in the subgeological surround influences. groundwater characteristics, which is relevant to the long-term safety of the Project.	Include	y	Y	N
2	2	9	Biological/biochemical processes and conditions (in geosphere)	Factors related to the biological and biochemical processes that affect the host rock and other rock units, and the overall evolution of conditions with time. This includes the effects of changes in condition (e.g., microbe populations), due to closure and long-term presence of the WRDF.	Biological/biochemical processes are considered in FEP # 2.1.10. The depth of the WRDF (near surface) excludes the potential for impacts from changes in host rock and substrate condition. This FEP is not relevant to the WRDF long-term performance and safety	Exclude	N	N	N
2	2	10	Thermal processes and conditions (in geosphere)	Factors related to the thermal processes that affect the host rock and other overburden units, and the overall evolution of conditions with time. This includes the effects of changes in condition (e.g., temperature), due to closure and long-term presence of the WRDF.	The depth of the WRDF (near surface but sufficiently deep to protect from frost penetration) excludes the potential for impacts from changes in host rock and substrate condition. This FEP is not relevant to the WRDF long-term performance and safety	Exclude	N	N	N
2	2	11	Gas sources and effects (in geosphere)	Factors related to natural gas sources and production of gas within the geosphere, and also the effect of natural and induced gas production on the geosphere, including the transport of bulk gases and the overall evolution of conditions with time.	Natural gas sources in the subgeological surround may influence the long-term performance and safety of the Project.	Include	y	N	N
2	2	12	Undetected features (in geosphere)	Factors related to natural or man-made features within the geology that may not be detected during the site investigation.	Examples of possible undetected features are faults, fracture zones, induced fractures, old boreholes, and unexpected branching of known fractures. Undetected features in the subgeological surround that could influence groundwater flow are relevant to the long-term safety of the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	2	13	Geological resources	Factors related to natural resources within the geosphere, particularly those that might encourage investigation or closure at or near the site. Geological resources could include oil and gas, solid minerals, water and geothermal resources. For a near-surface WRDF, quarrying of near-surface deposits (e.g., sand, gravel or clay), may be of interest.	There are no known mineral occurrences of economic significance on WL property and there is no significant mineral potential on the site. There are no mines or excavations on or adjacent to the WR site. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	**3**	**Surface Environment**							
2	3	1	Topography and morphology	Factors related to the relief and shape of the surface environment and its evolution. This FEPs refer to local land form and land form changes with implications for the surface environment (e.g., plains, hills, valleys), and effects of river and glacial erosion thereon. In the long term, such as changes may occur as a response to geological changes, see FEP# 1.3.	Changes in topography and/or morphology could affect surface water and groundwater flow. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	3	2	Soil and sediment	Factors related to the characteristics of the soil and sediment and their evolution.	Changes in the soil and sediment characteristics could affect contaminant mobility and transport in the environment. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N

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2	3	3	Aquifers and water-bearing features, near surface	Factors related to the characteristics of aquifers and water-bearing features within a few metres of the land surface and their evolution.	Changes in the characteristics of near surface aquifers and water-bearing features could affect the migration of contaminants in the environment. rise in the groundwater level could result in waste saturation and a contaminant release pathway. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	3	4	Lakes, river, streams, and springs	Factors related to the characteristics of terrestrial surface waterbodies and their evolution.	Changes in the characteristics of surface waterbodies could influence the migration of contaminants in the environment. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	3	5	Coastal features	Factors related to the characteristics of coasts and the near-shore, and their evolution. Coastal features include headlands, bays, beaches, spits, cliffs, and estuaries.	The Project site is not located near a coast. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	3	6	Marine features	Factors related to the characteristics of seas and oceans, including the sea bed, and their evolution. Marine features include oceans, ocean trenches, shallow seas, and inland seas.	The Project site is not located near an ocean. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	3	7	Atmosphere	Factors related to the characteristics of the atmosphere, including capacity for transport, and their evolution.	Changes in the characteristics of the atmosphere (e.g., wind direction) could influence the migration of contaminants during post-closure (e.g., dust deposition); however, no airborne effluent is anticipated to be released from the WRDF. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	3	8	Vegetation	Factors related to the characteristics of terrestrial and aquatic vegetation both as individual plants and in mass, and their evolution.	Vegetation growing in the vicinity of the WRDF may take up contaminated water through roots, thus shortening the groundwater pathway to surface soil, humans, and non-human biota. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	3	9	Animal populations	Factors related to the characteristics of the terrestrial and aquatic animals both as individual animals and as populations, and their evolution.	Animals may be exposed to contaminants that have been released and transported from the waste by airborne or groundwater pathways. Some members of the local animal population represent potential human food sources, and hence are components of contaminant transport through the human food chain. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	3	10	Meteorology / Weather and Climate	Factors related to the characteristics of weather and climate, and their evolution.	Changes in weather and climate characteristics could influence the long-term safety case of the WRDF by changing the migration of contaminants into the surrounding environment (e.g., reduced WRDF performance or increased groundwater flow). This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	3	11	Hydrological regime and water balance (near-surface)	Factors related to near-surface hydrology at a catchment's scale and also soil water balance, and their evolution.	The hydrological regime is a description of the movement of water through the surface and near-surface environment. It includes the movement of materials associated with the water such as sediments and particulates. Extremes such as drought, flooding, storms and snow melt may be relevant. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	3	12	Erosion and deposition	Factors related to all the erosion and deposition processes that operate in the surface environment, and their evolution.	Erosion of the engineered cover may impact the WRDF's long-term performance and may potentially result in contaminants being released to the environment. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	Y
2	3	13	Ecological/ biological/ microbial systems	Factors related to living organisms and relationship between populations of animals, plants and microbes, and their evolution.	Ecosystems are in a continuous process of adaptation and evolution, which could result in considerable change over long-time frames. Contaminant migration may occur within the ecosystems. Plants growing over or near the WRDF might draw up contaminated water through roots, thus shortening the groundwater pathway to surface soil (non-human biota) and humans. The ecosystems of the surface environment provide the system for potential exposure of non-human biota and provide a potential exposure pathway for humans. This FEP is related to the long-term performance and safety of the WRDF.	Include	Y	N	N
2	3	14	Animals/Plants Intrusion	Factors related to animal and plant intrusion.	Animals and plants intrude into the repository, promoting the release and spread of contamination. Plants growing over or near the WRDF might draw up contaminated water through roots, thus, shortening the groundwater pathway to surface soil and humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4		**Human Behaviour**						
2	4	1	Human characteristics (physiology, metabolism)	Factors related to characteristics (e.g., physiology, metabolism), of individual humans. Physiology refers to body and organ form and function. Metabolism refers to the chemical and biochemical reactions which occur within an organism or part of an organism, in connection with the production and use of energy.	Changes in the characteristics and habits of human receptors could influence the long-term safety of the WRDF. Further, there is a high degree of uncertainty when predicting future human characteristics and habits. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	2	Adults, children, infants and other variations	Factors related to considerations of variability, in individual humans, of physiology, metabolism and habits.	Human receptors considered in the safety assessment include an Adult, 10-year-old Child, 1 year old Infant (formula/milk), and 3-month-old Infant formula/nursing). This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	4	3	Diet and fluid intake	Factors related to intake of food and water by individual humans, and the compositions and origin of intake.	The human diet can vary greatly, both qualitatively and quantitatively. The diet and habits will be influenced by agricultural practices and human factors such as culture, religion, economics, and technology. The human consumption of contaminated foods and fluids is a potential exposure pathway. There is a potential for locally grown agricultural products, fish from local waters and game animals to be contaminated. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	4	Habits (non-diet-related behaviour)	Factors related to non-diet related behaviour of individual humans, including time spent in various environments, pursuit of activities and uses of materials.	Outdoor activities (e.g., fishing, swimming), agricultural practices, dwelling location and use of physical resources (e.g., wood, peat, and water) are examples of behaviour that might give rise to modes of exposure to environmental contaminants. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	5	Community characteristics	Factors related to characteristics, behaviour and lifestyle of groups of humans that might be considered as target groups in an assessment.	The habits and characteristics that are assumed for the human receptors are based on reasonably conservative and plausible assumptions that consider current lifestyles and available site-specific or region-specific information. Human habits and lifestyles will influence the exposure pathways to humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	6	Food and water processing and preparation	Factors related to treatment of food stuffs and water between raw origin and consumption.	Food and water, processing and preparation may influence contaminant concentrations in food and water, and result in the loss of contaminants. This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
2	4	7	Dwellings	Factors related to houses or other structures or shelter in which humans spend time.	The habits and characteristics that are assumed for the human receptors are based on reasonably conservative and plausible assumptions that consider current lifestyles and available site-specific or region-specific information. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	8	Natural / Semi-natural land and water use	Factors related to use of natural or semi-natural tracts of land and water such as forest, bush and lakes.	The habits and characteristics that are assumed for the human receptors are based on reasonably conservative and plausible assumptions that consider current lifestyles and available site-specific or region-specific information. Land and water use will influence the exposure pathways to humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	9	Rural and agricultural land and water use (including fisheries)	Factors related to use of permanently or sporadically agriculturally managed land and managed fisheries.	Agricultural activities at the WL site could occur after the institution control period, under the following two scenarios: the WL site is rezoned from industrial use, or unplanned land and resource use including use of groundwater for domestic purposes. Land and water use will influence the exposure pathways to humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
2	4	10	Urban and industrial land and water use	Factors related to urban and industrial developments, including transport, and their effects on hydrology and potential contaminant pathways.	The establishment of large water use systems for industrial activities, including use of groundwater, after the institutional control period could influence the behaviour and transport of contaminants in the environment, and exposure to humans and nonhuman biota. Land and water use will influence the exposure pathways to humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
2	4	11	Leisure and other uses of environment	Factors related to leisure activities, the effects on the surface environment and implications for contaminant exposure pathways.	Recreational activities (e.g., camping, canoeing, and fishing) may occur on the WL site after the institutional control period. Land use will influence the exposure pathways to humans. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3			**Radionuclide and Contaminant Factors**						
3	1		**Contaminant Characteristics**						
3	1	1	Radioactive decay and in-growth	Radioactivity is the spontaneous disintegration of an unstable atomic nucleus resulting in the emission of sub-atomic particles. Radioactive isotopes are known as radionuclides. Where a parent radionuclide decays to a daughter nuclide so that the population of the daughter nuclide increases, this is known as in-growth.	Radioactive decay and in-growth could influence the long-term safety of the WRDF. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	1	2	Chemical/organic toxin stability	Factors related to chemical stability of non-radiological (chemical) contaminants.	Chemical stability of the non-radiological waste could influence the long-term safety of the WRDF. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	1	3	Inorganic solids/solutes	Factors related to the characteristics of inorganic solids/solutes that may be considered.	Inorganic contaminants to be encapsulated could influence the long-term safety of the WRDF. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	1	4	Volatiles and potential for volatility	Factors related to the characteristics of radionuclide and chemical contaminants that are volatile or have the potential for volatility in the WRDF or the environment.	The WR-1 may include volatile contaminants, including gaseous radionuclides or chemical species may be generated during chemical interactions (e.g., grout pH and system alloy interaction). This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	1	5	Organics and potential for organic forms	Factors related to the characteristics of radionuclide and chemical contaminants that are organic or have the potential to form organics in the WRDF or the environment.	The WR-1 will include organic contaminants (i.e., HB-40). This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	1	6	Noble gases	Factors related to the characteristics of noble gases.	The production of radon gas may expose humans and non-human biota through the inhalation of radon daughters attached to dust particles. The FEP is related to the closure and long-term performance and safety of the WRDF.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
3	2		**Contaminant Release/Migration Factors**						
3	2	1	Dissolution, precipitation and crystallisation, contaminant	Factors related to the dissolution, precipitation and crystallization of radiological and non-radiological (chemical) contaminants in the WRDF or environmental conditions.	Dissolution is a relevant contaminant release pathway, since contaminants may dissolve into water that has infiltrated the WRDF. The dissolution process is dependent on the solubility of the contaminant. Precipitation and crystallization may occur due chemical reactions; these processes may impact the contaminant mobility and the contaminant transport pathway. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	Y	Y
3	2	2	Speciation and solubility, contaminant	Factors related to the chemical speciation and solubility of radiological and non-radiological (chemical) contaminants in the WRDF or environmental conditions.	Chemical speciation refers to the distribution of a chemical element among chemical species in a system. The solubility of a substance in aqueous solution (i.e., water) is an expression of the degree to which it dissolves. Contaminant speciation and solubility are important factors affecting the behaviour and transport of radionuclides and non-radiological chemicals. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	Y	Y
3	2	3	Sorption/desorption processes, contaminant	Factor related to sorption/desorption of radiological and non-radiological (chemical) contaminants in the WRDF or environmental conditions.	Sorption processes are important for determining the transport of radionuclides and chemical contaminants in groundwater. Sorption processes are important because sorption/desorption can impact the migration of contaminants (i.e., slow down), and contribute to the spread of contaminant releases as a function of time. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3	2	4	Colloids, contaminant interactions and transport with	Factors related to the transport of colloids and interaction of radiological and non-radiological (chemical) contaminants with colloids in the WRDF or environmental conditions.	Radionuclides and non-radiological contaminants can become strongly attached to colloidal particles and this can influence their behaviour and transport in the environment, particularly in surface water and soil. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3	2	5	Chemical/ complexing agents, effects on contaminant speciation/ transport	Factors related to the modification of speciation or transport of radiological and non-radiological (chemical) contaminants in the WRDF or environmental conditions due to association with chemical and complexing agents.	The WRDF is unlikely to contain chemical complexing agents in a quantity that could have a significantly impact on the contaminant solubility and transport. This FEP is not relevant to the long-term performance and safety of the WRDF.	Exclude	N	N	N
3	2	6	Microbial/biological plant-mediated processes, contaminant	Factors related to the modification of contaminant speciation or properties due to microbial/biological/plant activity.	Biological activity could change the physical and chemical environment in the WRDF, affecting corrosion, mobility of contaminants, and selective release of specific contaminants. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
3	2	7	Water-mediated transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants in groundwater and surface water in aqueous phase and as sediments in surface waterbodies.	Contaminants can be transported by the water-mediated transport processes including advection, molecular diffusion, dispersion, matrix diffusion, percolation and multiphase transport processes. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	Y
3	2	8	Solid-mediated transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants in solid phase, for example large-scale movements of sediments, landslide, solifluction, and volcanic activity.	Relevant contaminant solid-mediated transport processes including transport by suspended sediments and erosion may affect the exposure pathways, and the impacts on human and non-human biota. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3	2	9	Gas-mediated transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants in gas or vapour phase or as fine particulate or aerosol in gas or vapour.	There is a potential for gas generation and gas-mediated transport of contaminants, which may affect the exposure pathways, and the impacts on human and non-human biota. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3	2	10	Atmospheric transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants in the air as gas, vapour, fine particulate, or aerosol.	The atmospheric system may represent a source of dilution and may also provide exposure pathways (e.g., inhalation, immersion) to humans and nonhuman biota. This FEP is relevant to the closure phase.	Include	Y	N	N
3	2	11	Animal, plant, and microbe mediated transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants as a result of animal, plant and microbial activity.	Plants/animals may intrude in the waste damaging engineered cover and allowing the spread of contamination. In addition, animals/plants may take up contaminates and be an exposure pathway to human receptors. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	Y	N
3	2	12	Human-action-mediated transport of contaminants	Factors related to transport of radiological and non-radiological (chemical) contaminants as a direct result of human activity.	The human intrusion actions into the WRDF will impact the contaminant transport and exposure pathways (see FEP #1.4.4, FEP # 1.4.6, and FEP #1.4.13). This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	Y	N
3	2	13	Food chains, uptake of contaminants	Factors related to incorporation of radiological and non-radiological (chemical) contaminants into plant or animal species that are part of the possible eventual food chain to humans.	The contaminants in the vegetation and animals can be transported on in the food chain, resulting in an exposure pathway to humans and non-human biota. This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N
3	3		**Exposure Pathway Factors**						
3	3	1	Contaminant concentrations in drinking water, foodstuff, and drugs	Factors related to the presence of radiological and non-radiological (chemical) contaminants in drinking water, foodstuffs or drugs that may be consumed by humans.	Contaminant exposure through the food chain (i.e., ingestion) may result in an exposure pathway to humans and non-human biota. Exposure through contaminated drugs is considered a negligible exposure pathway as compared consumption of drinking water and foodstuff and therefore, was excluded. This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N

FEP #			FEP Name	FEP Description	Screening Analysis	Screening Decision	Normal Evolution Scenario	Disruptive Events/ Bounding Scenario	Sensitivity Analysis
3	3	2	Contaminant concentration in environmental media	Factors related to the presence of radiological and non-radiological (chemical) contaminants in environmental media other than drinking water, foodstuff or drugs.	The atmospheric system may represent a source of dilution and may also provide exposure pathways (e.g., inhalation, immersion) to humans and nonhuman biota. This FEP is relevant to the closure phase. Contaminated environmental media (e.g., soils, surface water, groundwater, vegetation) is a potential exposure pathway that could result in dose to human receptors and non-human biota. This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N
3	3	3	Contaminant concentrations in non-food products	Factors related to the presence of radiological and non-radiological (chemical) contaminants in human manufactured materials or environmental materials that have special uses (e.g., clothing, building materials and peat).	Various non-food products could be derived from contaminated material: clothing (e.g., hides, leather, linen, wool); furniture (e.g., wood, metal); building materials (e.g., stone, clay for bricks, wood); and fuel (e.g., wood, peat). This FEP is not relevant to the WRDF long-term performance and safety.	Exclude	N	N	N
3	3	4	Exposure modes	Factors related to the exposure of humans and non-human biota to radiological and non-radiological (chemical) contaminants.	The important modes of exposure affecting humans and non-human biota are: • Ingestion (internal exposure) from contaminated soil, and drinking or eating contaminated water or foodstuffs. • Absorption (internal exposure) by uptake through the skin. • Inhalation (internal exposure) from inhaling gaseous or particulate contaminated materials. • External exposure as a result of direct irradiation from radionuclides deposited on or present on, the ground, buildings or other objects. This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N
3	3	5	Dosimetry	Factors related to the dependence between radiation or chemical toxicity effect, and the amount and the distribution of radiation or chemical toxins in the organs of the body.	Doses to receptors will be based on site-specific source-terms and exposure durations, as well as the various relevant exposure pathways. This FEP is relevant to the WRDF long-term performance and safety.	Include	Y	N	N
3	3	6	Radiological toxicity/effects (human/biota)	Factors related to the effect of radiation on humans and non-human biota.	The radiological contaminants in the WRDF include fission products and activation products. This FEP is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N
3	3	7	Non-radiological (chemical) toxicity/effects (humans/biota)	Factors related to the effects of non-radiological (chemical) contaminants on humans or non-human biota.	Non-radiological effects to human and non-human biota is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N
3	3	8	Radon and radon daughter exposure	Factors related to exposure to radon and radon daughters.	Exposure to radon and radon daughters is relevant to the WRDF closure and long-term performance and safety.	Include	Y	N	N

Y = Yes; N = No

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APPENDIX D

Derived Release Limits for AECL's Whiteshell Laboratories

Canadian Nuclear Laboratories | Laboratoires Nucléaires Canadiens

Regulatory Requirement Document

DERIVED RELEASE LIMITS FOR AECL'S WHITESHELL LABORATORIES

WL-509211-RRD-001

Revision 3

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Regulatory Requirement Document

Derived Release Limits for AECL's Whiteshell Laboratories

Whiteshell Site Documentation

WL-509211-RRD-001
Revision 3

2016 August	août 2016
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Revision History Liste de révisions

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Page 1 of /de 1

CW-511300-FM-168 Rev. 2

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Derived Release Limits for AECL's Whiteshell Laboratories	71

For Release Information, refer to the Document Transmittal Sheet accompanying this document. / Pour des renseignements portant sur la diffusion, consultez la feuille de transmission de documents ci-jointe.

Revision History / Liste de révisions

Revision / Révision					
No./N°	Date (yyyy/mm/dd)	Details of Rev. / Détails de la rév.	Prepared by Rédigé par	Reviewed by Examiné par	Approved by Approuvé par
D1	2010/12/01	Issued for "Review and Comment."	S. Chouhan N. Scheier	M. Audet A. Ethier K. Ross	
0	2011/12/19	Issued for review by AECL Safety Review Committee.	S. Chouhan N. Scheier	K. Ross J. Bond D. Grondin	P. Barnsdale G. Dolinar
1	2013/12/13	Issued for review by AECL Safety Review Committee.	S. Chouhan N. Scheier	M. Klukas	K. Ross G. Dolinar
2	2015/02/12	Changed reference number 24 from a draft report to an issued report. Issued for review by CNSC.	S. Chouhan N. Scheier	M. Klukas	K. Rogers G. Dolinar
3	2016/08/31	Accepted by CNSC for implementation. Issued as "Approved for Use"	S. Chouhan	M. Klukas	K. Ross G. Dolinar

EXECUTIVE SUMMARY

This report provides revised Derived Release Limits (DRLs) for the operation of Atomic Energy of Canada's (AECL's) Whiteshell Laboratories (WL). These DRLs supersede the values established in 2001.

The DRLs were calculated based on Canadian Standards Association (CSA) Guideline N288.1-08, which was developed with Canadian Nuclear Safety Commission (CNSC) involvement. The DRL calculations were performed using the IMPACT computer code, which embodies the recommended methodology. This code has been validated against experimental data and has been confirmed to be compliant with CSA Standard N286.7, which addresses the quality assurance of computer programs. The results of the DRL assessment were extensively verified to ensure the accuracy of the calculations.

The assumptions regarding the locations and characteristics of population groups located around the WL site are documented and justified. Many assumptions were kept the same as in the previous WL DRL calculations. In following the current DRL modelling guidance, conservative assumptions and parameter values were adopted for exposures and intakes, and best-estimate values were used for many environmental transfer parameters and contaminated food source fractions. To the extent possible, site-specific values were used for parameters describing environmental conditions at the WL site, adding to the accuracy of the assessment.

DRLs were calculated for one stack location, two roof vent locations, and one waste management area location for airborne effluents, and for one liquid effluent release location. Most of the radionuclides considered in the assessment are those included the previous WL DRL calculations. Some included previously are no longer included and three new ones (Nb-94, Ni-63 and Tc-99) were added. The potential critical groups considered in the assessment include three farm groups having full-time occupancy, and a farm group that has limited occupancy. Within these groups, six different age classes were considered, and for the two infant age classes, three milk sources were assessed (cow milk, breast milk and formula milk). Considering the number of release locations, potential critical groups and age classes included in the modelling, the WL DRL study is deemed comprehensive.

For a select group of radionuclides, the revised DRLs were compared with those established in 2001. For the six radionuclides from airborne releases, three of the revised DRL values were higher than the previous. For the seven radionuclides from liquid releases, all of the revised DRL values were significantly lower than the previous, with the main reason for this being the difference in the methods used to calculate water concentrations.

ACRONYMS AND ABBREVIATIONS

AECL	Atomic Energy of Canada Limited
B100	Building 100
B200	Building 200
B300	Building 300
B401	Building 401
B402	Building 402
CDG	COG DRL Guidance
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owner's Group
CRL	Chalk River Laboratories
CSA	Canadian Standards Association
DRL	Derived Release Limits
ICRP	International Commission on Radiological Protection
N288.1	CSA Standard N288.1-08
OBT	Organically Bound Tritium
WL	Whiteshell Laboratories
WMA	Waste Management Area

TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

TABLES

TABLE OF CONTENTS

SECTION **PAGE**

FIGURES

TABLE OF CONTENTS

SECTION **PAGE**

APPENDICES

1. INTRODUCTION

1.1 Revision of WL DRLs

This report provides revised Derived Release Limits (DRLs) for emissions of radioactive materials from Atomic Energy of Canada Limited's (AECL) Whiteshell Laboratories (WL). The DRLs apply to emissions of both airborne and liquid effluents during normal operation and supersede the DRLs established in 2001 [1]. The WL DRLs have been revised because more than five years have elapsed since they were last updated, and there have been recent changes in the methodology recommended for calculating DRLs.

The recommended revised DRL calculation methodology is documented in the CSA Guideline N288.1-08 [2] (hereafter referred to as N288.1), which is based on the earlier CANDU Owner's Group (COG) DRL Guidance (hereafter referred to as the CDG) [3]. The Canadian Nuclear Safety Commission participated in the preparation of N288.1.

The 2001 WL DRLs were based upon the methodology in CSA Standard N288.1-M87 [4], with some updates to values for parameters such as food consumption rates, dose coefficients and dose limits. Improvements to the methodology since that time (as embodied in N288.1 and the CDG) include additional environmental compartments, additional inter-compartment transfers and additional exposure pathways. Additional age classifications are now included, and a model for a nursing infant is now incorporated. Some dose coefficients have been updated since the 2001 WL DRL report, based upon revisions to radiation protection requirements recommended by the International Commission on Radiological Protection (ICRP) in ICRP-60 [5].

Another general difference in the modelling recommendations relates to the approach to conservatism. Whereas most assumptions and parameter values were treated conservatively in CSA Standard N288.1-M87, only values associated with critical group exposure factors, occupancy factors and intake rates are treated conservatively in N288.1. All other parameters are assigned realistic values. In particular, the fractions of food and water intakes drawn from contaminated sources have been made more realistic. This change was intended to reduce the degree of conservatism in the DRLs, as it is broadly recognized that multiple conservatisms yield dose projections that are not representative of the critical group concept (i.e., the projections are representative of extreme individuals). On the same basis, transfer parameters have been refined to better represent typical conditions, and additional inter-compartment transfers included in the revised modelling enable radionuclide concentrations in different environmental media and compartments (including various food products) to be estimated more realistically. This should provide better agreement between model predictions and actual measured environmental concentrations.

However, in calculating revised DRLs for the WL site, some factors were treated somewhat more conservatively than is recommended by N288.1.

1.2 The IMPACT Computer Code

Another key change in DRL modelling methodology since 2001 is the availability of modelling software. Known as IMPACT [6], this tool implements almost all aspects of the methodology recommended in the N288.1. It includes a database of parameter values, as well as user-friendly interfaces to facilitate the input of scenario-specific information. It outputs compartmental radionuclide concentrations and dose rates as well as DRLs.

The DRL calculations reported here were carried out using Version 5.4.0 of the IMPACT code and database. This is the latest official release of the code and incorporates all the submodels required for application at WL, including methods for calculating dispersion in a river and air immersion dose rates from three-dimensional plumes of contaminated airborne material. The default database has been updated with error corrections as of 2010 July 9 [7]. The software has been subject to validation and verification testing as discussed in [8] and [9]. The development of the previous version of the code (Version 5.2.2) was analysed and found to be consistent [10] with the requirements of CSA Standard N286.7 [11], which relates to software quality assurance. The development of Version 5.4.0 was also guided by, and is expected to meet the requirements of, that standard.

1.3 The Whiteshell Laboratories Site

The 4375-hectare WL site is located in the Local Government District of Pinawa in southeastern Manitoba, about 100 km northeast of Winnipeg. Most of the site and all the facilities are located on the east bank of the Winnipeg River (Figure 1-1), which in this area, flows from south to north.

The WL site is in the zone of transition between farmland to the west and the exposed part of the Precambrian Shield to the east, and is overlain by glacial till and sediments. The surrounding terrain is relatively flat, except for the small hills on both sides of the river. Part of the surrounding land is used for farming, with the rest being wooded. Sport fishing is carried in the Winnipeg River, but there is no commercial fishing in the area.

The area surrounding WL is sparsely populated. The nearest population centres are Lac du Bonnet (population approximately 1000, located 8.6 km north), Pinawa (population approximately 1500, located 13.4 km east-southeast), River Hills (population less than 100, located 11.8 km south) and Seven Sisters (population less than 100, located 8.9 km south-southeast) (Figure 1-1). Of greater interest for this study are farms which are much closer to WL.

A near-field map of WL site is shown in Figure 1-2. There are four main sources of airborne radioactive effluents at WL: Building 100 (B100) (reactor building), Building 200 (B200) (Active – Liquid Waste Treatment Centre) and Building 300 (B300) (shielded facilities and other laboratories) which are located in the complex of buildings in the main part of the site (WL-Main); and, the Waste Management Area (WMA) (compactor/baler and incinerator). The only significant source of liquid radioactive effluents from the site is the process outfall from the Active –Liquid Waste Treatment Centre.

The WL site is currently being decommissioned by AECL. Other businesses are now established on site. In general, the workers from these other businesses are classified as Nuclear Energy Workers and their radiation exposure is monitored. The exceptions are farm workers on land leased from AECL on the west bank of the Winnipeg River (Farm E).

Farm E is the farming location which is closest to a source of WL effluents. However, farm workers are present at this location for only a limited duration each year. The closest farming property with year-round occupancy (Farm F) is also on the west bank of the Winnipeg River, but it is further downriver.

The DRLs presented in this report were calculated assuming the current WL site boundary and the supervised area as shown by a black line in Figure 1-2. If the site boundary and/or the supervised area are to be reduced as a result of ongoing decommissioning of the WL site, and there are changes in the use of affected land, then the impact on DRLs will need to be evaluated.

Figure 1-1 Location of WL Site

Figure 1-2 Near-Field Map of WL Site Showing Effluent Release Locations and Potential Critical Group Locations

2. DERIVED RELEASE LIMITS

AECL's nuclear facilities are required to operate in such a way that radionuclide releases to the environment are well below their DRLs. These limits represent release rates that correspond to critical group exposures at the public dose limit. They are calculated by the licensee from the combined radiation dose that a member of the public receives through all pathways of exposure to a radionuclide that is routinely released to the environment. The DRLs are based on individual doses to average members of a critical group. The critical group is defined so as to represent a group of individuals likely to receive the highest exposures to radionuclides released from a particular source.

Where two or more potential critical groups exist and it is not obvious which would receive the greatest dose, separate calculations are made for each group. Similarly, separate calculations are performed for each age class within a group. The DRL for the radionuclide in question is set equal to the smallest DRL across the age classes and potential critical groups.

N288.1 considers only three age classes (adult, 10-year-old child, and 1-year-old infant). However, in the current assessment, the six age classes defined in ICRP-72 [12] and the CDG were considered. These are adult, 15-year-old teenager, 10-year-old child, 5-year-old child, 1-year-old infant and 3-month-old infant.

A separate DRL is calculated for each radionuclide released. However, in order to simplify compliance monitoring, some radionuclides can be grouped. For example, the gross beta/gamma emitting radionuclides released to air (and similarly to water) can be grouped together and the DRL for the most restrictive radionuclide can be applied to that group.

Since the DRL for a given radionuclide (or radionuclide group) is calculated as though only that radionuclide was present in the effluent, facilities must operate to satisfy the following additional condition:

$$\sum \frac{R_i}{DRL_i} < 1.0 \qquad (1)$$

where: R_i is the release rate of the i^{th} radionuclide (or group), DRL_i is the derived release limit for that radionuclide, and the summation takes place over all n radionuclides for releases to both air and water from all effluents.

This condition ensures that all releases combined will not cause a member of the public to receive a dose in excess of the public dose limit.

In order to ensure that this condition is met, and in order to keep public doses as low as reasonably achievable, WL facilities operate with releases at a small fraction of the DRL.

DRLs are calculated assuming that releases from the facility are reasonably continuous, and that long-term steady-state is reached in the environment. Consequently, the doses and DRLs calculated in this report are not likely to be indicative of doses that would result from short-term incidents involving abnormal radioactive releases.

Since DRLs reflect the annual dose limit, they can be calculated as annual releases. However, for operational control purposes, airborne release limits are expressed in terms of a period of one week and liquid limits are expressed in terms of a period of one month. Therefore, the DRLs calculated in this assessment are also expressed in these terms.

3. DOSE LIMITS FOR MEMBERS OF THE PUBLIC

The dose limits for members of the public as set out in the CNSC Radiation Protection Regulations [13] are given in Table 3-1. These limits are based on the 1991 recommendations of the ICRP [5] and are intended to prevent deterministic effects and to limit the occurrence of stochastic effects to an acceptable level.

Table 3-1
Dose Limits for Members of the Public

Application		**Annual Dose Limits** ($mSv\ a^{-1}$)
Effective Dose ($D_{effective}$)		1
Equivalent Dose	Skin (D_{skin})	50
	Lens of the Eye (D_L)	15
	Hands and Feet	50

Paragraph S29 of ICRP Publication 60 [5] recommends that restrictions on effective dose are sufficient to ensure the avoidance of deterministic effects in all body tissues and organs except possibly the lens of the eye and the skin, which may be subject to localized exposures. Hence there is no equivalent dose limit for other body tissues and organs.

Section 2.1.2 of the CDG states: "It has been shown that the equivalent dose to the lens of the eye will not be limiting for the purpose of setting Derived Release Limits [14]. For the lens dose (D_L) to be limiting, it must be true that $D_L > 15*$ ($D_{effective}$) and $D_L > 0.3*$ (D_{skin}). This condition is met only for Kr-83m, and the dose from this radionuclide is insignificant in comparison to other noble gases. Thus, calculations of effective dose and skin dose are sufficient for determining facility DRLs." Accordingly, only the effective and the skin doses were calculated in this WL analysis.

The possibility of the release of some energetic beta-emitting radionuclides being limited by skin dose was checked. Skin dose calculations were made for all external dose situations (air immersion, groundshine, beachshine and water immersion) for all radionuclides. In no case was the DRL based on skin dose lower than the DRL based on effective dose; hence only the results for effective doses are discussed further.

4. CALCULATION OF DERIVED RELEASE LIMITS

The steps taken in this assessment to calculate the derived release limits for WL are as follows:

1. Identify the potentially most affected members of the public, determine their characteristics with respect to exposure to radionuclides released from WL to the environment, and select a set of potential critical groups that will form the basis for the DRLs. Determine the parameter values for these groups (Section 5).
2. Identify and characterize the sources of airborne and liquid effluents, the factors influencing atmospheric and aquatic dispersion, and the specific radionuclides to be included (Sections 6 and 7).
3. Identify the environmental pathways models to be used in calculating the DRLs and any assumptions to be made in applying them (Sections 5 and 8).
4. Specify values for the transfer parameters and other data used in the model calculations and any assumptions to be made in applying them (Sections 5, 6 and 8).
5. Set up the model scenarios with the appropriate modelling software (IMPACT) (Section 9).
6. Perform a screening analysis to reduce the number of potential critical groups and release locations for which detailed dose calculations are required (Section 10).
7. Execute the final DRL calculations for each combination of effluent type, radionuclide, critical group, age class and potentially bounding release location (Section 9).
8. Determine the most restrictive DRLs for each radionuclide for airborne and liquid effluents from the DRLs based on the different age classes of the critical group and potentially bounding release locations (Section 11).
9. Confirm the results (Section 12).

Site-specific data were used in the calculations, where possible.

5. CRITICAL GROUPS

5.1 General Discussion

A critical group is a relatively homogeneous group of members of the public who represent the people most highly exposed to radionuclides released from a facility. This may be by virtue of their location or characteristics. DRLs are calculated from the mean dose in the critical group per unit radionuclide release. Recently, N288.1 has replaced the term “average member of the critical group” with the term “representative person”. This is a purely cosmetic change: N288.1 states that the representative person “is the equivalent of … the average member of the critical group”. The term “critical group” will be used here.

In the current analysis, potential critical groups have been characterized based on site-specific information rather than by making hypothetical worst-case assumptions. Most of the assumptions related to potential critical groups were the same as for the previous 2001 WL DRL calculations [1], and are conservative. Assumptions related to exposure pathways, occupancy factors and fractions of the diet that consisted of local food and water were confirmed by interviewing some WL employees residing in surrounding communities. The local fractions applied in the current DRL assessment were checked against the default recommendations in Table G.9c of N288.1 and were found to be conservative. Assumptions related to newly added pathways in IMPACT were consistent with the recent 2007 Chalk River Laboratories (CRL) DRL calculations [8], and 2010 Nuclear Power Demonstration Site DRL calculations [15].

5.2 Potential Critical Groups

In applying the critical group concept discussed above, a range of types of potential critical groups were identified as being representative of different locations and characteristics of population groups residing in the vicinity of WL.

5.2.1 Potential Critical Groups for Airborne Effluents

For airborne effluents, two types of potential critical groups were considered: farms that have year-round occupants and raise livestock (Farms A, D and F in Figure 1-2); and a farm that has limited occupancy and grows canola (Farm E). Table 5-1 gives the distances and directions of these groups from Building 200 (B200), at WL. The livestock farms are located adjacent to the WL site boundary and lie in high-frequency wind-direction sectors (N, S and NNW) from the effluent sources. A potential critical group location closer to the WMA than Farm A was not selected because the terrain to the east of Farm A is not suitable for farming.

Table 5-1
Distance and Direction of Airborne Effluent Potential Critical Groups from Building 200

Potential Critical Group	Location relative to Building 200		
	Distance (m)	Direction	
		Degrees from North	Sector
Farm A (livestock)	2993	353	N
Farm D (livestock and honey)	2913	177	S
Farm E (canola)	1313	258	WSW
Farm F (livestock)	2708	335	NNW

A wide range of types and scales of farming exists on both sides of Winnipeg River from Seven Sisters Falls to Lac du Bonnet. While it is more common for a particular farm to specialize in one animal product, in this assessment it was conservatively assumed that the livestock farm groups grow most of the animal products that they consume. All the livestock farms were assumed to be identical except that honey is only produced at Farm D and is supplied to all others.

The canola farm (Farm E) was conservatively assumed to be occupied for only 16 hours per day for two weeks for planting, for two weeks for fertilizing and for two weeks for harvesting (total of 672 hours per year).

Individual members of the public who occasionally carry out recreational activities (e.g., boating, fishing and swimming) on the Winnipeg River closer to the WL site than the locations of the above mentioned potential critical groups are not explicitly considered in the DRL assessment. This is because these activities are not typical for population groups in the area, but are done by a few extreme individuals. In the recent CRL DRL calculations [8], a scoping analysis was carried out to show that the radiological risk from short-term, occasional occupancy of the river close to CRL is not significantly higher than that from the chronic exposure received by the more remote critical groups over extended periods of time. There is no reason to suspect that the risk for similar extreme individuals at WL would be significantly higher.

The workers from other organizations (including an automotive repair shop) present on the site are classified as Nuclear Energy Workers and their radiation exposure is monitored. Therefore, they are not included in this assessment. However, members of the public who bring their vehicles for repairs are not monitored. But given the very low duration of time they spend on site (estimated to be 12 hours per year, based on 4 visits of 3 hours each), their exposure while on site is insignificant. This has been determined by screening calculations (see Section 10.2 for details).

In view of the nature of the potential critical groups, screening calculations were performed to reduce the number of combinations of potential critical group and release location for which detailed dose calculations were required. The screening calculations identified Farm A as the critical group for all nuclides except HTO (see Section 10.1 for details). For HTO, Farm F was identified as the critical group. The characteristics of these two groups are the same, and only the characteristics of these groups are discussed further in subsequent sections.

5.2.2 Critical Group for Liquid Effluents

Dilution and dispersion studies by Merritt [16] and [17] have shown that effluents released from the process outfall move downstream along the east bank of the Winnipeg River and do not reach the west bank in the vicinity of WL. Therefore, the critical group for liquid effluents is obvious, being Farm A on the east bank of the Winnipeg River, adjacent to the site boundary and 2810 m downstream from the release point (Figure 1-2).

5.3 Characteristics of the Critical Groups

5.3.1 Critical Groups for Airborne Effluents (Farm A or F)

The group members:

- reside on a full-time basis at their assumed locations,
- maintain a large garden from which they obtain a significant fraction of their fruit and vegetable needs (see Section 5.4.2 for information on food sources),
- are self-sufficient in meeting their milk, poultry and egg requirements and semi self-sufficient in beef and pork,
- feed their animals entirely on forage grown on their farm,
- meet their honey requirements by acquiring it from another local farm,
- partake in hunting on their own property to fulfill their game (deer) meat requirement,
- obtain their water from a well located on the property, and
- use a backyard swimming pool filled with well water during four months in a year.

The exposure pathways applicable to the critical groups are summarized in Table 5-2.

Table 5-2
Airborne Exposure Pathways Applicable to the Critical Groups (Farm A or F)

Pathway	Comments
Air Inhalation	✓
Air Immersion	✓
Water Immersion	✓ (well)
Groundshine (airborne deposition)	✓
Incidental Soil Ingestion	✓
Water Ingestion	✓ (well)
Plant Ingestion	✓
a) Plant Uptake via Roots	✓
b) Plant Uptake via Foliar Deposition	✓
Animal Product Ingestion	✓ Beef (on site) Pork (on site) Poultry (on site) Eggs (on site) Game (on site) Milk (on site) Honey (from Farm D)
a) Animal Uptake via Forage Ingestion	✓
b) Animal Uptake via Water Ingestion	✓ Livestock – well Game – pond
c) Animal Uptake via Inhalation	✓
d) Animal Uptake via Soil Ingestion	✓

5.3.2 Critical Group for Liquid Effluents (Farm A)

The group members:

- reside on a full-time basis at their assumed location,
- obtain their water for domestic needs (drinking, washing) from the river,
- maintain a large garden from which they supply a significant fraction of their fruit and vegetable needs,
- irrigate their lawns and gardens (a total area of 2500 m^2) with river water,
- do not irrigate forage crops (hay, grain, corn),

- are self-sufficient in meeting their milk, poultry and egg requirements and semi self-sufficient for beef and pork,
- water their animals with river water,
- swim in the river during the summer months and in a pool filled with river water during the remainder of the year,
- spend a fraction of the time occupying the shoreline for recreational purposes, and
- fish in the Winnipeg River, from which they obtain a fraction of their fish ingestion needs.

The exposure pathways applicable to the critical group are summarized in Table 5-3.

Table 5-3
Liquid Exposure Pathways Applicable to the Critical Group (Farm A)

Pathway	Comments
Air Inhalation (from volatilized radionuclides following irrigation)	✓
Air Immersion (from volatilized radionuclides following irrigation)	✓
Water Immersion	✓
Groundshine (irrigation)	✓
Incidental Soil Ingestion	✓
Beach Shine	✓
Incidental Sediment Ingestion	✓
Water Ingestion	✓
Fish Ingestion	✓
Fruit/Vegetable Ingestion	✓
a) Plant Uptake via Roots (irrigation)	✓
b) Plant Uptake via Foliar Deposition (irrigation)	✓
Animal Produce Ingestion	✓ Beef (on site) Pork (on site) Poultry (on site) Eggs (on site) Milk (on site)
a) Animal Uptake via Water Ingestion	✓

5.4 Critical Group Parameters

5.4.1 Water Sources

The water source assumptions for the critical groups are summarized in Table 5-4, and are justified in the discussions below.

Table 5-4
Water Source Assumptions

Critical Group	Drinking		Washing and Bathing		Swimming		Irrigation		Animals	
	Source	**Percentage**	**Source**	**Percentage**	**Source**	**Percentage**	**Source**	**Percentage**	**Source**	**Percentage**
Airborne Effluents	Well	100	Well	100	Pool filled with well water	100*	Well**	100	Well for livestock, Pond for deer	100
Liquid Effluents	Winnipeg River	100	Winnipeg River	100	Winnipeg River beaches and swimming pools filled with Winnipeg River water	100	Winnipeg River	100	Winnipeg River for livestock Pond for deer	100

* An outdoor pool is assumed to be operated for only four summer months in a year.

** Grey shading indicates pathways and exposures that are not included in the calculations.

For the groups considered in this assessment, water was assumed to be used for the following applications:

- drinking by humans,
- showering, washing and other domestic uses,
- swimming,
- lawn and/or garden irrigation, and
- animal watering.

Some inhabitants of the banks of the Winnipeg River in the vicinity of WL use well water and some use river water. In the absence of detailed population survey information, average values for the usage of water that is radiologically contaminated by WL effluents could not be derived and applied. Instead, it was generally assumed that 100% of the water is obtained from sources that are radiologically contaminated by WL effluents.

In N288.1, the only surface water bodies that are assumed to become contaminated by airborne effluents are small ponds. This distinction is made because larger bodies (lakes and rivers) provide significant dilution of activity deposited locally from the atmosphere. Moreover, natural removal processes are more effective for larger water bodies. As a result, concentrations are lower in large water bodies, reducing the significance of the water exposure pathways.

Drinking Water Assumptions

For liquid effluent modelling, the critical group was conservatively assumed to draw all their water from the Winnipeg River. Thus, 100% of the drinking water was assumed to be contaminated.

For modelling airborne effluents, the critical groups were assumed to obtain their drinking water from wells, which were assumed to be contaminated.

Immersion Assumptions (External Exposures from Washing, Bathing and Swimming)

For modelling airborne effluents, the critical groups were conservatively assumed to obtain all their water for washing and bathing water from wells, which were assumed to be contaminated.

For liquid effluent modelling, 100% of the water for washing and bathing was assumed to be contaminated, because it was conservatively assumed to come from the Winnipeg River.

Immersion exposure from swimming in the Winnipeg River during three summer months was assumed for the critical group for liquid effluents. Immersion exposures from swimming in a pool supplied with water from the Winnipeg River for the remainder of the year was also assumed. In reality, a pool at a hotel in Lac du Bonnet (filled with municipal water taken from the River) is accessible to the public. However, in this assessment, a community pool was conservatively assumed to be located at the location of the critical group, where river water concentrations are much higher than those at Lac du Bonnet municipal water intake point.

For modelling airborne effluents, members of the critical groups were assumed to swim in a pool filled with well water during four months in a year.

Irrigation Assumptions

In the N288.1, lawn and garden irrigation with well water is not included in the modelling of airborne effluents. Inclusion is normally not warranted because the relative contribution of radioactivity to soil and plant tissue from irrigation is usually minor compared to the contribution from direct atmospheric deposition. On this basis, irrigation was not included in the modelling of the critical groups for airborne effluents.

In contrast, irrigation was included in the modelling of the critical group for liquid effluents. Lawn and garden watering was assumed to be done using water from the Winnipeg River.

Animal Watering Assumptions

It is discussed in the 2007 CRL DRL report [8] that a well (and not a river) is commonly used for watering of livestock. However, to be conservative, in this assessment it was assumed that livestock are watered from wells when modelling airborne effluents, and they receive water drawn from the Winnipeg River when modelling liquid effluents.

Deer are more likely to drink from small streams and ponds in forested areas than the exposed banks of the Winnipeg River. Therefore, it was assumed that game (deer) drink only from small contaminated ponds at the locations of the critical groups. Therefore, the ingestion of contaminated water by deer was modelled for airborne effluents only.

5.4.2 Food Sources

The percentages of the various food items in the diet of the critical groups that were assumed to come from contaminated sources are summarized in Table 5-5. The percentages are based upon site-specific information and judgement rather than statistical analysis, and are justified in the discussions below.

Table 5-5
Percentage of Food from Contaminated Sources

Critical Group	**Terrestrial Animal Products**							**Plant Products**				**Fish**
	Beef	**Pork**	**Poultry**	**Venison**	**Eggs**	**Cow or Breast Milk**	**Honey**	**Fruit**	**Above-Ground Vegetables**	**Potatoes**	**Grain**	
Airborne Effluents	50	50	100	100	100	100	100*	15	25	100	0**	0
Liquid Effluents	50	50	100	0	100	100	0	15	25	100	0	30

* Honey is produced on Farm D only and supplied to the critical group.

** Grey shading indicates pathways and exposures that are not included in the calculations.

In general, the percentages of food products from contaminated sources were assumed to be higher than those recommended in Table G9c of N288.1. The exceptions were: the contaminated fruit percentage was reduced from 20% to 15 %, the contaminated grain percentage was reduced from 1% to 0%, and the contaminated fish percentage was reduced from 100% to 30%.

<u>Plant Products</u>

The critical groups were assumed to grow 15% of the fruit that they eat. N288.1 gives a default value of 20%, which is appropriate for climate conditions in southern Ontario, Quebec and New Brunswick. Since southern Manitoba has a harsher climate, a lower percentage is considered reasonable.

The critical groups were assumed to grow 25% of the above ground vegetables and 100% of the potatoes (including other root vegetables) in their diet.

As stated in the 2001 DRL report, there are some farms in the neighbourhood of WL that produce grain (wheat and oats), but these crops are sold to large companies and the contamination in the final food products is diluted to negligible levels. Therefore consumption of contaminated grain was not included in the calculations.

The contaminated percentages discussed above apply for modelling both airborne and liquid effluents, for which crop contamination occurs through airborne deposition and irrigation, respectively.

Terrestrial Animal Products

Currently, no animal products are produced at the locations of the critical groups. However, within 10 km of WL there is a mix of beef and dairy farms, with some of their products being consumed on the farms. At about 20 km south-southeast from WL, there are farmers who are self-sufficient in milk, chicken and eggs and semi-self sufficient in beef and pork. It is possible that a farmer in the vicinity of WL could start producing beef, pork, poultry, eggs and milk, with some being for their own consumption. However, it is unlikely that they would be self-sufficient in beef and pork. Therefore, it was conservatively assumed that the critical groups are 50% self-sufficient in beef and pork, and 100% self-sufficient in poultry, eggs and milk, as was done in the 2001 DRL calculations.

The milk consumed by the 1-year-old infant was assumed to be either 100% cow milk or 100% formula milk. The formula milk was assumed to be prepared with local contaminated water. Either 100% formula milk or 100% breast milk was assumed to be the source of milk for the 3-month-old infant.

Deer is the main game animal in the area and the critical groups were assumed to hunt and get all the venison required from their own property. As the deer are assumed to drink from small ponds on the property and not from the Winnipeg River, the venison is contaminated by airborne effluents from WL but not by liquid effluents.

For modelling airborne effluents, it was assumed that 100% of the honey consumed is contaminated because it is available locally from Farm D. For modelling liquid effluents, the honey was assumed to be uncontaminated because the bees generally feed on forage crops which are not contaminated by liquid effluents.

Fish

The critical group exposed to liquid effluents was assumed to eat fish caught nearby in the Winnipeg River. This is reasonable given that fishing is a popular activity in the area. It was assumed that 30% of the total fish consumed is contaminated. This was based on interviews with local sportsmen.

5.4.3 Intake Rates for Humans

The assumed intake rates for food, water, soil, sediment and air are shown in Table 5-6. Most of these values are the recommended default values provided in Tables 17, 18, 19, and G9c, and Clause 7.10.2 of N288.1 for three of the age classes; and Tables 4.15, 4-16, 4-17, and G20c, and Clause 5.11 of the CDG for all of the age classes. These are the 90th or 95th percentiles of their respective distributions, which is consistent with the philosophy of using conservative values for intake rates. The adult intake rates are those for a male.

Table 5-6
Intake Rates of Food, Water, Soil and Air

Food Categories and Items	3-Month-Old Nursing Infant	3-Month-Old Formula-Milk-Drinking Infant	1-Year-Old Cow-Milk-Drinking Infant	1-Year-Old Formula-Milk-Drinking Infant	5-Year-Old Child	10-Year-Old Child	15-Year-Old Teenager	Adult (Male)	Nursing mother
Freshwater Fish (kg a^{-1})	0.31	0.31	0.91	0.91	2.69	3.1	3.48	7.41	4.75
Milk (mother's milk or cow's milk) (L a^{-1})	416	0*	371	0*	277	305	327	265	170
Beef + Beef Offal + Veal + Lamb + Rabbit (kg a^{-1})	6.1	6.1	5.4	5.4	9.6	15	20	34	22
Venison (kg a^{-1})	6.1	6.1	5.4	5.4	9.6	15	20	34	22
Pork (kg a^{-1})	0	0	3.2	3.2	7.3	11	15	29	19
Poultry (kg a^{-1})	0	0	4.6	4.6	7.7	9.8	11	20	13
Eggs (kg a^{-1})	2.9	2.9	8.4	8.4	9.6	11	15	30	19
Honey (kg a^{-1})	0.8	0.8	0.34	0.34	0.91	1.1	1.1	2	1.3
Fruit and Berries (kg a^{-1})	69	69	66	66	92	93	91	174	112
Above-Ground Vegetables + Mushrooms (kg a^{-1})	26	26	44	44	91	114	144	236	152
Potatoes (kg a^{-1})	4.6	4.6	23	23	47	63	80	104	67
Total Water Intake (L a^{-1})	0	347	0	358	365	511	657	840	840
Soil Intake (kg a^{-1})	0.044	0.044	0.044	0.044	0.12	0.12	0.12	0.12	0.12
Sediment Intake (kg a^{-1})	0.044	0.044	0.044	0.044	0.12	0.12	0.12	0.12	0.12
Inhalation Rates (m^{3} a^{-1})	1140	1140	2740	2740	6390	7850	8210	8400	8400

* Formula-milk-drinking infants (3-month-old and 1-year-old) have zero milk intake and proportionately higher water intake.

Since beef offal, veal, lamb, and rabbit are a small percentage of the diet in the WL area, they were combined into the “beef +” category in Table 5-6. Similarly, mushrooms were combined with above-ground vegetables.

The venison intake rates of the critical groups were assumed to be much higher than those recommended in N288.1 and the CDG because deer hunting is much more common among farmers in the WL area than in the general population. The intakes for the “beef +” category in Table 5-6 were assumed to be correspondingly lower, such that the total intake from these two categories is the same as in N288.1 and the CDG. This adjustment ensures a balanced energy intake for the receptors. Because of the abundance of deer in the area, it was assumed that the intakes from the venison and “beef +” categories were equal, which is similar to what was assumed in the 2007 CRL DRL calculations [8].

DRLs were not calculated for a nursing mother, but her intake rates were required to estimate radionuclide concentrations in breast milk fed to infants. The nursing mother was modelled in the same way as a terrestrial animal, but with intake rates being at the 90th percentiles rather than the median values which were used for other terrestrial animals. The higher values were used because of the increased energy requirements resulting from lactation. These intake rates are similar to those recommended by Wong [18].

5.4.4 Occupancy Factors

The occupancy factors applied to the critical groups and the different age classes were in general the recommended default values in N288.1, which include full time residential occupancy at the assumed receptor location.

An exception was with respect to swimming occupancy. For modelling liquid effluents, beach swimming was assumed to be based on a three-month period per year, rather than a four-month period, because the Winnipeg River water is colder than the average on which the N288.1 recommendations are based. N288.1 assumes that swimming takes place indoors during the period when beach swimming does not, so the pool occupancy is based on eight months per year. However, for the WL calculations it was assumed that the critical group swims in an indoor pool during nine months per year.

6. SOURCE CHARACTERISTICS AND DISPERSION

6.1 Airborne Effluents

6.1.1 Sources

Airborne effluents are discharged at four locations: the Building 100 (B100) stack, the Building 200 (B200) roof vent, the Building 300 (B300) roof vent, and the Waste Management Area (WMA). Although the WMA has two sources, the incinerator and the compactor/baler, they were treated as one, as was done in the 2001 DRL calculations. This is reasonable because they are close to each other and have similar release heights, so that at the downwind distances of the potential critical groups the differences have a marginal impact on radionuclide concentrations.

6.1.2 Atmospheric Dispersion Model

Atmospheric dispersion was modelled using the sector-averaged Gaussian model described in N288.1 and implemented in IMPACT 5.4.0.

The characteristics of the sources are shown in Table 6-1, together with the dimensions of the adjacent building. These parameter values are the same as those used in the 2001 DRL calculations.

Table 6-1
Source Characteristics and Building Dimensions used in the Atmospheric Dispersion Model

Parameter	B100	B200	B300	WMA
Physical Height of Release (m)	30.4	0	0	7.3
Stack Inside Diameter (m)	1.98	-	-	-
Stack Exit Velocity (m s^{-1})	4.6	-	-	-
Stack Gas Temperature (C°)	25	-	-	-
Ambient Air Temperature (C°)	0.4	-	-	-
Height of Nearby Building (m)	18.5	7.6	12	5.5
Smallest Horizontal Dimension of Nearby Building (m)	55	12.8	35	12.5
Cross-Sectional Area of Nearby Building (m^2)	1000	100	400	70

The B100 stack was treated as an elevated source with excess momentum and buoyancy and accompanying plume rise. The release was assumed to be affected to some extent by the wake of the adjacent building. The WMA was also treated as elevated source but without excess momentum and buoyancy. The incinerator stack is 12.2 m high while the roof vent of the compactor/baler is 7.3 m high. Therefore, a conservative value of 7.3 m was used for the release height in these calculations. The B200 and B300 roof vents were treated as ground-level sources because of building entrainment occurring at them.

Site-specific meteorological data collected routinely by AECL at WL in the past were used in the dispersion calculations. Temperature, wind speed, wind direction, and standard deviation of wind direction were measured at heights of 6, 25 and 61 m on the tower located within a 2-ha clearing about 300 m south-west of Building 300. Quality-assured values for each of these variables are available every hour from 1988 to 1995. The data from the 6-year period 1990-1995 inclusive were used for the present calculations. The 25-m level of the tower is at about the same height as the Building 100 stack. Meteorological conditions are therefore similar at the two locations and the 25-m data were used in calculations involving releases from Building 100. In contrast, Building 200 and Building 300 were treated as the ground-level sources and releases from the WMA occur from a short stack. The 6-m data represent best the meteorological conditions experienced by low-level releases and were used for these sources.

Limited, more-recent meteorological data is available for the WL site. Environment Canada routinely measures temperature and wind data at a single level above the ground surface. However, the older AECL data is more suitable for the current DRL calculations for the following reasons:

- Multi-level temperature measurements can be used to significantly improve estimates of the atmospheric stability class,
- Higher-level data better represents the meteorological conditions experienced by the release from Building 100,
- The quality assurance of the selected AECL data is believed to be better than that of the Environment Canada data,
- Although there may have been some changes in meteorological conditions over the last 15 years, these are relatively small considering the uncertainties in the atmospheric dispersion modelling.

The AECL data from the period 1990-1995 were also used in the 2001 DRL calculations.

For each tower level, triple joint frequency distributions of wind speed, wind direction and stability class were calculated from the hourly data. The full triple joint frequency distributions are reported in Appendix A. Wind roses for the two measurement levels are shown in Figures 6-1 and 6-2. The average wind speeds in each wind speed class, which were also required by the model, are listed in Table 6-2.

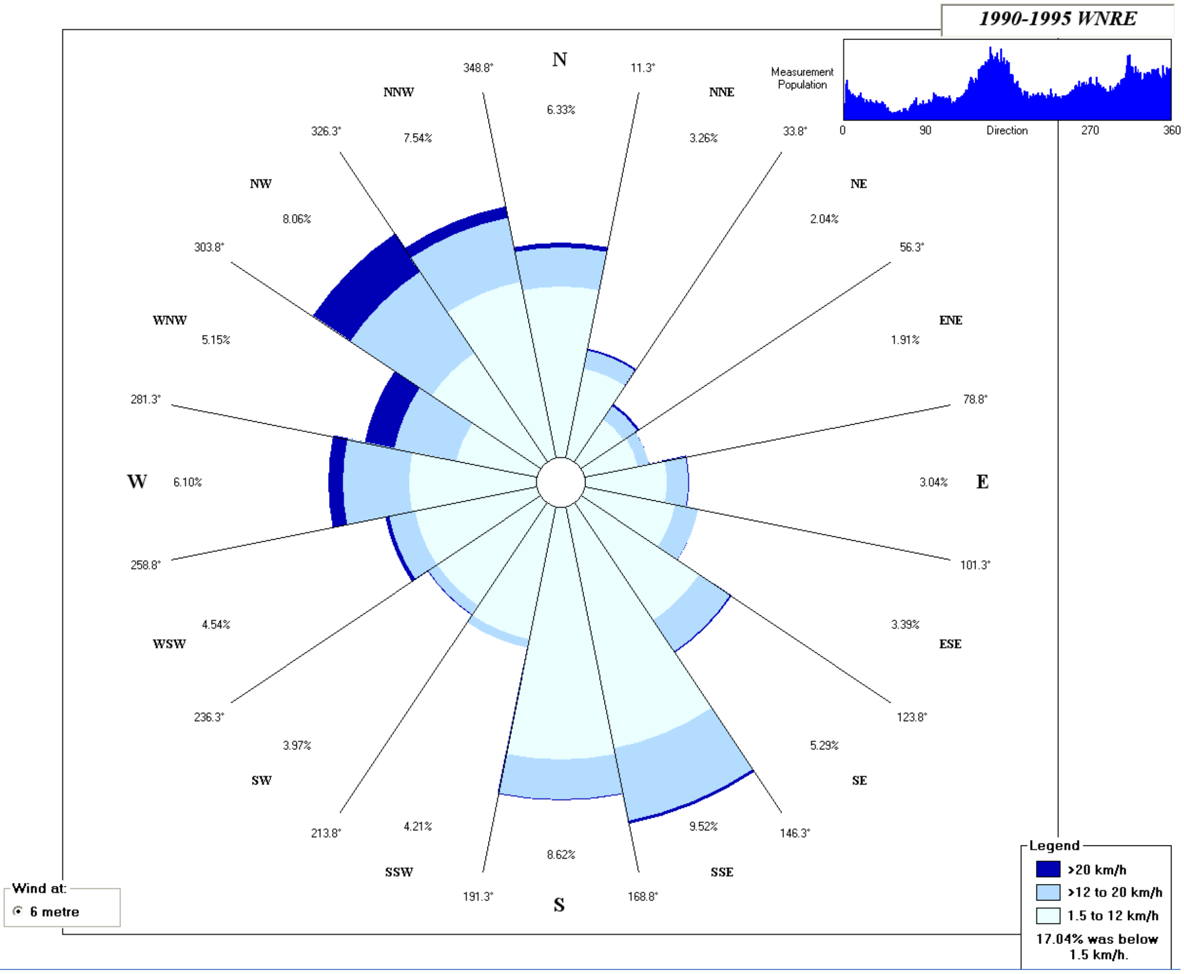

Figure 6-1 Wind Rose Diagram for the 6-m Level

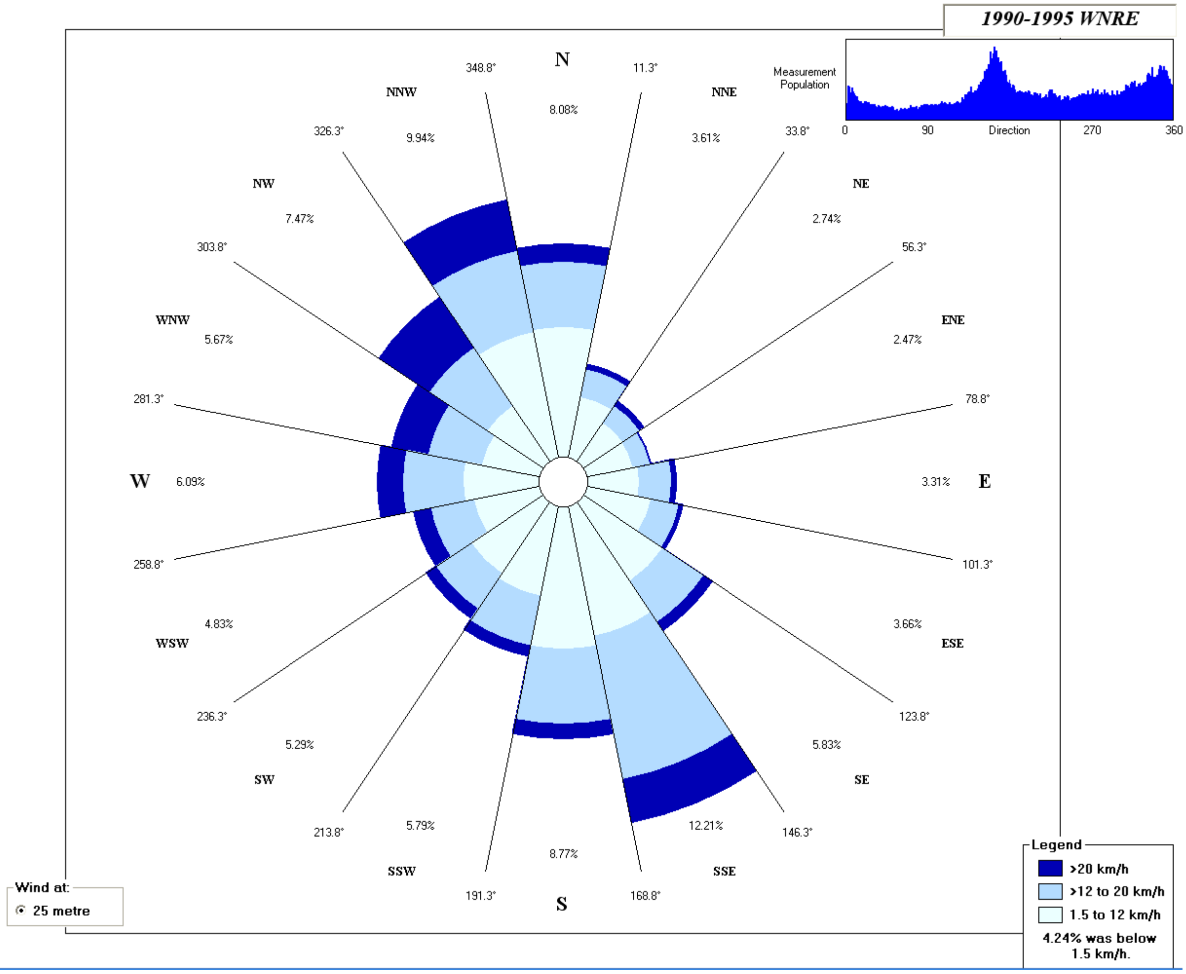

Figure 6-2 Wind Rose Diagram for the 25-m Level

Table 6-2
Mean Wind Speeds for each Wind Speed Class

Wind Speed Class	**Wind Speed Range** (m s^{-1})	**Mean Speed** (m s^{-1})	
		6-m level	**25-m level**
1	0-2	0.85	1.14
2	2-3	2.47	2.50
3	3-4	3.44	3.47
4	4-5	4.42	4.45
5	5-6	5.44	5.45
6	> 6	7.05	7.28

The land between the sources and the potential critical groups is partly wooded and partly farmland. Therefore, the meteorological roughness length was set equal to 0.4 m, as was done in the 2001 DRL calculations.

All other parameters required to calculate atmospheric dispersion were assigned the values recommended in N288.1.

6.2 Liquid Effluents

6.2.1 Sources

At the WL site, liquid effluents are discharged to the Winnipeg River continuously through the process outfall, twice a year from the sewage lagoon; and intermittently through small natural streams. Of these sources, only the process outfall (shown as "Outfall" in Figure 1-2), is significant enough for explicit inclusion in the calculation of DRLs. As has been done in the past, the DRLs calculated for the process outfall can be applied to the sewage lagoons, because the distance between the two sources is small compared to their distances from the critical group.

6.2.2 River Dispersion Model

The concentrations of radionuclides in the river water at the location of the water intake for the critical group have been calculated using the two-dimensional advection-dispersion model of N288.1. The model parameters include the river width, the river depth, the current velocity, the longitudinal and lateral dispersion coefficients, the offshore distance to the release point and the offshore distance to the point of water intake.

The river width was estimated from a topographic map to be 470 m. Based on the average river flow rate of 1.01E6 L s^{-1} for the period 2003-2008 [19] and the study of Merritt [16], the current velocity was estimated to be 0.28 m s^{-1}. Based on this width, flow rate and velocity, the river depth was estimated to be 7.7 m. The release point is located 8 m offshore. The offshore distance to the point of water intake was conservatively assumed to be 8 m also.

N288.1 recommends that values of longitudinal and lateral dispersion coefficients for the model are best determined from site-specific dispersion studies. In the previous DRL calculations [1], the dilution resulting at the location of the water intake was estimated based on the results of a short-term tracer test [16]. However, the radionuclide concentrations measured subsequently during routine monitoring of river water at a location 1930 m downstream from the release point and 880 m upstream from the point of water intake were consistently much higher than estimates based on the measured release rates and the dilution estimated from the tracer test. This has led to doubts as to the applicability of the results of the tracer test in DRL calculations.

Therefore, for the revised DRL calculations, it was decided to calibrate the river model using Sr-90 and C-137 concentration data obtained from ten years (2003-2012) of routine monitoring of the river water at the location 1930 m downstream from the release point, the river water upstream from the release point and the effluent in the process outfall [19, 20, 21, 22]. In this calibration, the longitudinal dispersion coefficient was set equal to 150 $m^2 s^{-1}$, the value recommended by N288.1 for the Ottawa River downstream of the CRL site. This was done because data limitations made it impossible to estimate independent values of the longitudinal and lateral coefficients, and because the model predictions are very insensitive to the value of the longitudinal dispersion coefficient. The value for the Ottawa River was selected because the Ottawa River is similar in size to the Winnipeg River. The lateral dispersion coefficient was calibrated to be 7.4E-7 $m^2 s^{-1}$.

7. RADIONUCLIDES

DRLs have been calculated for all thirty radionuclides that have been recently found or are reasonably expected to be found in WL's airborne and liquid effluents (see Table 7-1). Most of these radionuclides are the same as those included in the 2001 WL DRL calculations. Some radionuclides that were included in the previous calculations were not included in the current calculations, because they are not detected at the WL site any more. Three radionuclides (Nb-94, Ni-63 and Tc-99) have been added because parameter values for them were not available in the previous version of the CSA Standard N288.1 (N288.1-M87) [4] but are now available in the new version (N288.1-08) [2].

Table 7-1
Radionuclides Considered for Airborne and Liquid Effluents

Am-241	Fe-55	Pu-240
Am-243 (Np-239d, Pu-239dd)	HTO**	Pu-241 (Am-241d)
C-14*	I-129	Pu-242
Ce-144 (Pr-144d)	Mn-54	Sb-125 (Te-125md)
Cm-244	Nb-94	Sr-90 (Y-90d)
Co-60	Ni-63	Tc-99
Cs-134	Np-237 (Pa-233d)	U-234
Cs-137 (Ba-137md)	Pm-147	U-235 (Th-231d)
Eu-152	Pu-238 (U-234d)	U-238 (Th-234d, Pa-234mdd)
Eu-154	Pu-239	Zn-65

* For airborne effluents, C-14 was assumed to be released as CO_2.

** For airborne effluents, HTO was assumed to be released from Building 100 only.

In Table 7-1, radioactive daughters which are possibly significant (e.g., Np-239d, Pu-239dd) are given in parentheses after their parent (e.g., Am-243). They are not released directly, but the ingrowth of these daughters and their transfer through the environment were modelled explicitly in IMPACT, and were taken into account in determining the DRL for the parent. The letter 'd' following the radionuclide name indicates the first daughter, and the letters "dd" indicate the second daughter.

8. ENVIRONMENTAL PATHWAYS MODELS

8.1 General Discussion

The environmental pathways models described in the N288.1 were used in this analysis. These are illustrated in Figures 8-1 and 8-2 as flowcharts, which provide a summary of the environmental compartments and transfer mechanisms applied in the current modelling for the critical groups. Each compartment treated in the model is numbered and the quantity in compartment i is denoted by X_i. Transfer from compartment i to compartment j is characterized by a transfer parameter P_{ij}, such that the amount present in compartment j under steady-state conditions due to transfer from compartment i is $P_{ij}X_i$. The various compartments, transfer parameters and their units are summarized in Tables 8-1 and 8-2.

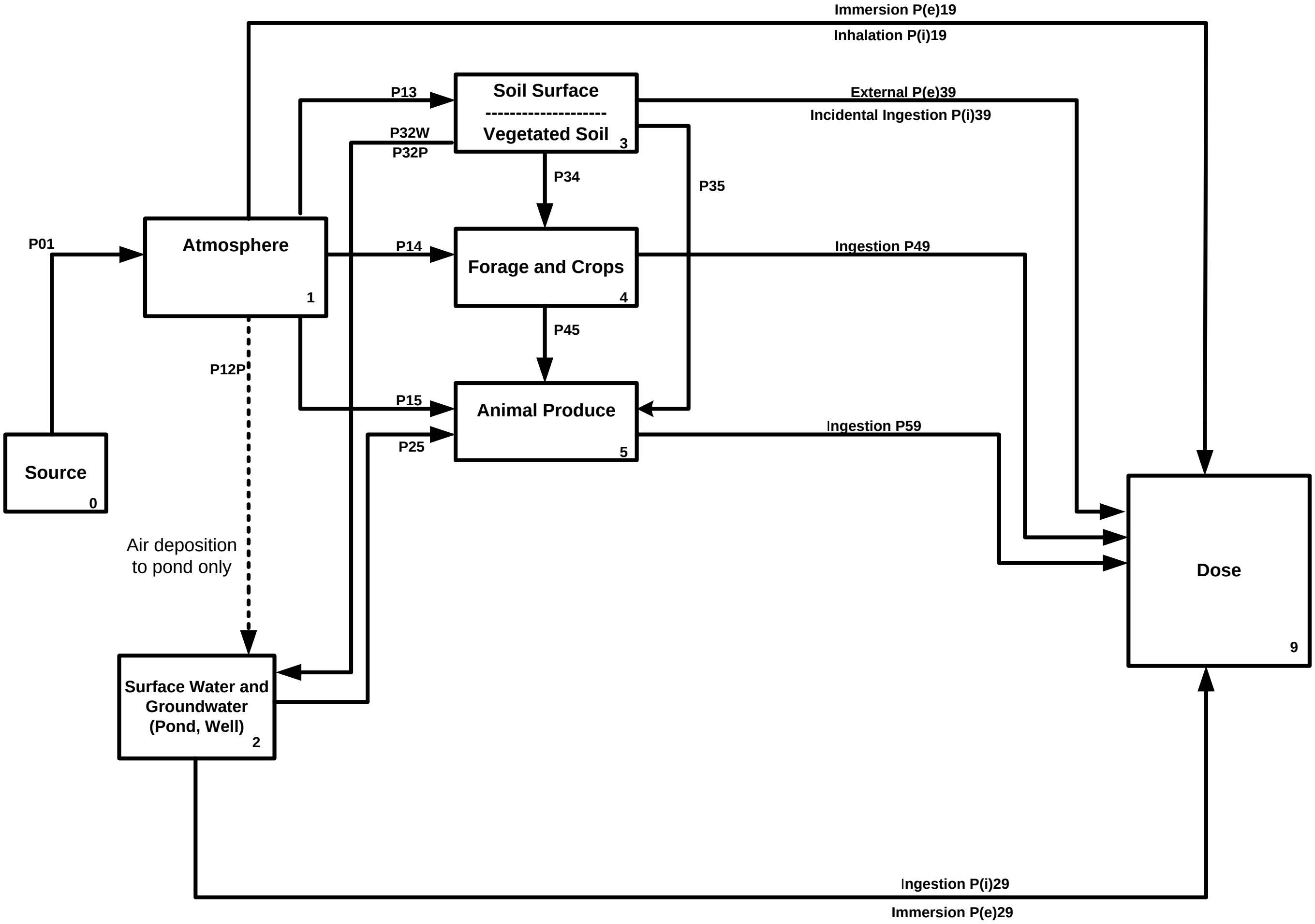

Figure 8-1 Environmental Transfer Model for Airborne Effluent Modelling

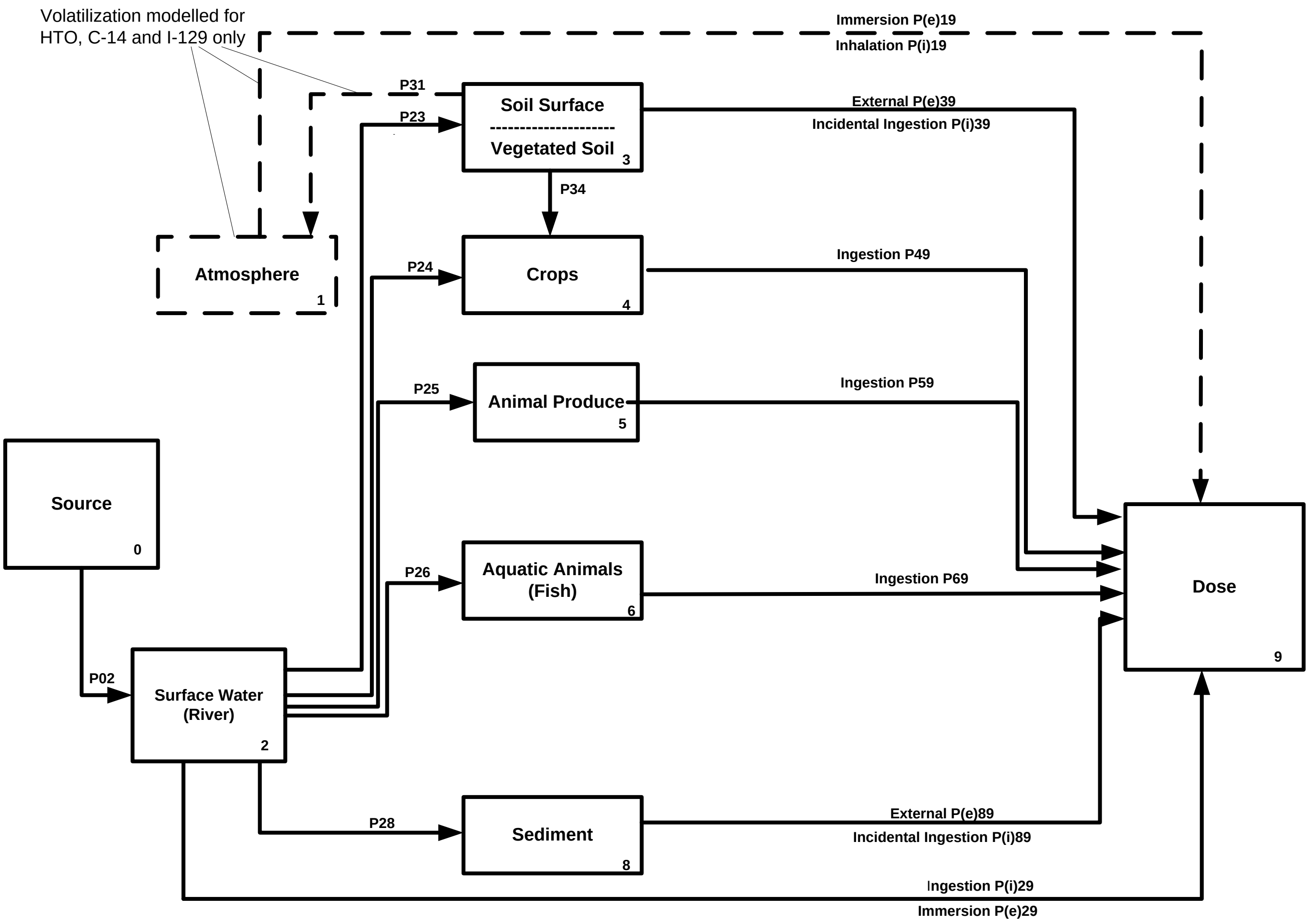

Figure 8-2 Environmental Transfer Model for Liquid Effluent Modelling

Table 8-1
Transfer Compartments and their Units

Compartment Number	Compartment Name	Units
0	Source	Bq • s^{-1}
1	Atmosphere	Bq • m^{-3}
2	Surface Water (river)	Bq • L^{-1}
2p	Surface Water (pond)	Bq • L^{-1}
2w	Ground Water (well)	Bq • L^{-1}
3area	Surface Soil	Bq • m^{-2}
3mass	Bulk Soil	Bq • kg^{-1} dw*
3spw	Soil Pore Water	Bq • L^{-1}
4	Forage and Crops	Bq • kg^{-1} fw†
5	Animal Produce	Bq • kg^{-1} fw
6	Aquatic Animals (fish)	Bq • kg^{-1} fw
8	Sediment	Bq • kg^{-1} dw
9	Dose	Sv • a^{-1}

[*] Dry weight

† Fresh weight

Table 8-2
Transfer Parameters and their Units

Transfer Parameter	Compartments		Parameter Units
	From	To	
P_{01}	Source	Atmosphere	s•m^{-3}
P_{3area1}	Surface Soil	Atmosphere	m^{2}•m^{-3}
P_{3mass1}^{*}	Bulk Soil	Atmosphere	kg dw • m^{-3}
P_{12p}	Atmosphere	Surface Water (pond)	m^{3}•L^{-1}
P_{13area}	Atmosphere	Surface Soil	m^{3}•m^{-2}
P_{13mass}	Atmosphere	Bulk Soil	m^{3}•kg^{-1} dw
P_{13spw}	Atmosphere	Soil Water	m^{3}•L^{-1}
P_{14}	Atmosphere	Forage and Crops	m^{3}•kg^{-1} fw
P_{15}	Atmosphere	Animal Produce	m^{3}•kg^{-1} fw
$P(i)_{19}$	Atmosphere	Dose (inhalation)	Sv•a^{-1}•Bq^{-1}•m^{3}
$P(e)_{19}$	Atmosphere	Dose (immersion)	Sv•a^{-1}•Bq^{-1}•m^{3}
P_{02}	Source	Surface Water (river)	s•L^{-1}
P_{3spw1}^{**}	Soil Water	Atmosphere	L • m^{-3}
$P_{3area2p}$	Surface Soil	Surface Water (pond)	m^{2}•L^{-1}
$P_{3area2w}$	Surface Soil	Groundwater (well)	m^{2}•L^{-1}
$P_{3area3spw}$	Surface Soil	Soil Water	m^{2}•L^{-1}
P_{3spw2w}	Soil Water	Groundwater (well)	unitless
P_{3spw2p}	Soil Water	Surface Water (pond)	unitless
P_{23area}	Surface Water	Surface Soil	L•m^{-2}
P_{23mass}	Surface Water	Bulk Soil	L•kg^{-1} dw
P_{23spw}^{**}	Surface Water	Soil Water	unitless
P_{24}	Surface Water	Forage and Crops	L•kg^{-1} fw
P_{25}	Surface Water (lake, river)	Animal Produce	L•kg^{-1} fw
P_{2p5}	Surface Water (pond)	Animal Produce	L•kg^{-1} fw
P_{2w5}	Well Water	Animal Produce	L•kg^{-1} fw
P_{26}	Surface Water	Aquatic Animal	L•kg^{-1} fw
P_{28}	Surface Water	Sediment	L•kg^{-1} dw
$P(i)_{29}$	Surface Water	Dose (ingestion)	Sv•a^{-1}•Bq^{-1}•L
$P(i)_{2w9}$	Well Water	Dose (ingestion)	Sv•a^{-1}•Bq^{-1}• L
$P(e)_{29}$	Surface Water	Dose (immersion)	Sv•a^{-1}•Bq^{-1}•L
$P(e)_{2w9}$	Well Water	Dose (immersion)	Sv•a^{-1}•Bq^{-1}•L
P_{3mass4}	Bulk Soil	Forage and Crops	kg dw•kg^{-1} fw
P_{3mass5}	Bulk Soil	Animal Produce	kg dw•kg^{-1} fw
$P(i)_{3mass9}$	Bulk Soil	Dose (ingestion)	Sv•a^{-1}•Bq^{-1}•kg dw

Transfer Parameter	**Compartments**		**Parameter Units**
	From	**To**	
$P(e)_{3area9}$	Surface Soil	Dose (groundshine)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet m^2$
P_{45}	Forage and Crops	Animal Produce	$kg\ fw \bullet kg^{-1}\ fw$
P_{49}	Forage and Crops	Dose (ingestion)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet kg\ fw$
P_{59}	Animal Produce	Dose (ingestion)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet kg\ fw$
P_{69}	Aquatic Animals	Dose (ingestion)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet kg\ fw$
$P(i)_{89}$	Sediment	Dose (ingestion)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet kg\ dw$
$P(e)_{89}$	Sediment	Dose (beachshine)	$Sv \bullet a^{-1} \bullet Bq^{-1} \bullet kg\ dw$

[*] For C-14 and radioiodine only

** For HTO only

The application of the models of the N288.1 to the WL assessment is discussed briefly in the sections below, emphasizing the few cases where it was necessary to deviate from the recommended models and parameter values.

8.2 Special Radionuclides

8.2.1 Tritium and Carbon 14

As recommended in N288.1, the models used to calculate DRLs for tritiated water (HTO) and C-14 were based mainly on specific activity (SA) concepts. For tritium, SA models were used for all pathways except for the final calculation of doses, where an uptake model was used instead. For C-14, SA models were used for all pathways except transfers to animals (where a transfer factor was used, which was still derived from SA consideration) and the calculation of dose (where an uptake model was used). HTO absorption by skin was taken into account by increasing the dose from HTO inhalation by 50%, as recommended in N288.1.

HTO can form stable bonds with carbon in plants and animals, in which case it is known as organically bound tritium (OBT). The DRLs for HTO take into account OBT formed in the environment.

The default parameter values recommended in N288.1 were used throughout the tritium and C-14 models.

8.2.2 I-129

I-129 was modelled in the same manner as other radionuclides except that, in the case of liquid effluents, its volatile nature was taken into account by including volatilisation following irrigation. Resulting air inhalation and immersion doses were calculated from the air concentrations estimated by the model.

8.3 Special Receptors

In order to facilitate the modelling of infant dose from the consumption of mother's breast milk, the concentrations of radionuclides in the milk were calculated. This modelling was carried out in IMPACT by modelling the lactating mother in the same way as a terrestrial animal and considering breast milk to be an animal product. The animal transfer models in N288.1 were used for the mother, as shown in Figure 8-3. The nursing mother's intakes of food, water and air were discussed and listed in Section 5.4.3 and Table 5-6. Values for all of the other parameters required by the model were left at the default values in the IMPACT database. Figure 8-3 is provided to demonstrate only the contribution from mother's milk to infant's ingestion dose. Infants receive additional doses from other pathways as shown in Figures 8-1 and 8-2.

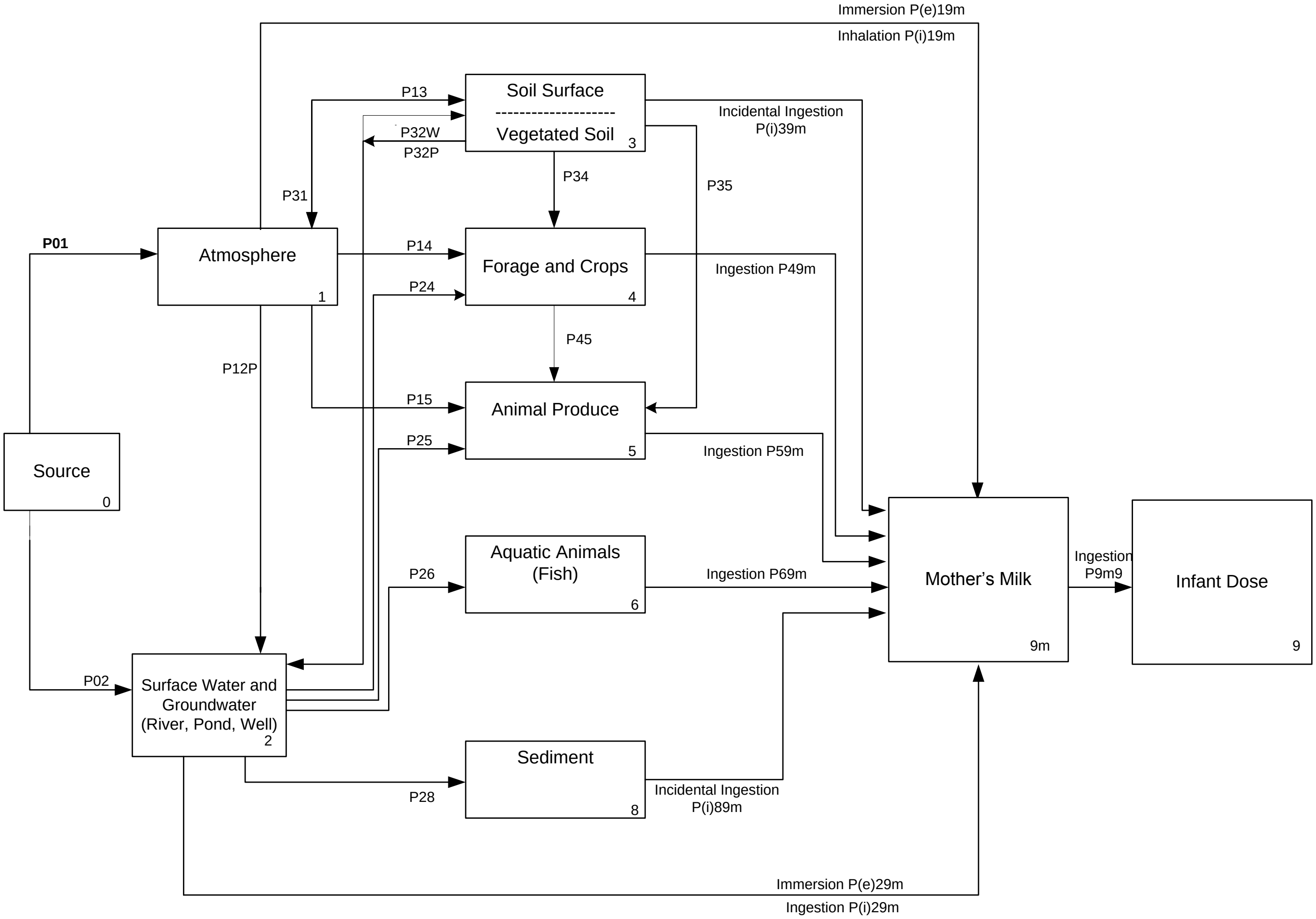

Figure 8-3 Supplemental Model for Mother's Milk

8.4 Other Parameter Values

8.4.1 Soil Types

The surface soil in the vicinity of WL was assumed to be clay, based on the recommendations of Killey [23]. The subsoil was conservatively assumed to be sand.

8.4.2 Well Depth

In areas having clay over bedrock, water supply wells are most likely to extend into the bedrock until some sufficiently permeable fracture zone is encountered, possibly at depths up to 100 m [21]. However, in some areas close to the WL site sands and sandy till are present, and it is possible to obtain a domestic water supply with a very shallow (5-10 m depth) dug well. Since the shallow supply well is conservative, a well depth of 6 m was assumed in this assessment.

The dose results generated from the DRL modelling demonstrate that the groundwater pathway is not of great importance for most radionuclides. However, for HTO, Tc-99 and Np-237, well water ingestion is one of the dominant pathways.

8.4.3 Pond Model

Although in reality ponds are fed by precipitation, groundwater and surface water inflows; only precipitation and groundwater inflow were included in the IMPACT pond model. This is conservative, because uncontaminated surface water inflow flushes radioactivity from the pond.

The parameter values assumed for the small ponds that were assumed to provide drinking water for deer are listed in Table 8-3. These are the values suggested in N288.1. In the IMPACT default database, the value for the sediment dry bulk density was found to be incorrect, and the database maintainers were notified of this error.

Table 8-3
Parameters Values used in the Pond Model

Parameter	Value
Effective Soil Porosity	0.2
Pond Surface Area (m^2)	5000
Pond Depth (m)	2
Horizontal Linear Groundwater Velocity ($m\ s^{-1}$)	1.58E-7
Groundwater Inflow Rate to Pond ($L\ s^{-1}$)	5.06E-3
Sediment Dry Bulk Density ($kg\ m^{-3}$)	400

8.4.4 Wet Deposition Velocity

The calculation of the wet deposition velocity involves a term f_{pj}, which is defined as the fraction of the time that precipitation occurs when the wind blows from sector j. Precipitation data routinely collected by AECL at WL in the past is incomplete, in that it does not include the contribution from snow. Therefore, the f_{pj} values were created using Environment Canada's meteorological data for the WL site, for the period of 2004 May to 2009 April inclusive (Table 8-4).

Table 8-4
Values of f_{pj}

Sector	f_{pj} Value
N	0.087
NNE	0.114
NE	0.111
ENE	0.084
E	0.072
ESE	0.064
SE	0.061
SSE	0.04
S	0.047
SSW	0.047
SW	0.052
WSW	0.053
W	0.057
WNW	0.071
NW	0.065
NNW	0.061

The calculation of the wet deposition velocity also requires a value for the annual average precipitation (rain + snow). The value measured at WL for the period of 1971 to 2000 [24], 565 mm, was used.

8.4.5 Volatilization from Irrigation Water

The critical group associated with liquid releases was assumed to irrigate its backyard gardens and lawns with contaminated water. In the calculations, the size of the irrigated area was assumed to be 50 m by 50 m, which was intended to account for a front yard, a back yard and a large garden plot. In calculating air concentrations following volatilization of volatile radionuclides (HTO, C-14 and I-129), the receptor was placed on the contaminated field, implying continuous exposure to the re-emitted activity.

8.4.6 Absolute Humidity

Values of annual average absolute humidity, average absolute humidity over the snow-free period and average absolute humidity over the growing season are required for modelling HTO. For the WL area, the snow-free period is approximately from May 15 to November 15 and the growing season is estimated to be from June 1 to September 30. Based on Environment Canada data measured at the WL site between 2008 Oct 1 and 2009 September 30, the following humidity values were used: annual average absolute humidity 0.00541 L m^{-3}; average absolute humidity over the snow-free period 0.0085 L m^{-3}; and average absolute humidity over the growing season 0.0102 L m^{-3}.

8.4.7 External Dose Coefficients

There are two approaches to modelling the contributions of daughter radionuclides to the dose resulting from the release of a parent radionuclide to the environment. The preferred approach is to explicitly model the ingrowth of these daughters, their transfer through the environment and the resulting dose from them. A second approach which is simpler, but in certain cases less accurate, is to explicitly model the parent only and include the contributions to dose from the daughters in the dose coefficients (DCFs). IMPACT has been designed so that either approach can be used. For the current DRL calculations, the explicit approach was used. However, the external DCFs in the default IMPACT database include the contributions of daughters. Since there is a potential for errors when correcting the many external DCFs; and since the external doses either are much less than the ingestion doses, or the contributions from the daughters to external doses are much less than those from the parent; the default external DCFs were used for the current calculations. This approach is slightly conservative.

8.4.8 Additional Changes to IMPACT Default Database

Table 8-5 shows changes to parameter values in the IMPACT default database that were made for the WL assessment and are not described elsewhere in this report.

Table 8-5
Other Parameter Values in the IMPACT Default Database that were Modified for the WL Calculations

Compartment	Parameter Name	Unit	IMPACT Default Database Value	Value Used in WL Model	Reference and Comment
Surface water (River)	Partition coefficient for Np	L kg^{-1}	25	40	N288.1 Clause 7.8.2
Surface water (Pond)	Partition coefficient for Np	L kg^{-1}	65	125	N288.1 Clause 6.6.2.2
Surface Water (Pond)	Net Precipitation Rate	mm a^{-1}	369	6	=WL precipitation rate – WL evapotranspiration rate, [25]
Groundwater (Well)	Rate of Infiltration to Aquifer	$m^3 m^{-2} s^{-1}$	4.757E-9	4.043E-9	Assumed same as soil infiltration rate below, although N288.1 Clause 6.5.2.2 suggests that it can be lower than soil.
Soil	Infiltration Rate	$m^3 m^{-2} s^{-1}$	1.142E-8	4.043E-9	= 0.5 (WL precipitation rate – 0.31) m a^{-1}, N288.1 Clause 6.3.6.3
Soil (link from water via air)	Annual Average Irrigation Rate	L $m^{-2} s^{-1}$	Not available	1.1E-5	N288.1 Clause 7.2.3.2.2
Terrestrial Plants	Fraction of Plant Carbon Derived from Air (all sources other than irrigation water)	-	1	0.7 or 1.0*	N288.1 Clauses 6.4.9.3 and 7.3.4.3
Sediment (River)	Partition coefficient for Np	L kg^{-1}	25	40	N288.1 Clause 7.8.2
Sediment (Pond)	Partition coefficient for Np	L kg^{-1}	65	125	N288.1 Clause 6.6.2.2
Dose	Fraction of Year Spent Swimming in a Surface Water Body (Beach Swimming)	-	0.014	0.011	Based on 3 months per year
Dose	Fraction of Year Spent Swimming in a Pool Filled with River Water	-	0.028	0.032	Based on 9 months per year
Full Simulation	Facility Life	years	Not available	57	WL site has been in existence for 47 years (opened in 1963) and the revised DRLs will be used for up to another 10 years.

* This value is set at 0.7 when calculating liquid effluent DRLs, but is set at 1 when calculating airborne effluent DRLs.

9. MODELLING SCENARIOS

9.1 General Discussion

At the beginning of the calculations, a scenario file was set up using IMPACT. A map covering the region of interest was imported into the model and calibrated to ensure the UTM coordinates were properly aligned. All effluent release locations and potential critical group locations were then entered into the model. Links between compartments were set up, including transfers between adjacent environmental compartments and between the potential critical groups and their food supply locations. All site-specific and scenario-specific parameters needed by the pathways models were then incorporated.

Once the scenario file was finalized, eight sub-scenario files were created. Four of these were for screening calculations for airborne effluents. Two were for detailed calculations for airborne releases of all radionuclides, except HTO, one for each potentially bounding release location. Another was for detailed calculations of the release of HTO from B100. The final one was for detailed calculations for liquid effluents. For each combination of effluent type, radionuclide, critical group, age class and potentially bounding release location, calculations of dose rate per unit release were carried out by running IMPACT. The results were searched to identify the highest dose rates per unit release for each combination of effluent type and radionuclide among all age classes and potentially bounding release locations. These dose rates per unit release were then used to calculate the DRLs for each combination of effluent type and radionuclide. For each of these combinations, the dominant exposure pathway and its percent contribution to the dose rate were also determined. The results of this analysis are presented in Section 11.

9.2 Scenarios for Airborne Effluent Modelling

Table 9-1 lists the UTM coordinates of the release locations and potential critical group locations considered in the analysis of airborne effluents.

Table 9-1
Release and Potential Critical Group Locations for Airborne Effluent Modelling

Location	**Easting** (m)	**Northing** (m)
Release		
Building 100 Stack	709909	5562658
Building 200	709928	5562834
Building 300	709714	5562668
WMA Gate	711322	5564260
Potential Critical Group		
Farm A	709547	5565803
Farm D	710067	5559924
Farm E	708642	5562571
Farm F	708783	5565288

The exposure pathways for the critical groups for airborne effluents are air inhalation, air immersion, water immersion, groundshine, soil ingestion, water ingestion, plant ingestion and ingestion of terrestrial animal products.

9.3 Scenario for Liquid Effluent Modelling

Table 9-2 lists the UTM coordinates of the release and critical group locations considered in the analysis of liquid effluents.

Table 9-2
Release and Critical Group Locations for Liquid Effluent Modelling

Location	**Easting (m)**	**Northing (m)**
Release		
WL Process Outfall	709474	5562997
Critical Group		
Farm A	709547	5565803

The exposure pathways for the critical group for liquid effluents are air inhalation, air immersion, water immersion, groundshine, soil and sediment ingestion, water ingestion, plant ingestion, ingestion of terrestrial animal products, fish ingestion and beach shine.

10. SCREENING CALCULATIONS FOR AIRBORNE EFFLUENTS

10.1 Potential Critical Groups and Release Locations

Screening calculations for airborne effluents were performed to reduce the number of combinations of potential critical group and release location for which detailed dose calculations were required.

In view of the similar nature of the potential critical groups, the screening was based on:

1. The predicted annual-average air concentrations at each group location, resulting from a unit release of each radionuclide at a release location. Concentrations at Farm E were multiplied by a factor of 0.08 to account for the fact that it has limited occupancy (672 hrs per year).
2. The predicted annual-average soil concentration at each group location, resulting from a unit release of each radionuclide at a release location, except for C-14 and HTO, for which doses do not depend on soil concentration. Concentrations at Farm E were multiplied by a factor of 0.08 to account for the limited occupancy.

Table 10-1 shows the predicted air concentrations for a subset of the radionuclides released, along with the predicted air concentrations of their daughters. Predictions for all released radionuclides having daughters are included to illustrate the differing effects of ingrowth for different combinations of potential critical group and release location. Also included in Table 10-1 are predictions for the slowest- and fastest-decaying released radionuclides that have no daughters, I-129 and Zn-65. Predictions for HTO, which is released only from B100, are also included in this table. The highest concentrations for each radionuclide are indicated by yellow shading.

For all radionuclides except Pu-239dd (second daughter of released radionuclide Am-243) and HTO, the highest predicted air concentrations are for the combination of Farm A and release from B200. For Pu-239dd, the air concentration is also highest at Farm A, but in this case in combination with release from B300. However, the air concentrations of Pu-239dd are ten orders of magnitude lower than those of its parent, Am-243, so the dose contribution from Pu-239dd will be negligible compared to that of Am-243 and it need not be considered further. Although Farm E is the potential critical group closest to release locations B100, B200 and B300, it has limited occupancy and is not located in one of the high-frequency wind-direction sectors relative to them (Figures 1-2, 6-1 and 6-2). Therefore, the scaled predicted air concentrations at this group location are lower than the concentrations at other group locations. Although the WMA is closer to Farm A than B200 is, the WMA is not the bounding release location for air concentrations because Farm A is in a low-frequency wind-direction sector relative to it (Figure 1.2 and 6.1).

For HTO, the highest predicted air concentration is at Farm F, not at Farm A. This is because HTO is released only from B100, whereas the other radionuclides are also released from B200, B300 and the WMA. Releases from B100 were assumed to be driven by the meteorological conditions typical of a height of 25 m, whereas releases from the other locations were assumed to be driven by the meteorological conditions typical of a height of 6 m. The meteorological conditions at these two heights differ significantly (Figures 6-1 and 6-2).

Table 10-1
Predicted Air Concentrations (Bq m^{-3}) at Potential Critical Group Locations as a Result of the Unit Release of Radionuclide (1 Bq s^{-1}) to the Atmosphere at a Release Location

Radionuclide	**Release from B100**				**Release from B200**				**Release from B300**				**Release from WMA**			
	Farm A	**Farm D**	**Farm E ***	**Farm F**	**Farm A**	**Farm D**	**Farm E ***	**Farm F**	**Farm A**	**Farm D**	**Farm E ***	**Farm F**	**Farm A**	**Farm D**	**Farm E ***	**Farm F**
Am-243	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Np-239d	1.05E-09	9.85E-10	3.08E-11	1.03E-09	8.44E-09	4.06E-09	2.82E-10	7.21E-09	8.38E-09	4.10E-09	3.32E-10	7.35E-09	5.93E-09	1.86E-09	1.05E-10	3.68E-09
Pu-239dd	1.03E-18	8.64E-19	1.15E-20	8.20E-19	1.32E-17	6.14E-18	1.92E-19	1.01E-17	1.38E-17	5.88E-18	1.84E-19	1.05E-17	7.27E-18	4.36E-18	1.73E-19	5.27E-18
Ce-144	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.85E-07	1.16E-07	9.31E-09	3.65E-07
Pr-144d	1.14E-07	1.12E-07	4.67E-09	1.24E-07	6.59E-07	3.26E-07	3.51E-08	6.14E-07	6.29E-07	3.42E-07	4.46E-08	6.13E-07	5.45E-07	1.07E-07	7.98E-09	3.06E-07
Cs-137	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Ba-137md	1.85E-07	1.86E-07	1.26E-08	2.30E-07	7.63E-07	3.87E-07	5.85E-08	7.53E-07	7.16E-07	4.13E-07	8.41E-08	7.45E-07	6.83E-07	1.16E-07	9.29E-09	3.64E-07
HTO	1.87E-07	1.90E-07	1.46E-08	2.37E-07												
I-129	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Np-237	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Pa-233d	9.20E-11	8.60E-11	2.68E-12	8.99E-11	7.39E-10	3.56E-10	2.46E-11	6.31E-10	7.34E-10	3.59E-10	2.90E-11	6.43E-10	5.18E-10	1.64E-10	9.23E-12	3.23E-10
Pu-238	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
U-234d	2.78E-17	2.60E-17	8.10E-19	2.71E-17	2.23E-16	1.07E-16	7.44E-18	1.91E-16	2.22E-16	1.09E-16	8.75E-18	1.94E-16	1.57E-16	4.94E-17	2.79E-18	9.74E-17
Pu-241	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Am-241d	1.58E-14	1.47E-14	4.60E-16	1.54E-14	1.27E-13	6.10E-14	4.22E-15	1.08E-13	1.26E-13	6.16E-14	4.97E-15	1.10E-13	8.88E-14	2.80E-14	1.58E-15	5.53E-14
Sb-125	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.85E-07	1.16E-07	9.31E-09	3.65E-07
Te-125md	4.28E-11	4.00E-11	1.25E-12	4.18E-11	3.44E-10	1.66E-10	1.15E-11	2.94E-10	3.41E-10	1.67E-10	1.35E-11	2.99E-10	2.41E-10	7.61E-11	4.29E-12	1.50E-10
Sr-90	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Y-90d	9.30E-10	8.69E-10	2.72E-11	9.09E-10	7.45E-09	3.59E-09	2.49E-10	6.37E-09	7.40E-09	3.62E-09	2.93E-10	6.49E-09	5.23E-09	1.65E-09	9.30E-11	3.25E-09
U-235	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Th-231d	2.32E-09	2.17E-09	6.78E-11	2.27E-09	1.85E-08	8.92E-09	6.21E-10	1.58E-08	1.84E-08	9.01E-09	7.32E-10	1.61E-08	1.30E-08	4.07E-09	2.31E-10	8.09E-09
U-238	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.86E-07	1.16E-07	9.31E-09	3.65E-07
Th-234d	1.03E-10	9.64E-11	3.01E-12	1.01E-10	8.28E-10	3.99E-10	2.76E-11	7.07E-10	8.22E-10	4.03E-10	3.25E-11	7.21E-10	5.81E-10	1.83E-10	1.03E-11	3.61E-10
Pa-234mdd	9.68E-11	9.00E-11	2.53E-12	9.27E-11	8.02E-10	3.86E-10	2.56E-11	6.82E-10	7.98E-10	3.89E-10	2.95E-11	6.96E-10	5.58E-10	1.79E-10	1.00E-11	3.49E-10
Zn-65	1.87E-07	1.90E-07	1.46E-08	2.37E-07	7.64E-07	3.89E-07	5.97E-08	7.56E-07	7.17E-07	4.14E-07	8.70E-08	7.47E-07	6.85E-07	1.16E-07	9.31E-09	3.65E-07

* The values listed for Farm E are the predicted concentrations multiplied by a factor of 0.08 to account for the limited occupancy.

Table 10-2 shows the predicted soil concentrations for a subset of the radionuclides released (except for C-14 and HTO), along with the predicted soil concentrations of their daughters. The highest concentrations for each radionuclide are indicated by yellow shading.

For all released radionuclides except I-129, the highest predicted soil concentrations are for the combination of Farm A and release from the WMA. This group location is the one having the highest predicted air concentrations, but the release location resulting in the highest air concentrations is different (B200). The difference is because precipitation occurs more frequently when the wind is blowing toward Farm A from the WMA than when it is blowing from B200 (see Figure 1-2 and Table 8-4), with the net result being more wet deposition and higher soil concentrations, even though the air concentrations are lower.

For I-129, the highest predicted soil concentration is for the same combination as the highest predicted air concentration (Farm A and B200). This is because, for I-129, wet deposition is less significant than dry deposition, whereas the opposite is true for the other radionuclides of interest. The washout ratio for I-129 is much lower than that for the other radionuclides (1.6e5 vs 5.5e6), and the dry deposition velocity for I-129 is higher (7.5e-3 m s^{-1} vs. 1.4E-3 m s^{-1}).

For many of the daughter radionuclides, the highest predicted soil concentrations are for the combination of Farm A and release from B300. However, the soil concentrations of the daughters are several orders of magnitude lower than those of the parents, so they can be ignored.

In summary, based on the results of these screening calculations, detailed dose calculations were required for only three combinations of potential critical group and release location: Farm F and B100 for HTO; Farm A and B200 for radionuclides other than HTO; and, Farm A and the WMA for radionuclides other than HTO.

Table 10-2
Predicted Soil Concentrations (Bq kg^{-1}dw) at Potential Critical Group Locations as a Result of the Unit Release of Radionuclide (1 Bq s^{-1}) to the Atmosphere at a Release Location

Radionuclide	Release from B100				Release from B200				Release from B300				Release from WMA			
	Farm A	Farm D	Farm E *	Farm F	Farm A	Farm D	Farm E *	Farm F	Farm A	Farm D	Farm E *	Farm F	Farm A	Farm D	Farm E *	Farm F
Am-243	5.97E-03	9.99E-03	6.93E-04	7.01E-03	2.43E-02	2.05E-02	2.97E-03	2.30E-02	2.33E-02	2.07E-02	4.17E-03	2.22E-02	2.80E-02	7.51E-03	5.51E-04	1.54E-02
Np-239d	6.42E-09	9.88E-09	2.80E-10	5.82E-09	5.12E-08	4.11E-08	2.69E-09	4.20E-08	5.20E-08	3.92E-08	3.04E-09	4.17E-08	4.62E-08	2.31E-08	1.19E-09	2.98E-08
Pu-239dd	3.27E-14	4.54E-14	5.48E-16	2.43E-14	4.20E-13	3.25E-13	9.55E-15	3.07E-13	4.48E-13	2.94E-13	8.82E-15	3.13E-13	2.97E-13	2.83E-13	1.02E-14	2.23E-13
Ce-144	1.37E-04	2.30E-04	1.60E-05	1.61E-04	5.59E-04	4.73E-04	6.84E-05	5.30E-04	5.36E-04	4.77E-04	9.61E-05	5.11E-04	6.44E-04	1.73E-04	1.27E-05	3.55E-04
Pr-144d	3.57E-09	5.75E-09	2.18E-10	3.58E-09	2.05E-08	1.69E-08	1.71E-09	1.83E-08	2.00E-08	1.67E-08	2.10E-09	1.78E-08	2.18E-08	6.80E-09	4.63E-10	1.27E-08
Cs-137	3.43E-03	5.74E-03	3.98E-04	4.03E-03	1.39E-02	1.18E-02	1.71E-03	1.32E-02	1.34E-02	1.19E-02	2.40E-03	1.27E-02	1.61E-02	4.32E-03	3.17E-04	8.85E-03
Ba-137md	8.48E-10	1.41E-09	8.64E-11	9.80E-10	3.49E-09	2.95E-09	4.20E-10	3.31E-09	3.35E-09	2.98E-09	5.81E-10	3.19E-09	4.02E-09	1.08E-09	7.93E-11	2.22E-09
I-129	2.45E-03	2.53E-03	1.93E-04	3.10E-03	1.00E-02	5.16E-03	7.91E-04	9.88E-03	9.38E-03	5.49E-03	1.15E-03	9.76E-03	9.03E-03	1.55E-03	1.24E-04	4.81E-03
Np-237	4.85E-03	8.10E-03	5.62E-04	5.69E-03	1.97E-02	1.67E-02	2.41E-03	1.87E-02	1.89E-02	1.68E-02	3.39E-03	1.80E-02	2.27E-02	6.10E-03	4.47E-04	1.25E-02
Pa-233d	6.44E-09	9.91E-09	2.80E-10	5.83E-09	5.15E-08	4.13E-08	2.69E-09	4.22E-08	5.23E-08	3.94E-08	3.05E-09	4.19E-08	4.64E-08	2.33E-08	1.20E-09	2.99E-08
Pu-238	4.87E-03	8.15E-03	5.66E-04	5.72E-03	1.98E-02	1.68E-02	2.43E-03	1.88E-02	1.90E-02	1.69E-02	3.41E-03	1.81E-02	2.28E-02	6.13E-03	4.50E-04	1.26E-02
U-234d	8.82E-13	1.36E-12	3.84E-14	7.99E-13	7.06E-12	5.66E-12	3.69E-13	5.79E-12	7.17E-12	5.40E-12	4.18E-13	5.75E-12	6.36E-12	3.20E-12	1.64E-13	4.10E-12
Pu-241	2.17E-03	3.63E-03	2.52E-04	2.55E-03	8.84E-03	7.47E-03	1.08E-03	8.38E-03	8.47E-03	7.54E-03	1.52E-03	8.07E-03	1.02E-02	2.73E-03	2.01E-04	5.60E-03
Am-241d	4.83E-10	7.43E-10	2.10E-11	4.37E-10	3.86E-09	3.10E-09	2.02E-10	3.17E-09	3.92E-09	2.96E-09	2.29E-10	3.15E-09	3.48E-09	1.75E-09	8.99E-11	2.24E-09
Sb-125	4.77E-04	7.98E-04	5.54E-05	5.60E-04	1.94E-03	1.64E-03	2.37E-04	1.84E-03	1.86E-03	1.66E-03	3.34E-04	1.77E-03	2.24E-03	6.01E-04	4.41E-05	1.23E-03
Te-125md	6.43E-09	9.90E-09	2.80E-10	5.83E-09	5.15E-08	4.12E-08	2.69E-09	4.22E-08	5.23E-08	3.94E-08	3.05E-09	4.19E-08	4.64E-08	2.33E-08	1.20E-09	2.99E-08
Sr-90	3.12E-03	5.22E-03	3.62E-04	3.66E-03	1.27E-02	1.07E-02	1.55E-03	1.20E-02	1.22E-02	1.08E-02	2.18E-03	1.16E-02	1.46E-02	3.93E-03	2.88E-04	8.05E-03
Y-90d	6.42E-09	9.89E-09	2.80E-10	5.82E-09	5.13E-08	4.11E-08	2.69E-09	4.21E-08	5.21E-08	3.93E-08	3.04E-09	4.18E-08	4.63E-08	2.32E-08	1.19E-09	2.98E-08
U-235	5.95E-03	9.94E-03	6.90E-04	6.98E-03	2.42E-02	2.04E-02	2.96E-03	2.29E-02	2.32E-02	2.06E-02	4.16E-03	2.21E-02	2.79E-02	7.48E-03	5.49E-04	1.53E-02
Th-231d	6.39E-09	9.84E-09	2.79E-10	5.80E-09	5.09E-08	4.08E-08	2.68E-09	4.18E-08	5.16E-08	3.90E-08	3.04E-09	4.15E-08	4.60E-08	2.29E-08	1.18E-09	2.96E-08
U-238	5.95E-03	9.94E-03	6.90E-04	6.98E-03	2.42E-02	2.04E-02	2.96E-03	2.29E-02	2.32E-02	2.06E-02	4.16E-03	2.21E-02	2.79E-02	7.48E-03	5.49E-04	1.53E-02
Th-234d	6.44E-09	9.91E-09	2.80E-10	5.83E-09	5.15E-08	4.13E-08	2.69E-09	4.22E-08	5.23E-08	3.94E-08	3.05E-09	4.19E-08	4.64E-08	2.33E-08	1.20E-09	2.99E-08
Pa-234mdd	2.04E-13	3.12E-13	7.95E-15	1.81E-13	1.68E-12	1.35E-12	8.43E-14	1.37E-12	1.71E-12	1.28E-12	9.37E-14	1.37E-12	1.51E-12	7.70E-13	3.92E-14	9.75E-13
Zn-65	1.18E-04	1.97E-04	1.37E-05	1.38E-04	4.80E-04	4.06E-04	5.87E-05	4.55E-04	4.60E-04	4.09E-04	8.25E-05	4.38E-04	5.53E-04	1.48E-04	1.09E-05	3.04E-04

* The values listed for Farm E are the predicted concentrations multiplied by a factor of 0.08 to account for the limited occupancy.

10.2 Radiation Exposure at the Automotive Repair Shop

A screening calculation for airborne effluents was also performed to determine the significance of the dose received on-site by members of the public who bring their vehicles to the WL site for repairs.

Such a dose would be most significant for a member of one of the previously-described potential critical groups. However, it is unlikely that members of the Farm A and Farm F groups would have vehicle repairs done at the WL repair shop because there are several garages in Lac du Bonnet, which are closer by road than the WL repair shop. The additional dose would be more significant for the Farm D group than the Farm E group because the predicted occupancy-weighted air concentrations, not accounting for trips to the WL repair shop, are much higher at Farm D than at Farm E (see Table 10-1). Therefore, the Farm D group would likely be bounding with respect to the significance of doses received at the WL repair shop.

For each radionuclide, the dose received at the WL repair shop is determined mainly by the air concentration at the repair shop. Therefore the significance of the doses received by members of the public at the repair shop can be estimated in terms of the predicted air concentrations at the repair shop and Farm D.

The WL repair shop is located 380 m south of B300. Therefore, for all radionuclides except HTO, the bounding release location with respect to doses at the repair shop is B300, the same as for doses at Farm D. For a unit release of any of the radionuclides released from B300, the predicted air concentration at the repair shop is 1.03E-05 Bq m^{-3}. Therefore, for these nuclides, the predicted occupancy-weighted air concentration experienced by Farm D group members who go to the repair shop for 12 hours per year is 4.29E-07 Bq m^{-3}, which is only 3% higher than that experienced by those who do not. This higher concentration is still significantly lower than the highest concentration predicted for any combination of potential critical group location and release location without accounting for trips to the repair shop (see Table 10-1).

For HTO, which is released only from B100, the predicted occupancy-weighted air concentration experienced by Farm D group members who go to the repair shop for 12 hours per year is less than 3% higher than that experienced by those who do not. This higher concentration is also significantly lower than the highest concentration predicted for any potential critical group location without accounting for trips to the repair shop (see Table 10-1)

Therefore, the dose received on-site by members of the public who bring their vehicles to the WL site for repairs is not significant.

11. DRL RESULTS

For each effluent type, the DRL for a given radionuclide was calculated from

$$DRL_i = \frac{DL_{eff}}{D_i} \tag{2}$$

where DRL_i is the derived release limit for radionuclide i (Bq s^{-1}), DL_{eff} is the annual effective dose limit for members of the public (0.001 Sv a^{-1}), D_i is the dose rate per unit release rate ((Sv a^{-1})/(Bq s^{-1})) for radionuclide i, summed over all applicable exposure pathways, for the age class leading to the highest dose.

The calculated DRLs for each radionuclide are summarized in Tables 11-1 and 11-2 for airborne and liquid effluents respectively. For airborne effluents, the DRLs are expressed on a weekly basis, and for liquid effluents they are expressed on a monthly basis. The tables also provide information on the bounding age classes, the dominant exposure pathways and the percent contributions from the dominant pathways to the total dose rates.

Table 11-1
DRLs for Airborne Effluents Released from WL

Radionuclide (and daughters)	DRL (Bq week^{-1})	Bounding Release Location	Bounding Age Class*	Dominant Pathway**	Percent Contribution from Dominant Pathway to Total Dose Rate
Am-241	2.07E+09	B200	Adult	AI	92
Am-243 (Np-239d, Pu-239dd)	2.04E+09	B200	Adult	AI	89
C-14 (CO_2)	8.61E+11	B200	1y CMDI	TAMM	97
Ce-144 (Pr-144d)	3.52E+11	B200	1y CMDI	TP	45
Cm-244	3.20E+09	B200	Child-5y	AI	96
Co-60	1.82E+10	WMA	1y CMDI	SLE	83
Cs-134	1.39E+10	WMA	Adult	TAMM	78
Cs-137 (Ba-137md)	1.51E+10	WMA	Adult	TAMM	65
Eu-152	2.00E+10	WMA	3mo NI	SLE	98
Eu-154	2.57E+10	WMA	3mo NI	SLE	97
Fe-55	1.74E+12	B200	3mo NI	TAMM	74
HTO †	1.65E+15	B100	Adult	WI	39
I-129	4.71E+08	B200	1y CMDI	TAMM	98
Mn-54	3.11E+11	WMA	1y CMDI	SLE	83
Nb-94	5.05E+09	WMA	1y CMDI	SLE	100
Ni-63	1.53E+11	WMA	1y CMDI	TAMM	97
Np-237 (Pa-233d)	1.66E+09	WMA	3mo FMDI	WI	78
Pm-147	5.18E+12	B200	3mo NI	TAMM	47
Pu-238 (U-234d)	1.89E+09	B200	Adult	AI	92
Pu-239	1.73E+09	B200	Adult	AI	92
Pu-240	1.74E+09	B200	Adult	AI	92
Pu-241 (Am-241d)	9.60E+10	B200	Adult	AI	92
Pu-242	1.80E+09	B200	Adult	AI	92
Sb-125 (Te-125md)	2.07E+11	WMA	1y CMDI	SLE	92
Sr-90 (Y-90d)	6.92E+09	WMA	3mo NI	TAMM	81
Tc-99	1.21E+11	WMA	3mo FMDI	WI	86
U-234	4.78E+09	B200	3mo FMDI	TAMM	84
U-235 (Th-231d)	4.67E+09	B200	3mo FMDI	TAMM	78
U-238 (Th-234d, Pa-234mdd)	4.92E+09	B200	3mo FMDI	TAMM	80
Zn-65	1.99E+10	B200	1y CMDI	TAMM	95

*** Acronyms for Age Class:**

1y CMDI: 1-year-old cow-milk-drinking infant

3mo NI: 3-month-old nursing infant

3mo FMDI: 3-month-old formula-milk-drinking infant

**** Acronyms for Pathway:**

AI: air (inhalation)

TAMM: terrestrial animals + mother's milk (ingestion)

TP: terrestrial plants (ingestion)

SLE: soil external (groundshine)

WI: water (ingestion)

† For all radionuclides except HTO, the critical group is Farm A. For HTO, the critical group is Farm F.

Table 11-2
DRLs for Liquid Effluents Released from WL

Radionuclide (and daughters)	**DRL** (Bq month^{-1})	**Bounding Age Class ***	**Dominant Pathway ****	**Percent Contribution from Dominant Pathway to Total Dose Rate**
Am-241	1.04E+09	3mo FMDI	WI	74
Am-243 (Np-239d, Pu-239dd)	1.04E+09	3mo FMDI	WI	72
C-14	7.67E+10	3mo NI	TAMM	89
Ce-144 (Pr-144d)	6.50E+10	3mo FMDI	WI	82
Cm-244	1.08E+09	3mo FMDI	WI	60
Co-60	2.09E+10	3mo FMDI	SLE	69
Cs-134	8.94E+09	Adult	FI	73
Cs-137 (Ba-137md)	1.16E+10	Adult	FI	65
Eu-152	2.37E+10	3mo FMDI	SLE	84
Eu-154	2.78E+10	3mo FMDI	SLE	75
Fe-55	6.05E+11	3mo FMDI	WI	88
HTO	6.80E+13	3mo FMDI	WI	91
I-129	8.94E+09	Child-10y	WI	48
Mn-54	2.41E+11	3mo FMDI	SLE	46
Nb-94	6.59E+09	3mo FMDI	SLE	94
Ni-63	1.09E+12	1y CMDI	TAMM	87
Np-237 (Pa-233d)	2.40E+09	3mo FMDI	WI	92
Pm-147	1.37E+12	3mo FMDI	WI	94
Pu-238 (U-234d)	1.16E+09	3mo FMDI	WI	89
Pu-239	1.11E+09	3mo FMDI	WI	89
Pu-240	1.11E+09	3mo FMDI	WI	89
Pu-241 (Am-241d)	8.32E+10	3mo FMDI	WI	89
Pu-242	1.16E+09	3mo FMDI	WI	89
Sb-125 (Te-125md)	1.71E+11	3mo FMDI	SLE	54
Sr-90 (Y-90d)	1.30E+10	3mo NI	TAMM	80
Tc-99	4.38E+11	3mo FMDI	WI	84
U-234	1.34E+10	3mo FMDI	WI	95
U-235 (Th-231d)	1.17E+10	3mo FMDI	WI	78
U-238 (Th-234d, Pa-234mdd)	1.25E+10	3mo FMDI	WI	82
Zn-65	3.29E+10	Adult	FI	79

*** Acronyms for Age Class:**

3mo FMDI: 3-month-old formula-milk-drinking infant

3mo NI: 3-month-old nursing infant

1y CMDI: 1-year-old cow-milk-drinking infant

**** Acronyms for Pathway:**

WI: water (ingestion)

TAMM: terrestrial animals + mother's milk (ingestion)

SLE: soil external (groundshine)

FI: fish (ingestion)

For airborne effluents, B200 is the bounding release location for seventeen of the thirty radionuclides. The WMA is the bounding release location for all others, except for HTO which is released only from B100. The adult is the bounding age class for ten radionuclides, with air inhalation being the dominant pathway for seven of these. For all other radionuclides except one, infant age classes are bounding, with ingestion of terrestrial animal products and mother's milk being the dominant pathway for over half of these.

For liquid effluents, the 3-month-old formula-milk-drinking infant is the bounding age class for twenty-three radionuclides, with water ingestion being the dominant pathway for seventeen of these and groundshine being the dominant pathway for the remainder.

12. VERIFICATION

The analysis assumptions, the selection of input parameter values, the contents of IMPACT code input files and the determination of DRLs were verified in several ways. Draft assumptions and the selection of some key input parameter values were reviewed by appropriate AECL staff prior to the start of calculations. As the DRL results were generated, those for selected radionuclides (C-14, Co-60, Cs-134, Cs137 (Ba-137m), HTO (OBT), I-129, Pu-239, and Sr-90 (Y-90), and Ce-144 (Pr-144)) were checked using the independent code CSA-DRL. Finally, the assumptions, input files and calculated DRLs were independently verified through a comprehensive internal AECL review [26].

The initial checking using the CSA-DRL code did not identify any problems with the results generated using the IMPACT code. The final AECL verification identified two errors with the selection of input parameter values, one of which was significant. This review also identified one minor transcription error in the input files. These errors were subsequently corrected.

Subsequently it was realized by the authors that there was an error in the screening calculations. This was corrected, although it did not significantly change the results.

13. COMPARISON OF REVISED DRLS WITH PREVIOUS VALUES

The revised DRLs have been compared with those established in 2001 [1]. Reasons for the differences were mentioned briefly in Section 1 and 5 of this report. For a select group of radionuclides, a detailed analysis of the reasons for the differences was performed [27]. For the combinations of effluent type and radionuclide included in this analysis, Table 13-1 compares the revised and previous DRLs, bounding release locations, critical groups, bounding age classes, dominant exposure pathways, and percentages of total dose from the dominant exposure pathway.

The previous DRL calculations for airborne effluents included potential critical groups that were not included in the revised calculations. These were future industrial park employees in existing Buildings 401 (B401) and 402 (B402) (see Figure 1) in what is currently the Controlled Area of the WL site, and individuals in possible future residential areas (Residences B and C in Figure 1) in what is currently the Supervised Area of the WL site. The previous final DRLs were determined for four combinations of type of release location and type of potential critical group. These were for:

- releases from WL-Main and industrial park groups.
- releases from WL-Main and other potential critical groups.
- releases from the WMA and industrial park groups.
- releases from the WMA and other potential critical groups.

The information tabulated in Table 13-1 for the previous calculations corresponds to the lowest DRL determined for the four combinations of type of release location and type of potential critical group. This provides consistency with the approach used in the revised calculations.

For the thirteen combinations of effluent type and radionuclide included in the detailed comparison, ten of the revised DRLs are lower than the previous. The greatest decrease is 99.8%, which occurs for the liquid release of Cm-244. The greatest increase is 1096%, which occurs for the airborne release of HTO (OBT).

For the six radionuclides from airborne releases, three of the revised DRL values are higher than the previous. The main reason for the increases for Cs-137 (Ba-137md), HTO (OBT) and Pu-239 is the exclusion of the industrial park group from the revised calculations. For I-129 and Zn-65, the main reason for the decrease is the higher rate of food intake. The main reason for the decrease in DRL for Sr-90 (Y-90d) is the addition of the 3-month-old nursing infant (higher ingestion dose coefficient, high transfer from mother’s milk) to the revised calculations.

For the seven radionuclides from liquid releases, all of the revised DRL values are significantly lower than the previous. In all cases, the main reason for the decrease is the difference in the methods used to calculate water concentrations.

For the combinations of effluent type and radionuclide that were not included in the detailed comparison, the factors discussed above or in the memorandum describing the detailed comparison [25], likely account for most of the differences between the previous and revised DRLs.

Table 13-1
Comparison of Revised and Previous DRLs

Radionuclide (and Daughter Product)	% Change in DRL	Bounding Release Location*		Critical Group*		Bounding Age Class*		Dominant Exposure Pathway*		% of Total Dose from Dominant Exposure Pathway	
		Revised	**Previous**	**Revised**	**Previous**	**Revised**	**Previous**	**Revised**	**Previous**	**Revised**	**Previous**
Airborne Effluents											
Cs-137 (Ba-137md)	508.9	WMA	B300	Farm A	B402	Adult	Adult	TAMM	SLE	65	100
HTO (OBT)	1095.7	B100	B300	Farm F	B402	Adult	Adult	WI	AI	39	100
I-129	-66.6	B200	B300	Farm A	Farm E	1y CMDI	Adult	TAMM	TP	98	63
Pu-239	1645.7	B200	B300	Farm A	B402	Adult	Adult	AI	AI	92	100
Sr-90 (Y-90d)	-60.0	WMA	WMA	Farm A	Res B	3mo NI	Adult	TAMM	TP	81	91
Zn-65	-74.5	B200	B300	Farm A	Farm E	1y CMDI	1y	TAMM	TAMM	95	65
Liquid Effluents											
Am-241	-99.6			Farm A	Farm A	3mo FMDI	Adult	WI	WI	74	41
Am-243 (Np-239d, Pu-239dd)	-99.6			Farm A	Farm A	3mo FMDI	Adult	WI	WI	72	38
Cm-244	-99.8			Farm A	Farm A	3mo FMDI	Adult	WI	WI	60	47
Cs-137 (Ba-137md)	-95.2			Farm A	Farm A	Adult	Adult	FI	FI	65	48
HTO (OBT)	-98.4			Farm A	Farm A	3mo FMDI	1y	WI	WI	91	64
Pu-239	-99.7			Farm A	Farm A	3mo FMDI	Adult	WI	WI	89	59
Sr-90 (Y-90d)	-99.1			Farm A	Farm A	3mo NI	Adult	TAMM	TP	80	48

* see next page for a description of acronyms

* **Acronyms for release locations:**

B200: Building 200
B100: Building 100
B300: Building 300
WMA: Waste Management Area

* **Acronyms for critical groups:**

B402: Building 402
Res B: Residence B

[*] **Acronyms for age classes:**

1y CMDI: 1-year-old cow-milk-drinking infant
3mo NI: 3-month-old nursing infant
3mo FMDI: 3-month-old formula-milk-drinking infant

[*] **Acronyms for exposure pathways:**

TAMM: terrestrial animals + mother's milk (ingestion)
WI: water (ingestion)
AI: air (inhalation)
FI: fish (ingestion)
SLE: soil external (groundshine)
TP: terrestrial plants (ingestion)

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Appendix A

Triple Joint Frequency Distribution of Wind Speed, Wind Direction and Stability Class

Table A-1
Triple Joint Frequency Distribution data obtained at the 6-m Level

Wind from Sector	Stability Class	Wind Speed Classes					
		0 to 2 m/s	2 to 3 m/s	3 to 4 m/s	4 to 5 m/s	5 to 6 m/s	>6 m/s
		Average Wind Speed (m/s) for Each Wind Speed Class					
		0.853882	2.466141	3.439265	4.417175	5.43651	7.052884
		Triple Joint Frequencies					
N	A	0.00536	0.002556	0.000887	0.000103	0.000062	0
NNE	A	0.002948	0.001237	0.000371	0.000041	0.000021	0
NE	A	0.001649	0.000598	0.000392	0.000082	0	0
ENE	A	0.001567	0.00066	0.00033	0.000021	0	0
ENE	A	0.001835	0.000845	0.00035	0.000021	0	0
ESE	A	0.002165	0.000474	0.000186	0.000041	0	0
SE	A	0.002783	0.000763	0.000247	0	0.000021	0
SSE	A	0.003381	0.000825	0.000206	0.000041	0.000021	0
S	A	0.004371	0.001134	0.000206	0.000021	0	0
SSW	A	0.004165	0.001299	0.000227	0.000041	0	0
SW	A	0.004494	0.001567	0.000515	0.000062	0	0
WSW	A	0.00369	0.001587	0.000763	0.000103	0.000021	0.000021
W	A	0.003525	0.001814	0.000969	0.000227	0.000062	0.000103
WNW	A	0.002618	0.001237	0.00099	0.000309	0.000206	0.000124
NW	A	0.004247	0.002433	0.001587	0.000536	0.000268	0.000165
NNW	A	0.005298	0.002783	0.001258	0.000412	0.000082	0.000041
N	B	0.005855	0.004865	0.004	0.001567	0.000618	0.000309
NNE	B	0.002082	0.001794	0.001051	0.000371	0.000103	0
NE	B	0.001319	0.00099	0.000515	0.000309	0.000041	0.000021
ENE	B	0.001031	0.000433	0.000412	0.000103	0.000041	0
ENE	B	0.000825	0.000639	0.000598	0.000206	0.000062	0
ESE	B	0.001464	0.000866	0.000618	0.000247	0.000041	0
SE	B	0.002515	0.001752	0.001072	0.000206	0.000041	0.000021
SSE	B	0.002598	0.001876	0.001051	0.000268	0.000021	0
S	B	0.005195	0.003567	0.001855	0.000907	0.000247	0.000021
SSW	B	0.004927	0.00503	0.002103	0.000639	0.000062	0
SW	B	0.004639	0.003484	0.001464	0.00033	0.000062	0
WSW	B	0.003381	0.002453	0.00134	0.000515	0.000082	0.000041
W	B	0.002247	0.001443	0.001361	0.001113	0.000289	0.000082
WNW	B	0.001732	0.001691	0.00202	0.001505	0.000742	0.00033
NW	B	0.004247	0.00367	0.004391	0.003072	0.002185	0.001443
NNW	B	0.007236	0.006989	0.006721	0.004783	0.002144	0.00099

Wind from Sector	**Stability Class**	**Wind Speed Classes**					
		0 to 2 m/s	**2 to 3 m/s**	**3 to 4 m/s**	**4 to 5 m/s**	**5 to 6 m/s**	**>6 m/s**
		Average Wind Speed (m/s) for Each Wind Speed Class					
		0.853882	2.466141	3.439265	4.417175	5.43651	7.052884
		Triple Joint Frequencies					
N	C	0.010721	0.006948	0.004762	0.002082	0.000536	0.000392
NNE	C	0.005484	0.004618	0.002783	0.001319	0.000474	0.000206
NE	C	0.002041	0.002371	0.001402	0.000928	0.000289	0.000021
ENE	C	0.002474	0.001278	0.000804	0.000495	0.000186	0
ENE	C	0.00167	0.001773	0.00202	0.000928	0.000227	0.000041
ESE	C	0.003257	0.002433	0.002474	0.001134	0.000289	0.000021
SE	C	0.005505	0.00501	0.00402	0.002144	0.000825	0.000144
SSE	C	0.008638	0.009648	0.007855	0.004206	0.001113	0.00035
S	C	0.013009	0.009669	0.007628	0.002928	0.000763	0.000021
SSW	C	0.012906	0.004371	0.001402	0.000144	0.000021	0
SW	C	0.009917	0.003958	0.002206	0.000536	0.00033	0.000041
WSW	C	0.005546	0.004783	0.002866	0.001464	0.000701	0.00035
W	C	0.003773	0.003567	0.004288	0.002763	0.001691	0.001175
WNW	C	0.003299	0.002886	0.003628	0.003938	0.003051	0.004288
NW	C	0.00703	0.005422	0.006164	0.00635	0.005546	0.005999
NNW	C	0.010473	0.007463	0.005298	0.003134	0.001361	0.000742
N	D	0.005752	0.001216	0.000495	0.000309	0.000124	0.000041
NNE	D	0.004268	0.002185	0.000763	0.000371	0.000227	0.000021
NE	D	0.001608	0.002123	0.001464	0.000866	0.000763	0.00035
ENE	D	0.003154	0.002206	0.001196	0.000577	0.000103	0.000082
ENE	D	0.004577	0.00468	0.00369	0.001072	0.00035	0.000289
ESE	D	0.006391	0.004103	0.002845	0.001155	0.00033	0.000041
SE	D	0.008494	0.006577	0.003938	0.001587	0.000371	0
SSE	D	0.015174	0.015091	0.009071	0.004082	0.000969	0.000165
S	D	0.017751	0.006494	0.003505	0.000598	0.000082	0.000021
SSW	D	0.005999	0.000186	0	0	0	0
SW	D	0.006577	0.000577	0.000227	0.000041	0	0
WSW	D	0.007628	0.003814	0.001423	0.000928	0.000289	0.000433
W	D	0.006102	0.007175	0.006206	0.003649	0.002082	0.001402
WNW	D	0.002742	0.003051	0.003381	0.002783	0.001484	0.001711
NW	D	0.004144	0.002144	0.002845	0.003031	0.002288	0.002206
NNW	D	0.005608	0.001587	0.000845	0.000392	0.000247	0.000227
N	E	0.003196	0.000495	0.000845	0.000639	0.000433	0.000103
NNE	E	0.001051	0.000082	0.000021	0	0	0
NE	E	0.000866	0.000186	0.000124	0	0	0
ENE	E	0.002474	0.000309	0.000041	0	0	0
ENE	E	0.004103	0.000804	0.000041	0	0	0
ESE	E	0.004618	0.000577	0.000082	0.000021	0	0
SE	E	0.005731	0.000887	0.000268	0.000082	0.000021	0

Wind from Sector	**Stability Class**	**Wind Speed Classes**					
		0 to 2 m/s	**2 to 3 m/s**	**3 to 4 m/s**	**4 to 5 m/s**	**5 to 6 m/s**	**>6 m/s**
		Average Wind Speed (m/s) for Each Wind Speed Class					
		0.853882	2.466141	3.439265	4.417175	5.43651	7.052884
		Triple Joint Frequencies					
SSE	E	0.008329	0.001484	0.000412	0.000041	0	0
S	E	0.010741	0.000227	0	0	0	0
SSW	E	0.005237	0.000021	0	0	0	0
SW	E	0.004845	0.000021	0	0	0	0
WSW	E	0.004927	0.00035	0.000124	0.000041	0.000021	0
W	E	0.003587	0.001299	0.00101	0.000144	0.000103	0.000021
WNW	E	0.001876	0.000454	0.000144	0.000103	0.000082	0.000144
NW	E	0.001876	0.000041	0	0	0	0
NNW	E	0.002618	0.000124	0	0	0	0.000021
N	F	0.005711	0.000309	0.000186	0.000082	0	0.000021
NNE	F	0.002227	0	0	0	0	0
NE	F	0.001464	0	0	0	0	0
ENE	F	0.002206	0	0	0	0	0
ENE	F	0.003628	0.000021	0	0	0	0
ESE	F	0.005855	0.000021	0	0	0	0
SE	F	0.010143	0	0.000062	0	0	0
SSE	F	0.01004	0.000062	0.000062	0.000041	0.000021	0.000021
S	F	0.014514	0.000124	0.000021	0	0	0
SSW	F	0.013792	0	0	0	0	0
SW	F	0.014225	0.000124	0	0	0	0
WSW	F	0.011772	0	0	0	0	0
W	F	0.009875	0	0.000021	0.000021	0.000041	0.000021
WNW	F	0.007257	0.000041	0.000041	0.000082	0.000062	0.000021
NW	F	0.00635	0	0	0	0	0
NNW	F	0.006123	0.000021	0	0	0	0

Table A-2
Triple Joint Frequency Distribution data obtained at the 25-m Level

Wind from Sector	Stability Class	Wind Speed Classes					
		0 to 2 m/s	**2 to 3 m/s**	**3 to 4 m/s**	**4 to 5 m/s**	**5 to 6 m/s**	**>6 m/s**
		Average Wind Speed (m/s) for Each Wind Speed Class					
		1.136575	2.49576	3.465201	4.452005	5.446335	7.277126
		Triple Joint Frequencies					
N	A	0.002515	0.001718	0.000736	0.000245	0.000082	0.000102
NNE	A	0.001411	0.000634	0.000368	0.000143	0.000102	0
NE	A	0.001043	0.000409	0.000286	0.000123	0.000061	0
ENE	A	0.001043	0.000491	0.000225	0.000061	0.000061	0
ENE	A	0.001268	0.000757	0.000348	0.000164	0.00002	0
ESE	A	0.001043	0.000573	0.000389	0.000061	0	0
SE	A	0.001636	0.000634	0.000123	0.000041	0.000061	0.00002
SSE	A	0.001902	0.00092	0.000429	0.000143	0.000041	0
S	A	0.001984	0.000838	0.000348	0.000164	0.000061	0.000041
SSW	A	0.002413	0.000777	0.000348	0.000184	0.000041	0.000041
SW	A	0.002208	0.001043	0.000348	0.000204	0.000041	0.000123
WSW	A	0.0018	0.000716	0.000716	0.000123	0.000061	0.000102
W	A	0.002147	0.001084	0.000941	0.000368	0.000164	0.000204
WNW	A	0.00182	0.000757	0.000838	0.00047	0.000184	0.000225
NW	A	0.002249	0.001104	0.000798	0.000491	0.000184	0.000143
NNW	A	0.002945	0.001431	0.000675	0.000245	0.000164	0.000204
N	B	0.001963	0.002597	0.001145	0.000777	0.000204	0.00002
NNE	B	0.000941	0.000982	0.000654	0.000164	0.000082	0
NE	B	0.000859	0.000695	0.000654	0.000184	0.000061	0.00002
ENE	B	0.000838	0.000552	0.000389	0.000225	0	0
ENE	B	0.000552	0.00047	0.000429	0.000245	0.00002	0
ESE	B	0.000654	0.000573	0.000327	0.000184	0.000082	0.00002
SE	B	0.001288	0.000777	0.000552	0.000204	0.000102	0
SSE	B	0.00135	0.000982	0.000491	0.000286	0.000143	0.00002
S	B	0.002024	0.001063	0.000634	0.000511	0.000266	0.000143
SSW	B	0.001922	0.001247	0.001125	0.000818	0.000348	0.000184
SW	B	0.001636	0.001247	0.001534	0.000777	0.000286	0.000123
WSW	B	0.00137	0.001329	0.001043	0.000757	0.000348	0.000348
W	B	0.001309	0.000961	0.001043	0.000798	0.000327	0.000245
WNW	B	0.001513	0.000818	0.001104	0.0009	0.00045	0.000348
NW	B	0.001615	0.001125	0.001309	0.001206	0.000941	0.000613
NNW	B	0.002086	0.002352	0.001575	0.000818	0.000593	0.000511
N	C	0.003619	0.00638	0.005726	0.00499	0.003026	0.002699
NNE	C	0.001615	0.002536	0.001963	0.00135	0.000695	0.00047
NE	C	0.00182	0.001636	0.001309	0.0009	0.000348	0.000204
ENE	C	0.001493	0.001022	0.000777	0.000654	0.000348	0.000041

Wind from Sector	**Stability Class**	**Wind Speed Classes**					
		0 to 2 m/s	**2 to 3 m/s**	**3 to 4 m/s**	**4 to 5 m/s**	**5 to 6 m/s**	**>6 m/s**
		Average Wind Speed (m/s) for Each Wind Speed Class					
		1.136575	2.49576	3.465201	4.452005	5.446335	7.277126
		Triple Joint Frequencies					
ENE	C	0.001288	0.000982	0.001268	0.000941	0.000368	0.000245
ESE	C	0.001575	0.001677	0.001043	0.000859	0.000654	0.000266
SE	C	0.003006	0.002372	0.002045	0.001738	0.000695	0.000409
SSE	C	0.003047	0.003517	0.003619	0.003026	0.002311	0.001227
S	C	0.00454	0.004315	0.004499	0.003742	0.002454	0.001922
SSW	C	0.003844	0.003742	0.004744	0.003824	0.002842	0.002045
SW	C	0.003865	0.003415	0.003354	0.00272	0.002045	0.001411
WSW	C	0.002393	0.002352	0.002208	0.001902	0.000859	0.00135
W	C	0.003026	0.001881	0.00272	0.002127	0.001227	0.002127
WNW	C	0.002699	0.002147	0.001963	0.00227	0.001984	0.003211
NW	C	0.003497	0.003129	0.004253	0.003804	0.003988	0.006441
NNW	C	0.005419	0.007014	0.007341	0.008364	0.007341	0.010265
N	D	0.006973	0.009141	0.006728	0.004887	0.002311	0.002086
NNE	D	0.002495	0.00503	0.005071	0.002147	0.001452	0.001125
NE	D	0.002924	0.003599	0.003354	0.002147	0.001247	0.001431
ENE	D	0.002904	0.004253	0.00274	0.001595	0.0009	0.000389
ENE	D	0.002433	0.003456	0.00411	0.003476	0.001902	0.001145
ESE	D	0.003988	0.005337	0.004785	0.003149	0.00135	0.000757
SE	D	0.005174	0.006666	0.008282	0.005705	0.003558	0.002045
SSE	D	0.004744	0.01094	0.018261	0.018241	0.013374	0.009325
S	D	0.007178	0.011922	0.009611	0.006891	0.003231	0.001697
SSW	D	0.00499	0.005992	0.005726	0.003395	0.001206	0.000675
SW	D	0.00499	0.005705	0.00454	0.002945	0.001922	0.001677
WSW	D	0.004642	0.004213	0.005071	0.003538	0.002249	0.003088
W	D	0.004192	0.004417	0.00548	0.005215	0.003517	0.005092
WNW	D	0.003333	0.003108	0.004335	0.004785	0.003313	0.006687
NW	D	0.003211	0.004458	0.004785	0.005726	0.005808	0.011615
NNW	D	0.006585	0.008814	0.006421	0.004785	0.003906	0.004233
N	E	0.003865	0.003517	0.002208	0.000777	0.000184	0.000143
NNE	E	0.002536	0.002106	0.000941	0.000061	0.00002	0
NE	E	0.001022	0.001022	0.000634	0.000061	0.000061	0.000041
ENE	E	0.001738	0.001452	0.000941	0.000225	0.000041	0
ENE	E	0.002352	0.002413	0.00182	0.000757	0.000061	0
ESE	E	0.002495	0.003067	0.00137	0.000409	0.000041	0
SE	E	0.003967	0.004172	0.001227	0.000204	0.000041	0.000041
SSE	E	0.004213	0.00638	0.006176	0.003088	0.001063	0.000225
S	E	0.004621	0.007239	0.003517	0.000716	0.000286	0.000082
SSW	E	0.002515	0.002699	0.001166	0.000286	0.000061	0
SW	E	0.002474	0.001534	0.001186	0.000204	0	0

Wind from Sector	**Stability Class**	**Wind Speed Classes**					
		0 to 2 m/s	**2 to 3 m/s**	**3 to 4 m/s**	**4 to 5 m/s**	**5 to 6 m/s**	**>6 m/s**
		Average Wind Speed (m/s) for Each Wind Speed Class					
		1.136575	2.49576	3.465201	4.452005	5.446335	7.277126
		Triple Joint Frequencies					
WSW	E	0.002168	0.001186	0.001247	0.000634	0.000204	0.000204
W	E	0.00274	0.002372	0.002842	0.001493	0.000695	0.000327
WNW	E	0.001513	0.001472	0.001718	0.001513	0.000757	0.000777
NW	E	0.001922	0.000654	0.000573	0.000409	0.000327	0.000164
NNW	E	0.00364	0.003006	0.00092	0.000286	0.000061	0.000061
N	F	0.002536	0.000879	0.00047	0	0	0
NNE	F	0.001166	0.000204	0.00002	0	0	0
NE	F	0.001227	0.00002	0.00002	0	0	0
ENE	F	0.001125	0.000082	0.000061	0	0	0
ENE	F	0.001329	0.00045	0.000368	0	0	0
ESE	F	0.001391	0.000389	0.000102	0	0	0
SE	F	0.002413	0.000593	0	0	0	0
SSE	F	0.002352	0.000961	0.000532	0.00002	0	0
S	F	0.002045	0.001002	0.000368	0.000041	0	0
SSW	F	0.001452	0.000204	0.00002	0	0	0
SW	F	0.002208	0.000123	0.00002	0	0	0
WSW	F	0.001738	0.000082	0.000082	0	0	0
W	F	0.002188	0.000164	0.000061	0.00002	0	0
WNW	F	0.00184	0.000102	0.000184	0.00002	0	0
NW	F	0.00182	0.00002	0	0	0	0
NNW	F	0.002393	0.000307	0.000266	0	0	0

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