LIMITED REPORT

SRC SLOWPOKE-2 FACILITY
LICENSE # NPROL-19.01/2023

ANNUAL COMPLIANCE REPORT

for the period from January 1, 2019 to December 31, 2019

Prepared for:

Canadian Nuclear Safety Commission

Prepared by:

Saskatchewan Research Council
Environment and Biotech Division

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Report Limitations and Use of Report

EXECUTIVE SUMMARY

The Saskatchewan Research Council (SRC) SLOWPOKE-2 Facility began the reporting period January 1 to December 31, 2019 under licence NPROL 19.00/2023. From January 1 to April 30, the reactor was operated for the same purposes as it has been for the past 38 years. Following the final day of operation on April 30, the reactor was maintained in a shut-down state for approximately three months until it was defueled. In December, the licence was amended to NPROL 19.01/2023 to authorize the full decommissioning of the Facility, which will take place in 2020.

From January 1 to April 30, usage of the facility was consistent with past years. Operational activities remained the same with the majority of operations being irradiation of samples for instrumental neutron activation analysis on a commercial basis. No isotopes were generated for off-site use.

Saskatchewan Research Council (SRC) has an active Occupational Health and Safety Program in place based on a management system. SRC's Occupational Health and Safety Management System recognizes radiation safety as one of the critical risks and addresses it by having detailed policies and procedures to reduce radiation exposure to a minimal level, as explained in the Safety Management section of this document. Quality Management is an integral part of the operations. The facility, under the umbrella of SRC Environmental Analytical Labs (EAL), has a continuous improvement program which adheres to international standard ISO/IEC17025:2017.

Initial and continuing training programs in place at the facility ensure that staff are properly trained and equipped to perform their duties with the safety of themselves, co-workers and the general public at the forefront.

Operations at the facility are performed in accordance with the ALARA principle. Potential radiation hazards are identified, and appropriate procedures and tools are in place to prevent unnecessary exposure to staff and general public. As a result of strict adherence to radiation safety policies and procedures including regular radiation and contamination monitoring, doses to staff have been kept well below the annual occupational exposure limit for members of the general public and, in most cases, to non-detectable levels.

A comprehensive aging management program has enabled the facility to operate trouble-free throughout its history. Minor problems are detected and fixed before they can have a detrimental effect on the operations, equipment, personnel or the environment.

An effective environmental monitoring program is in place to minimize negative impact to the environment and public by limiting air and water releases to the absolute minimum. Wastes and by-products are managed such that only materials with no measurable radiation field are discarded by normal waste disposal. As a result, the maximum estimated potential radiation dose from all the various types of reactor releases is well below that which is permitted for the annual occupational radiation exposure of a member of the general public.

A Public Information Program and Disclosure Protocol has been established and is reviewed regularly to ensure relevant information is disseminated in a timely manner to stakeholders and residents living in the vicinity of the facility.

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1.0 INTRODUCTION

1.1 General Introduction

The reporting period of January 1 to December 31, 2019 saw the Saskatchewan Research Council (SRC) SLOWPOKE-2 reactor facility (the facility) transition from fully operational to maintenance in a shut-down state and subsequent defueling. From January 1 to April 30, the facility was operated in a routine and trouble-free manner as it has been since 1981. While fully operational, the reactor was operated an average of 1.8 days per week. The primary purpose of operations is for instrumental neutron activation analysis (INAA) and delayed neutron counting (DNC) on a commercial basis. Due to the anticipated shutdown of the facility and the relatively short portion of the year that it was in operation, no isotopes were produced for off-site clients and no students made use of the facility in 2019.

The facility began the reporting period under License No. NPROL-19.00/2023, valid to June 30, 2023. On December 6, 2019, an amended licence, NPROL 19.01/2023, valid to June 30, 2023, was issued. The amended licence authorizes the full decommissioning of the facility. Activities undertaken to ensure compliance include maintenance of all operational logs, performance of routine maintenance as described in “CPSR-362 Rev.2, SLOWPOKE-2 Nuclear Reactor Operation and Routine Maintenance”, inspections of the reactor container, external components, and auxiliary systems, routine radiation and contamination monitoring, and maintenance of dosimetry records.

There were no reportable incidents at the facility in 2019.

1.2 Facility Operation

1.2.1 Operational Items

The facility operated in a safe and reliable manner for the reporting period. There were no problems experienced with the critical components – control rod, control rod motor, flux detector, and shutdown systems –or any other component of the reactor.

1.2.1.1 Modifications to the Facility

Following the final operational shutdown of the reactor on April 30, several modifications were made to the facility prior to the commencement of defueling. These modifications included removal of casework and cabinets in room 143, installation of HEPA filters on all HVAC exhaust vents, and disconnection and removal of the delayed neutron counting system. Concrete pool covers were removed and a guard rail was installed around the pool. A gantry crane was erected over the pool. Lighting, fencing and additional security cameras were installed around the exterior of the facility. Locks were added or modified on certain interior doors to prevent access from areas of the Resources Research Center building not occupied by SRC.

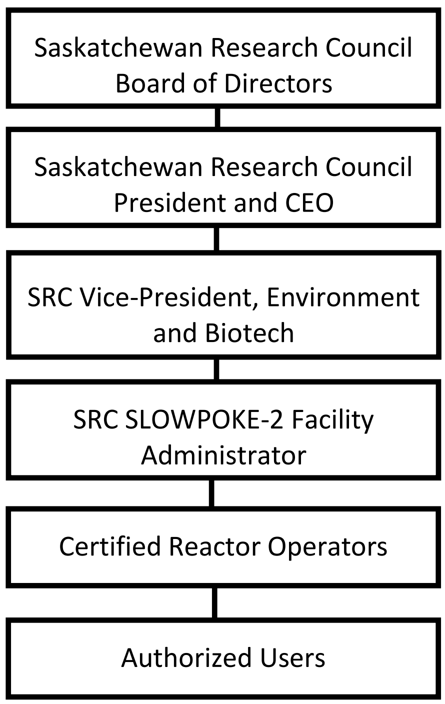
1.2.2 Audits/Inspections

The facility had two safety inspections during the reporting period - a radiation safety inspection conducted by the SRC Radiation Safety Officer (RSO) and a fire inspection conducted by Saskatoon Fire and Protective Services. The radiation safety inspection did not produce any action items related to the facility. The fire inspection report stated that the building was satisfactory at the time of the inspection.

Internal audits of the facility are conducted by EAL Quality Assurance on a three-year cycle. The current audit cycle is for the period of 2019-2021. Action items identified included minor revisions to standard operating procedures.

1.2.3 Organizational Structure

Figure 1: Levels of Management Control Responsibility for the Saskatchewan Research Council SLOWPOKE-2 Facility.



1.2.4 Key Personnel

1.2.4.1 SLOWPOKE-2 Committee

Members of the SLOWPOKE-2 committee are primarily selected from staff of the Saskatchewan Research Council (SRC). Normally, the SLOWPOKE-2 Committee chairperson is the Vice-President responsible for the SRC Environmental Analytical Laboratories. The President and CEO, and the Radiation Safety Officer (RSO) of the Saskatchewan Research Council are ex officio members of the committee.

Membership as of December 31, 2019 in the committee is as follows:

- Chairperson: Dr. Joe Muldoon, Vice-President, Environment Division, SRC
- Vice-chairperson: Mr. Dave Chorney, Senior Technologist, Environmental Analytical Laboratories, and SLOWPOKE-2 Facility Administrator and Operator, SRC
- Mr. Mike Crabtree, President and CEO, SRC
- Mr. Jeff Zimmer, Manager, Environmental Analytical Laboratories, SRC
- Mr. Sunil Sohani, Safety Services Manager, SRC
- Ms. Gloria Drader, Radiation Safety Officer, SRC
- Ms. Debbie Frattinger, Safety and Compliance Officer, Fedoruk Center

1.2.4.2 Certified Reactor Operators

The facility currently has five certified operators:

- Mr. Dave Chorney (automatic and manual modes)
- Mr. Jeff Zimmer (automatic and manual modes)
- Ms. Jenna Smith-Windsor (automatic mode)
- Ms. Kelcey Reding (automatic mode)
- Ms. Karina Knorr (automatic mode)

Certifications for Mr. Chorney and Mr. Zimmer are valid until August 22, 2021.

Certification for Ms. Smith-Windsor was renewed in 2018 and is valid until May 3, 2023.

Certifications for Ms. Reding and Ms. Knorr are valid until May 16, 2022

1.2.4.3 Authorized Users

Authorized Users as of December 31, 2019:

- Vicky Snook (Technologist, SRC)

1.2.4.4 Nuclear Energy Workers

There are no employees at the facility classed as nuclear energy workers.

1.3 Reactor Utilization

During the first four months of the reporting period (up to April 30) the SRC SLOWPOKE-2 reactor was operated for a total of 103.7 hours during 31 operating days. During defueling, the reactor was operated on August 7 and 8 for an additional 6.43 hours at a flux of 5×10^{9} $n.cm^{-2}.sec^{-1}$ to verify the excess reactivity of the reactor and the negative reactivity worth of the cadmium auxiliary shutdown capsules; and to monitor the response of the reactor as the beryllium shims and irradiation tubes were removed. The total integrated flux for the reporting period was 507.2×10^{11} $n.cm^{-2}. sec^{-1}$ hours. The monthly totals are listed in Table 1. Since commissioning, the reactor has been operated a total of 20651.5 hours with a total integrated flux of $100{,}335.1 \times 10^{11}$ $n.cm^{-2}.sec^{-1}$ hours.

1.3.1 Production

During the reporting period, 731 capsule irradiations were performed. Approximately 63% (459) were solids and liquids for delayed neutron counting (DNC) of uranium. The remainder (272) were solid and liquid samples for instrumental neutron activation analysis (INAA) of such sample types as waste oils, solvents and soils for organic halide analysis. The monthly totals are listed in Table 1.

Activated materials that are produced as a result of the facility's commercial analysis activities are stored on-site for a minimum of six months to allow for decay of activity to background levels (<0.3µSv/h). Once the activity has decayed to this level the capsules are discarded via a local waste collection company. Materials which do not decay to background (e.g., samples which contain naturally occurring long-lived radionuclides, such as uranium ore specimens) are returned to the client if they are licensed to possess them, or sent to an approved waste facility.

No isotopes were produced for off-site use during 2019.

Table 1: Summary of SRC SLOWPOKE-2 Operations for the period January 1, 2019 to December 31, 2019.

Month	Days Operated	Total Hours	Flux-Hours $n/cm^2.sec\text{-}h\ x10^{11}$	Capsules Irradiated
January	11	34.3	170.6	225
February	7	25.3	126.0	169
March	6	19.8	98.3	161
April	7	24.3	112.0	176
May	0	0	0	0
June	0	0	0	0
July	0	0	0	0
August	2	6.4	0.3	0
September	0	0	0	0
October	0	0	0	0
November	0	0	0	0
December	0	0	0	0
2019 TOTAL	**33**	**110.1**	**507.2**	**731**
Total to December 31, 2018	4693	20541.4	98827.9	241791
Total to December 31, 2019	**4726**	**20651.5**	**100335.1**	**242522**

1.3.2 Samples Which Could Create Unusual Hazards

No samples which could create unusual hazards were irradiated during the reporting period.

1.3.3 Manual Operation

The reactor was not operated under manual control during the reporting period.

1.3.4 Remotely Attended Operation

The reactor was not operated in remotely attended mode during the reporting period.

1.3.5 Reactivity Adjustments

On the last day of regular operation, April 30, the excess reactivity was measured at 2.078 mk. The shim tray contained ten half-inch semicircular beryllium shims.

On August 7 just prior to the commencement of defueling the excess reactivity was measured at 2.01 mk by the period measurement method and 2.10 mk by low power critical balance. Over the next two days, the irradiation tubes and shims were removed in a predetermined sequence, overseen by the CNSC-certified SLOWPOKE reactor engineer. The reactor was brought sub-critical on August 8 at 9:49 AM.

2.0 SAFETY AND CONTROL AREAS

2.1 Management

2.1.1 Management System

2.1.1.1 Safety Management

Safety Management at the facility is governed by the SRC Occupational Health and Safety (OH&S) Program. The program is based on a management system. Its philosophy is illustrated by the model diagram shown in Figure 2.

Figure 2: SRC OH&S Management System.

At the core of SRC's OH&S Management System is the most important element, "Leadership & Commitment". The other elements around this core element indicate that they are driven by management leadership and commitment. The "Hazards and Risk Management" element recognizes radiation safety as one of the critical risks along with other health and safety risks of SRC activities. The Audits and Management Reviews serve to monitor all the elements and to provide recommended actions for continuous improvement of the OH&S Management System.

One of the supporting documents of the OH&S Management System is the SRC Radiation Safety Manual. The Radiation Safety Manual describes implementation of the OH&S Management System including established policies and procedures, minimum requirements, expectations, and roles and responsibilities for activities involving radiation, which follow the ALARA principle to reduce radiation risks to minimum levels.

At the beginning of every year corporate OH&S objectives are set by the Executive Team. The objectives are monitored every quarter and actions are implemented to ensure the objectives are met.

Based on the corporate objectives, employees receive individual annual objectives specific to the risks in their work area.

SRC's overall OH&S performance during the reporting period was positive. There was one lost time injury Incident – not related to the SLOWPOKE - but there was a reduction in the number of total incidents compared to the previous year. Activities at the SLOWPOKE facility did not result in any safety incident and there was no exposure related to radiation safety.

The facility had two safety inspections during the reporting period; a radiation safety inspection conducted by the SRC Radiation Safety Officer (RSO) and a fire inspection conducted by Saskatoon Fire and Protective Services. The radiation safety inspection did not produce any action items related to the facility. The fire inspection report stated that the building was satisfactory at the time of the inspection.

2.1.1.2 Quality Management

Quality Management at the facility is governed by the SRC Environmental Analytical Laboratories Quality Manual which adheres to international standard ISO/IEC17025:2017.

The laboratory has a continuous improvement program to routinely examine current systems, identify non-conformances and potential non-conformances, and identify areas where efficiencies can improve.

External audits are conducted on a biannual basis by the Canadian Association for Laboratory Accreditation Inc. (CALA). The lab was audited by CALA in October 2019, and was found to conform to the requirements of ISO/IEC 17025:2017. The accreditation is valid through August 2020.

EAL Quality Assurance personnel conduct an audit of the facility on a three-year cycle. The current audit cycle is for the period of 2019 to 2021. Action items identified included minor revisions to standard operating procedures.

A management review of the Laboratory Quality System is conducted in January of each year. At the most recent review (2019) there were no items pertaining to SLOWPOKE which required corrective action.

The management review determined that the current quality policy and objectives as outlined in revision 14 of the quality manual (effective date November 1, 2019) are adequate and do not need revision, and that the medium and long term goals of the quality assurance staff are adequate to address the medium and long term goals of the laboratory's quality policy and objectives.

2.1.2 Human Performance Management

2.1.2.1 Training Programs

Several levels of training programs are in place at SRC EAL. The training provided is in accordance with the extent to which an employee's job requires presence in the facility, use of irradiation systems and operation of the reactor.

All EAL staff receives site-specific training, including basic radiation safety, safe handling of low-level radioactive materials, use of appropriate personal protective equipment (PPE), and location and use of survey meters. Further, all EAL staff are required to complete the SRC General Radiation Safety Course within one year of commencement of employment. A general radiation safety refresher course was implemented in 2016. Staff are required to take the refresher course every three years.

Candidates for the designation as authorized users of the irradiation facilities must have successfully completed the SRC General Radiation Safety Course. They must also complete the SRC Training Program for Authorized Users. The objectives of this training program are for the candidate:

- To understand the purpose of the Sample Irradiation Forms. Know how to complete a Sample Irradiation Form and why the information on the form is required.
- To be able to demonstrate the ability to operate the irradiation controllers and sample transfer devices. Know what to do in the event of a malfunction with sample controllers and sample transfer systems.
- To understand the principles of radiation safety. Understand the hazards of handling irradiated samples. Know what preventive measures are available for authorized users to reduce radiation exposures.
- To know how to perform any necessary radiochemical manipulations of isotopes produced in the reactor in a safe manner.
- To successfully complete any necessary continuing training requirements for knowledge of radiation safety as specified in the SRC Radiation Safety Manual. Maintain knowledge and capabilities of authorized users as outlined in the Authorized Users Handbook.
- To know how to recognize and respond to emergency situations.

Due to the shutdown and decommissioning of the facility, no new authorized users were trained during the reporting period and the authorized user training program has been archived.

A candidate for becoming a certified reactor operator is trained as per *Saskatchewan Research Council Training Program for SLOWPOKE-2 Reactor Operators – Automatic Mode*. Due to the shutdown and decommissioning of the facility, no new operators were trained during the reporting period and the reactor operator training program has been archived

Current certified reactor operators maintain their status by fulfilling the requirements as per *Licence Conditions Handbook for Saskatchewan Research Council SLOWPOKE-2 Non-Power Reactor, Appendix F, Section 3. Saskatchewan Research Council Training Program for SLOWPOKE-2 Reactor Operators – Automatic Mode, contains a section on continuing training*.

Training for the positions of reactor technician and reactor engineer is not done at the facility, as it does not employ any personnel in these positions. Work which requires a certified reactor technician or engineer is contracted out to Canadian Nuclear Laboratories.

2.1.2.2 Number of Qualified Workers

At the beginning of 2019 there were five certified reactor operators and 11 authorized users on staff. This level of staffing ensured that there are always sufficient personnel on duty to perform reactor operations and sample irradiations as needed. Following the final shutdown of the reactor on April 30, access to the facility was restricted to personnel who would require access during decommissioning. As such, authorization was revoked for ten of the users, leaving one authorized user.

2.1.3 Operating Performance

"The SLOWPOKE philosophy has been to tailor the engineering design and operating procedures of a reactor exhibiting typical inherent safety characteristics, such that these inherent characteristics assure the reactor's safety".[1] These safety characteristics are described in detail in CPR-26, Rev. 1, Description and Safety Analysis for the SLOWPOKE-2 Reactor and in Saskatchewan Research Council SLOWPOKE-2 Facility Site Description and Operating Manual for the SLOWPOKE-2 Reactor (the operating manual).

Operations at the facility are performed in accordance with the ALARA principle. That is, all procedures are performed in a manner that keeps exposure as low as reasonably achievable with emphasis on the safety of one's self, co-workers, and the general public. Radiation hazards are discussed in detail in the operating manual.

Several procedures are in place for the monitoring of contamination and radiation throughout the facility. These procedures and results of monitoring will be discussed in Section 2.3.1. Radiation Protection.

An aging management program has been established to ensure that all required routine monitoring, maintenance and inspection is done as per schedules. This systematic procedure is documented in EAL Standard Operating Procedure (SOP) SLO-160. Performance of these tasks ensure the continued safe operation of the reactor by maintaining systems in optimal condition, identifying problems before they become serious, and providing trend analysis for component aging.

2.2 Facility and Equipment

2.2.1 Safety Analysis

The original safety analysis for the SLOWPOKE-2 Reactor is detailed in CPR26, *Description and Safety Analysis for the SLOWPOKE-2 Reactor.* The facility has remained largely unchanged since its construction. There have been no renovations or redesigns of the reactor room or supporting labs up to the time of final shutdown. Following final shutdown, certain modifications - described in section 1.2.1.1 of this report - were made to the facility in preparation for decommissioning.

2.2.2 Physical Design

The activities for which the facility is licensed - operation of the reactor and production, possession, transfer, usage, packaging, managing and storage of nuclear substances required for, associated with or arising from operation of the reactor - have not changed since its commissioning. The frequencies with

[1] "CPR-26, Rev. 1, Description and Safety Analysis for the SLOWPOKE-2 Reactor". M.E. wise and R.E. Kay, Atomic Energy of Canada, Ltd. February, 1981

which these activities are performed have remained steady or decreased slightly over the years.
The overall design basis for the facility remains valid for its purpose. There were no changes during the reporting period that impacted the ability of the systems, structures or components (SSCs) to meet and maintain their design basis, other than those required for decommissioning.

2.2.3 Fitness for Service

An aging management program has been established to ensure that all required routine monitoring, maintenance and inspection is done as per schedules. This systematic procedure is documented in SOP SLO-160. Performance of these tasks ensure the continued safe operation of the reactor by maintaining systems in optimal condition, identifying problems before they become serious, and providing trend analysis for component aging.

2.2.3.1 Summary of Routine Maintenance and Inspection

Scheduled testing of alarms, monitors, the overflow sump and auxiliary power system has been effective at verifying operation and provides early indications of malfunctions so they can be fixed as soon as possible.

Aging management strategies include the regular inspections of auxiliary systems including the pool water cooling system, pool deionizer, irradiation systems, the control rod motor and wire (where visible), the outside of the reactor container, and pool walls. These inspections have revealed problems from the relatively minor, e.g., the need to replace a capsule transfer tube (external to the reactor), to the need to replace the pool deionizer system, which was done in 2009.

Water Deionizer Systems

The reactor container water deionizer system was operated weekly as part of routine maintenance until after final shutdown. Between final shutdown and commencement of decommissioning, it was operated bi-weekly. The system operated normally throughout the reporting period. The reactor water level is maintained by manually adding make-up water through the deionizer system. Indicator lights on the control console illuminate if the water level is outside limits. The system did not require the addition of water in 2019.

In preparation for defueling, the reactor container water deionizer system was operated continuously for approximately one week to ensure the reactor water was as pure as possible prior to separation of the upper and lower sections of the container. Separation of the container sections causes the reactor container water to blend with the pool water. Following defueling the reactor container water deionizer system was modified to circulate and deionize the blended pool and reactor water and allowed to run continuously until authorization to discharge the pool water was obtained.

The pool water deionizer system circulates the pool water continuously. The pool water deionizer system can operate with the deionizer beds ‘on line’ or ‘off line’. In the on line mode, water is circulated through the deionizer beds to increase water purity. In the off line mode, water bypasses the deionizer beds because the water is above the minimum purity setpoint. The system automatically toggles between the two modes to maintain water purity between 2 and 20 times the minimum required purity of 0.1 MΩ. (10 µS) . Water quality and operational data are recorded as part of the weekly maintenance routine.

The pool deionizer was shut down and disconnected on August 7, just prior to defueling the reactor. While the deionizer was in operation, makeup water was added at an average rate of 5.55 L/day, which is consistent with historical rates. The 20-inch carbon filter was replaced February 15, 2019

The pH of the water can have a significant effect on corrosion rates. Water is considered to be corrosive if the pH is less than 4.5 or greater than 8.3. Thus, it is prudent to monitor the pH of the reactor container and pool water on a regular basis. The pH measurements for the reporting period are summarized in Table 2. The pH and conductivity measurements provide additional evidence of the integrity of the reactor container and nuclear fuel.

Table 2: pH of pool and reactor container water.

Date	Pool Water pH	Reactor Container Water pH
02-Jan-19	6.3	6.0
14-Jan-19	6.1	6.1
21-Jan-19	6.5	6.1
28-Jan-19	6.9	5.8
04-Feb-19	6.1	5.9
11-Feb-19	6.2	6.4
19-Feb-19	6.5	6.0
25-Feb-19	6.4	6.1
04-Mar-19	6.4	6.0
11-Mar-19	6.5	5.9
18-Mar-19	6.9	6.2
25-Mar-19	6.6	6.0
01-Apr-19	6.8	6.1
08-Apr-19	6.5	6.0
15-Apr-19	7.1	5.8
23-Apr-19	6.2	6.0
29-Apr-19	6.5	5.8
06-May-19	6.3	6.0
21-May-19	6.4	6.0
03-Jun-19	6.3	5.6
17-Jun-19	5.8	6.0
15-Jul-19	6.2	5.9
06-Aug-19	5.8	5.7
01-Oct-19	6.1*	NR

* Sample taken after reactor container was separated and reactor water mixed with pool water.

Headspace Gas Sampling System

The headspace gas sampling system is operated during routine maintenance. No problems were experienced with the headspace gas sampling system during the reporting period.

Auxiliary Power System

The batteries that provide power to the auxiliary power system were replaced on September 5, 2017. The new batteries are sealed units, so the fluid level and specific gravity tests that were part of routine maintenance can no longer be performed. The auxiliary power system operation is tested during routine maintenance. In addition to the other tests described in "CPSR-362 Rev.2, SLOWPOKE-2 Nuclear Reactor Operation and Routine Maintenance", the batteries are "exercised" once every four to six weeks by turning the main power off during routine maintenance and allowing the auxiliary power system to operate for several hours. This test provides information on the state of health of the batteries by giving an indication of how long they may operate before becoming depleted. The longest auxiliary power test operation was seven hours, with auxiliary power reserves still adequate at the end of the test. During routine maintenance, battery voltage is read with the auxiliary power system in both operating and stand-by modes and recorded in the routine maintenance log. No power outage occurred during the reporting period.

Control Console

The control console functioned normally during the reporting period. As part of the routine maintenance, the flux and control point switches were exercised by rotating amongst the switch positions several times.

Alarm Systems

Radiation alarms were tested weekly until July 16, 2019 when they were disconnected. After July 16, radiation levels in the facility were monitored by the contractors performing the decommissioning. Security alarms were tested monthly through July. Once decommissioning activities began, the facility had manned security 24/7. The high and low water level alarms can only be tested with the pool covers removed and are tested at intervals of no greater than six months. The water level alarms were last tested March 19, 2019 and were disconnected prior to defueling. All alarm tests are documented. Systems were functional throughout the reporting period.

Shutdown Systems

The remote shutdown and control shutdown systems are tested weekly. The auxiliary shutdown system is tested annually. All functioned normally during the reporting period.

Pool Overflow Sump

The pool overflow sump is tested quarterly to verify operation. No problems were observed during the reporting period.

Irradiation Systems

No problems were observed with the irradiation systems during the reporting period.

2.3 Core Control Processes

2.3.1 Radiation Protection

Management at SRC is committed to an effective radiation protection program that eliminates unnecessary exposures to radiation and reduces unavoidable exposures to levels that are as low as reasonably achievable (ALARA). Adopting this attitude when working with radioactive material places a limitation on the risk associated with its use. The ALARA principle is a formal concept of the CNSC Radiation Protection Regulations and the best practice when working with radioactive materials.

2.3.1.1 Dose Control Data

Employees assigned to work in the facility wear dosimeters as a means of monitoring dose.
The dosimeters monitor X-ray, beta, and gamma radiation. Dosimeters are worn on the torso, providing whole body monitoring. Facility employees are grouped into two categories – certified operators and authorized users. The facility administrator is a certified operator and is included in that category. Senior management (Vice-President and President/CEO) do not work in the facility and do not wear dosimeters. Dosimetry services are provided by Mirion Technologies. Mirion's quarterly reporting periods coincide with the reporting period used by the facility (the calendar year). All dosimetry records for the year were below the minimum detectable level of 0.1 mSv.

Table 3: Summary of dose control data for the period (January 1 to December 31, 2019).

Worker Category (number of workers)	Minimum Total Effective Dose (mSv)	Maximum Total Effective Dose (mSv)	Mean Total Effective Dose (mSv)
Certified Operator (5)	<0.1	<0.1	<0.1
Authorized User (11)	<0.1	<0.10.12	<0.1

Visitors to the facility, if they do not have their own dosimeters, are issued a personal alarm dosimeter (PAD) which is read at the beginning and end of the visit. The minimum dose detectable by the PAD is 1 µSv. No detectable doses were recorded for any visitor during the reporting period.

Dose monitoring for workers performing the defueling and decommissioning of the reactor was conducted by a radiation safety officer subcontracted by the general contractor for the decommissioning of the reactor and reported to CNSC by the general contractor.

The greatest radiation hazard to personnel in the facility is the handling of irradiated samples. Procedures for the handling of radioactive material are based on the ALARA principle. Adherence to these procedures has proven very effective as doses to facility personnel have been consistently non-detectable. There have been only five detectable doses recorded within the past ten years, the maximum of which was 0.24 mSv (less than three times the minimum detectable level of 0.1 mSv). From these considerations, it is shown that the routine irradiation and handling of normal samples does not pose a significant risk if the foregoing procedures are followed.

2.3.1.2 Contamination Control Data

Routine surface contamination monitoring was historically conducted three times per year at prescribed locations throughout the facility, as detailed in SOP SLO-130 “Contamination and Radiation Monitoring”. Due to the decommissioning schedule, contamination monitoring was conducted only once during the reporting period, in March.
Additionally, if contamination is suspected the area is monitored immediately. Areas routinely monitored are work and irradiation stations in Room 143, work and irradiation stations and the fume hood in Room 144, the floor in Room 145 and the console desk and pool covers in Room 146. Additional contamination monitoring is conducted at each irradiation station on a weekly basis or whenever the irradiation station is used, whichever is less frequent, as detailed in SOP SLO-131 “Contamination Monitoring at Irradiation Stations”. Surface contamination monitoring is done using a portable survey meter and probe. The facility action level for contamination on all surfaces is 0.3 Bq/cm^2.

Routine contamination monitoring is conducted at 18 prescribed locations. Monitoring is conducted using a hand-held Geiger counter equipped with a pancake probe which measures counts per minute (cpm). Activity in Bq/cm^3 is calculated from the reading in cpm, taking the window area and efficiency of the probe into account. For the majority of the locations, there was no detectable contamination. The highest measured activity, 0.03 Bq/cm^2, was recorded at one location on the floor of room 143. This reading is less than 2 times background (background averages about 50 cpm) and does not exceed the action level. Results of monitoring during the reporting period are consistent with historical levels. There has been no upward trend in contamination levels over the life of the facility. This is a strong indication that contamination control procedures in place at the facility are functional and sufficient.

Contamination monitoring at the irradiation stations following use did not produce any readings in exceedance of the action levels. The presence of residual ^{41}Ar in the transfer lines when the reactor is or was recently operating will influence the radiation readings when doing direct monitoring. For this reason, direct monitoring is not done for at least 5.5 hours (about three half-lives of ^{41}Ar) after the reactor is shut down.

There were no personnel contamination events at the facility during the reporting period.

2.3.1.3 Facility Radiological Conditions

Routine radiation monitoring was historically conducted three times per year at prescribed locations throughout the facility, as detailed in SOP SLO-130 “Contamination and Radiation Monitoring”. Due to the decommissioning schedule, contamination monitoring was conducted only once, in March 2019.

Action levels are 0.1 mR/h (1µSv/h) at locations outside the facility perimeter, 0.2 mR/h (2 µSv/h) at work and irradiation stations and various locations in room146 (except for the three fixed monitors in room 146 and near the reactor deionizer), 2 mR/h (20 µSv/h) in the fume hood in room 144, and 2.5 mR/h (25 µSv/h) in the storage room and near the reactor deionizer. The action levels for the three fixed radiation monitors in Room 146 are: Reactor (under pool covers) 10 mR/h (0.1 mSv/h), Area (ceiling) 3 mR/h (30 µSv/h) and Reactor water deionizer 25 mR/h (0.25 mSv/h).

Radiation monitoring in March 2019 was performed with the reactor not operating and the pool covers closed. Prior to monitoring, background was measured in an open area just outside the entrance to the facility and was 0.02 mR/h (0.2 µSv/h). Measurements taken at locations outside the facility perimeter, at work and irradiation stations, the fume hood, and various locations in room 146 were consistently at or near background. Levels in the storage room were slightly elevated due to the presence of radioactive materials, but did not exceed 25% of the action level. The reading from the fixed monitor under the pool covers was less than 1% of the action level with the reactor not operating. The reading from the ceiling monitor was 0.02 mR/h (0.2 µSv/h), less than1% of the action level. The reading from the reactor water deionizer monitor was 3.2 mR/h (432 µSv/h) The field from the deionizer varies depending on reactor use during the previous week and seasonally. Increased reactor use results in a higher concentration of radioisotopes in the water, which are subsequently adsorbed by the resin bed during purification. The result for the reporting period is typical for the level of reactor use at the time of monitoring and is less than 15% of the action level.

2.3.1.4 Radioactivity of Reactor Container Water

Liquid Scintillation Measurements

Liquid scintillation measurements of the reactor container water are performed monthly, in-house. In general, activity of the container water is relative to the number of flux-hours the reactor was operated in the week prior to testing. The results of the 2019 measurements are provided in Table 4.

Table 4: Gross radioactivity of reactor container water as measured by liquid scintillation counting.

Date	Reactor Container Water Gross Radioactivity (Bq/L)
02-Jan-19	12907
04-Feb-19	24081
04-Mar-19	28024
01-Apr-19	12704
06-May-19	8455
03-Jun-19	5407
02-Jul-19	16603
06-Aug-19	4560
02-Oct-19	216*

* Sample taken after reactor container was opened and reactor water mixed with pool water.

Gamma Spectroscopy Measurements

Annual gamma spectroscopy measurements of radionuclides in the reactor container water and pool water provide additional confirmatory evidence of the integrity of the reactor container and nuclear fuel. The reactor container water and pool water were measured on July 29, 2019, approximately three months after final operational shutdown. A summary of the measurements is provided in Table 5.

Table 5: Gamma spectroscopy measurements of the concentration of radionuclides in the reactor container and pool water.

Nuclide	Concentration in Container Water (Bq/L)	Concentration in Pool Water (Bq/L)	Nuclide	Concentration in Container Water (Bq/L)	Concentration in Pool Water (Bq/L)
Ar-41	<0.01	<3	I-135	<0.4	<0.8
Xe-133	<5	<0.7	Sr-85	4	<0.2
Xe-133M	<40	<8	Sr-91	<2	<0.6
Xe-135	<2	<0.2	Y-88	<0.1	<0.2
Na-24	<0.3	<0.1	Y-91M	<2	<30
Be-7	<20	<2	Rh-106	<10	<1
Cr-51	<20	<2	Ru-103	<2	<0.2
W-187	<0.5	<1	Cs-134	10	<0.3
Mn-54	<0.2	<0.06	Cs-136	<0.3	<0.2
Mo-90	<1	<0.4	Cs-137	3950	<0.2
Mo-99	<1	<0.1	Ba-140	<8	<0.6
Tc-99M	<1	<0.2	La-140	0.9	<0.08
Fe-59	<0.5	<0.1	Ce-144	35	<2
Co-57	<2	<0.2	Cd-109	<40	<6
Co-58	<1	<0.07	Ce-139	<1	<0.2
Co-60	0.5	<0.1	Hg-203	<2	<0.2
Zn-65	<0.6	<0.4	Sn-113	4	<0.2
Nb-94	<0.2	<0.3	K-40	8	<6
Zr-95	2	<0.5	Sb-124	<0.5	<0.3
Nb-95	2	0.2	Sb-125	<6	<0.6
Zr-97	<0.2	<0.2	Se-75	<2	<0.3
Nb-97	<2	<2	Eu-152	<5	<0.7
I-131	<1	<0.2	Ra-226	34	<5
I-132	<0.3	<2	U-235	<2	<0.4
I-133	<3	<0.2	Te-132	<1	<0.2

2.3.1.5 Exceedances of Regulatory Limits or Action Levels

There were no exceedances of regulatory limits or action levels during the reporting period.

2.3.1.6 Radiation Protection Program Effectiveness

No facility personnel received a detectable dose during the reporting period. No personnel contamination events, and no exceedances of regulatory limits or action levels occurred during the reporting period. This is a strong indicator that the radiation protection program in place at the facility is well designed and well managed to keep exposure to a minimum and to prevent unreasonable risk to the health and safety of personnel.

2.3.1.7 Radiation Protection Program Improvements

The radiation protection program in place at the facility is a mature program. The program is based on the ALARA principle, “to keep all exposures As Low As Reasonably Achievable, with social and economic factors taken into account”. Regular reviews of the program and safety analysis of new activities, if applicable, are conducted to ensure the program remains current. The effectiveness of the program, as stated in Section 2.3.1.6 indicates that the program is well suited for the purpose it serves. Continuous improvement of the radiation protection program is included as part of the continuous improvement of the OH&S Management System. A radiation safety refresher course, to be taken every three years, has been implemented.

2.3.1.8 Radiation Protection Program Performance

Specific numerical goals and targets (e.g., reduce exposures by 0.05 mSv) are not set by the radiation protection program. Rather, the goals and targets of the radiation protection program are more global in nature, (i.e., to keep employee exposure to a minimum and minimize or prevent releases to the environment). As evidenced by data presented in Sections 2.3.1.1 and 2.3.3.1, the program is meeting its goals. Decommissioning activities planned for 2020 will not alter the goals of the radiation protection program. Radiation protection measures that are in line with the current goals are built into the detailed decommissioning plan.

2.3.1.9 Summary of Continuous Improvements under ALARA Performance

The facility is not required to have an ALARA committee. The Radiation Protection Program is based on the ALARA principle. Discussion of improvements and performance of the program are discussed in Sections 2.3.1.7. and 2.3.1.8.

2.3.1.10 Summary of Radiation Protection Training Program and Effectiveness

The initial EAL site-specific training provided to new employees followed by the SRC general radiation safety course, and the facility-specific safety training embedded in the authorized user training manual, equips the employees with the knowledge and skills necessary to work safely with radioactive substances and devices. The effectiveness of this training is reflected in the results of dose control, radiation monitoring and contamination monitoring data discussed earlier in this report.

2.3.1.11 Summary of Radiation Device and Instrument Performance

The only radiation device in the possession of EAL that falls under the licensed activities of the facility is the 11.77 mg ^{235}U sealed source which is used as a reference when taking fissile monitor readings of encapsulated material prior to irradiation. The activity of the device is 940 Bq. The device is doubly sealed – that is the tiny chip of ^{235}U is inside a heat-sealed 1.5 cc polyethylene vial, which is in turn heat-sealed inside a 7cc polyethylene vial. The purpose of this packaging is to mimic a sample which has been prepared for irradiation by similar encapsulation. The only time the device is handled is to place it in the fissile monitor reader (FMR). It is not subjected to any stress, force or heat. The device is adequately robust for the purpose it serves.

Radiation instruments in use in the facility include the three radiation monitors in room 146, rate meters at each of the irradiation stations in room 144 and several portable survey meters which are available for use throughout the facility and EAL.

The three radiation monitors in room 146 provide a continuous readout of radiation levels in the room (area monitor), under the pool covers (reactor monitor) and next to the reactor water deionizer column (deionizer monitor). The three monitors are tested weekly by holding a source next to the detector and observing response. For the pool and area monitors, it is verified that the alarms were received at the alarm monitoring stations (the deionizer monitor does not have a remote alarm associated with it). Alarm trip levels and verification of receipt are recorded in the alarm testing log. Calibration of the area, pool and deionizer monitors is done in-house, using a procedure which satisfies the criteria in Regulatory Document R-117 and which was accepted by CNSC in 2011 (email from E. Thanabalasingam to W. Yuen, subject: Response to Action Item RC2007-AN03). The monitors were calibrated on June 11, 2019.

The purpose of the rate meters at the irradiation stations in room 144 is to detect abnormally high levels of radiation from freshly irradiated samples. This is an early warning system for the reactor operators or authorized users that an elevated level of radiation is emanating from the sample that has just been irradiated in the SLOWPOKE-2 reactor. These monitors are used in the rate mode, counts per minute. The monitors are not intended to be used as survey meters and thus are not required to be calibrated. The monitors are checked quarterly to verify that they are functioning by holding a source close to the detector and noting the level at which the audible alarm is tripped. Functionality checks are recorded on Form SLO-160, Inspection and Maintenance Checklist.

The portable survey meters are calibrated annually by an external service provider. The calibration schedule is staggered such that no more than two meters are away for calibration at a time. The RSO maintains the calibration schedule and calibration certificates. Verification of performance is conducted prior to the first usage of the day for each meter and recorded in the Portable Radiation Survey Meter Logbook.

2.3.1.12 Summary of Inventory Control Measures

Following final operational shutdown and prior to commencement of decommissioning, all sealed sources were removed from the facility and stored at a different location within SRC. No sealed sources are currently stored at the facility.

2.3.2 Conventional Health and Safety

2.3.2.1 Discussion on Conventional Health and Safety Program Effectiveness

Conventional health and safety at the facility falls under the umbrella of the SRC corporate OH&S program. The SRC corporate OH&S program has evolved over the years and is based on a management system. Description of the management system structure and its implementation is discussed in Section 2.1.1.1. Inspections, audits and management reviews serve the purpose of monitoring OH&S performance as well as ensuring continuous improvement of the management system.

In 2019 the following inspections and audits were carried out:

- Fire inspection by Saskatoon Fire and Protective Services
- Radiation Safety Inspection conducted by SRC RSO

All the above inspections were formally recorded. Action items resulting from these inspections were tracked and completed as per target dates.

The effectiveness of the OH&S program is evidenced strongly by:

- The consistent minimal or non-detectable dose exposure to facility employees, as discussed in Section 2.3.1.1.
- No lost time incidents in the history of the SLOWPOKE-2 facility.

2.3.2.2 Summary of Occupational Health and Safety Committee Performance

SRC's Occupational Health and Safety committees meet every three months to conduct meetings and inspections as mentioned above. The committees are actively involved in the implementation of the OH&S Management System and also conducting promotional campaigns and events such as NAOSH week celebrations during May of every year.

The committees' performance is measured against expectations described in the Saskatchewan Occupational Health and Safety Act and Regulations. Serving on the committees is also considered during performance evaluation of employees at the end of the year.

2.3.2.3 Summary of Conventional Health and Safety Program Improvements

During the year 2019 SRC OH&S program continued to improve with specific initiatives as listed below:

- A Pre-trip inspection process for all business trips with load was introduced and is being monitored.
- Safety Awards for the year 2018-19 were presented to the winners in three different categories – Everyday Role Model, Special Project and Top SPOT (unsafe condition or unsafe act) report. Winners were selected by the OH&S Committee members.
- Mock drill exercises for spill emergency scenarios were conducted for the remaining BUs from last year.
- SRC won "Mission Zero" award by WorkSafe Saskatchewan. This is the 4th year SRC received the award in the 4 years that it has been a charter member.
- Latest technology sealed safety glasses were tested by the Environmental Analytical Lab (EAL) employees to address risks related to the incidents of chemical splashes.
- An internal audit of the Air and Climate Business Unit was conducted as part of the annual requirement to maintain Certificate of Recognition (COR). The audit resulted in a score of 88% which is considered satisfactory.

2.3.2.4 Discussion of Hazardous Occurrences

There were no hazardous occurrences related to the facility during the reporting period.

2.3.3 Environmental Protection

2.3.3.1 Air and Water Release Monitoring Results

Reactor Container Head Space Purge

The volume of the head space in the reactor container is 108 L. During normal reactor operation, hydrogen - from the radiolytic decomposition of water - and fission and activation product gases accumulate in the reactor container head space. These gases are released when the head space is purged. Head space purge is part of routine reactor maintenance scheduled on the first workday of each week. This schedule reduces the concentration of radionuclides released in the head space purge since it allows additional time over the weekend for the short-lived radionuclides to decay. The radioactive gases in the reactor head space have relatively short half-lives and decay fairly quickly when the reactor is not operating. In accordance with CNSC requirements, a 10 L sample of the headspace gas is collected in an aluminum Marinelli type container prior to the weekly purge and analyzed by gamma spectroscopy. Purging is not done until the sample has been analyzed and the results have been compared to historical values and release limits. In 2019, purging was done weekly (unless the reactor was not used in the previous week) until the first Monday following final shutdown, then bi-weekly until decommissioning began in August. The results of monitoring measurements taken during the reporting period are summarized in Table 6.

Variations in the concentration of the radionuclides released depend primarily on two factors; the extent of reactor usage during the week prior to the release and the elapsed time between the last reactor use and the release.

Table 6: Concentration of gaseous fission and activation products in the reactor container headspace prior to the weekly purge.

Radionuclide	^{41}Ar	^{131M}Xe	^{133}Xe	^{133M}Xe	^{135}Xe	^{85M}Kr	^{87}Kr	^{88}Kr
Date	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**	**(Bq/L)**
02-Jan-19	<0.07	35	1460	4.7	0.07	<0.1	<0.3	<0.4
14-Jan-19	<0.1	58	9630	270	469	<0.6	<0.2	<2
21-Jan-19	<0.1	30	5100	87	149	<0.4	<0.5	<0.8
28-Jan-19	<0.1	32	6410	123	178	<0.4	<0.5	<0.6
04-Feb-19	<0.06	33	5310	77	30	<0.2	<0.3	<0.7
11-Feb-19	<0.2	36	4990	87	76	<0.2	<0.4	<0.8
19-Feb-19	<0.08	23	3220	56	37	<0.3	<0.4	<0.5
25-Feb-19	<0.08	19	2460	36	22	<0.2	<0.3	<0.4
04-Mar-19	<0.07	17	3270	63	8.1	<0.2	<0.3	<0.3
11-Mar-19	<0.1	23	4830	84	155	<0.4	<0.4	<0.9
18-Mar-19	<0.3	16	2330	30	1.1	<0.09	<0.4	<0.5
25-Mar-09	<0.2	17	2000	36	34	<0.2	<0.3	<0.6
01-Apr-19	<0.2	21	2480	44	92	<0.3	<0.3	<0.9
08-Apr-19	<0.2	16	2070	37	31	<0.2	<0.4	<0.4
15-Apr-19	<0.2	20	2820	60	228	<0.5	<0.4	<0.8
23-Apr-19	<0.05	13	1880	34	6.0	<0.1	<0.3	<0.3
29-Apr-19	<0.1	18	4890	123	225	<0.5	<0.2	<1
06-May-19	<0.1	18	2880	30	0.5	<0.2	<0.3	<0.4
21-May-19	<0.06	11	328	<0.4	<0.07	<0.1	<0.1	<0.3
03-Jun-19	<0.07	8.9	92	<0.4	<0.05	<0.1	<0.2	<0.4
17-Jun-19	<0.06	6	17	<0.3	<0.06	<0.09	<0.2	<0.3
02-Jul-19	<0.1	5	3.5	<0.5	<0.02	<0.1	<0.3	<0.2
15-Jul-19	<0.1	3	0.7	<0.4	<0.06	<0.1	<0.3	<0.2
29-Jul-19	<1	<2	<0.3	<0.5	<0.09	<0.1	<8	<1
2019 Average*		**25**	**3780**	**71**	**97**			
2018 Average		42	8030	140	156			
2017 Average		54	9850	165	169			
2016 Average		38	8210	141	140			
2015 Average		52	10800	180	177			
2014 Average		49	10400	176	183			
2013 Avg.**		46	8980	148	168			
2012-13 Avg.		45	8130	131	122			

*2019 averages include only the data up to May 6 (first sampling after final shutdown). Data for samples collected from May 21 forward show that the concentrations (with the exception of ^{131m}Xe) do not exceed 10% of the average concentration while the reactor was operational and are considered insignificant in the calculation of the total quantity released.

**2013 data is for April to December only. Previous years are April to March (e.g. April 2012 to March 2013). Data for 2014 and forward are January to December.

Notes:

1) Concentrations are in units of Bq per liter of headspace gas. Since the total volume of the headspace is 108 L, the total quantity (in Bq) of any radionuclide released during the weekly purge can be calculated by multiplying the

concentration by 108. However, this is a slight overestimate since 10 L of the headspace gas is taken for the gamma spectroscopy sample. This 10 L sample is typically not released until the following week when the sampling container is needed again. This practice yields some further minor reduction in the quantities of radionuclides released to the environment.

2) Any fission or activation gases that are not listed were not detected.
3) The symbol < means “less than”. Results with such a symbol indicate that the radionuclide was not detected. The numeric value beside the symbol indicates the Minimum Detectable Activity (MDA) of the measurement system for that radionuclide. If the radionuclide was not detected and greater than 12 half-lives of that radionuclide had elapsed between sampling and counting, then no result (---) is reported. If no result was reported or if the result was less than the MDA, then a value of zero was used to calculate the average quantity released per week. No average was calculated for isotopes with results that were typically either not reported or less than the MDA. Reported averages include only the detectable values.
4) Since its inception, the weekly monitoring program has been applied quite consistently. However, due to a variety of reasons there are some measurements or measurement results that are not available. In most cases, this is because the reactor was not used in the previous week. If the reactor is not used during the week, then routine maintenance is not required and therefore there is no release. In other cases where measurement results are not available, the headspace purge is not performed and there is no release for that week.

Other Air Releases

Other releases of fission and activation product gases occur during normal operation of the reactor. The facility has developed a formal environmental protection program including procedures for estimating the quantity of fission and activation products in releases from transfer operations and diffusion processes. These procedures are based on the results of an environmental monitoring report submitted to CNSC in May 2001 and environmental monitoring results from previous years. In accordance with the facility's environmental protection program, periodic measurement of the exhaust gas flow rate and radionuclide concentrations under various reactor operating conditions provide a basis for estimating the quantity of radionuclides in these releases. The annual sampling program historically took place in the fall. With the final shutdown of the reactor occurring on April 30, 2019 no exhaust gas samples were taken in 2019. The last sampling period was in September – October 2018, the results of which were discussed in the 2018 Annual Compliance Report. The average concentration of each radionuclide over the previous six years will be used to estimate the quantities released during 2019. A description of each sample is given in Table 7.

Table 7: Description of environmental monitoring samples.

Sample Identification	Description
RNO	This sample is taken while the reactor is not operating. This sample is intended to provide information on any releases that may occur from diffusion when the reactor is not operating. From 2013 to 2018 the average time between reactor shutdown and the collection of this sample was 1.6 days
ROWT	This sample is taken with the reactor operating at half power (5×10^{11} $n.cm^{-2}.s^{-1}$). Sampling is coordinated with other staff so that a total of 10 pneumatic transfer operations (five transfers in, five transfers out) are performed while the sample is collected. A small (inner) irradiation site is used for the transfer operations. This sample is intended to provide information on releases that may occur during transfer operations.
RONT	This sample is taken with the reactor operating at half power (5×10^{11} $n.cm^{-2}.s^{-1}$). No transfer operations are performed during sampling. From 2013 to 2018 the average elapsed time between the last transfer operation and sample collection was 106 minutes. The reactor was in operation for an average of 4.3 hours prior to sampling. This sample is intended to provide information on releases that may occur due to diffusion processes while the reactor is operating.

^{41}Ar is produced from the irradiation of air in the transfer tubes. Results from samples 2013 to 2018-ROWT indicate an average of 42.0 Bq/L of ^{41}Ar (420 Bq/L ÷ 10 transfers) is released in a single transfer operation. Measurement results also indicate that there are traces of ^{133}Xe released during transfer operations. A summary of environmental monitoring results from previous years is presented in Table 8.

Table 8: Summary of environmental monitoring program results over time.

Estimated concentration of radionuclides released in a transfer operation (ROWT)	41Ar (Bq/L)	133Xe (Bq/L)	135Xe (Bq/L)
2013	26.7	0.15	<0.02
2014	53.9	0.23	<0.1
2015	59.3	0.26	<0.2
2016	35.6	0.18	<0.1
2017	33.2	0.19	<0.2
2018	43.4	0.26	0.08
Average	42.0	0.21	--
Estimated concentration of radionuclides released due to diffusion processes while the reactor is operating (RONT)	**41Ar (Bq/L)**	**133Xe (Bq/L)**	**135Xe (Bq/L)**
2013	60.0	2.0	<0.08
2014	3.7	1.2	<0.1
2015	40.0	0.8	0.20
2016	67.0	1.8	<0.1
2017	81.0	2.9	0.9
2018	85.0	0.8	0.4
Average	56.1	1.6	--
Estimated concentration of radionuclides released due to diffusion processes while the reactor is not operating (RNO)	**^{41}Ar (Bq/L)**	**^{133}Xe (Bq/L)**	**^{135}Xe (Bq/L)**
2013	<0.4	2.2	<0.05
2014	<0.4	<2	<0.08
2015	<0.1	1.2	<0.06
2016	<0.1	2.3	0.2
2017	<0.2	0.4	<0.7
2018	<0.07	1.4	<0.06
Average	--	1.5	--

As outlined in the facility's environmental protection program[2], the measured concentrations of radioisotopes in the exhaust gases from samples taken during transfer operations are used to calculate an estimate of environmental releases due to transfer operations. For this estimate the number of capsules irradiated is multiplied by a factor which takes into account the exhaust gas flow rate, the number of transfer operations per capsule irradiation and the concentration of radionuclides measured in samples of

[2] A Means of Estimating Releases of Fission and Activation Products to the Environment From Sample Transfer Operations and Diffusion Processes, April 2001, Jeff Zimmer, SRC Environmental Analytical Laboratories SLOWPOKE-2 Facility.

the exhaust gas. The multiplication factor uses the radionuclide with highest measured concentration to ensure that a conservative estimate is calculated. The estimated quantities of radionuclides released via transfer operations are presented in Table 9.

Table 9: Estimated quantity of gaseous radionuclides released from transfer operations.

Month (2019)	Capsules Irradiated	Estimated Quantity Released		
		^{41}Ar (MBq)	^{133}Xe (MBq)	^{135}Xe (MBq)
January	225	133.3	0.67	<0.03
February	169	100.1	0.50	<0.02
March	161	95.4	0.48	<0.02
April	176	104.3	0.52	<0.02
TOTAL	**731**	**433.1**	**2.17**	**<0.09**

Notes:

1) Any gaseous radionuclides not listed were not detected in the exhaust gas samples taken. All gaseous emissions from the reactor exit through the same exhaust vent. Measurements at the vent during the 2013 to 2018 sampling programs indicated an average flow of 3527 L of air per minute.
2) The estimated quantity released is based on the number of transfer operations per irradiation (2), the measured concentration of the radionuclide in the exhaust gas samples, the exhaust gas flow rate, the time of sampling and the number of transfer operations during sampling. For example, , the 2013 – 2018 average observed concentration of ^{41}Ar in the exhaust gas during sample transfer operations was 420 Bq/L. There were 10 transfers over a period of 2.00 minutes so the estimated maximum quantity released per transfer is:

$$420\text{Bq/L} \times 3527 \text{ L/min} \times 2.00 \text{ min} \div 10 \text{ transfer operations} = 296{,}268 \text{ Bq/transfer}$$

In January 2019 there were 225 irradiations and the maximum estimated amount of ^{41}Ar released from the related transfer operations is:

$$296{,}268\text{Bq/transfer} \times 2 \text{ transfers/irradiation} \times 225 \text{ irradiations} = 133.3\text{MBq}$$

The CNSC has requested that SRC calculate the ^{41}Ar production rate for comparison purposes. Calculations[3] indicate that there are approximately 4.8 MBq of ^{41}Ar produced for each flux-hour of reactor operation. Table 1 lists a total of 507.2 flux-hours of reactor operation during the year. From these values the total quantity of ^{41}Ar produced during the year is calculated as 2434.6 MBq. It is notable that all of the ^{41}Ar produced in the reactor is not released to the environment. This is because diffusion of the ^{41}Ar out of the transfer tubes into the exhaust duct is hindered by at least three factors:

- A lack of make-up air in the transfer tubes (that is, air does not flow in a transfer tube unless there is a transfer operation)
- A HEPA filter in the connection lines where the transfer tubes meet the exhaust duct

[3] Calculations are given in Appendix A.

- Mechanical devices which limit the number of transfer tubes that can be connected to the exhaust system at any one time.

Sample RONT is taken to provide information on releases that may occur due to diffusion processes out of the transfer tubes while the reactor is operating, but no transfer operations taking place. The results are used to estimate releases from diffusion processes while the reactor is operating.

The estimated quantity of ^{41}Ar released from diffusion processes during reactor operation is calculated[3] to be 2,374,400 Bq per flux hour of reactor operation. The total number of flux hours during the year was 507.2. Therefore, the estimated quantity of ^{41}Ar released from diffusion processes is 1204.3 MBq. ^{41}Ar is also released during capsule transfer operations. The estimated quantity of ^{41}Ar released from transfer operations is listed in Table 9 as 433.1 MBq. An estimate of the total quantity of ^{41}Ar released during the year is the sum of these two quantities or 1637.4 MBq. This total is approximately 67% of the estimated quantity of ^{41}Ar produced.

Sample RNO is taken to provide information on releases that may occur due to diffusion processes out of the transfer tubes while the reactor is **not** operating. Traces of ^{133}Xe were detected in sample RNO taken annually from 2013-2018. This indicates that traces of ^{133}Xe diffuse out of the transfer tubes when the reactor is not operating. A simplified estimate of the total quantity of ^{133}Xe released from diffusion processes can be calculated from the measurement results.

Results presented in Table 9 for Sample RONT indicate that the average concentration of ^{133}Xe due to diffusion in the exhaust gases when the reactor is operating is 1.6 Bq/L. During the reporting period, the reactor was operated for a total of 110.1 hours. An estimate of the quantity of ^{133}Xe released from diffusion processes while the reactor is operating is then:

$$1.6 \text{ Bq/L} \times 3527 \text{ L/min} \times 60 \text{ min/hr} \times 110.1 \text{ hr} = 37.3 \text{ MBq}$$

Sample RNO indicates that the 2013 – 2018 average concentration of ^{133}Xe due to diffusion in the exhaust gases when the reactor is not operating is 1.5 Bq/L. The concentration of ^{133}Xe in exhaust gases is expected to be highest immediately following reactor operation and to gradually diminish between periods of reactor operation. Immediately after purging, the concentration of ^{133}Xe in the headspace is expected to be non-detectable. Following purging, ^{133}Xe builds up when the reactor is operating. The concentration of ^{133}Xe present at the time of sampling is influenced by the total flux-hours of operation in the week preceding sampling, whether a majority of the flux-hours were accumulated earlier or later in the week, and the elapsed time between shutdown and sampling. Since the inception of the environmental monitoring program, these measurements have been taken with variations in the amount of time elapsed between shutdown and sampling; the results are summarized in Table 10.

Table 10: Relationship between time since reactor shutdown and concentration of ^{133}Xe released from diffusion.

Time Elapsed between Reactor Shutdown and Sampling (Days)	Estimated concentration of ^{133}Xe released from diffusion while the reactor is not operating (Bq/L)
1 (in year 2018)	1.4
1(2017)	0.4
1 (2016)	2.3
1 (2015)	1.2
1 (2014)	<2
4 (2013)	2.2
Average	**1.5**

For simplicity, the average of these values (1.5 Bq/L) is used to estimate the concentration of ^{133}Xe released from diffusion processes when the reactor is not operating. The total quantity of ^{133}Xe released from diffusion processes when the reactor is not operating can then be estimated based on the amount of time the reactor was not operated. During the reporting period, the reactor was operated from January 1 to April 30. After April 30, no ^{133}Xe was produced and by May 6 the concentration of ^{133}Xe in the container head space had decayed to a negligible concentration. Thus, only the period of January 1 to May 6 is considered in the calculation of ^{133}Xe released while the reactor was not operating. The reactor was not operated for 2913.9 hours (3024 hours from January 1 to May 6 minus the 110.1 hours of reactor operation). The estimated quantity of ^{133}Xe released while the reactor was not operating is:

$$1.5 \text{ Bq/L} \times 3527 \text{ L/min} \times 60 \text{ min/hr} \times 2913.9 \text{ hr} = 925.0 \text{ MBq}$$

An estimate of the total quantity of ^{133}Xe released through diffusion processes during the period of this report is the sum of the estimated amount released due to diffusion while the reactor is operating (37.3 MBq) and the estimated amount released due to diffusion when the reactor is not operating (925.0 MBq) which equals 962.3 MBq.

Water Releases

There were no releases of reactor container water at the facility during the reporting period. During weekly pH testing of the pool water, approximately 200 mL is discharged to the city sewer system. This was the only release of pool water during the reporting period.

2.3.3.2 Significance of Air and Water Release Monitoring Results

The measured and estimated quantities of radionuclides released to the environment from reactor operations were used to estimate the maximum potential radiation dose to a member of the general public from facility operations. The estimates were prepared using a scenario that would clearly lead to a conservative estimate of the radiation dose. A complete description of the scenario used to estimate the maximum radiation dose and the related dose calculations is given in Appendix A. Since the scenario chosen is very conservative and unlikely to occur, any radiation doses actually received by a member of the public from reactor operations would be much less than the values estimated here. The estimated doses are summarized in Table 11. The total estimated maximum radiation dose from all the various types of reactor releases is 3.90×10^{-6} Sv which is less than 0.4% of the annual occupational radiation exposure limit for a member of the general public.

Table 11: Estimated maximum radiation dose from various radionuclide releases at the SRC SLOWPOKE-2 Facility

Type of Release	Estimated Maximum Annual Radiation Dose
Weekly reactor container headspace purge	3.92×10^{-10} Sv
Diffusion of ^{41}Ar while reactor is operating	2.83×10^{-6} Sv
Releases of ^{41}Ar from pneumatic transfer operations	1.02×10^{-6} Sv
Diffusion of ^{133}Xe	5.14×10^{-8} Sv
Releases of ^{133}Xe from pneumatic transfer operations	1.16×10^{-10} Sv
Estimated Maximum Dose from all Releases	**3.90×10^{-6} Sv**

2.3.3.3 Exceedances of Regulatory Limits or Action Levels

There were no exceedances of regulatory limits or action levels at the facility during the reporting period.

2.3.3.4 Environmental Protection Program Effectiveness

The facility's environmental protection program was established in 2002 as a means to monitor and minimize the impact of the facility's operations on the environment.

The data generated by the environmental protection program are documented in the annual compliance report, which is reviewed internally by the SLOWPOKE-2 Committee, and externally by CNSC. The total estimated maximum radiation dose from all the various types of reactor releases is much lower than the annual occupational radiation exposure limits for the general public. It should be noted that the potential dose estimations are based on very conservative scenarios and that the actual dose risks are likely much lower than the estimates. This is strong evidence that the managerial controls on the prevention of unreasonable risk to the environment are effective and adequate.

2.3.3.5 Summary of Environmental Protection Program Improvements

The facility's environmental protection program is a mature program, having been deployed effectively for 18 years. SOPs related to environmental monitoring are audited on a three-year cycle and are updated as necessary. The licensed operations and activities at the facility have not changed since its commissioning, nor have the design of the facility or the scope and nature of the sample irradiations performed. If any changes to the above were proposed, a change review would be conducted, including an assessment of impact to the environment.

2.3.3.6 Summary of Environmental Protection Program Performance

The environmental protection program uses the ALARA principle to minimize the impact of the facility's operations on the environment.
The objectives of the program are:

- To minimize wastes, especially radioactive and other hazardous wastes
- To measure the quantities and concentrations of any necessary radioactive releases and to investigate any unusual increases in the quantities of such releases
- To control radioactive and hazardous substance releases.

Results of air and water release monitoring show that these objectives were achieved.

2.3.3.7 Comment on Well and Soil Sampling and Measuring/Monitoring

Well and soil sampling is not applicable to the facility.

2.3.4 Emergency Management and Response

Emergency response is administered by SRC Facilities Department and is detailed in the Emergency Management Program which is posted on the SRC intra-web. The Emergency Management Program is an active program that is reviewed and updated on an ongoing basis rather than at predetermined static dates. All incidents are recorded and reviewed by the team, and any areas that are found to have performance deficiencies are updated. The ongoing process ensures that the Emergency Management Program is and remains current.

2.3.4.1 Review of Emergency Preparedness Program Activities

Since the relocation of EAL to its new lab space in April 2018, the portion of the building outside the reactor facility has been unoccupied. Since the final shutdown of the reactor, reactor personnel have only been required to enter the facility sporadically and for short periods of time. Because of this, evacuation and shelter-in-place drills have been discontinued at the facility. Fire alarms remain active and fire extinguishers are still present. A personal distress alarm is available for personnel who need to enter the facility.

2.3.4.2 Summary of Emergency Preparedness Training Program and Effectiveness

Emergency response preparedness training includes the activities described in Section 2.3.4.1. In addition, EAL conducts semi-annual reviews of the site-specific emergency response procedures, and an annual safety review which includes emergency response procedures. All staff are required to attend these training sessions.

2.3.4.3 Summary of Fire Protection Program Improvements

The facility fire protection was submitted to CNSC (letter from D. Chorney to I. Erdebil, January 17, 2012, Subject: Fire Protection Program for SRC SLOWPOKE Facility). Acceptance of the program was acknowledged in a letter dated June 18, 2012 from I. Erdebil to D. Chorney, Subject: CNSC Staff Review of Saskatchewan Research Council SLOWPOKE-2 Facility Fire Protection Program. Annual inspections by Saskatoon Fire and Protective Services help ensure the program remains current and effective.

2.3.5 Waste and By-Product Management

Liquid and solid wastes from the SLOWPOKE-2 facility are irradiated sample capsules. All irradiated samples are stored for at least six months and very often for more than one year before disposal. Consequently, sample radiation levels are normally indistinguishable from background (0.2 to 0.3 µSv/h) so that the samples can be considered inactive and may be disposed of by normal waste disposal techniques.

Occasionally there are samples, primarily from uranium mining operations, which contain naturally occurring radioactive materials that are irradiated in the SLOWPOKE-2. These samples have radiation levels above natural background levels prior to irradiation. Although the artificially created radionuclides in these samples decay fairly rapidly, the naturally occurring radionuclides have long half-lives and therefore these samples will not decay to background levels after irradiation. These samples are retained and eventually returned to the client for disposal. During the past year of operation, there were no radioactive samples that were returned for disposal.

2.3.6 Nuclear Security

The facility security plan was updated in March 2017.

2.3.7 Safeguards and Non-Proliferation

Physical Inventory Taking (PIT) was conducted October 31, 2019. The facility was not selected by IAEA for physical inventory verification (PIV) in 2019. Nuclear Material Reporting Forms are submitted as required, as per *RD-336 – Accounting and Reporting of Nuclear Material*.

2.3.8 Packaging and Transport of Nuclear Substances

Following defueling, the reactor core was transferred to the United States Department of Energy under the US Global Threat Reduction Initiative. Packaging and transport met all requirements of the *Packaging and Transport of Nuclear Substances Regulations, 2015,* section 5 of the *Nuclear Security Regulations (SOR/2000-2009)*, and *Regulatory Guide G-208 Transportation Security Plans for Category III Nuclear Material.* No other packaging or transport of nuclear substances was conducted during the reporting period.

3.0 OTHER MATTERS OF REGULATORY INTEREST

3.1 Public Information Program

3.1.1 Summary of Public Information Program Activities

SRC SLOWPOKE Public Information Plan revision 9 was published in March 2016.

3.1.2 Summary of Public Information Program Initiatives

Information on commercial analytical services available through the facility is promoted along with all of EAL's services. The SLOWPOKE page on SRC's website received 543 visits (excluding visits from within SRC) during the reporting period. Most visitors were from Canada and found the webpage via Google.

The SLOWPOKE-2 video (released in 2014) and blog post (published in 2015) continue to attract viewers through either the website or YouTube; the video had 150 viewers in 2019 (1100 views lifetime) and the blog had 25 viewers.

The decommissioning of the reactor generated more interest from the media than the facility has had in previous years, with a total of seven media inquiries pertaining to the reactor in 2019.

As a matter of policy SRC does not offer tours to the general public of any of its facilities. In alignment with policy, tours of the reactor are offered to government officials, clients, university students conducting relevant work and other interested stakeholders as long as business needs are not interrupted and safety/security criteria are met. SRC often offers tours of the reactor in conjunction with events at the University of Saskatchewan or relevant industry conferences being held in the Saskatoon area.

In 2019, a majority of the visitors to the facility were stakeholders in the decommissioning of the reactor. On April 30, an open house was held for SRC employees to commemorate the final day of operation. 42 employees attended the event.

3.2 Site-Specific

3.2.1 Nuclear Criticality Safety Program

The facility is not required to have a nuclear criticality safety program due to the inherent safety features of the reactor.

3.2.2 Financial Guarantee

The financial guarantee did not require any revisions during the reporting period.

3.3 Improvement Plans and Future Outlook

The facility ceased operations on April 30, 2019. Defueling took place in August, and the licence to decommission was granted December 6. Full decommissioning will be completed in 2020 with the expectation that a licence to abandon will be issued by the end of 2020.

3.4 Safety Performance Objectives for Following Year

Safety performance objectives are no lost time incidents, no significant doses to employees or members of the general public, and to keep releases to a minimum as per the ALARA principle. Safety considerations specific to decommissioning are described in the Detailed Decommissioning Plan.

4.0 CONCLUDING REMARKS

After 38 years the SRC SLOWPOKE-2 reactor's final day of operation was April 30, 2019. It was defueled in August and the facility is expected to be fully decommissioned in 2020. Through a combination of the inherent safety of the design of the reactor and dedication of SRC staff to safe, responsible operation, the reactor was operated for 20,651.5 hours over 4726 operating days with no safety or radiation related incident during its lifetime. Although the SRC SLOWPOKE is gone, its proven track record, along with that of the other SLOWPOKEs still operating or previously decommissioned, will stand out as a shining example of the safe, responsible and peaceful use of nuclear energy for Canada and the rest of the world.

5.0 CLOSURE

This, *Annual Compliance Report 2019*, has been prepared by the Saskatchewan Research Council, Saskatoon, Saskatchewan for the Canadian Nuclear Safety Commission.

Any use that a third party makes of this report, or any reliance on or decisions to be made based on it, are the responsibility of such parties. SRC accepts no responsibility for damages, if any, suffered by any third party as a result of decisions made or actions based on this report.

We trust this report meets your current requirements. Please do not hesitate to contact us with questions or comments.

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APPENDIX A
CALCULATIONS

A.1 ^{41}AR PRODUCTION RATE

^{41}Ar is produced by the irradiation of air in the transfer tubes. To determine the ^{41}Ar production rate, the volume of air (and hence the quantity of argon) in the irradiation tubes is calculated. The quantity of ^{41}Ar produced is then determined from the quantity of argon irradiated, the rate of neutron flux, and the nuclear cross-section of ^{40}Ar. It is assumed that only air in close proximity to the reactor core is irradiated.

The SRC SLOWPOKE-2 has seven small irradiation tubes and two large tubes. One small tube is outside the reactor container (site 11) so ^{41}Ar production in this tube is ignored. The two large tubes are outside the beryllium annulus and only receive ½ the neutron flux as the small tubes. Therefore, the volume of air in a single large tube is determined, but this air is treated as though it received the same flux as one of the small tubes:

$$\text{Volume}_{\text{big tube}} \text{ x 2 tubes x ½ flux of small tube} = \text{Volume}_{\text{big tube @ flux of small tube}}$$

The small tubes have a 20.42 mm ID with a 10.92 mm Return Air Line (RAL). A large tube has a 32.1 mm ID with a 14.12 mm ID RAL. It is assumed that the length of tube subject to irradiation is 220 mm (the same length as a fuel pin). The volume of air irradiated is therefore the sum of the volume from six small tubes and their associated RALs plus the volume of one large tube and its associated RAL as calculated below:

Large tube (32.1 mm ID)	πr^2 x 220 mm ➔	178.0 cm^3
Large RAL (14.12 mm ID)	πr^2 x 220 mm ➔	34.4 cm^3
Six small tubes (20.42 mm ID)	πr^2 x 220 mm ➔	6 x 72.0 cm^3
Six small RALs (10.92 mm ID)	πr^2 x 220 mm ➔	6 x 20.5 cm^3
Total volume		767.4 cm^3

Assuming a temperature of 273.15K and pressure of 100 kPa, the ideal gas law yields 22.7 L/mol so the quantity of air irradiated is:

$$0.7674 \text{ L} \div 22.7 \text{ L/mol} = 0.0338 \text{ mol of air}$$

The concentration of argon in air is 0.94% and the natural abundance of ^{40}Ar is 99.6%. The concentration of ^{40}Ar is therefore taken as 0.94% and the quantity of ^{40}Ar subject to irradiation is:

$$0.0338 \text{ mol air} \times \frac{0.0094 \text{mol } ^{40}\text{Ar}}{\text{mol air}} = 0.000318 \text{ mol } ^{40}\text{Ar}$$

The quantity of ^{41}Ar produced per flux hour of reactor operation is therefore:

$$0.000318 \text{ mol } ^{40}\text{Ar} \times \left(\frac{6.022 \times 10^{23} \text{ atoms}}{\text{mol}}\right) \times \left(3.6 \text{ x } 10^{14} \frac{\text{n}}{\text{cm}^2}\right) \times \left(0.65 \text{ x } 10^{-24} \text{ cm}^2\right) = 4.47 \text{ x } 10^{10} \text{ atoms}$$

where 0.65 x 10^{-24} cm^2 is the nuclear cross-section of ^{40}Ar, and 3.6 x 10^{14} n/cm^2 is the neutron flux in one flux-hour of operation (1 x 10^{11} n.cm^{-2}.s^{-1} x 3600s per hour). To convert the number of atoms of ^{41}Ar produced to a quantity in terms of Bq is as follows:

$$(4.47 \text{ x } 10^{10} \text{ atoms}) \div (6.022 \text{ x } 10^{23} \text{ atoms/mol}) \text{ x } (41\text{g/mol}) = 3.064 \text{ x } 10^{-12}\text{g}$$

$$(3.064 \text{ x } 10^{-12}\text{g}) \text{ x } (1.55 \text{ x } 10^{18}\text{Bq/g}) = 4.8 \text{ x } 10^{6}\text{Bq}$$

Thus, the quantity of ^{41}Ar produced is 4.8 MBq per flux-hour of operation. It is notable that releases of ^{41}Ar from reactor operations will be less than the quantity produced.

A.2 ESTIMATE OF RADIATION DOSE FROM RADIONUCLIDE RELEASES

A.2.1 Underlying Assumptions

A scenario to arrive at a conservative estimate of radiation dose to members of the public from gaseous radionuclides was developed. This scenario assumes that a person spends eight hours each day in a park located just east of the building that houses the reactor. It is assumed that all the gaseous radionuclides from a given period of reactor operation (say one day) are released immediately at the start of the reactor operation. These gaseous radionuclides mix gently, but quickly and homogeneously with the air and drift into the park area forming a cloud that covers the entire park area (~15,000 m^2). This cloud extends from the ground to 50 m in height. Thus, the cloud has a volume of 750,000 m^3. It is further assumed that the air is quite stagnant so that the cloud containing the gaseous radionuclides remains in the park for the entire 8-hour period.

This scenario provides a very conservative dose estimate for several reasons. The first of these is a large overestimate of the occupancy rate of the park. Secondly, the meteorological data given in the SRC operating manual[4] indicate that the climatic conditions at the facility would lead to much greater dilution and dispersion of the radionuclides released. In addition, the rate at which the radionuclides are released will be slower than assumed. The higher dilution and dispersion expected combined with the slower release rate mean that the actual concentrations of radionuclides in air near the reactor facility would be only a small fraction of the concentrations used to estimate dose.

During routine reactor operations, there are three different types of gaseous radionuclide releases: releases from the weekly reactor container headspace purge; releases from diffusion of gaseous radionuclides out of the transfer tubes; and releases from transfer operations. The types and quantities of radionuclides from each type of release are tabulated in Section 2.3.3.1. The radiation dose from each type of release is calculated separately. The estimated dose from each type of release are added together to obtain the total dose.

A.2.2 Estimate of Dose from Weekly Purge of Headspace Gases

The quantity of radionuclides released during the purge of the reactor container headspace during routine reactor maintenance is calculated from the information given in Table 6 illustrates that on average more than 95% of the radioactivity in the headspace gas is from ^{133}Xe. Although the quantity released each week varies somewhat, the upper and lower extremes are generally within an order of magnitude of each other. The weekly quantity released depends mainly on the extent of reactor usage in the previous week.

[4] Site Description and Operating Manual for the SLOWPOKE-2 Reactor, Revision 7, Revised by Dave Chorney, November, 2015

The average quantity of ^{133}Xe released each week is used to estimate the radiation dose from these releases. The average quantity of ^{133}Xe released each week is the average ^{133}Xe concentration (given in Table 6) multiplied by the volume of the headspace (108 L):

$$3780 \text{ Bq/L} \times 108 \text{ L} = 408{,}240 \text{ Bq } ^{133}\text{Xe}$$

Using the underlying assumptions, this quantity mixes evenly into a "cloud" of volume 750,000 m^3 and a person is exposed to this cloud for 8 hours (28,800 s). For releases of ^{133}Xe, Health Canada[5] suggests an external dose coefficient of 1.39×10^{-15} $Sv.s^{-1}Bq^{-1}m^3$. Therefore, the estimated radiation dose under our assumptions from one average weekly purge release is as follows:

$$\frac{408{,}240\ Bq}{750{,}000\ m^3} \times 28{,}800s\ \times 1.39x10^{-15}\ \frac{Sv\ m^3}{s\ Bq} = 2.18 \times 10^{-11} \text{ Sv}$$

Multiplying this by 18 weekly releases in 2019 produces an accumulated annual radiation dose of 3.92×10^{-10} Sv.

A.2.3 Estimate of Dose from Diffusion of ^{41}Ar

Environmental monitoring results presented in Table 8 indicate that ^{41}Ar is released through diffusion processes while the reactor is operating. The results presented in Table 9 indicate that the concentration of ^{41}Ar in the exhaust gas is 56.1 Bq/L while the reactor is operating at a flux of 5×10^{11} $ncm^{-2}s^{-1}$. The average exhaust gas flow rate from measurements taken from 2013 to 2018 is 3527 L/min (211,620 L/hour). Thus, the quantity of ^{41}Ar released in one hour is:

$$56.1 \text{ Bq/L} \times 211{,}620 \text{ L/hour} = 11{,}781{,}882 \text{ Bq/hour}$$

Since this is at a flux of 5×10^{11} $ncm^{-2}s^{-1}$, the quantity released per flux-hour of reactor operation is one-fifth of this value or 2,374,400 Bq.

Health Canada[5] suggests a dose coefficient for ^{41}Ar of 6.13×10^{-14} $Sv.s^{-1}Bq^{-1}m^3$. During the period of this report, the reactor was operated on 33 days for a total of 507.2 flux hours. This is an average of 15.37 flux hours each day the reactor is operated. On an average operating day, the quantity of ^{41}Ar released through diffusion processes is:

$$2{,}374{,}400 \text{ Bq/flux-hour} \times 15.37 \text{ flux-hours} = 36{,}494{,}528 \text{ Bq}$$

[5] Recommendations on Dose Coefficients for Assessing Doses from Accidental Radionuclide Releases to the Environment, Health Canada, 1999.

Using the underlying assumptions that this quantity of ^{41}Ar is all released at the same time, is evenly distributed within a volume of 750,000 m^3, and a person is exposed to this cloud for a full eight (8) hours (28,800 s) the dose estimate is:

$$\frac{36{,}494{,}528\ Bq}{750{,}000\ m^3} \times 28{,}800\ s \times \left(6.13x10^{-14}\frac{Sv\ m^3}{s\ Bq}\right) = 8.59 \times 10^{-8}\ \text{Sv}$$

Since the reactor was operated for 33 days, the estimated annual radiation dose would then be 33 times the value given above, or 2.83 x 10^{-6} Sv

A.2.4 Estimate of Dose from Diffusion of ^{133}Xe

Measurement results from the environmental monitoring program indicate that traces of ^{133}Xe are released through diffusion whether the reactor is operating or not. For the purpose of this estimate, only the period from January 1 to May 6 (126 days) is considered. The final operational shutdown of the reactor occurred on April 30, so no ^{133}Xe was produced after that date. Measurements of the head space gas show that after May 6 the amount of ^{133}Xe available for diffusion was negligible. For releases of ^{133}Xe, Health Canada[6] suggests an external dose coefficient of 1.39 x 10^{-15} Sv.s^{-1}Bq^{-1}m^3. An estimate of the total quantity of ^{133}Xe released via diffusion processes is given in Section 2.3.3.1 Other Air Releases as 962.3 MBq. This is an average of 7,637,300 Bq released per day. Using the underlying assumptions that this quantity of ^{133}Xe is all released at the same time, is evenly distributed within a volume of 750,000 m^3, and a person is exposed to this cloud for a full eight (8) hours (28,800 s) the daily dose estimate is:

$$\frac{7{,}637{,}300\ Bq}{750{,}000\ m^3} \times 28{,}800\ s \times \left(1.39x10^{-15}\frac{Sv\ m^3}{s\ Bq}\right) = 4.08 \times 10^{-10}\ \text{Sv}$$

The estimated radiation dose is then 126 times this value or 5.14 x 10^{-8} Sv, which is again many orders of magnitude lower than the 1 mSv permitted for occupational radiation exposure to a member of the general public under current CNSC regulations.

A.2.5 Estimate of Radiation Dose from Pneumatic Transfer Operations

Measurement results presented in Table 8 for sample 2013 - 2018-ROWT indicates the average concentration of ^{41}Ar in the exhaust gas during irradiation capsule transfer operations was 420 Bq/L (42 Bq/L per transfer x 10 transfers/sample). There was a total of 10 capsule transfer operations during a sampling period of two (2) minutes. The flow rate from the exhaust vent is 3527 L/min. The estimated maximum quantity of ^{41}Ar released per capsule transfer operation is:

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[6] Recommendations on Dose Coefficients for Assessing Doses from Accidental Radionuclide Releases to the Environment, Health Canada, 1999.

420 Bq/L x 3527 L/min x 2.00 min ÷ 10 transfer operations = 296,268 Bq

From Table 1, the total number of capsules irradiated during the previous year was 731. Each irradiation requires two transfer operations (in and out). The reactor was operated 33 days. The average number of transfer operations in a day was (731 x 2 ÷ 33) = 44.3. The average quantity of ^{41}Ar released from pneumatic transfers during a day of reactor operation is then:

44.3 x 296,268 = 13,124,700 Bq

Using the underlying assumptions that this quantity of ^{41}Ar is all released at the same time, is evenly 8distributed within a volume of 750,000 m^3, and a person is exposed to this cloud for a full eight (8) hours (28,800 s) the dose estimate is:

$$\frac{13{,}124{,}700}{750{,}000\ m3}\ x\ 28{,}800s\ x\ \left(6.13x10^{-14}\frac{Sv\ m3}{s\ Bq}\right) = 3.09 \times 10^{-8}\ Sv$$

Multiplying this value by the total number of days of reactor operation (33) yields an estimated annual dose of $1.02x10^{-6}$ Sv.

Measurements of sample ROWT from 2013 to 2018 indicate the average concentration of ^{133}Xe in the exhaust gas during irradiation capsule transfer operations was 2.1 Bq/L. There was a total of 10 capsule transfer operations during a sampling period of two (2) minutes. The flow rate from the exhaust vent is 3527 L/min. The estimated maximum quantity of ^{133}Xe released per capsule transfer operation is:

2.1 Bq/L x 3527 L/min x 2.00 min ÷ 10 transfer operations = 1481 Bq

From Table 1 the total number of capsules irradiated during the previous year was 731. Each irradiation requires two (2) transfer operations (in and out). The reactor was operated 33 days. The average number of transfer operations in a day was (731 x 2 ÷ 33) = 44.3. The average quantity of ^{133}Xe released from pneumatic transfers during a day of reactor operation is then:

44.3 x 1481 = 65,608 Bq

Using the underlying assumptions that this quantity of ^{133}Xe is all released at the same time, is evenly distributed within a volume of 750,000 m^3, and a person is exposed to this cloud for a full eight (8) hours (28,800 s) the dose estimate is:

$$\frac{65{,}608}{750{,}000\ m3}\ x\ 28{,}800s\ x\ \left(1.39x10^{-15}\frac{Sv\ m3}{s\ Bq}\right) = 3.50 \times 10^{-12}\ Sv$$

Multiplying this value by the total number of days of reactor operation (33) yields an estimated annual dose of 1.16×10^{-10} Sv.