SASKATCHEWAN RESEARCH COUNCIL

LIMITED REPORT

SRC SLOWPOKE-2 FACILITY
LICENSE # NPROL-19.00/2023

ANNUAL COMPLIANCE REPORT

for the period from January 1, 2016 to December 31, 2016

Prepared for:

Canadian Nuclear Safety Commission

Prepared by:

Saskatchewan Research Council
Environment Division

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Report Limitations and Use of Report

EXECUTIVE SUMMARY

The Saskatchewan Research Council (SRC) is managing the SLOWPOKE-2 Facility and it operated in a safe and reliable manner in 2016. No major problems were experienced; no modifications were made to the facility in 2016.

Usage of the facility down slightly from 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 with only a few isotopes generated for off-site use. Reactor time was made available to students from the University of Saskatchewan for classes and research projects.

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.

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 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 byproducts 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 will be 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 Saskatchewan Research Council (SRC) SLOWPOKE-2 reactor facility (the facility) operated in a routine and trouble-free manner during the period of January 1 to December 31, 2016 (the review period) as it has done since 1981. The reactor was operated an average of 2.3 days per week during the review period. The primary purpose of operations is for instrumental neutron activation analysis (INAA) and delayed neutron counting (DNC) on a commercial basis. Other purposes are isotope production and use of the reactor as a teaching tool in conjunction with the University of Saskatchewan.

The facility operates under License No. NPROL-19.00/2023, valid to June 30, 2023. Activities undertaken to ensure compliance include maintenance of all operational logs, performance of weekly 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 was one reportable incident at the facility in 2016. On July 20, during conductance of the auxiliary power check (part of routine weekly maintenance) the control rod became frozen in the fully out position, resulting in a neutron flux exceeding $1.4x10^{12}$ $n.cm^{-2}.s^{-1}$ for a period of approximately five minutes. Appendix A, paragraph 3 of NPROL-19.00/23 states “The licensee shall not operate the reactor at neutron flux levels exceeding 1.05×10^{12} $n.cm^{-2}.s^{-1}$, except that while increasing power under automatic control a peak power of no more than 1.4×10^{12} $n.cm^{-2}.s^{-1}$ may be permitted for a time of no more than one minute”.

The cause of the incident was that on Monday, July 18, 2016 both the main and auxiliary power systems were switched off while the reactor was fully shut down in order to perform maintenance on the water temperature chart recorder. Both power systems were switched off to prevent any chance of electrical shock to the persons performing the maintenance of the recorder. Upon completion of the maintenance the main power was switched back on, but the auxiliary power system was not. As a result of this error the auxiliary power system was non-operational at the time of the weekly system check. The reactor was safely shut down by restoring the main power and turning the automatic control switch to off.
An incident investigation was initiated, following protocol set out by SRC’s Occupational Health and Safety Program. Corrective actions arising from this investigation were:

1. the implementation of a ‘tag-out’ system which requires a “do not operate’ tag to be placed on the reactor console whenever the auxiliary power is disabled The tag-out system will also be used for other situations where work is being performed that may affect the safe operation of the reactor
2. Facility administrator to review the incident with all reactor operators and trainee operators

The corrective actions were completed on August 31, 2016 and September 7, 2016 respectively.
A detailed incident report was sent to CNSC September 9, 2016 (CNSC e-Doc 5080552). CNSC indicated they will follow up through compliance activities (e-Doc 5145444).

1.2 Facility Operation

1.2.1 Operational Items

The facility operated in a safe and reliable manner for the review 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. One auxiliary component - the temperature chart recorder – ceased to function in July 2016 and was replaced with a paperless digital recorder.

1.2.1.1 Modifications to the Facility

No modifications were made to the facility during the review period.

1.2.2 Audits/Inspections

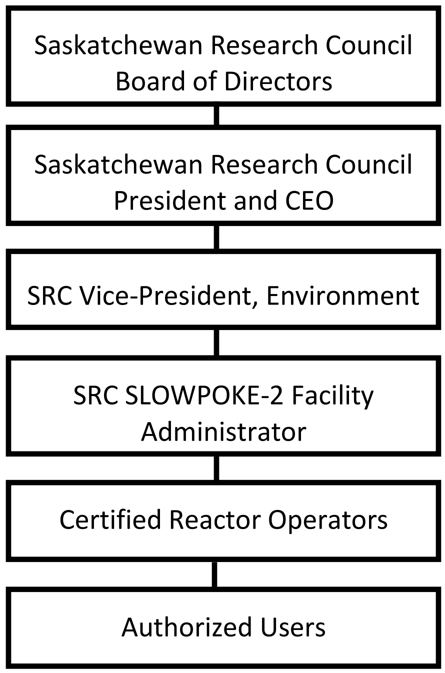
The facility had three safety inspections during the review period - a health and safety inspection conducted by SRC Occupational Health and Safety Committee (OHSC), a radiation safety inspection conducted by the SRC Radiation Safety Officer (RSO) and a fire inspection conducted by Saskatoon Fire and Protective Services. The OHSC committee inspection produced two action items pertaining to the facility. Both items were classified as minor and were addressed within a day of the inspection.The radiation safety inspection produced two action items related to the facility, both of which were addressed within a day of the inspection. The fire inspection report stated that the building was satisfactory at the time of the inspection.

Canadian Nuclear Safety Commission (CNSC) conducted a Type 2 Compliance Inspection August 4 and 5, 2016. The facility was found to be in compliance with the CNSC regulatory requirements for the areas and the activities inspected. No action items were identified.

Internal audits of the facility are conducted by EAL Quality Assurance on a three-year cycle. All action items arising from the 2013 to 2015 audit cycle were completed by October 2015. The current audit cycle is for the period of 2016 to 2018. To date, no action items have been identified.

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, 2016 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
- Dr. Laurier Schramm, 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 Consultant, Wellness & Safety Resources, University of Saskatchewan

1.2.4.2 Certified Reactor Operators

There were three certified reactor operators on staff during the review period:

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

Certifications for Mr. Chorney and Mr. Zimmer were renewed in 2016 and are valid until August 22, 2021.

Certification for Ms. Smith-Windsor is valid until May 16, 2018.

1.2.4.3 Authorized Users

Authorized Users as of December 31, 2016:

- Vicky Snook (Technologist, SRC)
- Karina Knorr (Technologist, SRC)
- Mohammad Yaquoob (Technologist, SRC)
- Kelcey Reding (Technologist, SRC)
- Cherie Shah (Technologist, SRC)
- Jamie Caldwell (Technologist, SRC)
- Kristy James (Technologist, SRC)
- Maegan Ponak (Technologist, SRC)
- Danna Polanska (Technologist, SRC)
- Michael Mudri (Technologist, SRC)
- Tejas Patel (Technologist, SRC)
- Devon Bechtel (Technologist, SRC)
- Ivan Charles (Technologist, SRC)
- Deanna Becker (Technologist, SRC)
- Jenna Ewanchuk (Technologist, SRC)
- Ennessa Hiestad (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 review period the SRC SLOWPOKE-2 reactor was operated for a total of 462.9 hours during 123 operating days. The total integrated flux for this period was 2268.8 x 10^{11} n.cm^{-2}. sec^{-1} hours. The reactor utilization down approximately 15% from the previous review period; the monthly totals are listed in Table 1. Since commissioning, the reactor has been operated a total of 19428.4 hours with a total integrated flux of 94478.1 x 10^{11} n.cm^{-2}.sec^{-1} hours.

1.3.1 Production

During the review period, 3155 capsule irradiations were performed. Approximately 59% (1874) were solids and liquids for delayed neutron counting (DNC) of uranium. The remainder (1281) 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; and student projects. 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.

Isotopes were produced for off-site use on two occasions during 2016. Both were for researchers at the University of Saskatchewan

Table 1: Summary of SRC SLOWPOKE-2 Operations for the period January 1, 2016 to December 31, 2016.

Month	Days Operated	Total Hours	Flux-Hours n/cm^2.sec-h x10^{11}	Capsules Irradiated
January	10	36.3	180.9	241
February	6	25.4	126.6	158
March	7	20.0	99.6	175
April	9	31.3	161.0	311
May	15	61.5	295.6	587
June	9	32.0	159.1	240
July	7	23.4	116.3	114
August	11	41.5	204.9	191
September	12	49.4	230.1	302
October	11	38.6	187.3	248
November	14	61.1	300.8	339
December	12	42.4	206.6	249
2016 TOTAL	**123**	**462.9**	**2268.8**	**3155**
Total to December 31, 2015	4304	18965.5	92209.3	231451
Total to December 31, 2016	**4427**	**19428.4**	**94478.1**	**234606**

1.3.2 Samples Which Could Create Unusual Hazards

There were no unusual samples irradiated during the review period.

1.3.3 Manual Operation

The reactor was operated under manual control on August 15, 2016 (Zimmer) and August 16, 2016 (Chorney), as a continuing training exercise for the operators certified for manual operation.

1.3.4 Remotely Attended Operation

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

1.3.5 Reactivity Adjustments

No reactivity adjustment was made during the review period. The shim-tray currently contains ten half-inch semi-circular beryllium shims. On December 6, 2016 the reactivity was measured at 2.59 mk.

1.3.6 Facility Modification

No modifications were made to the facility during the review period.

2.0 SAFETY AND CONTROL AREAS

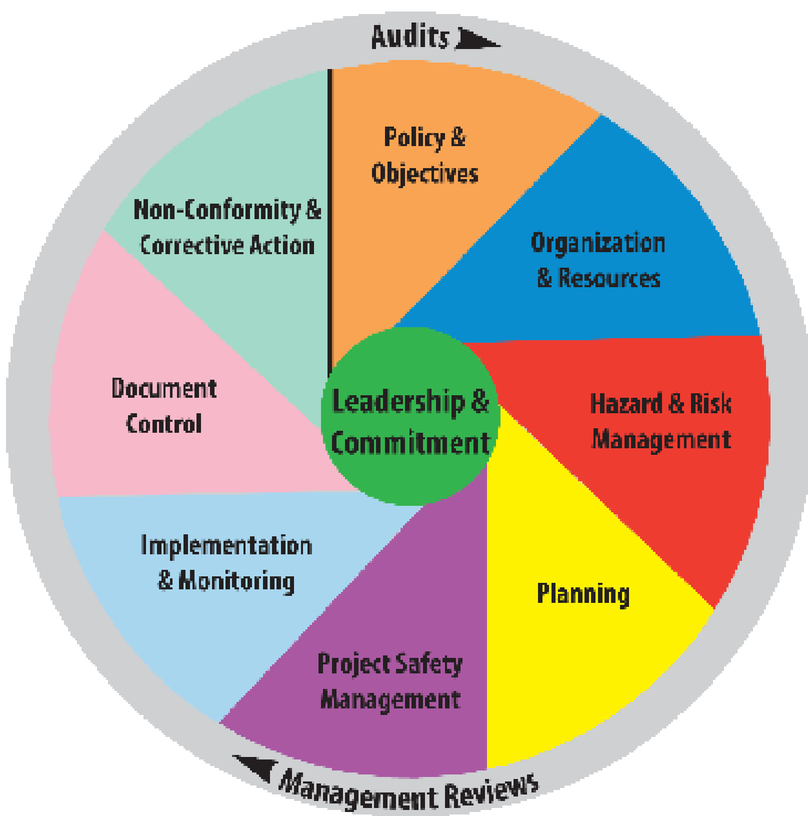
2.1 Management

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 below.

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 review period was positive. No lost time incidents were recorded. Activities at the SLOWPOKE facility did not result in any incident related to radiation safety.

The facility had three safety inspections during the review period - a health and safety inspection conducted by SRC Occupational Health and Safety Committee (OHSC), a radiation safety inspection conducted by the SRC Radiation Safety Officer (RSO) and a fire inspection conducted by Saskatoon Fire and Protective Services. The OHSC committee inspection produced two action items pertaining to the facility. Both items were classified as minor, and both were addressed within a day of the inspection.The radiation safety inspection produced two action items related to the facility, both of which were addressed within a day of the inspection. 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.

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 September 2015, and was found to conform to the requirements of ISO/IEC 17025. The accreditation is valid through August 5, 2018.

EAL Quality Assurance personnel conduct an audit of the facility on a three year cycle. All action items for the period of 2013 to 2015 were completed by October 2015. The current audit cycle is for the period of 2016 to 2018. To date, no action items have been identified.

A management review of the Laboratory Quality System is conducted in January of each year. At the most recent review (2016) 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 12 of the quality manual 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. To date, 1 certified operator and 3 authorized users have completed the refresher course..

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.

A candidate for becoming a certified reactor operator is trained as per *Saskatchewan Research Council Training Program for SLOWPOKE-2 Reactor Operators – Automatic Mode*.

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

There were three certified reactor operators on staff during the review period. Historically, three operators has been sufficient to ensure that at least one operator is on site every working day and at least one is reachable and within a two hour response time outside of regular working hours. Two of the current operators have held their certifications for over 25 years. As such, succession planning is underway and two candidates began training in 2016 to become certified operators. The 16 authorized users on staff ensure that there are always sufficient personnel on duty to perform sample irradiations as needed.

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.

[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

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.

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 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 review period that impacted the ability of the systems, structures or components (SSCs) to meet and maintain their design basis.

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 have been effective at verifying operation, and provide 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 is operated weekly. The system operated normally throughout the review period. The deionizer resin cartridge, in service since September, 2000, reached the end of its useful life in July 2016 and was replaced with a new one. 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. 10L of water was added August 24, 2016.

The pool water deionizer system circulates the water continuously. The 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 system operated normally during the review period. Routine maintenance included the replacement of the two in-line filters. Makeup water was added at an average rate of 3.16 L/day, which is consistent with historical rates.

The pH of the water can have a significant effect on corrosion rates. Thus it is prudent to monitor the pH of the reactor container and pool water on a regular basis. The pH measurements for the review 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	Date	Pool Water pH	Reactor Container Water pH
04-Jan-16	6.1	5.7	04-Jul-16	6.4	5.8
11-Jan-16	5.0	5.8	11-Jul-16	5.8	5.4
18-Jan-16	6.1	5.9	18-Jul-16	5.8	5.7
25-Jan-16	6.0	5.8	25-Jul-16	5.9	5.5
01-Feb-16	5.5	5.7	02-Aug-16	6.2	5.6
08-Feb-16	5.7	5.8	08-Aug-16	6.6	5.7
16-Feb-16	6.5	6.0	15-Aug-16	7.0	5.6
22-Feb-16	6.1	5.9	22-Aug-16	5.8	5.7
29-Feb-16	5.9	5.8	29-Aug-16	6.0	5.5
07-Mar-16	5.4	5.6	06-Sep-16	6.2	5.4
14-Mar-16	5.5	5.6	12-Sep-16	5.8	5.7
29-Mar-16	5.8	6.0	19-Sep-16	6.1	5.7
04-Apr-16	5.8	5.8	26-Sep-16	5.9	5.4
11-Apr-16	6.6	5.8	03-Oct-16	6.0	5.6
18-Apr-16	6.2	5.8	11-Oct-16	5.8	5.6
25-Apr-16	6.0	5.6	17-Oct-16	5.8	5.3
02-May-16	5.7	5.8	24-Oct-16	5.8	5.6
09-May-16	6.2	5.8	31-Oct-16	5.4	5.6
16-May-16	6.2	5.8	07-Nov-16	5.7	5.7
24-May-16	6.1	5.8	14-Nov-16	6.4	6.3
30-May-16	6.2	5.4	21-Nov-16	6.6	5.7
06-Jun-16	6.2	5.6	28-Nov-16	6.4	5.5
13-Jun-16	5.9	5.8	05-Dec-16	6.4	5.7
20-Jun-16	6.1	5.5	12-Dec-16	6.4	5.8
27-Jun-16	6.2	5.8	19-Dec-16	5.8	6.1

Headspace Gas Sampling System

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

Auxiliary Power System

The auxiliary power system operation is tested during weekly maintenance. In addition to the 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 eight hours, with auxiliary power reserves still adequate at the end of the test. During weekly maintenance, battery voltage is read and recorded in the routine maintenance log. Two power outages occurred during the review period. The reactor was not operating at the time of either outage.

Control Console

The control console continues to function normally. As part of the weekly maintenance, the flux and control point switches are exercised by rotating amongst the switch positions several times.

Alarm Systems

Radiation alarms are tested weekly. Security alarms are tested monthly. 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. All alarm tests are documented. Systems were functional, with one exception on September 7, 2016; the radiation and water level alarms were received at the security desk in the Galleria Building but not at the building maintenance alarm console at the energy centre (water level and radiation alarms are classified as building maintenance alarms and are normally received at both stations). Innovation Place maintenance was notified and the problem was corrected by September 8, 2016. During the one day period when the energy centre could not receive the alarms, the security desk in the Galleria could still receive the alarms so there was never a time when the alarms could not be received by at least one station.

In 2016, a decision was made at the the corporate level for SRC to assume control of its security monitoring, rather than to rely on the Innovation Place alarm monitoring services. On December 20, the SLOWPOKE facility security alarms were switched over to the new system. The testing schedule remains the same.

Shutdown Systems

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

Pool Overflow Sump

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

Irradiation Systems

During routine inspections, two short lengths of pneumatic transfer tubing were found to have developed cracks and were replaced.

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 will place 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 well below regulatory limits; in fact all measurements 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, 2016).

Worker Category (number of workers)	Minimum Total Effective Dose (mSv)	Maximum Total Effective Dose (mSv)	Mean Total Effective Dose (mSv)
Certified Operator (3)	<0.1	<0.1	<0.1
Authorized User (16)	<0.1	<0.1	<0.1

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 two detectable doses recorded within the past ten years, and these doses were less than two 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 is conducted three times per year at prescribed locations throughout the facility, as detailed in SOP SLO-130 "Contamination and Radiation Monitoring". 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 was conducted tri-annually 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, 0.11 $Bq/cm^{2,}$ was recorded at two locations in March and at a different two locations in July. These readings are only 2 times background (background averages about 50 cpm and did not exceed the action level. All four areas showed no detectable activity the next time they were monitored.. Results of monitoring during the review 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 exceedence 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 review period.

2.3.1.3 Facility Radiological Conditions

Routine radiation monitoring is conducted three times per year at prescribed locations throughout the facility, as detailed in SOP SLO-130 "Contamination and Radiation Monitoring".

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 and July 2016 was performed with the reactor not operating and the pool covers closed. Monitoring in December 2016 was performed with the reactor not operating and the pool covers open. 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 20% of the action level. The readings from the fixed monitor under the pool covers were <1% of the action level. The readings from the ceiling monitor were 0.02 mR/h (0.3 µSv/h), less than 1% of the action level. The readings from the reactor water deionizer monitor were 4.0 mR/h (40 µSv/h in March,) to 1.3 mR/h (13 µSv/h)in July and 6.0 mR/h (60 µSv) in December. 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 low reading in July is attributable to the fact that the resin cartridge had been replaced 10 days before the reading was taken so there were almost no long-lived isotopes in the resin. The results for the review period are typical for the level of reactor use at the time of monitoring and are less than 25% 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 2016 measurements are provided in Table 4 below.

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

Date	Reactor Container Water Gross Radioactivity (Bq/L)
4-Jan-16	49560
1-Feb-16	33004
14-Mar-16	64768
4-Apr-16	24849
2-May-16	62917
6-Jun-16	69238
4-Jul-16	39974
2-Aug-16	21369
6-Sep-16	53271
3-Oct-16	91260
7-Nov-16	98776
5-Dec-16	71365

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 was measured on August 22, 2016 and the pool water on September 1, 2016. A summary of the measurements are provided in Table 5 below.

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	<20	<1	I-135	<30	<1
Xe-133	64200	<0.2	Sr-85	<4	<0.2
Xe-133M	<140	<0.5	Sr-91	88	<0.4
Xe-135	802	<0.1	Y-88	<4	<0.1
Na-24	169	1	Y-91M	<20	<0.1
Be-7	<60	<2	Rh-106	<60	<1
Cr-51	<60	<1	Ru-103	67	<0.08
W-187	<20	<0.4	Cs-134	<5	<0.2
Mn-54	<6	<0.1	Cs-136	19	<0.2
Mo-90	9	<0.3	Cs-137	107	<0.2
Mo-99	125	<0.1	Ba-140	7310	<0.8
Tc-99M	13200	<0.2	La-140	3720	0.4
Fe-59	<10	<0.4	Ce-144	114	<0.7
Co-57	<7	<0.1	Cd-109	100	<0.1
Co-58	<7	<0.2	Ce-139	<5	<0.1
Co-60	<10	<0.2	Hg-203	<7	<0.1
Zn-65	<10	<0.2	Sn-113	<8	<0.1
Nb-94	<2	<0.2	K-40	<80	<2
Zr-95	48	<0.4	Sb-124	<6	<0.07
Nb-95	21	<0.2	Sb-125	<20	<0.5
Zr-97	12	<0.2	Se-75	<9	<0.2
Nb-97	28	<0.2	Eu-152	<30	<0.3
I-131	3240	<0.1	Ra-226	<120	<3
I-132	75	<0.7	U-235	<8	<1
I-133	1240	<0.2	Te-132	254	<0.1

2.3.1.5 Exceedences of Regulatory Limits or Action Levels

There were no exceedences of regulatory limits or action levels during the review period.

2.3.1.6 Radiation Protection Program Effectiveness

There were no detectable doses received by facility personnel, no personnel contamination events, and no exceedences of regulatory limits or action levels during the review 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. One certified operator ane three authorized users completed the refresher course in 2016.

2.3.1.8 Radiation Protection Program Performance

Specific numerical goals and targets (e.g., reduce exposures by 0.05 uSv) 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. As there are no new initiatives planned for the upcoming year that would result in changing the program's goals, the goals will remain the same.

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 (ceiling 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. Alarm trip levels and verification of receipt are recorded in the alarm testing log. The deionizer monitor is moveable and can be sent out for annual calibration. The monitor was calibrated in September 2016. The pool and area monitors are fixed in place. They cannot be sent out for calibration and there is no local service provider who will come on-site to calibrate the monitors. Calibration of the pool and area 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 20, 2016. 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 Log Book.

2.3.1.12 Summary of Inventory Control Measures

Sealed sources – primarily sealed, individual or mixed-gamma reference sources for the calibration of gamma spectrometers – and unsealed sources – calibrated aqueous reference solutions used for the calibration of, or tracers in, a variety of analytical procedures not related to the facility – are used and/or stored within the facility. These sources fall under Consolidated License 01784-5-19.6. Lists of these sources are provided in Table 6 and Table 7.

Table 6: Inventory of sealed sources located within the SLOWPOKE-2 Facility.

Manufacturer	Model	Serial Number	Nuclear Substance	Activity at June 20, 2016	Activity Units	Comments
Amersham	QCR1 Gamma Reference Source	1Q568	Am-241	378.9	kBq	
Amersham	QCR1 Gamma Reference Source	1R474	Ba-133	38.03	kBq	
Amersham	QCR1 Gamma Reference Source	1S610	Cs-137	204.0	kBq	
Amersham	QCR1 Gamma Reference Source	1U553	Co-60	3.898	kBq	
Amersham	QCR1 Gamma Reference Source	1X528	Na-22	0.0306	kBq	
Amersham	QCR1 Gamma Reference Source	1Y701	Y-88	0	kBq	Ref date Feb 1, 1981. No activity remaining today.
Amersham	QCR1 Gamma Reference Source	1V589	Mn-54	0	kBq	Ref date Feb 1, 1981. No activity remaining today.
Amersham	QCR1 Gamma Reference Source	1W722	Hg-203	0	kBq	Ref date Feb 1, 1981. No activity remaining today.
Amersham	QCR1 Gamma Reference Source	2T154	Co-57	0	kBq	Ref date Feb 1, 1981. No activity remaining today.
Amersham	AMR-33 Mixed Nuclide Source	9754RA	Cm-244	1.43	kBq	
Amersham	AMR-33 Mixed Nuclide Source	9754RA	Pu-239	5.543	kBq	
Amersham	AMR-33 Mixed Nuclide Source	9754RA	Am-241	5.243	kBq	
ICN	Bi-210 Source	none	Bi-210	0.290	kBq	

Manufacturer	Model	Serial Number	Nuclear Substance	Activity at June 20, 2016	Activity Units	Comments
Unknown	Reference Source	S-2370	Tc-99	648	kBq	
		S-2369	Th-230	816.3	kBq	
		S-2371	Cs-137	19.24	kBq	
Exploranium	none	none	Cs-137	12.80	kBq	
Exploranium	none	GRS-101s	Th-232	unknown		Limited History Available
Spectrum Techniques	none	none	Ba-133	16.7	kBq	
Unknown	none	7514	Am-241	0.230	kBq	
NRD	2U500	none	Po-210	0.0339	kBq	
NRD	2U500	none	Po-210	0.978	kBq	
NRD	2U500	none	Po-210	1.314	kBq	
NRD	2U500	none	Po-210	20.55	kBq	
North American Scientific	CAL-2600	118769	Cs-137	3.026	kBq	
Amersham	CDR8122	FS 909	Cs-137	25.26	kBq	
AEA Technology	QCRX1229 Mixed Nuclide Source	KF518	Am-241	2.814	kBq	CD-109, Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported.
			Cs-137	1.909	kBq	
			Co-60	0.444	kBq	
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1555-93-1	Pb-210	9.99	kBq	Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported
			Am-241	1.12	kBq	
			Cd-109	1.43	kBq	
			Cs-137	2.34	kBq	
			Co-60	1.73	kBq	

Manufacturer	Model	Serial Number	Nuclear Substance	Activity at June 20, 2016	Activity Units	Comments
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1577-64-1	Pb-210	9.95	kBq	Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported
			Am-241	1.14	kBq	
			Cd-109	1.57	kBq	
			Cs-137	2.24	kBq	
			Co-60	1.78	kBq	
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1577-64-2	Pb-210	10.17	kBq	Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported
			Am-241	1.16	kBq	
			Cd-109	1.60	kBq	
			Cs-137	2.29	kBq	
			Co-60	1.82	kBq	
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1743-20-1	Pb-210	10.02	kBq	Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported
			Am-241	10.08	kBq	
			Cd-109	4.69	kBq	
			Cs-137	2.42	kBq	
			Co-60	2.23	kBq	
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1743-20-2	Pb-210	10.31	kBq	Co-57, Ce-139, Hg-203, Sn-113, Sr-85 & Y-88 decayed to below exemption quantities so not reported
			Am-241	1.11	kBq	
			Cd-109	4.83	kBq	
			Cs-137	2.42	kBq	
			Co-60	2.32	kBq	

Manufacturer	Model	Serial Number	Nuclear Substance	Activity at June 20, 2016	Activity Units	Comments
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1850-93	Pb-210	11.33	kBq	
			Am-241	1.112	kBq	
			Cd-109	13.53	kBq	
			Co-57	0.3971	kBq	
			Ce-139	0.3599	kBq	
			Hg-203	0.2825	kBq	
			Sn-113	1.188	kBq	
			Sr-85	0.7585	kBq	
			Cs-137	2.349	kBq	
			Y-88	2.454	kBq	
			Co-60	2.835	kBq	
Eckert & Ziegler Isotope Products Labs	EG-ML Mixed Nuclide Calibration Source	1850-92-1	Pb-210	10.64	kBq	
			Am-241	1.120	kBq	
			Cd-109	13.75	kBq	
			Co-57	0.4036	kBq	
			Ce-139	0.3656	kBq	
			Hg-203	0.1524	kBq	
			Sn-113	1.207	kBq	
			Sr-85	0.7704	kBq	
			Cs-137	2.386	kBq	
			Y-88	2.493	kBq	
			Co-60	2.88	kBq	
Perkin Elmer	Tri-carb 2810TR	H441	Ba-133	581	kBq	

Table 7: Inventory of Unsealed Sources Located Within the SLOWPOKE-2 Facility.

UNSEALED SOURCE INVENTORY DATE: 2016/06/20	
Nuclear substance	**Total quantity in possession**
Ba-133	5.878 MBq
C-14	3.127 MBq
Co-60	15.89 kBq
Cs-137	205.33 kBq
H-3	127.05 kBq
Ni-63	1.984 MBq
Pb-210	78.48 kBq
Po-209	41.17 kBq
Ra-226	21.46 kBq
Sr-90	95.62 kBq
Th-230	12.17 kBq
Ra-228	260.8 Bq
Am-241	37.64kBq
Cd-109	337 Bq

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 2016 the following inspections and audits were carried out:

- OH&S inspections, conducted by the Occupational Health and Safety Committee
- Fire inspection by Saskatoon Fire and Protective Services
- Radiation Safety Inspection conducted by SRC RSO
- Monthly laboratory walk-throughs.

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 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 every year during May.

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 2016 SRC OH&S program continued to improve with specific initiatives as listed below:

- Launching of a system of reporting of unsafe acts and conditions in the form of an app made available to all employees on smartphones, IPADS or computers.
- A new initiative of "Rotating term position in Safety Services" was started in which a selected R&T employee will work for a term position to learn more about OH&S standards.
- Two mock exercises were arranged to test response to medical injury emergencies.
- A newly updated generic "Spill Response Training" course was made available to employees who work with chemicals.

2.3.2.4 Discussion of Hazardous Occurrences

There were no hazardous occurrences related to the facility during the review 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 during the weekly purge. 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. The results of monitoring measurements taken during the review period are summarized in Table 8.

Coinciding with the beginning of the 2012/2013 fiscal year, a new gamma spectroscopy software package was deployed. This software package includes a greatly improved method of determining the counting efficiency of the headspace sample in the 10 L aluminum Marinelli container. Through the use of this software it was determined that results prior to 2012/2013 have been biased low by a factor of six . When compared to previously reported data, it may appear that releases from 2012/2013 forward are greatly increased from previous years. However, to make a true comparison the previous years data must be corrected by a factor of six . Historical data for the years prior to 2012/2013 contained in this report has been corrected by this factor. Even with the discovery and subsequent correction of the low bias, the quantities of radioactive gaseous releases remain well below regulatory and dose limits.

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 8: 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)
04-Jan-16	<0.07	35	3200	18	<0.07	<0.08	<0.4	<0.4
11-Jan-16	<0.2	41	5860	121	211	<0.5	<0.2	1
18-Jan-16	<0.05	39	8860	162	37	<0.2	<0.4	<0.7
25-Jan-16	<0.08	57	17400	299	105	<0.4	<0.2	<0.9
01-Feb-16	<0.1	33	5720	55	<0.2	<0.2	<0.2	<0.5
08-Feb-16	<0.1	64	10600	204	180	<0.3	<0.4	<1
18-Feb-16	<0.2	34	6750	100	36	<0.3	<0.3	<0.9
22-Feb-16	<0.1	37	7740	133	268	<0.5	<0.4	<2
29-Feb-16	<0.08	24	3520	39	1	<0.1	<0.2	<0.4
07-Mar-16	<0.08	37	6810	132	147	<0.4	<0.2	<0.6
14-Mar-16	<0.06	28	5650	114	261	<0.5	<0.2	<1
29-Mar-16	<0.09	24	2180	21	0.3	<0.1	0.3	<0.2
04-Apr-16	<0.2	25	2720	44	10	<0.2	<0.2	<0.4
11-Apr-16	<0.2	20	4210	82	39	<0.2	<0.2	<0.8
18-Apr-16	<0.1	20	5090	95	50	<0.2	<0.2	<0.7
25-Apr-16	<0.1	33	10900	248	298	<0.6	<0.4	<1
02-May-16	<0.2	23	7560	104	37	<0.2	<0.2	<0.5
09-May-16	<0.06	43	13900	257	466	<0.6	<0.5	<2
16-May-16	<0.1	25	14800	308	385	<0.5	<0.6	<1
24-May-16	<0.1	31	12800	223	44	<0.2	<0.3	<0.8
30-May-16	<0.09	56	15400	218	276	<0.6	<0.3	<2
06-Jun-16	<0.1	51	9370	112	44	<0.3	<0.4	<0.7
13-Jun-16	<0.1	54	6050	53	8	<0.2	<0.2	<0.4
20-Jun-16	<0.2	40	7300	166	274	<0.5	<0.4	2
27-Jun-16	<0.1	53	9840	195	263	<0.5	<0.4	<0.8
04-Jul-16	<0.2	42	5390	55	5	<0.1	<0.2	<0.5
11-Jul-16	<0.3	51	8160	171	239	<0.4	<0.4	<1
18-Jul-16	<0.2	156	23200	373	44	<0.4	<0.4	<0.7
25-Jul-16	<0.06	32	4060	40	5	<0.2	<0.2	<0.6
02-Aug-16	<0.06	21	1720	13	0.2	<0.1	<0.2	<0.4
08-Aug-16	<0.2	34	5350	132	217	<0.4	<0.2	<1
15-Aug-16	<0.1	24	6300	109	18	<0.2	<0.5	<0.6
22-Aug-16	<0.3	29	10200	198	55	<0.2	<0.2	<1
29-Aug-16	<0.08	26	5030	50	1	<0.2	<0.4	<0.2
06-Sep-16	<0.1	25	4970	79	10	<0.2	<0.4	<0.4
12-Sep-16	<0.08	30	4200	60	185	<0.4<	<0.6	<1
19-Sep-16	<0.3	33	9400	199	116	<0.2	<0.7	<1
26-Sep-16	<0.1	27	7490	132	89	<0.3	<0.3	<0.5

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)**
03-Oct-16	<0.3	35	10300	212	285	<0.5	<0.6	<1
11-Oct-16	<0.3	32	7650	105	18	<0.2	<0.4	<0.6
17-Oct-16	<0.1	36	6420	72	10	<0.1	<0.4	<0.6
24-Oct-16	<0.2	28	5440	101	139	<0.4	<0.6	<1
31-Oct-16	<0.06	38	9490	182	232	<0.5	<0.5	<0.9
07-Nov-16	<0.06	56	14200	258	203	<0.5	<0.5	<0.8
14-Nov-16	<0.2	36	7550	117	40	<0.3	<0.4	0.7
21-Nov-16	<0.2	15	10600	217	273	<0.5	<0.4	<2
28-Nov-16	<0.05	41	9060	135	130	<0.3	<0.4	<1
05-Dec-16	<0.05	51	8680	146	354	1	<0.2	<1
12-Dec-16	<0.06	48	10800	217	412	0.8	<0.2	<2
19-Dec-16	<0.05	46	10800	180	199	<0.5	<0.3	<1
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			
2011-12 Avg.†		40	6420	114	149			
2010-11 Avg		87	16400	319	368			

*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.
† Data for 2010-11 and 2011-12 has been corrected from previous reports to take into account the more accurate way of calculating activities that was introduced in 2012-13.

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 the isotopes 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 weekly 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. As with the analysis of the headspace gas, the introduction of the new gamma spec software in April, 2012 revealed a low bias by a factor of six in the calculation of radionuclides in the exhaust gases in previous years. Data from 2012-13 forward presented in this report has been processed using the new software, and data from previous years has been adjusted by a factor of six for comparison purposes. Monitoring results for 2016 are given in Table 9 and a description of each sample is given in Table 10.

Table 9: Environmental monitoring results.

Sample Identification	Sampling time (minutes)	Measured flow rate (ft^3/min)	^{41}Ar (Bq/L)	^{133}Xe (Bq/L)	^{133M}Xe (Bq/L)	^{135}Xe (Bq/L)
2016-RNO	2.50	136	<0.1	2.3	<0.2	0.2
2016-ROWT	2.00	129	356	1.8	<1	0.3
2016-RONT	2.00	130	67	1.8	<0.8	<0.1
Average	2.17	131.7 (3729 L/min)	--	--	--	--

Table 10: Description of environmental monitoring samples.

Sample Identification	Description
2016-RNO	This sample was taken while the reactor was not operating. The reactor was shut down for one day before sampling. This sample is intended to provide information on any releases that may occur from diffusion when the reactor is not operating.
2016-ROWT	This sample was taken with the reactor operating at half power (5×10^{11} $n.cm^{-2}.s^{-1}$). Sampling was coordinated with other staff so that a total of 10 pneumatic transfer operations (five transfers in, five transfers out) were performed while the sample was collected. Site 1 was used for the transfer operations. This sample is intended to provide information on releases that may occur during transfer operations.
2016-RONT	This sample was taken with the reactor operating at half power (5×10^{11} $n.cm^{-2}.s^{-1}$). There were no transfer operations during sampling and the sample was taken about 70 minutes after the last transfer operation. The reactor was in operation for about three 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 sample 2016-ROWT indicate an average of 35.6 Bq/L of ^{41}Ar (356 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, including this year's data is presented in Table 11. Data prior to 2012 has been adjusted by a factor of six from previously reported values to take into account the more accurate way of calculating activities that was introduced in 2012.

Table 11: 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)
2009	24.6	0.33	<0.004
2010	1.4	0.66	<0.003
2011	13.8	0.20	<0.01
2012	36.9	0.14	0.03
2013	26.7	0.15	<0.02
2014	53.9	0.23	<0.10
2015	59.3	0.26	<0.20
2016	35.6	0.18	<0.1
Estimated concentration of radionuclides released due to diffusion processes while the reactor is operating (RONT)	**41Ar (Bq/L)**	**133Xe (Bq/L)**	**135Xe (Bq/L)**
2009	7.2	1.00	<0.02
2010	0.9	1.60	<0.02
2011	80.4	<0.50	0.54
2012	56.0	0.27	0.08
2013	60.0	2.00	<0.08
2014	3.7	1.20	<0.10
2015	40.0	0.80	0.20
2016	67	1.8	<0.1
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)**
2009	<0.04	4.56	<0.003
2010	<0.006	0.30	<0.005
2011	<0.10	2.20	0.12
2012	<0.08	1.60	<0.05
2013	<0.40	2.20	<0.05
2014	<0.40	<2.00	<0.08
2015	<0.10	1.20	<0.06
2016	<0.1	2.3	0.2

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

[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.

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 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 12.

Table 12: Estimated quantity of gaseous radionuclides released from transfer operations.

Month (2016)	Capsules Irradiated	Estimated Quantity Released		
		^{41}Ar (MBq)	^{133}Xe (MBq)	^{135}Xe (MBq)
January	241	128.0	0.6	<0.04
February	158	83.9	0.4	<0.02
March	175	92.9	0.5	<0.03
April	311	165.1	0.8	<0.05
May	587	311.7	1.6	<0.09
June	240	127.4	0.6	<0.04
July	114	60.5	0.3	<0.02
August	191	101.4	0.5	<0.03
September	302	160.4	0.8	<0.05
October	248	131.7	0.7	<0.04
November	339	180.0	0.9	<0.05
December	249	132.2	0.7	<0.04
TOTAL	**3155**	**1675.3**	**8.5**	**<0.47**

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 period of this report indicated an average flow of 3729 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, from Table 9 and Table 10, the observed concentration of ^{41}Ar in the exhaust gas during sample transfer operations was 356 Bq/L.There were 10 transfers over a period of 2.00 minutes so the estimated maximum quantity released per transfer is:

   356Bq/L x 3729 L/min x 2.00 min ÷ 10 transfer operations = 265,504 Bq/transfer

   In January 2016 there were 241 irradiations and the maximum estimated amount of ^{41}Ar released from the related transfer operations is:

   265,504Bq/transfer x 2 transfers/irradiation x 241 irradiations = 128.0MBq

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. Lists a total of 2268.8 flux-hours of reactor operation during the year. From these values the total quantity of ^{41}Ar produced during the year is calculated as 10,890 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
- Mechanical devices which limit the number of transfer tubes that can be connected to the exhaust system at any one time.

Sample 2016-RONT was 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[4] to be 1,789,920 Bq per flux hour of reactor operation. The total number of flux hours during the year was 2268.8. Therefore the estimated quantity of ^{41}Ar released from diffusion processes is 4061 MBq. ^{41}Ar is also released during capsule transfer operations. The estimated quantity of ^{41}Ar released from transfer operations is listed in

as 1675.3 MBq. An estimate of the total quantity of ^{41}Ar released during the year is the sum of these two quantities or 5736 MBq. This total is approximately 53% of the estimated quantity of ^{41}Ar produced.

Sample 2016-RNO was 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 2016-RNO. This sample and similar samples from previous years indicate 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 2016-RONT indicate that the concentration of ^{133}Xe due to diffusion in the exhaust gases when the reactor is operating is 1.8 Bq/L. During the review period, the reactor was operated for a total of 462.9 hours. An estimate of the quantity of ^{133}Xe released from diffusion processes while the reactor is operating is then:

$$1.8 \text{ Bq/L} \times 3729 \text{ L/min} \times 60 \text{ min/hr} \times 462.9 \text{ hr} = 186.4 \text{ MBq}$$

Sample 2016-RNO indicates that the concentration of ^{133}Xe due to diffusion in the exhaust gases when the reactor is not operating is 2.3 Bq/L. The concentration of ^{133}Xe in exhaust gases is expected to be highest

[3] Calculations are given in Appendix A.

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 13.

Table 13: 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)*
0.9 (in year 2002)	3.6
1 (2016)	2.3
1 (2015)	1.2
1 (2010)	0.3
1 (2011)	2.2
1 (2014)	<2
2 (2004)	0.4
3 (2007)	2.3
4 (2003)	0.3
4 (2009)	4.6
4 (2013)	2.2
5 (2005)	0.4
5 (2006)	0.7
5 (2008)	1.7
5 (2012)	1.6
Average	**1.7**

*Historical data has been corrected from previous reports to take into account the more accurate way of calculating gaseous concentrations that was introduced in 2012.

For simplicity, the average of these values (1.7 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 review period, the reactor was not operated for 8297.1 hours (8760 hours in a year minus the 462.9 hours of reactor operation). The estimated quantity of ^{133}Xe released while the reactor was not operating is:

$$1.7 \text{ Bq/L} \times 3729 \text{ L/min} \times 60 \text{ min/hr} \times 8297.1 \text{ hr} = 3155.9 \text{ MBq}$$

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

(186.4 MBq) and the estimated amount released due to diffusion when the reactor is not operating (3155.9 MBq) which equals 3342.3 MBq.

Water Releases

There were no releases of reactor container water at the facility during the review 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 review 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 14. The total estimated maximum radiation dose from all the various types of reactor releases is 1.37×10^{-5} Sv which is less than 2% of the annual occupational radiation exposure limit for a member of the general public.

Table 14: 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	2.46×10^{-9} Sv
Diffusion of ^{41}Ar while reactor is operating	9.53×10^{-6} Sv
Releases of ^{41}Ar from pneumatic transfer operations	3.95×10^{-6} Sv
Diffusion of ^{133}Xe	1.78×10^{-7} Sv
Releases of ^{133}Xe from pneumatic transfer operations	4.52×10^{-10} Sv
Estimated Maximum Dose from all Releases	**1.37×10^{-5} Sv**

2.3.3.3 Exceedences of Regulatory Limits or Action Levels

There were no exceedences of regulatory limits or action levels at the facility during the review 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 that is 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 15 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 Safety Services and is detailed in the SRC Emergency Response Plan, OHS-STD-02, a copy of which was submitted to CNSC as part of the fire protection program (letter from D. Chorney to I. Erdebil, January 17, 2012, Subject: Fire Protection Program for SRC SLOWPOKE Facility).

2.3.4.1 Review of Emergency Preparedness Program Activities

Evacuation drills are conducted on a quarterly basis. Drills are coordinated between the EAL emergency response team, Innovation Place maintenance and alarm monitoring station, and Saskatoon Fire and Protective Services. Drills are evaluated and corrective actions are implemented as necessary.

In the event of any emergency requiring evacuation, the fire alarm is activated as the signal to evacuate, which also alerts the alarm monitoring station and emergency response personnel.

On August 2, members of the Saskatoon Fire Protective Services were provided with a tour to familiarize them with the facility in the event that they ever have to respond to an emergency situation.

Fire extinguisher training is offered annually by SRC Safety Services. EAL ensures that several staff receive the training each year.

Shelter in place drills are conducted every two years. The urgency of shelter in place is not as acute as an evacuation, so the drills are not conducted with the same frequency. The purpose of shelter in place is to provide personnel with a safe area of the building in which to gather in the event of an external hazard such as a tornado. Shelter in place is announced via the building intercom and emergency personnel is not summoned unless necessary.

Management review of the emergency response plan is done as a part of the overall review of the SRC OH&S Program. The current version of the emergency response plan, OHS-STD-02 was issued in April 2016.

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 quarterly 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. During the review period, 1682 irradiated capsules whose radiation levels had decayed to background were disposed of.

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 May 2016.

2.3.7 Safeguards and Non-Proliferation

Physical Inventory Taking (PIT) was conducted October 31, 2016. The facility was not selected by IAEA for physical inventory verification (PIV) in 2016. 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

The only nuclear substances transported off-site were Isotopes produced for researchers at the University of Saskatchewan. The isotopes were packaged and transported by personnel from the University Waste Management Facility.

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. This version contained revisions made in response to CNSC's review of revision 8. Revision 9 was submitted to CNSC for review in March. An informal response was received from CNSC within days. The program was discussed further during CNSC's Type II Compliance Inspection August 5.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 118 visits (excluding visits from within SRC) during the review 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 163 viewers and the blog had 77 viewers in 2016.

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.

On August 2, members of the Saskatoon Fire Protective Service were given a tour to familiarize them with the facility in the event that they ever have to respond to an emergency situation.

On December 3, an open-house and tour was given to students of the University of Saskatchewan department of Physics and Engineering Physics, plus a few interested stakeholders.
SRC also provided nine other tours of the reactor during the review period.

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.

There was one reportable incident during the review period. This incident is described in section 1.1 of this document.

3.2.2 Financial Guarantee

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

3.3 Improvement Plans and Future Outlook

Currently, there are no plans for any improvements or changes for which the Commission's approval may be requested during the next year.

3.4 Safety Performance Objectives for Following Year

Facility operations, equipment, procedures, production capacity and organization are expected to remain unchanged for the upcoming 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.

4.0 CONCLUDING REMARKS

The SLOWPOKE-2 reactor is, as the acronym implies, a Safe, LOW Power reactor. Through the implementation of appropriate policies, programs, and procedures discussed in this report, the facility is operated and maintained in a manner that produces no risk to the health and well-being of staff, the general public or the environment.

5.0 CLOSURE

This, *Annual Compliance Report 2016*, 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 40Ar, and 3.6 x 1014 n/cm^2 is the neutron flux in one flux-hour of operation (1 x 1011 n.cm-2s-1 x 3600s per hour). To convert the number of atoms of 41Ar 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 a variety of 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 weekly reactor maintenance is calculated from the information given in Table 8. Table 8 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

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

the previous week. 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 8) multiplied by the volume of the headspace (108 L):

$$8210 \text{ Bq/L} \times 108 \text{ L} = 886{,}680 \text{ Bq } ^{133}\text{Xe}$$

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 xenon-133, 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:

$$\frac{886{,}680\ \text{Bq}}{750{,}000\ \text{m}^3} \times 28{,}800\text{s} \times \left(1.39 \times 10^{-15}\ \frac{\text{Sv m}^3}{\text{s Bq}}\right) = 4.73 \times 10^{-11} \text{ Sv}$$

Multiplying this by a maximum of 52 weekly releases in a year produces an accumulated annual radiation dose of 2.46×10^{-9} Sv.

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

Environmental monitoring results presented in Table 9.

Table 10 and Table 11 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 67 Bq/L while the reactor is operating at a flux of
5×10^{11} $ncm^{-2}s^{-1}$. From Table 9 the exhaust gas flow rate is 3729 L/min (223,740 L/hour). Thus, the quantity of ^{41}Ar released in one hour is:

$$40 \text{ Bq/L} \times 223{,}740 \text{ L/hour} = 8{,}949{,}600 \text{ Bq/hour}$$

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 1,789,920 Bq.

Health Canada 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 123 days for a total of 2,262.8 flux hours. This is an average of 18.40 flux hours each day the reactor is operated. On an average operating day, the quantity of ^{41}Ar released through diffusion processes is:

$$1{,}789{,}920 \text{ Bq/flux-hour} \times 18.40 \text{ flux-hours} = 32{,}934{,}500 \text{ Bq}$$

[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:

$$\frac{32{,}934{,}500\ \mathrm{Bq}}{750{,}000\ \mathrm{m}^3} \times 28{,}800\mathrm{s} \times \left(6.13 \times 10^{-14}\ \frac{\mathrm{Sv\ m}^3}{\mathrm{s\ Bq}}\right) = 7.75 \times 10^{-8}\ \mathrm{Sv}$$

Since the reactor was operated for 123 days, the estimated annual radiation dose would then be 123 times the value given above, or $^{-6}9.53 \times 10^{-6}$ 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 releases of ^{133}Xe, Health Canada[6] suggests an external dose coefficient of 1.39×10^{-15} Sv.s^{-1}Bq^{-1}m^3. An estimate of the total quantity of ^{133}Xe released during the year via diffusion processes is given in Section 2.3.3.1 **Other Air Releases** as 3342.3 MBq. This is an average of 9.13 MBq 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:

$$\frac{9{,}130{,}000\ \mathrm{Bq}}{750{,}000\ \mathrm{m}^3} \times 28{,}800\mathrm{s} \times \left(1.39 \times 10^{-15}\ \frac{\mathrm{Sv\ m}^3}{\mathrm{s\ Bq}}\right) = 4.87 \times 10^{-10}\ \mathrm{Sv}$$

The estimated annual radiation dose is then 366 times this value or 1.78×10^{-7} 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 9 for sample 2013-ROWT indicates the concentration of ^{41}Ar in the exhaust gas during irradiation capsule transfer operations was 356 Bq/L. There were a total of 10 capsule transfer operations during a sampling period of two (2) minutes. The flow rate from the exhaust vent is 3729 L/min. The estimated maximum quantity of ^{41}Ar released per capsule transfer operation is:

356 Bq/L x 3729 L/min x 2.00 min ÷ 10 transfer operations = 265,504 Bq

From Table 1 the total number of capsules irradiated during the previous year was 3155. Each irradiation requires two transfer operations (in and out). The reactor was operated 123 days. The average number of

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

transfer operations in a day was (3155 x 2 ÷ 123) = 51.3. The average quantity of ^{41}Ar released from pneumatic transfers during a day of reactor operation is then:

51.3 x 265,504 = 13,620,355 Bq

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:

$$\frac{13{,}620{,}355\ \mathrm{Bq}}{750{,}000\ \mathrm{m}^3} \times 28{,}800\mathrm{s} \times \left(6.13 \times 10^{-14}\ \frac{\mathrm{Sv\ m}^3}{\mathrm{s\ Bq}}\right) = 3.21 \times 10^{-8}\ \mathrm{Sv}$$

Multiplying this value by the total number of days of reactor operation (123) yields an estimated annual dose of 3.95×10^{-6} Sv.

Measurement results presented in Table 9 for sample 2016-ROWT indicate the concentration of ^{133}Xe in the exhaust gas during irradiation capsule transfer operations was 1.8 Bq/L. There were a total of 10 capsule transfer operations during a sampling period of two (2) minutes. The flow rate from the exhaust vent is 3729 L/min. The estimated maximum quantity of ^{133}Xe released per capsule transfer operation is:

1.8 Bq/L x 3729 L/min x 2.00 min ÷ 10 transfer operations = 1342 Bq

From Table 1 the total number of capsules irradiated during the previous year was 3155. Each irradiation requires two (2) transfer operations (in and out). The reactor was operated 123 days. The average number of transfer operations in a day was (3155 x 2 ÷ 123) = 51.3. The average quantity of ^{133}Xe released from pneumatic transfers during a day of reactor operation is then:

51.3 x 1342 = 68845 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:

$$\frac{68845\ \mathrm{Bq}}{750{,}000\ \mathrm{m}^3} \times 28{,}800\mathrm{s} \times \left(1.39 \times 10^{-15}\ \frac{\mathrm{Sv\ m}^3}{\mathrm{s\ Bq}}\right) = 3.67 \times 10^{-12}\ \mathrm{Sv}$$

Multiplying this value by the total number of days of reactor operation (123) yields an estimated annual dose of 4.52×10^{-10} Sv.