Title: WR-1 Reactor Radiological Characterization Summary and Radionuclide Inventory Estimates			Date: 2020/10/28
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1. INTRODUCTION

1.1 Scope and Purpose

This technical document presents a summary of existing characterization information on the Whiteshell Reactor 1 (WR-1) research reactor that may be of relevance in assessing the decommissioning strategy of in-situ decommissioning (ISD).

Characterization reports and data evaluations of radionuclide inventories are provided for the following reactor systems:

- Reactor vessel
- Biological shield
- Outside core reactor systems and rooms

Additionally, radiation dose rate hazards have been evaluated for the calandria and fuel channels, and a summary of reactor rooms' workplace radiological hazards are provided.

A comprehensive characterization campaign was performed during the calendar years of 2017 and 2018. The general characterization objectives were to address data gaps and provide quantitative, unbiased estimates of residual radionuclide content remaining within the WR-1 systems [1]. The study results provided necessary information to either validate existing inventory estimates as bounding case scenarios or otherwise refine the inventories that will be used as inputs to the ISD environmental and safety assessments as part of the decommissioning safety assessment report.

Specific study objectives included:

- Establishing whether previous investigations and/or modelling on reactor core components, biological and thermal shields, and reactor systems provided defensible radionuclide inventory upper bounds for the safety assessment or future computer-based radioactivity dose calculation inputs.
- Verifying the presence or absence of radionuclides of concern (ROCs) and confidently establishing whether the assumed radioactivity releases from fuel failures were conservative.
    - Confirming process knowledge regarding specific fission products and actinides that are present in the primary heat transport (PHT) or fuel handling systems due to previous operational fuel failures.
    - Confirming that the relative activity ratios between fission products and actinides support historical radionuclide inventories.
    - Determining the degree to which activated corrosion products (e.g. ^{55}Fe, ^{60}Co), ^{3}H, and ^{14}C have impacted systems.
- Collecting characterization field data for future applications in:
    - Assessing the data for a correlation between system exterior gamma dose rate measurements and system internal gamma-emitting radionuclide activity concentrations per unit area.

- Assessing the data for potential surrogate ratios between the gamma-emitting and hard-to-detect radionuclides to allow for inference calculations from direct gamma radiation field measurements.
- Assessing the data to establish correlations between the hard-to-detect radionuclides and ^{137}Cs or other gamma-emitters that would enable post-characterization refining of the inventory estimates based on system external gamma dose rate measurements.

A summary of the characterization design approach is discussed in Section 6.1. The results of this recent characterization are discussed within the specific reactor system subsections together with comparison to the previous inventory estimates as provided in the 2016/08/22 version of this report.

1.2 Background

WR-1 was placed in service in 1965 to demonstrate the organic-cooled reactor concept using heavy water (D_2O) as the moderator. Using oil as the primary coolant allowed the primary heat transport (PHT) system to operate at lower pressures and correspondingly higher temperatures than a similarly constructed light water PHT system. The reactor was used to perform research on the development and testing of experimental reactor fuel failure modes and detection systems, coolant materials, fuel channel materials, and fuel cladding materials.

The reactor was permanently shut down in 1985 and then defueled. Decommissioning was to be performed in two phases [1][2]. The first phase started in 1989 and was completed in 1995[1] placing the reactor in a shutdown with surveillance operational state [3]. Phase 2 was to be performed following a 40-year deferment period and involved the removal of the remaining reactor systems as well as final decommissioning and remediation for unrestricted or restricted release [1][2][3].

The concept of in-situ decommissioning (ISD) has recently been proposed as a new strategy for the Phase 2 Decommissioning of WR-1 [4]. This strategy involves permanently encapsulating the below-grade reactor systems, components, structure, and associated radiological and non-radiological hazards with a specially engineered grout. Under this approach the above-grade structure will be removed and an engineered cover will be constructed over the below-grade structure. This will significantly reduce the timeline for WR-1 decommissioning[2], worker and public risks[3,4], and costs to the Canadian public and government while providing a sound and technically-proven means of managing and disposing of the WR-1 radioactive systems[5].

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1 Phase 1 involved defueling the reactor, draining the organic coolant, removing the heavy water, removing spent reactor fuel stored in the fuel storage bays, and full or partial decommissioning and removal of certain reactor systems. This included: the C circuit building annex, the organic supply system and sampling rooms, the organic purification system, emergency coolant injection tanks, selected equipment from the boron addition systems, and selected alarm and monitoring systems.

2 ISD will allow the completion of WR-1 decommissioning in as little as 8 years.

3 ISD will reduce radiological and industrial hazard exposures to workers during the removal, handling, and packaging of highly radioactive reactor systems and while work within contaminated, confined, and highly restricted work spaces during dismantling and removal activities.

4 ISD will eliminate the need to transport a large number of radioactive waste shipments over public roads to an off-site location for interim storage and future disposal.

5 ISD technology and methods have been developed, approved, and successfully performed for the decommissioning of research reactors operated at U.S. Department of Energy national laboratory sites including Savannah River and Idaho Falls.

2. REACTOR DESCRIPTION

2.1 General

WR-1 is an organic-cooled, heavy water moderated (99.78% isotropic purity), vertical pressure tube, thermal neutron research reactor with an output rating of 60 MW (thermal). It was in operation from November 1965 until May 1985 when it was permanently shut down and defueled. Fuel used in WR-1 was enriched to 1-5% ^{235}U by weight.

WR-1 extends approximately 18 m below-grade. Level 600 of the reactor building (B100) is the main reactor floor and is located at-grade. Five of the B100 levels are below-grade (Levels 100-500). Level 100 (sub-basement) is located approximately 18.5 m below Level 600. The six levels cover a total area of 6,488 m^2. B100 is an industrial type structural steel frame building with double corrugated steel sheathing above the Level 600 and is reinforced.

The primary remaining reactor systems include:

- Reactor vessel
- Reactor biological shield
- Primary heat transport system
- Auxiliary organic and gas systems
- Helium and heavy water systems
- Spent fuel handling and storage system
- Experimental loops

Figure 1 provides a cross-sectional illustration of B100 and WR-1, and Table 1 gives a brief description of each of the main reactor systems.

Following the completion of Phase 1 decommissioning activities in 1995, WR-1 was divided into two general access areas:

- WR-1 unrestricted access
- WR-1 restricted access

The WR-1 unrestricted access area consists of rooms that underwent decommissioning and decontamination under the Phase 1 decommissioning activities and had radiological hazards reduced to background or minimal levels. This includes the majority of rooms on the 600 and 500 levels.

The WR-1 restricted access area is comprised of rooms that have not undergone any decommissioning activities or had remaining elevated radiological hazards following the completion of Phase 1 decommissioning. This includes rooms and areas on the 100 to 400 levels, and some rooms on the 500 to 600 levels. Table 2 provides a list of rooms and areas that are restricted access. Radiological hazards within the rooms are discussed in Section 6.2 and summarized in Table 3.

2.2 Reactor Core [5]

The major components still remaining in the reactor core area (Room 401; see Figure 2, Figure 3, Figure 4) are:

- Calandria vessel
- Calandria tubes
- Fuel channels
- Thermal shields

The calandria vessel is cylindrical and stainless with a torispherical upper head and a dished lower head (see Figure 3). It is approximately 503 cm tall with a diameter of 270 cm and wall thickness of 1.27 cm. The dump ring assembly divided the calandria into an upper and lower section. The upper section, or the core space, contained the fuel and the moderator/reflector. The lower section, or the dump space, contained helium gas and collected the moderator spillage from the core space. If the reactor was tripped, the dump space received the moderator from the core space and drained it to the moderator dump tank (Room 107).

Fifty-four (54) vertical calandria tubes (30 large aluminum tubes[6] and 24 small aluminum tubes[7]) penetrate the calandria vessel forming the reactor lattice (see Figure 5). Fifty-three (53) pressure tubes (referred to as fuel channels for WR-1) are located concentrically within these tubes for accommodation of the fuel assemblies with one location used by used by the pneumatic capsule facility. There are three fuel channel materials: ozhennite (28 locations), Zr-2.5%Nb (20 locations), and stainless steel (4 positions).

Also within the reactor vault, are auxiliary structures and systems such as moderator spray nozzles, helium purge lines, moderator sampling and drain lines, and rods suspending the calandria from the radial or side thermal shield.

2.3 Biological Shield [5]

The reactor vault is surrounded in the radial direction by poured heavy concrete approximately 7 feet thick which forms the biological shielding. The thickness was chosen to attenuate radiation fields to 2 mR/h in the adjacent areas that are typically occupied by personnel while the reactor is running at 60 MW. Heavy concrete or Ilmenite (titanium iron ore concentrate) with a density of 220 lbs/ft^3 was used. Heavy concrete was also used in the vicinity of the upper and lower access rooms as well as for the shutdown shields. Figure 4 details the shielding system.

2.4 Primary Heat Transport System [5]

The primary heat transport (PHT) system removed the heat produced in the reactor. The system was comprised of three separate circuits (A, B, & C) of approximately 20 MW (thermal) heat removal capacity each. The heat removed was dissipated to the Winnipeg River through three conventional tube-and-shell heat exchangers using an organic primary coolant and river water as the secondary coolant.

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6 12.1 cm inner diameter

7 9.9 cm inner diameter

The PHT system consisted of three circuits for experimental research flexibility. Each circuit was independent of the others and had its own coolant, circulation system, and degassing and particulate removal system. The A and B circuits were located in Rooms 506 and 602 whereas the C circuit occupied Rooms 528 and 647.

The initial installation consisted of only the A and B circuits with the core limited to 37 available sites. In 1972, the C circuit was installed bringing the core size to 53 sites. The design of the A and B circuits are identical in equipment, circuitry, and allowable operating conditions. The C circuit is similar in design, but differs in these regards.

The coolant used in WR-1, Monsanto HB-40 (renamed OS-84), was a complex mixture of organic compounds varying in molecular weight from 2 to over 1000. It consisted of terphenyls treated catalytically with hydrogen to produce 40% saturated hydrocarbons. At high operating temperatures (350-400°C), the coolant became de-hydrogenated resulting in an increase in the carbon:hydrogen ratio and a higher viscosity. The rate of de-hydrogenation increased with temperature necessitating an increase in coolant make-up and feed-bleed rates to control the high boiler concentrations. The coolant high boiler concentrations in the PHT circuits were limited to a maximum of 33% because at higher concentrations the viscosity of the coolant increased to the point where flow problems occurred during shutdown and low temperature operation.

2.5 Auxiliary Organic and Gas Systems [5]

A degassing system was provided for each of the PHT circuits. Each degassing system consisted of two pressurizing pumps, a degassing tank, an off-gas condensing circuit, a volatile recycle circuit, and a particulate removal circuit. Details on the components of the systems are provided in Table 4.

The degassing systems for the A and B circuits were essentially identical. The C circuit system was installed in 1972 with some differences in circuitry and equipment. The following description applies to all three systems unless specified otherwise.

Approximately 15% of the degassing return flow was channeled through a particulate removal circuit to remove organic particulates, metal, and other fouling substances from the coolant. The circuit consisted of a heat exchanger, two attapulgus clay adsorption columns, and a backup filter unit. Special heat exchangers were provided for the A and B circuits whereas the C circuit used the degassing system standby heat exchanger. The coolant was cooled to 300°C before entering the adsorption columns to achieve maximum efficiency. One adsorption column and a filter were normally in service when the degassing system was operating.

The degassing tank provided storage for surplus coolant from swells of the PHT and degassing systems due to temperature changes, and reserve coolant necessary for purification transfers, sampling, and minor system leakage. The coolant in the degassing tank also provided the necessary suction head for the pressurizing pumps.

2.6 Helium and Heavy Water Systems [5]

Heavy water was used in WR-1 as a moderator/reflector and as a coolant for the removal of gamma and thermal heat picked up by the calandria vessel and tubes. The moderator was supported in the core space by the helium blowers of the helium system that created a differential helium pressure between the core and dump spaces. The moderator level, and hence reactivity, was controlled by regulating the differential pressure with two control valves operated by a regulating system. The helium also provided an inert cover gas which

prevented air and light water vapour from entering the heavy water system and thus contaminating and downgrading the heavy water.

The main components of the heavy water system were a dump tank, a helium accumulator tank, three circulation pumps, a heat exchanger, the calandria, piping, and instrumentation (see Figure 6). All components, other than the calandria, are located in Room 107. All piping and material in the system that came into contact with heavy water were constructed from corrosion resistant materials. Most of piping in Room 107 was aluminum while all embedded or concealed heavy water and helium piping was seamless stainless steel. All valves, pumps, and special assemblies, other than the main heat exchanger, were stainless steel. The heat exchanger was carbon steel with Inconel tubes and an Inconel liner on the channel side. Expansion bellows were provided on the main piping connections between equipment in Room 107 and the calandria. There were three, single-stage, horizontally mounted, centrifugal pumps connected in parallel to common supply and discharge headers.

The helium system consisted of two helium pumps, two helium control valves, six reactor dump valves, a helium accumulator tank, five heavy water vapour condensers, a recombination unit, oxygen and helium addition stations, a sampling station, and system piping and instrumentation. All components of the system were in Room 107 except the addition and sampling stations which were in Room 106. The piping and equipment were constructed from stainless steel with the exception of some aluminum piping in Room 107.

The heavy water collection system located in Room 106 consisted of a 0.365 m^3 stainless steel tank, a pump, piping, and instrumentation. The system was used to drain the moderator pumps before maintenance, the helium purge lines, and the heavy water sample station lines. Other components and lines in the heavy water and helium system could be drained to the collection tank through temporary connections.

2.7 Reactor Fuel [5]

Initially, the reactor used a uranium oxide (UO_2) reference or “driver” fuel enriched up to 2.4 wt. % ^{235}U and sheathed in Zr-2.5%Nb, or ozhennite as an alternative. However, this was switched to a uranium monocarbide (UC) fuel due to its higher thermal conductivity and density which enabled the fuel to operate at higher power ratings.

WR-1 was operated at approximately 54 MW(t) using the UC fuel which was slightly enriched with ^{235}U. Two enrichments were used to allow the reactor thermal neutron flux to be maximized without overdriving the centre fuel rods, 1.33 wt. % ^{235}U in the centre sites (E-8, E-10, D-7, & F-7) and 1.95 wt. % ^{235}U in the remainder of the core.

In addition to the driver fuels, a variety of experimental fuels and experimental loop fuels were irradiated in WR-1 as individual experiments. The fast neutron (FN) loops (1L4, 1L5, 1L6) used 4.95 wt. % enriched UC fuel with elements in a ring configuration to provide a high fast neutron flux test facility for irradiation experiments inside the large central support tubes.

A maximum of 53 fuel rods could be accommodated in the 55-position reactor lattice. Two positions, A-8 and J-8 were used for in-core flux detection and the pneumatic capsule facility, respectively. Positions temporarily left unfuelled were fitted with flow restrictors to restrict the flow to a value equivalent to that of the fuelled sites to maintain a proper coolant distribution throughout the core.

2.8 Spent Fuel Handling and Storage Systems [5]

Irradiated fuel was removed from the core with the 45-ton fuel transfer flask approximately six hours after the reactor had been shut down. The initial transfer was from the reactor to an organic coolant filled storage tube immersed in a water-filled fuel storage block where it remained for a decay period of usually two to six weeks. The fuel remained immersed in organic coolant throughout this transfer.

The fuel storage block provided a facility for storing and washing irradiated fuel, storing irradiated fuel channels and experimental equipment, and loading and unloading the 21-ton fuel transfer flask.

When the fuel had decayed sufficiently to transfer it dry without exceeding the maximum allowable sheath temperature, it was transferred to the fuel storage bays. The transfer was made at the transfer station on the reactor hall floor. The fuel rod was lowered through a shielded transfer passage into an organic-filled can suspended in the transfer trolley positioned below. The can was immersed in a water-filled bay for shielding and heat removal with only its top above water.

The transfer trolley was then moved out of the transfer passage and into the work bay to a point where the spent fuel room crane could be used to transfer the can containing the fuel to a storage rack in either the north or south storage bays.

Spent fuel was stored in the organic-filled cans until it could be safely shipped out. The UO_2 fuel was shipped using a special shipping flask which was lowered onto the floor of the work bay and loaded under water. The UC spent fuel remained in the storage bays unless removed and transferred to the WL Hot Cells Facility.

The 21-ton transfer flask was used to transfer irradiated fuel to the WL Hot Cells Facility for examination, bundle shuffling, disassembly, or waste packaging. Before being shipped, the fuel was washed in a xylene-filled, converted storage tube in the fuel storage block to remove the opaque, irradiated organic coolant film from the sheathing.

The fuel wash down system supplied organic coolant needed for the fuel transfer flask tubes and the fuel storage block/storage bays cans, and provided a used coolant drainage facility. Decay heat from the stored fuel was dissipated to the water in the circulation system of the spent fuel bay which in turn dissipated to the Winnipeg River through a tube-and-shell heat exchanger cooled by process water.

2.9 Experimental Loops [5]

2.9.1 General

There were four experimental loops in WR-1 and one out-of-reactor hydraulic test loop. Each in-reactor loop consisted of a fuelled test section in a reactor lattice position, and piping equipment and instrumentation in an adjacent room to maintain the required operating conditions. A fuel position was converted to a loop by disconnecting the inlet and outlet feeders from the primary heat transport system and connecting the feeders to the loop inlet and outlet piping.

2.9.2 1L1 Loop

The 1L1 Loop was an out-of-reactor hydraulic test facility capable of handling full sized fuel channels and fuel assemblies. The loop consisted of a circulation pump, a pressurizing pump, three test sections, three electric heaters, a makeup tank/degassifier, a condenser circuit, a purification circuit, a loop cooler, piping, and

instrumentation. Organic coolant was circulated through the test section(s) at operating conditions determined by the experiment being performed.

The three test sections extended between a hatch in the reactor hall floor and the 1L1 Loop room (Room 410). Each test section consisted of an inner WR-1 fuel channel and an outer insulated 6 in. carbon steel pipe jacket suspended from the building. The fuel channel was sealed top and bottom to the jacket similar to the in-reactor seals to the calandria extension tubes. A CO_2 annulus between the fuel channel and the jacket provided heat insulation. The test sections were arranged in parallel, enabling multiple tests to be carried out simultaneously. The outlet feeder connections were variable to adapt to the various outlet feeder elevations of the WR-1 fuel channels.

2.9.3 1L2 Loop

The 1L2 Loop was primarily located in Room 539. The loop was designed to provide a light water-cooled research facility with an environment similar to that found in a pressurized water reactor (PWR) or a boiling water reactor (PWR). The loop was modified in 1974 to simulate the Bruce and post-Bruce CANDU power reactor conditions for studies of activity transport in two-phase flow to establish a Canadian design code for CANDU power reactors.

2.9.4 Fast Neutron Loops – 1L4, 1L5, 1L6

The fast neutron (FN) loops were designed to:

1. Provide test facilities to study the creep rate and various other physical property changes of metals and other materials in a fast neutron flux.
2. Develop fuels, fuel channels, and reactor components for the CANDU reactors.

Each loop was designed to remove 4.5 MW(t) using organic coolant from either the A or B PHT circuits. The loop systems and components occupied Rooms 537, 538, and 546.

2.10 Process and Active Drainage Systems [5]

The process water (PW) system provided cooling water for WR-1 under normal operating conditions. Process water pumps, located in the site pump house, discharged into a 24 in. diameter underground concrete pipe that conveyed the PW to WR-1. A strainer was located at the inlet to WR-1 at an elevation of 826 ft. 2 in. Downstream of the strainer was a pressure control station after which the system divided into the PW and standby water (SW) systems. The SW system provided cooling water to essential components of the reactor which could not withstand a loss of cooling water caused by a failure of the PW system. Process and standby water was returned to the Winnipeg River through the process drainage (PD) system. An organic trap was provided on the discharge line to precipitate out any organic coolant entrained in the water being discharged into the river.

The active drainage (AD) system was designed to collect liquid effluent from the various areas of the WR-1 plant and the groundwater from around the building base. The system had five sumps:

1. Active Drainage Sump A – collected active and potentially active liquid wastes from all B100 rooms except those that contained organic or heavy water equipment.
2. Organic Drainage Sump A – collected liquid wastes from areas in B100 where the effluent might have contained organic coolant.

3. Active Drainage Sump B – collected floor drain effluent from the heavy water rooms.
4. WR-1 Extension Active Drainage Sump – collected active or potentially active effluent from the WR-1 extension, other than the C circuit and 1L6 Loop rooms.
5. Sub-Surface Drainage Sump – collected effluent from the weeping tiles located under and around the periphery of B100.

Collected wastes were transferred for disposal to the active waste disposal building.

3. POTENTIAL CONTAMINANTS OF CONCERN

This section outlines the main radiological and non-radiological contaminants of concern that are known to or potentially remain in B100. Table 5 provides a summary of common activation products, fission products, and actinides that can be found in shutdown reactors and how they are produced. Table 6, Table 8, and Table 9 give a summary of known or potential non-radiological contaminants and hazards of concern.

3.1 Radiological

The main sources and mechanisms of radiological contamination and hazard expected to be present following the shut down and defueling of WR-1 are:

- Activation products in the reactor vessel and biological shield, and corrosion products in the PHT and heavy water moderator systems.
- Fuel failures resulting in the release of fission products and actinides to the PHT system and/or experimental loops.
- Contaminated components due to contact with radioactivity transported through the PHT and heavy water moderator systems.
- Reactor rooms and materials that became contaminated as a result of spills or system leaks.
- Fuel handling and storage systems that became contaminated.

Table 10 provides the primary radionuclides of concern (ROCs) that were accessed during the 2017/2018 characterization campaign. Additional discussion of the ROC origins are provided in Table 5.

3.1.1 Activation Products [6]

Activation products are primarily produced when components and structures are in close proximity to the high neutron flux of the reactor core. For WR-1, this includes the fuel channels (pressure tubes and calandria tubes), the calandria vessel, the thermal shields, the biological shield, and any other structures or instrumentation inside the reactor vault. These components and structures absorb neutrons and, as a result, may become radioactive, i.e. activated.

Activation products make up the vast majority of radioactivity remaining in WR-1. Activation products of concern include ^{14}C, ^{36}Cl, ^{55}Fe, ^{63} Ni, ^{59}Ni, ^{60}Co, and ^{94}Nb (see Table 10).

3.1.2 Corrosion Products [6]

Corrosion products are mobile activation products that are produced when debris from corrosion or damage is transported through the reactor and becomes radioactive by absorbing neutrons. During the operation of a nuclear reactor, most metallic surfaces oxidize and form a layer of corrosive film. This layer erodes and is transported through the reactor core where it is exposed to a high neutron flux. Common corrosion products include ^{55}Fe and ^{60}Co.

Minimal corrosion products are expected in the WR-1 reactor systems due to the use of organic coolant which is less corrosive than a water-based system.

3.1.3 Fission Products and Actinides [6]

Fission products are produced in the reactor fuel as a result of nuclear fission. The process of fission involves a fissile atom (^{233}U, ^{235}U, ^{239}Pu, or ^{241}Pu) absorbing a neutron into its nucleus to create an unstable isotope that subsequently splits (fissions) into lighter nuclei while releasing neutrons, which make it possible for another fission to occur, and energy. The most abundant fission products are ^{137}Cs and ^{90}Sr. Longer-lived fission products of greater concern for waste disposal are ^{99}Tc and ^{129}I.

Actinides are a group of radionuclides of potential concern that are generated in irradiated reactor fuel as a result of atoms absorbing neutrons and not undergoing fission. Common actinides include ^{241}Am, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{244}Cm.

Fission products and actinides are normally contained within the fuel cladding; however, they may be released by a fuel cladding breach (referred to as a fuel failure). This results in deposited contamination on fuel handling systems and in the PHT system.

3.1.4 Contaminated Systems, Components, and Surfaces [6]

Contamination deposited on internal and external surfaces of the reactor systems and components can occur as a result of the transport and/or leaching out of activated corrosion products or fission products and actinides. Reactor core hardware, PHT circuits piping, and auxiliary circuits and associated equipment often become contaminated. Also, contamination is generally present on all surfaces near the fuel discharging equipment, the storage pools, and the processing and storage facilities for radioactive effluents and wastes (e.g. ion exchange resins, clay columns, filters).

Contamination within system piping is likely to be concentrated at the discontinuity points of the fluid flow, e.g. elbows or tees. It also tend to settle preferentially on rough surfaces, for example, on the oxidized parts of metals.

Except for in concrete, contamination does not generally penetrate deeply. The surface distribution of contamination is typically not homogeneous. Accumulation may occur on the walls or on the floors behind or beneath the equipment (such as motors, cables, pipes, and vessels) particularly when leakages or flows may have occurred.

Surface contamination of concrete is generally limited to areas of the facility where radioactive liquids and aerosols have been released; ^{137}Cs, ^{90}Sr, and ^{60}Co are the most common radionuclides for this type of contamination. Radiocaesium isotopes are preferentially absorbed onto bare concrete (or concrete surfaces that have lost their paint coatings) relative to other radionuclides because of the ability of caesium to undergo ion exchange with mineral phases in the concrete.

3.2 Non-Radiological

In 2006, Pinchin Environmental Ltd. conducted an industrial characterization of B100 [7]. The hazardous materials surveyed for included:

- Asbestos-containing materials (ACMs)
- Lead in both paint and solid form
- Polychlorinated biphenyls (PCBs)
- Mercury
- Mould

Other potential hazardous materials in B100 include residual amounts of:

- Potassium hydroxide
- Boron
- Xylene
- Chromium
- Cadmium
- Mercury
- Organic coolant

Table 6, Table 8, and Table 9 summarize some of the non-radiological hazardous material in B100.

3.2.1 Asbestos-Containing Materials

Friable asbestos-containing mechanical insulation was found in B100. The term friable is applied to a material that can be reduced to dust or powder by hand or moderate pressure. ACMs that are friable have a much greater potential to release airborne asbestos fibres when disturbed. Overall, the 2006 characterization found that the general condition of the ACMs within B100 was good; however, there are some sections of pipe and mechanical insulation that did show some damage.

Room 506 and the associated mezzanine have a significant amount of friable asbestos. Vinyl composite floor tiles were observed throughout B100. These tiles are considered to contain asbestos.

Insulation on straight runs of pipe systems in B100 are insulated with fibreglass, magnesia block, sweatwrap, or armaflex. Insulation on elbows, valves, tees, and fittings of pipe systems are either insulated with a hand applied parging cement, fibreglass, or not insulated at all. Some of the parging cement contained up to 10% chrysotile asbestos. This material is considered to be a friable product and requires a protective covering. The parging cement is covered with a canvas jacket. Approximately 858 parging cement fittings were rated as in good condition, 33 fittings were rated as in fair condition, and one fitting is rated as being in poor condition.

Nonfriable ACMs (countertops, fume hoods, etc.) are also present within B100.

Table 6 details the locations of accessible ACMs within B100.

In 2013, a comprehensive survey specific to asbestos containing materials in B100 was performed [8]. This characterization investigated areas of the below grade reactor building that were not inspected during the 2006 industrial characterization [7]. The additional surveys identified ACMs in the following forms:

- Thermal System Insulation (TSI) / mechanical pipe insulation (friable)
- Mastic associated with thermal system insulation (non-friable)
- Vinyl floor tiles (non-friable)
- Vermiculite (friable)
- Laboratory counter top (non-friable)
- Tar (building) paper cover (friable)
- Cement board (transite; non-friable)

A summary of the sampling results of the 2013 investigation are given in Table 7.

3.2.2 Other Hazardous Materials

Lead-based paint is present within B100 at the locations shown in Table 8. Light ballasts in B100 fluorescent light fixtures are suspected of containing PCBs. Identification of mercury-containing materials was conducted on a visual basis and lightbulbs present in the Slowpoke Demonstration Reactor Hall (Room 690) are suspected of containing mercury vapour. Identification of mould-containing materials was generally conducted on a visual basis. Minor amounts of mould were observed in B100 on pipe insulation and ceiling tiles. A number of other hazardous chemicals and materials were used during the operation of WR-1. Table 9 lists these hazards and their expected locations within B100.

4. REACTOR CORE: ESTIMATED AND CHARACTERIZATION INVENTORIES

Inaccessibility of the reactor calandria and core was a limiting factor for acquiring a representative sample population for radionuclide-specific analyses. Therefore, a multi-pronged approach was required to conservatively estimate and cross-validate the total inventory. The estimation and validation methods consisted of:

- Using computer modelling codes to determine the neutron flux and resulting activation of the calandria vessel, thermal shield, dump floor, and core fuel channels and tubes.
- Using gamma dose rate measurements to validate estimated gamma radiation dose rates based on the modelling.
- Reviewing the results of the modelling against changes made to the codes and code libraries since the time of the modelling.
- Using laboratory analytical data from fuel channel samples to validate the modelled activation inventory.

4.1 Modelled Reactor Core Radioactivity Inventory Estimates

Evaluations were performed in the early 1990s to determine the reactor core radionuclide inventory in various core components and the activity of specific radionuclides, both over a 100-year period beginning after shutdown. The evaluations and results are given in References [9] and [10].

The first evaluation [9] calculated photon decay curves and the radioactive inventory over time for the fuel channels, calandria tubes, calandria vessel, and thermal shields. The second evaluation [10] calculated the total reactor inventory for the remaining reactor core components at 10, 50, and 100 years after shutdown.

The calculated radioactivity inventory in the reactor core components are shown in Table 11 and Table 12 for 10, 50, and 100 years after shutdown and decayed to 1,000 and 10,000 years. The principal radionuclides are low energy and low radiotoxic beta emitters ^{14}C ($T_{1/2}$ = 5,730 y), ^{59}Ni ($T_{1/2}$ = 76,000 y), and ^{63}Ni ($T_{1/2}$ = 100 y) and moderate radiotoxic gamma emitters ^{60}Co ($T_{1/2}$ = 5.3 y) and ^{94}Nb ($T_{1/2}$ = 20,300 y). The majority of the radioactivity is expected to be contained within the stainless steel calandria vessel and stainless steel fuel channels. Additional data for ^{108m}Ag was brought forward in Table 13 due to the importance of ^{108m}Ag in the Nuclear Power Demonstration Reactor Decommissioning project and to demonstrate the negligible contributions of nuclides outside of the principal radionuclides.

The total reactor core activity, estimated based on the modelling, was 6,000 TBq (~160,000 Ci) at 10 years following reactor shutdown (1995). The activity decays steadily by a factor of about ten to 610 TBq (~16,500 Ci) at 100 years following shutdown (2085). ^{63}Ni accounts for about 98% of the radioactivity during these first 100 years. ^{60}Co dominates the gamma radioactivity during the first 50 years following shutdown, but decays away representing only a small fraction of the gamma activity after 100 years. The radioactivity decays to 16 TBq (~422 Ci) after 1,000 years and to 10.6 TBq (~286 Ci) after 10,000 years where it is dominated by activity due to ^{59}Ni and ^{94}Nb, with a small fraction of activity due to ^{14}C.

4.2 Modelling Code Validation [11]

The activation of the reactor components and the decay of the activation products were modelled using WIMS-CRNL, ONEDANT, ORIGEN-S, AND MCNP Rev. 2 codes. This modelling provided an estimate of the total activity of the WR-1 core and has provided the basis for the overall estimate of remaining activity in WR-1. This work was complete in 1992 and as such there have been several revisions to the codes used. To validate the estimate, a review of the modelling was carried out examining three aspects of the work:

1. The modelling assumptions and approach.
2. Changes in the codes since 1992 that may affect results.
3. Changes in information databases, such as nuclide cross-sections and half-lives, since 1992 that may affect results.

The review found that the calculations have an order of magnitude accuracy. Since the various EIS modelling scenarios are all several orders of magnitude below the safety case, the 1992 modelling is sufficient and new calculations do not need to be performed.

4.3 2017/2018 Fuel Channel Characterization [12][13]

Empirical determination of radionuclide concentrations within core components was achieved through the collection of shavings from the fuel channels. Although the number of samples that could be collected was

severely restricted due to access considerations, four samples were obtained from four fuel channels (D13: Zr-2.5%Nb fuel channel, D7: ozhennite fuel channel, and E10 & F5: stainless steel fuel channels). The analytical results provided necessary evidence to validate the conservatism of the modelled inventory.

The analysis of the samples focussed on the expected primary contributors to total activity, as predicted by the ORIGEN-S model: ^{14}C, ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, and ^{94}Nb. Combined, these nuclides make up the overwhelming majority (95-99%) of activity due to activation products of concern.

When the results of the samples were compared to the model estimated inventories, a ratio for each material and nuclide was calculated, and applied to the inventory calculation. This 'Modified Total' is lower than the inventory estimate calculated from the ORIGEN-S model (see Table 14), providing additional confidence in the original modelling results being conservative.

5. BIOLOGICAL SHIELD: ESTIMATED AND CHARACTERIZATION INVENTORIES [14]

As part of a CANDU Operators Group research project, a study was performed in 1996 to sample and characterize irradiated concrete from the WR-1 biological shield. A 9.5 cm diameter, 212.6 cm long concrete sample was obtained by cutting through the biological shield from the ion chamber in Room 402. The sample was taken from 1 m below the elevation of the maximum neutron flux.

Concrete samples and separated samples of cement, aggregate, and reinforcing bars were analyzed for ^{3}H, ^{14}C, ^{36}Cl, ^{41}Ca, ^{63}Ni, and gamma-emitting radionuclides as well as to determine the elemental composition of the concrete components. Comparison of experimental results with theoretical neutron flux profiles, plus additional information on concrete composition, allowed estimates of the 90% radionuclide inventories (per unit cross-sectional area) for ^{14}C, ^{36}Cl, ^{41}Ca, ^{63}Ni, ^{60}Co, and ^{152}Eu in the sample (see Table 15)[8]. There was generally good agreement between experimental measurements and scaled thermal-neutron fluxes, plus agreement between experimental and calculated ^{60}Co activities in rebar steel which suggested that most of the estimated inventories are accurate within an order of magnitude. The estimated inventory of ^{36}Cl was an exception and remained uncertain due to challenges in measuring low chlorine concentrations.

The inventory estimates are applicable to irradiated biological shield concrete near the mid-core elevation of WR-1. Determination of the variation of the neutron flux over the entire surface of the biological shield and estimation of the overall radionuclide inventory was not undertaken because of the termination of the waste management research program at WL in 1997.

Table 15 provides a total radioactivity estimate of the shield by multiplying the 90% radionuclide inventories per unit cross-sectional area by the shield's outer surface area of 140.5 m^2 [(9)]. The total biological shield activity was calculated to be 58 GBq (1,600 mCi) at 10 years following shutdown and decays steadily by a factor of about 100 to 0.58 GBq (~16 mCi) at 100 years following shutdown. ^{60}Co initially accounts for most of the activity, but decays to a small fraction by 100 years where ^{14}C, ^{41}Ca, and ^{63}Ni are the main contributors. The radioactivity decays to 0.20 GBq (5.3 mCi) after 1,000 years and to 0.15 GBq (4.0 mCi) after 10,000 years where the activity is due to ^{14}C and ^{41}Ca with a very small fraction due to ^{36}Cl.

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[8] Inventory estimates are for the radioactivity contained within the first 22.9 cm of the biological shield (corresponding to 90% of the inventory).

[9] Assumes 5.2 m in height and 8.6 m in diameter.

6. OUTSIDE CORE REACTOR SYSTEMS: ESTIMATED AND CHARACTERIZATION INVENTORIES

6.1 Methodology

The characterization of the outside core reactor systems followed two general steps. The first was the review of historical information. Information was collected from a variety of sources including modelling reports, survey results, operational logs, and staff anecdotes. The suite of information available is comprehensive and was used as a baseline for direction of the second phase of characterization work and sample collection.

The 2017/2018 outside core reactor systems characterization followed an optimized rank set sampling[10] (RSS) design [1]. The intrusive sample characterization plan for these systems was designed and implemented in a manner that ensured a robust primary data set from which unbiased estimates of various radiological parameters were developed at specified confidence levels.

The RSS methods were used for most systems to determine the required number of field measurement locations from which the physical sample locations were selected. The sample population size was determined based on a desired 80 or 90% confidence level of the 80 or 90% confidence interval (80/80, 80/90, 90/80, or 90/90) mean radionuclide of concern (ROC) concentrations. The selected interval and confidence was dependent upon the expected contribution of activity hold-ups within a given system. In other words, a greater tolerance in the uncertainty of the mean confidence intervals was acceptable for systems anticipated to have lesser contributions to the total inventory.

RSS locations were randomly distributed over each system primary decision unit, direct gamma dose rate and gamma radiation measurements were performed, measurement results were ranked, and sample locations were selected based on the RSS process. Intrusive samples, consisting of coupons cut from larger system components or sections of small diameter piping, were collected from each selected location.

The active and process drain systems were sampled based on a simple random sampling plan due to accessibility restrictions. Other systems that were not conducive to a RSS design, specifically the components and systems within the reactor vault and biological shield, were selectively sampled.

Samples were packaged and shipped to a contracted laboratory for initial analysis by gamma spectroscopy and, for samples collected outside of the reactor vault, acid surface leaching. Any samples from within the vault, and therefore likely to be volumetrically activated, were to be subject to total dissolution. The gamma spectroscopy results were reviewed and additional analyses for the non gamma-emitting ROCs were completed in accordance with Table 6.1 of the characterization plan [1]. When these additional analyses were required, sample leachates were either composited or processed individually and selectively analyzed as applicable for ^{3}H, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{99}Tc, ^{238}Pu, $^{239/240}Pu$, ^{241}Pu, and $^{243/244}Cm$.

The analytical results of the samples provided the necessary inputs to calculate the system population per unit area activity (Bq/cm^2) mean and uncertainty at the 90% confidence level for each ROC. These parameters were calculated as follows for RSS-based investigations (from Reference [15], Page 96):

[10] Rank set sampling is an innovative design that can be highly useful and cost efficient in obtaining better estimates of mean concentration levels in soil and other environmental media by explicitly incorporating the professional judgement of a field investigator or a field screening measurement method to pick specific sampling locations in the field [15].

a. **Mean (Bq/cm^2) @ 90% Confidence:**

$$\bar{x}_{\text{RSS,balanced}} = \frac{1}{rm}\sum_{i=1}^{m}\sum_{j=1}^{r} x_{i,j}$$

where *r* = number of cycles,

m = set size, and

$x_{i,j}$ is the measurement of the sample collected from the field location that had rank *i* that was collected in the j^{th} cycle of sampling.

b. **The standard deviation (Bq/cm^2)** is calculated by taking the square root of the variance for a balanced RSS sample. The variance is calculated with the following equation:

$$\text{Var}\left(\bar{x}_{\text{RSS,balanced}}\right) = \sum_{i=1}^{m}\sum_{j=1}^{r}\frac{\left(x_{i,j} - \bar{x}_i\right)^2}{m^2 r(r-1)}$$

where $\bar{x}_i = \frac{1}{r}\sum_{j=1}^{r} x_{i,j}$ or the arithmetic mean of the *r* laboratory measurements of the *m* samples from field locations that had rank *i* collected during the j^{th} cycle of sampling.

The 90, 95, and 99% upper confidence levels (UCLs) of the radionuclide-specific per unit area activity (Bq/cm^2) for each system were, in most cases, constructed based on the above RSS population mean and uncertainty concentrations. The specific UCLs generated were based upon the data distribution—i.e. normal vs. no discernable distribution and skewness—and population variability in accordance with the following (from Reference [16]):

a. **UCL (Bq/cm^2):** It was determined by ProUCL[11] if a data set appeared normal, and if it did not, then the Chebyshev inequality for non-parametric distributions was used:

$$UCL_{1-\alpha} = \bar{X} + \sqrt{\frac{1}{\alpha} - 1}\left(\frac{s}{\sqrt{n}}\right)$$

where $\bar{X}$ is the mean as calculated above,

s is the standard deviation as calculated above,

n is the total number of samples (*rm*), and

α = 0.1, 0.05, or 0.01 for 90%, 95%, or 99%, respectively.

The results were then multiplied by the total surface area of the respective system to obtain the total inventory (Bq) for each ROC.

There were cases where the UCLs for specific radionuclides required estimation, specifically when the sample population was small (i.e. fewer than five results) and/or for composite sample results. The secondary methods used were as follows:

[11] ProUCL is a statistical software package for analysis of environmental data sets with and without nondetect observations.

b. **No measurement with uncertainty:** For cases where the laboratory did not report an analytical uncertainty and all results were less than the minimum detectable activity (MDA), the analyte was listed as "Undetected". If no measurements with uncertainty were available, but some results were above the MDA, the minimum, maximum, and average for the analyte are presented.

c. **One measurement with analytical uncertainty:** When only one measurement with analytical uncertainty was available, the assumption is that the measurement represents the mean of the population. The UCL was calculated by:

$$UCL = x + z\sigma$$

where *x* is the measurement which represents the population mean,

z is the critical value for the confidence interval (1.282 for 90% one-tailed, 1.645 for 95% one-tailed, and 2.326 for 99% one-tailed), and

σ is the analytical uncertainty of the measurement (representing the population standard deviation).

d. **Two, three, or four measurements with analytical uncertainty:** When two to four measurements with analytical uncertainty were available, the assumption is that the mean of the measurements represents the mean of the population. The mean is calculated using a weighted average:

$$\bar{x} = \sum_{i=1}^{n} w_i x_i$$

where *w* is the fractional weight and *x* is the measurement. For example, if a system contained 39 samples where three samples were analyzed individually and the remaining 36 were composited, each individual sample result would have a weight of 1/39th (0.0256) and the composite result would have a weight of 36/39th (0.9231) in the summed ROC inventory. A specific example is shown in Table 21.

The UCL was calculated by:

$$UCL = \bar{x} + z\bar{\sigma}$$

where $\bar{x}$ is the weighted average of the measurements,

z is the critical value for the confidence interval (1.282 for 90% one-tailed, 1.645 for 95% one-tailed, and 2.326 for 99% one-tailed), and

$\bar{\sigma}$ is the propagated analytical uncertainty of the measurements calculated by:

$$\bar{\sigma} = \sqrt{\sum_{i=1}^{n} w_i \sigma_i^2}$$

where *w* is a fractional weighting factor the measurement standard deviation represents and *σ* is the uncertainty of the measurement.

Secondary data produced through the RSS field screening and post-sampling—direct gamma radiation measurements on system component exterior locations and alpha and beta direct measurements on the

system interior side of sampled coupons—are available for use to establish a correlation between the radionuclide-specific analytical results and the field screening measurements. Data fit equations generated by this correlation could be used to refine inventories by using the larger unsampled field measurement population.

6.2 B100 Rooms Radiological Hazards

Radiological hazard characterization surveys of the B100 rooms have been performed in recent years in preparation for the planning and execution of physical decommissioning activities. These surveys obtained updated information on gamma radiation dose rates in accessible areas of the rooms as well as surface contamination levels on the room surfaces. The elevated dose rate information provides indication of the locations within the systems where there is an internal build-up of contamination.

Table 3 provides a summary of the room hazard surveys that were performed and a detailed listing of each WR-1 room, its use, systems, and components as well as the measured gamma radiation and surface contamination hazard levels.

Radiation dose rate hazards in most rooms range from minimal to low, with moderate hazards limited to only a few rooms. Rooms and areas with elevated gamma radiation levels were associated with the primary heat transport system and components, experimental loops, process drain lines, and include the reactor core lower and upper access rooms. General area radiation fields in various rooms are typically less than 1 mrem/h. Some rooms have general fields ranging from 1-5 mrem/h and from 10-25 mrem/h. Maximum general radiation fields were measured to be 55 mrem/h[12]. Near contact gamma radiation dose rates on rooms and system components range from 30-80 mrem/h (pumps, tanks, process drains, fuel flasks) and from 200-600 mrem/h (drain tanks, degassing tanks, circuit piping valves, surge tank piping). Maximum near contact gamma readings are from 800-1,000 mrem/h[13]. Table 3 also provides the B100 rooms which were either sealed or not accessible where information on the hazard conditions does not exist.

Surface contamination hazards in most rooms range from minimal to low, with moderate hazards limited to a few rooms. The primary contaminants are beta/gamma emitting and are associated with the fission products ^{90}Sr and ^{137}Cs. Minimal alpha contamination was detected in the rooms. ^{241}Am was occasionally detected on surface swipes with ^{137}Cs to ^{241}Am ratios in the range of 20:1 to 25:1 and 30:1 to 50:1. There was one room with low-level surface contamination where a ratio of 7:1 was encountered.

6.3 Primary Heat Transport System

6.3.1 Fuel Failures

Experimental fuels were subject to high burn-ups and extreme operating conditions which resulted in frequent fuel failures during the operating lifetime of WR-1. There were also several fuel failures of the reactor driver fuel. Fuel failure reports are given in References [17][18][19][20][21] and are summarized in Table 16. Between 1966 and 1983 there was a total of ~150 documented fuel failures, most notably in 1983 there were seven separate fuel failure events.

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[12] An exception is a room with radioactive sources stored in a drum where general fields up to 100 mrem/h exist.

[13] An exception is a room with a radioactive source stored in a drum where near contact dose rate of 5 rem/h exist on the drum.

Depending on the fuel location and testing being performed, these fuel failures released short-lived radioactive noble gases and irradiated uranium fuel to piping and system components of: either the A, B, or C circuit primary heat transport systems; experimental loops; and the associated degassing or particulate removal systems. During operation, it was not uncommon for gamma radiation fields on piping to be 25 R/h or higher and fields at the entrance to the rooms to be 10 R/h due to short-lived radioactivity. The degassing and particulate removal systems removed the majority of particulate radioactivity with some fraction of particulates being deposited on piping surfaces, trapped in pumps or values, and in system tank sludge.

6.3.2 Estimate of Fission Products and Actinides Released into PHT

Based on the fuel failure records, the amount of irradiated fuel released from an individual failure ranged from milligrams to grams. Burn-up values for failed fuel typically ranged from 75-260 MW-h/kg (270-940 GJ/kg) with a maximum of 470 MW-h/kg (1,690 GJ/kg). To obtain an upper bound for the amount of radioactivity released into the primary heat transport (PHT) system and experimental loops due to fuel failures, it is assumed that a total of 1 kg of irradiated uranium fuel was released (~6.7 g per failure) with an average fuel burn-up of 190 MW-h/kg (685 GJ/kg). Fuel burn-up tables are given in Reference [22] were then used to give an estimated radionuclide specific radioactivity released at the time of failure and Microshield was used to decay radioactivity over various periods of time.

Table 17 shows the estimated amount of released radionuclide for radionuclides of potential interest including: ^{90}Sr, ^{137}Cs, ^{154}Eu, ^{155}Eu, ^{99}Tc, ^{129}I, ^{237}Np, ^{239}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm, and remaining activity following radioactive decay periods ranging from 5 to 10,000 years. Table 18 shows the percentage of estimated radionuclide specific activity remaining following various periods of radioactive decay. The upper bound radioactivity in the system decreases from ~2.2 TBq (~60 Ci) to ~0.62 TBq (~17 Ci) from decay period of 5 to 50 years with radioactivity being dominated by ^{90}Sr and ^{137}Cs (~65-80% of the total radioactivity inventory). The ^{137}Cs to ^{241}Am ratio during this period varies from ~140:1 to 12:1 as the ^{137}Cs radioactivity decays and ^{241}Am radioactivity increases through the decay of ^{241}Pu. From 100 to 1,000 years of decay, the radioactivity inventory decreases from ~0.2 TBq (~5.5 Ci) to ~0.02 TBq (~0.54 Ci) with the radioactivity becoming dominated by longer-lived actinides ^{239}Pu, ^{240}Pu, and ^{241}Am. The total radioactivity inventory after 10,000 years is ~7.7 GBq (0.21 Ci) with radioactivity dominated by ^{239}Pu and ^{240}Pu. The radioactivity inventory of longer-lived fission products ^{99}Tc and ^{129}I remains constant over time with estimated values of ~0.13 TBq (~3.4 mCi) for ^{99}Tc and ~0.28 MBq (~7.6 μCi) for ^{129}I.

6.3.3 1994 Estimate of PHT System Radioactivity Inventory [23]

A rough estimate of the radioactivity inventory within the PHT system and associated components was made in 1994. The evaluation involved inferring radioactivity inside piping and other closed systems using gamma radiation survey records of externally measured fields and an activity conversion factor derived with Microshield[14]. Survey records were used to make activity estimates for close systems in the following rooms:

- 103 (A & B Circuit Drain Tanks Room)
- 104 (A & B Circuit Degassing Room)

[14] The derived activity conversion factor was Activity (Bq) = 1.20 x 10^{10} Bq per mR/h, where mR/h was the measured gamma exposure rate near contact with a pipe or tank surface.

- 302 (A & B Circuit Degassing Room)
- 306 (Fuel Storage Block)
- 404 (Active Monitoring Room)
- 506 (A & B Circuit Header Room)
- 602 (A & B Circuit Primary Pumps Room)

Measured gamma fields were assumed to be due to ^{137}Cs activity and radionuclide specific radioactivity was inferred based on the relative activity of other nuclides to ^{137}Cs found through the analysis of surface swipes in the various rooms.

Table 19 shows measured gamma radiation fields on closed systems in these rooms and the estimated radioactivity inventories. The total estimated amount of ^{137}Cs radioactivity was 432 GBq (11.7 Ci). In comparison, the upper bound estimate of ^{137}Cs radioactivity due to fuel failures, given in Table 17, is in the range of 480-760 GBq (~13-21 Ci) assuming a 10 to 30 year decay period.

6.3.4 2017/2018 In-System Characterization Inventory

The PHT circuits, in combination with the co-located auxiliary organic and gas (AOG) systems, were expected to contribute the majority of the residual inventory outside of the reactor vault. This expectation is based on process knowledge and the prior estimates discussed in this report. As such, the characterization measurement and sampling stratification required the greatest sample density to achieve an increased confidence in the mean ROC concentrations. The PHT/AOG characterization unit required 117 field assessment locations from which 39 intrusive samples were collected.

The following ROCs were identified—results greater than the laboratory reported MDA—for the PHT and AOG systems in one or more samples:

^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{108m}Ag, ^{137}Cs, ^{235}U, ^{238}U, ^{238}Pu, $^{239/240}$Pu, ^{241}Pu, ^{241}Am, and $^{243/244}$Cm

A summary of inventory results in provided in Table 20. The inventories shown are for the 90th, 95th, and 99th UCLs of the mean. The sample population mean and standard deviation used in the UCL determination were developed based on the reported analytical result for each radionuclide and either the non-parametric (Chebyshev) calculation or, for small sample populations/composites, one of the previously described secondary methods.

The characterization 99th percentile UCL total inventory of 4.70 x 10^{10} Bq shown in Table 20 was compared with the 30-year post-shutdown theoretical inventory of 1.0 x 10^{12} Bq presented in Table 17. The comparison indicates that the theoretical total activity is conservative by a factor of 20. The characterization inventory estimates for ^{90}Sr (27%) and ^{137}Cs (62%) compare closely to the historical inventory estimates that approximately 80% of the activity would be from ^{90}Sr (31%) and ^{137}Cs (48%) combined—a ^{137}Cs:^{90}Sr ratio of 1.6:1.

The ^{90}Sr and ^{137}Cs results for the individual PHT characterization samples and the tallied per unit area inventories were further assessed for correlation to this historical ratio. Table 21 summarizes the measurement comparisons together with the resulting activity ratios for the individual samples that were analyzed for both ^{90}Sr and ^{137}Cs, and the single composite sample result. The assessment confirmed the

historical ratio. Additionally, the table provides an example of the previously discussed weighted mean determination.

6.4 Contaminated Process Drain

6.4.1 Historical Information [24]

In the mid to early 1990s, the process drain line (PDL-1) was found to be contaminated. Radiation surveys indicated that the contamination originated from the B200 Active Liquid Waste Treatment Centre (ALWTC) as there was no significant contamination upstream of the ALWTC. Figure 7 shows a gamma radiation dose rate survey map of the process drain prepared at the time. The surveys also found that the contamination had largely settled in the portions of the PDL-1 that run horizontally within Rooms 108, 109, 203, 408, and 508. The contamination in the process drain was estimated in 1994 to be 3 x 10^{10} Bq (~0.8 Ci) with 95% being ^{137}Cs and ^{90}Sr, and the rest being shorter-lived radionuclides.

6.4.2 2017/2018 In-System Characterization

Eight random samples each were collected from the process and active drain systems. The following ROCs were detected (at concentrations greater than the MDA) in one or more samples:

Process Drain System—^{3}H, ^{55}Fe, ^{59}Ni, ^{60}Co, ^{90}Sr, ^{94}Nb, ^{137}Cs, $^{239/240}$Pu, ^{241}Am, $^{243/244}$Cm, and ^{238}U

Active Drain System—^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{90}Sr, ^{137}Cs, ^{235}U, ^{238}U, ^{238}Pu, $^{239/240}$Pu, and ^{241}Am

The summary inventory results for the process and active drain systems are provided in Table 22 and Table 23, respectively. The 99th percentile UCL total inventories are 5.32 x 10^{8} Bq for the process drain system and 5.19 x 10^{8} Bq for the active drain system. The combined total for both systems is 5.73 x 10^{9} Bq and is a factor of 5 lower than the historical estimate of 3 x 10^{10} Bq. ^{137}Cs accounted for 83% of the characterization inventory within the process drain system and 97% within the active drain system.

6.5 Helium and Heavy Water Systems

6.5.1 Heavy Water and Tritium Inventory [25]

At the time of WR-1 permanent shutdown, the heavy water system had an estimated heavy water inventory of 17,236 kg. During reactor operation, the heavy water was continuously exposed to neutron radiation resulting in the production of tritium in the form of deuterium tritium oxide (DTO) molecules. The DTO concentration in the heavy water rose as reactor operation continued over the years. At the same time, tritium slowly absorbed into the metal piping and tanks of the system just as stable hydrogen is slowly absorbed over time. For WR-1, the tritium level was about 100 Ci/kg at the time of shutdown (based on back calculations from the 2004 heavy water analysis) and would currently be around 17 Ci/kg due to the radiological decay process.

In 1988, the heavy water was drained from the system and sent to Chalk River Laboratories (CRL) for storage and disposition. Dry-out of the system was started in 1990 with the help of a closed loop air circulating mechanical dryer system. The dryer system operated for approximately ten months removing 156 kg of heavy water. In 1991, air purging of the system began. The purge rate was increased slowly on a weekly basis and raised from 3 L/min to 85 L/min over a period of 25 months and collect seven more litres of heavy water. This

extra heavy water was collected and stored in drums in B100 before being sent to CRL for storage and disposition in early 2016.

The system has been under negative ventilation since 1993. This negative ventilation is provided by connecting the helium and heavy water systems (H&HWS) loop to the exhaust duct using an elephant truck. Throughout this period of time, the administrative level of tritium allowed to be released through the stack has been used as an operating limit. The current tritium releases are well below the administrative level of 8 GBq/wk (~2.1 Ci/wk).

6.5.2 H&HWS Radiological Characterization Surveys [25]

In 2015, radiological scoping and characterization surveys were performed on the H&HWS. The objectives of the surveys were to determine: whether there was heavy water remaining within the systems and the levels of internal ^{3}H, ^{14}C, ^{60}Co, and fission product and actinide surface contamination[15].

Below summarizes the characterization findings; the full characterization report and results are provided in Reference [25].

The components of the H&HWS showed no indication of heavy water; however, tritium contamination was found on their inner surfaces. Based on the measurements taken, removable tritium surface contamination was estimated to range from 60-80 Bq/cm^2 within the heavy water dump tank and helium accumulator tank. The total tritium surface contamination could not be determined due to the limitations of the analytical methods. Internal ^{14}C surface contamination was also present. Removable levels were not determined, but total ^{14}C surface contamination levels were estimated to range from 10 Bq/cm^2 (dump tank) to 125 Bq/cm^2 (accumulator tank).

In-situ gamma spectroscopy of piping valves with elevated dose rates[16] identified ^{60}Co as the dominant gamma emitter with trace amounts of ^{137}Cs detected. This was consistent with gamma spectroscopy measurements on drums storing removed heavy water which had measured radioactivity concentrations ranging from 500-1,500 Bq/kg of ^{60}Co and less than 50 Bq/kg of ^{137}Cs. There was no indication of the actinide ^{241}Am.

The inner surfaces of the heavy water dump tank were found to have total surface contamination levels of about 5 Bq/cm^2 ^{60}Co and 0.3 Bq/cm^2 ^{137}Cs. There was only trace levels of internal ^{60}Co surface contamination within the helium accumulator tank (0.07 Bq/cm^2) and no detection of ^{137}Cs.

6.5.3 2017/2018 In-System Characterization

The H&HWS were characterized separately. The helium gas system (HGS) had 27 field assessment locations measured and nine intrusive samples collected. The heavy water/moderator demineralizer/heavy water

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15 ^{14}C is produced when air or water is exposed to neutrons and can slowly be deposited on internal surfaces during the operational phase of a reactor. The level of ^{14}C contamination on the internal surfaces of the WR-1 H&HWS was previously unknown. Operating Experience (OPEX) documents suggested that at Point Lepreau (New Brunswick) and Pickering (Ontario) nuclear power generating plants contamination of these systems was more than 99% ^{14}C. For that reason, the radiological characterization included the collection of coupon samples from component walls to measure the contamination levels of ^{14}C and other radionuclides [26].

16 Near contact gamma dose rates ranged from 5-25 mrem/h.

collection systems (HWS) had 36 field assessment locations measured with twelve intrusive samples collected. The following ROCs were detected in one or more samples[17]:

HGS—^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{137}Cs, ^{235}U, and ^{238}U

HWS—^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{94}Nb, ^{137}Cs, ^{235}U, ^{238}U, and ^{241}Am

Table 24 and Table 25 provide summary results for the HGS and HWS characterization. ^{3}H contributed approximately 91% of the total 4.52 x 10^{8} Bq 99th percentile UCL inventory for the HGS. The remaining predominant activities are from ^{14}C (4-6%) and smaller percentages of ^{63}Ni and ^{55}Fe. The 99th percentile UCL total inventory for the HWS was 1.26 x 10^{9} Bq and consisted of 88% ^{3}H.

The analysis of the HWS showed significantly less tritium remaining within the system (see Table 25) when compared to the preliminary tritium estimate (see Table 26). The belief at the time was that the mechanical means of sample extraction generated sufficient heat and vibrations to release the absorbed tritium as during both the 2015 and 2017 campaigns a notable increase in tritium emissions to the ventilation stack was observed.

Various methods were then used to estimate the remaining tritium inventory of the system [27]. Most of the estimations rely on the assumption that the samples were liberated of their tritium content during the collection process. These estimates further assume that all of the tritium released to the ventilation stack was from the sampled area only.

It is also possible that the samples were not liberated of their content [28] and that not all of the tritium on or in the samples was accessible for analysis. An additional estimate that calculates the maximum amount of tritium that could be present in the metal components of the moderator system was made to account for this possibility.

The results of the estimation methods and the absorbed tritium analysis are presented in Table 26.

It was proposed that the heating and vibrations experienced by the samples may not have been sufficient to release substantial amounts of tritium and that the increase in tritium emissions was a result of increased airflow through the system.

To test this theory, a series of experiments were performed [28]. First, one of the sampling holes was opened up to increase to increase airflow. Second, a sample-sized portion of the system was heated up to 500°C. For both experiments the tritium emissions were monitored, but neither produced comparable emissions to the sampling campaigns.

Based on the above hypothesis testing, it is concluded that neither increased airflow nor heating caused the increased tritium emissions during sample collection and, as such, no inferences will be drawn to which tritium activity best represents the actual conditions. To maintain a consistent conservative approach for the selection of model parameters, the source term values (used for modelling tritium in the environmental assessment)

[17] The identification of ^{235}U and ^{238}U in the H&HWS samples may be due to an analytical false positive as these radionuclides would not be expected to be present in these systems. ICP-MS analyses for uranium were not performed on samples from these systems as was done for other systems and so the gamma spectroscopy analytical results could not be confirmed. ^{238}U quantification was based on the concentration reported for ^{234}Th, one of the short-lived gamma-emitting immediate ^{238}U progeny, and was considered detected when the ^{234}Th result exceeded the detection limit. For conservatism, the calculated total activities for ^{235}U and ^{238}U were carried through as an ROC for the total system inventories.

will be modified to use the most conservative estimate (2.47 x 10^{15} Bq) and the results will be updated as necessary.

6.6 Experimental Loops

6.6.1 Historical Information

The historical upper bound estimate of the residual activity within the experimental loops (EL) was included as part of the estimate for the PHT system which was presented in Section 6.3 and is not repeated here.

6.6.2 2017/2018 In-System Characterization

The EL were combined into a single characterization unit. There were 81 field assessment measurements, from which the 27 intrusive samples were collected. Laboratory analyses detected the following ROCs in at least one sample:

^{3}H, ^{14}C, ^{60}Co, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{108m}Ag, ^{137}Cs, ^{235}U, ^{238}U, ^{238}Pu, $^{239/240}$Pu, ^{241}Am, and $^{243/244}$Cm

The inventory results summary for the EL is provided in Table 27. The 99th percentile UCL total inventory is 1.71 x 10^{9} Bq, of which ^{137}Cs, ^{60}Co, and ^{63}Ni comprise 65%, 21%, and 5% of the total activity, respectively. As the historical inventory for the EL was included with the PHT system, a direct comparison of the characterization results with the historical inventory is not provided. Rather, the characterization inventories for the systems were summed. The result of 4.87 x 10^{10} Bq did not alter the conservatism of the historical upper bound estimate of the combined PHT system, AOG system, and EL inventories (see Table 17).

6.7 Fuel Transfer System

There was no historical estimated inventory for the fuel transfer system.

The 2017/2018 characterization included 27 field assessment measurement locations from which nine intrusive samples were collected. Laboratory analyses detected the following ROCs in at least one sample:

^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{90}Sr, ^{137}Cs, ^{235}U, ^{238}U, ^{238}Pu, $^{239/240}$Pu, and ^{241}Am

The inventory results summary for the fuel transfer system is provided in Table 28. The 99th percentile UCL total inventory is 3.20 x 10^{10} Bq, of which ^{137}Cs and ^{90}Sr comprise 85% and 8% of the total activity, respectively.

6.8 Concrete and Thermal Shield Cooling Water Systems

There was no historical inventory estimate for the concrete and thermal shield cooling water systems.

The 2017/2018 characterization consisted of four intrusive samples. These samples were submitted for screening analysis by gamma spectroscopy at the CNL Whiteshell Environmental Compliance Laboratory. The only gamma-emitting contaminant identified by the analysis was ^{60}Co at a maximum concentration of 5.34 Bq/cm^{2}. Based on the low-level screening results, further analysis was not warranted and an expanded inventory was not calculated as it would provide inconsequential contributions to the WR-1 inventory.

Sampling of the accessible piping in this system revealed the presence of water. Futher investigation indicated that the thermal shield and concrete cooling systems were not fully drained, and remain largely full of water containing detectable levels of tritium. The tritium results from the use of lithium hydroxide as a pH control measure in the cooling water. It is estimated that up to 21 m^{3} of water remains in the system. Laboratory

analysis of the water indicates a tritium concentration between 2.5 x 10^6 and 3.4 x 10^6 Bq/L [29][30]. This translates to a conservative total tritum inventory of 7.1 x 10^{10} Bq, and represents an insignificant contribution to total tritium in the overall facility.

7. NON-RADIOLOGICAL HAZARDOUS MATERIALS

7.1 Order of Magnitude Estimates [31]

An estimate of the non-radiological inventory that is proposed to be encapsulated and disposed of in-situ with WR-1 was completed in March of 2017. The results shown in Table 29 are an order of magnitude estimate based on:

- A review of system descriptions in the WR-1 Handbook [5].
- A review of system specifications in the WR-1 Design Manual [32].
- Anecdotal evidence from reactor operations staff.

The estimation only considers items that are expected to be encapsulated as part of ISD.

For contaminants that were present in liquid form and have since been drained from reactor systems, an estimated residual percentage is assumed to incorporate any small amounts of the contaminants that could reasonably be expected to remain in the system after draining/flushing.

Some items could not be readily estimated due to various reasons. For example, chromium plating was used on many components without any indication of such in available documentation. In such cases, conservative assumptions have been made to provide a reasonable estimate.

7.2 Lead and PCBs [33]

To verify the conservatism of the previously completed order of magnitude estimates, a more extensive examination was performed for lead and polychlorinated biphenyl (PCB) containing materials [33]. This new inventory was based on a desktop review of documentation, analysis of paint samples, and a thorough walkthrough of B100. A total of 21,677 kg of lead is expected to be encapsulated and disposed of in-situ with a variable 514 kg in which a decision has yet to be made. Since this is just over half of the order of magnitude estimate, the conservatism of the previous estimate is confirmed and will continue to be used.

Aside from the light ballasts already identified for removal, the results of the PCB analysis identified no PCB sources above exemption quantities within the ISD envelope. If suspect materials are encountered during ISD, additional samples may be taken to confirm their exemption or identify them for removal.

8. SUMMARY AND DISCUSSION

This document provided a summary of the WR-1 theoretical inventory estimates and compared them with recent characterization sampling and laboratory analysis. The culmination of these evaluations is a comprehensive picture of the current conditions of WR-1 and the radionuclide inventory contained within it. The historical estimates and validation sampling generally align well with one another. The historical estimates typically provided higher total activities and thus are a more conservative upper bound for the remaining radiological inventory of WR-1.

Table 30 provides an overall summary of the historical WR-1 upper bound estimated radioactivity inventory. The inventory is based on the information and evaluations presented in this technical document after adjusting for decay times.

The historical total WR-1 activity decreases by a factor of ~10 from 6,000 TBq (162,000 Ci) to 630 TBq (17,000 Ci) during the decay period between 10 and 100 years after reactor shutdown. The predominant radionuclides are expected to be ^{55}Fe, ^{60}Co, and ^{63}Ni during this time period. The total activity decreases by another factor of ~42 to 15 TBq (408 Ci) at 1,000 years after shutdown with ^{14}C, ^{59}Ni, ^{63}Ni, and ^{94}Nb being the expected predominant radionuclides.

At 10,000 years after shutdown, the activity decreases by another factor of ~4 to 3.4 TBq (95.5 Ci) with the predominant radionuclides being ^{14}C, ^{59}Ni, and ^{94}Nb. The reactor core (calandria vessel and fuel channels) accounts for over 99.7% of the total radioactivity up to 10,000 years. Approximately half of the reactor core activity is due to activation products in the calandria vessel and the other half due to activation products in the fuel channels, primarily the stainless steel fuel channels.

The long-lived activation product ^{36}Cl is a particular ROC for long-term disposal and is expected only to be contained within the WR-1 biological shield. Available characterization information indicates that quantities of ^{36}Cl are trivial (~4,200 Bq (110 µCi)). Long-lived ^{14}C is another radionuclide of interest for long-term disposal and is expected to be found primarily in the reactor core with low levels of surface activity within the components of the H&HWS. ^{14}C radioactivity is dominated by the reactor core with calculated quantities being ~3.0 TBq (~80 Ci) at 100 years after reactor shutdown. The long-lived fission product ^{99}Tc is another radionuclide of interest for long-term disposal. ^{99}Tc radioactivity is only expected to be found in the PHT system and EL as a result of fuel failures and surface deposition. The fission yield for ^{99}Tc is small and the total ^{99}Tc activity that could be remaining in the reactor is not expected to be higher than 0.00013 TBq (3.5 mCi) at 100 years following reactor shutdown. These small quantity assumptions were substantiated during the recent characterization campaign: all composite sample ^{99}Tc analytical results were less than the analytical detection limit.

Figure 8 and Figure 9 provide graphical and tabular summarizes of the 99th percentile UCL ROC inventory calculations, totalling 8.90 x 10^{10} Bq, that were developed using the 2017/2018 characterization campaign laboratory analytical results. The graphs and tables illustrate the percent distribution for all ROCs to the individual system inventory and also the percent that each system contributes to the total inventory. As expected, the characterization verified that the PHT system contain approximately 54% of the residual activity inventory outside of the reactor vault, the majority of which (approximately 90%) consists of ^{90}Sr and ^{137}Cs.

Table 31 summarizes and compares both the inventory of WR-1 calculated through historical work and through recent sampling and validation work. It demonstrates that refinement to the estimated inventory through validation sampling work reduces the total estimated inventory. For conservatism, the more bounding estimates shall be used in any safety or environmental assessment modelling.

Table 32 is a list of historical documents that provide additional reference information on WR-1 that may have relevance in support of WR-1 decommissioning planning activities.

9. ACKNOWLEDGEMENTS

The authors would like to acknowledge the contributions of Tim Vitkus of Oak Ridge Associated Universities.

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Tables

Table 1: Summary description of the primary WR-1 systems

Primary WR-1 Systems	
System	**Description**
Experimental Loops (Rooms 410, 414, 537, 538, 539, 540, 541)	There are four experimental loops in WR-1 and one out-of-reactor hydraulic test loop in the reactor building. Each in-reactor loop consisted of a fuelled test section in a reactor lattice position, and piping equipment and instrumentation in an adjacent loop room to maintain required operating conditions of flow, pressure, and temperature in the test section. A fuel position was converted to a loop by disconnecting the inlet and outlet feeders from the PHT and connecting the feeders to the loop inlet and outlet piping. The WR-1L1 loop was an out-of-reactor hydraulic test facility capable of handling full-sized fuel channels and fuel assemblies. The WR-1L2 loop provided a light water cooled research facility in WR-1. Three fast neutron (FN) loops (WR-1L4/1L5/1L6) occupied the reactor lattice positions D-9, F-9, and E-6.
Primary Heat Transport System (Rooms 103, 104, 302, 409, 503, 506, 535, 602)	Designed to remove the heat produced in the reactor core. The system is divided into three circuits (A, B, and C). The removed heat was dissipated to the Winnipeg River through three conventional tube-and-shell heat exchangers using an organic primary coolant and river water for the secondary coolant. The PHT system had three similar circuits to achieve flexibility for experimental research.
Auxiliary Organic and Gas Systems (Rooms 103, 104, 302, 409, 535, 602)	Systems that supported the operation of the PHT system included degassing and particulate removal systems, the purification system, the relief exhaust system, the organic supply system, and the nitrogen supply system.
Helium and Heavy Water Systems (Rooms 107, 108, 502)	Heavy water was used in WR-1 as a moderator and as a coolant for removal of the gamma and thermal heat picked up by the calandria vessel and tubes. The moderator was supported in the core space by differential helium pressure between the core and dump spaces provided by the helium blowers of the helium system. The moderator level was controlled by regulating the differential pressure by the regulating system. The helium system also provided an inert cover gas which prevented air and water vapour from entering the heavy water system. The following auxiliary systems are included as part of the helium and heavy water systems: the boron addition system, the demineralizer system, the heavy water leak detection system, the heavy water collection system, and the helium gas chromatograph.
Thermal Shield and Concrete Cooling System (Room 504)	Heat was generated in the thermal and radiation shielding immediately surrounding the reactor from nuclear radiation and thermal radiation from the PHT system. The thermal shield cooling system consists of two circulation pumps, a tube-and-shell heat exchanger, a storage/deaerator head tank, piping, instrumentation, and miscellaneous equipment. Circulation pumps were operated with one running and one on standby to circulate water through the radial shield, the top and bottom shutdown shields, and the cooling coils of the ion chamber window. The pumps were driven by motors connected to the Class III bus. Heat was dissipated to the Winnipeg River by the two-bank, tube-and-shell heat exchanger located on the inlet line to the shields.
Spent Fuel Handling and Storage System (Rooms 103, 109, 111, 114, 203)	Irradiated fuel was removed from the reactor core with the large fuel transfer flask. The initial transfer was from the reactor to an organic coolant filled storage tube immersed in the water-filled fuel storage block. The fuel remained immersed in organic coolant throughout this transfer. The fuel storage block was a wet transfer and a short-

Primary WR-1 Systems	
System	**Description**
	term storage facility for freshly irradiated fuel. Irradiated fuel was washed with xylene prior to transfer to the block for inspection. It was also a transfer and storage facility for irradiated fuel channels and experimental equipment. The spent fuel station provided shielding in the reactor hall during fuel transfer operations at the fuel storage block and supported the fuel transfer flask and other equipment over the block. The purpose of the active dry storage system was to provide a storage facility for irradiated or contaminated hardware. The storage facility consists of a concrete shielded pit. The fuel wash down system provided a facility to wash fuel, encase defective fuel, and supply and drain organic coolant used by the fuel transfer flask, the fuel storage block, and the spent fuel storage bays. The wash down system consists of a fuel wash down station, an organic storage and pumping facility, and an organic drainage facility.
Active Drainage System (Rooms 103, 107, 415)	The active drainage system collected liquid effluent from the various areas of the WR-1 facility and the groundwater from around the building base. The system is sub-divided as follows: active drainage sump A - general drainage (1.68 m^3 concrete tank), active drainage sump B - heavy water drainage (2.73 m^3 concrete tank), organic drainage sump A - organic coolant leakage (1.68 m^3 concrete tank), WR-1 extension sump - general drainage WR-1 extension (3.63 m^3 concrete tank), and sub-surface drainage sump - ground water around WR-1 basement (15 m^3 concrete tank).
Heating and Cooling Systems (Rooms 104, 408, 516, 601)	The heating system provided process and building heating for the WR-1 facility. High temperature, high pressure water (HHW) is supplied to WR-1 from the Powerhouse. One main supplies heating for systems in the WR-1 main building and one supplies heating for systems in the building extension. The system consists of several closed loop systems circulating heated propylene glycol solutions or water through ventilation system in-duct heater coils, room heater units, or wall convectors. The main building glycol system (HBY) consists of a closed loop in which a propylene glycol solution is circulated by one of two pumps. The system includes a heat exchanger and pre-heater coils. Chilled building water can be circulated through a heat exchanger. An expansion tank maintains the loop pressure constant during temperature fluctuations and provides make-up to offset minor leakage losses.

Table 2: List of B100 restricted access rooms and areas

B100 Restricted Access Rooms and Areas			
Room	**Description**	**Systems**	**Components**
103	Corridor to Room 111		
104	Degassing	Primary Heat Transport System Heating and Cooling System	Circuit A & B Degassing Tanks, Circuit A & B Volatiles/Water Collection Tanks, Circuit A & B Seal Drain Tanks COW Tanks, Condensers
107	Moderator	Heavy Water System Helium System Auxiliary Organic and Gas Systems Active Drainage System	Dump tank, Helium Accumulator tank, Moderator Heat Exchanger, Moderator Demineralizer System, Heavy Water Priming Flow Cooler Helium Condenser Organic Supply System – Dechlorination Hold Tank Active Drainage Sump B - Heavy Water Drainage
108	Boron Addition Room	Helium and Heavy Water System	Boron Addition
109	Fuel Wash Down Drain Tank Room	Spent Fuel Handling and Storage System Active Dry Storage/Fuel Wash Down System	Fuel Wash Down Systems Heaters
110	Service Pipe Trench		
111	Active Solvent	Spent Fuel Handling and Storage System Active Dry Storage/Fuel Wash Down System Heating and Cooling System	Dump Tank, Storage Tank, Dump Tank, Storage Tank COW Tanks, Condenser
113	Strainer Room		
114	Fuel Wash Down Drain Tank (Upper Extension)	Spent Fuel Handling and Storage System	Fuel Wash Down Systems
115	Access Shaft (NW Corner of RM 601 to NE corner of RM 302 & 104)		
116	Access Shaft (SW Corner of Rm 601 to SW Corner of Rm 107)		
201	Lower Access		
202	CO2 Sample Room – Lower Level		

B100 Restricted Access Rooms and Areas			
Room	**Description**	**Systems**	**Components**
203	Bay Clean-Up	Spent Fuel Bay Circulation System	Change Volume Tank, Bay Water Heat Exchanger, Sand Bed Filter, Ion Exchange Column
301	Flask Maintenance – Upper Level		
301	Flask Maintenance – Lower Level		
302	Degassing	Primary Heat Transport	Circuit A & B Degassing Tanks, Circuit A & B Primary and Secondary Condensers, Circuit A & B Low Boiler Condensers
303	Spent Fuel Bay South		
304	Spent Fuel Bay North		
305	Spent Fuel Work Bay		
306	Fuel Storage Block		
402	Ion Chamber	Neutron Power System	
403	Service Corridor		
404	Activity Monitoring		
406	Transmitter Room		
407	Vault Ventilation		
408	Reactor Building Ventilation	Heating and Cooling System	Condenser
409	Surge Tank & Pipe Shaft	Primary Heat Transport	Circuit A and B Surge Tank
410	1L1 Loop	WR-1L1 Experimental Loop	Make-Up Tank, Loop Cooler
412	Active Dry Storage Pit		
413	Fuel Wash-down Pit		
414/415	Crawlspaces	WR-1L2 Experimental Loop Active Drainage System	Loop Catch Tank WR-1 Extension Sump - General Drainage
501	Upper Access		
502	Moderator Ion Exchange Pit	Heavy Water System	Ion Exchange Columns
503	Organic Purification Pit	Primary Heat Transport	Circuit A & B Absorption Columns, Circuit A & B Filters

B100 Restricted Access Rooms and Areas			
Room	**Description**	**Systems**	**Components**
504	Auxiliaries	Thermal Shield Cooling System	Head Tank, Chemical Addition Tank, Head Tank, Chemical Addition Tank, Heat Exchanger
506	Header	Primary Heat Transport	Circuit A & B Inlet Header, Circuit A & B Outlet Header
508	Spent Fuel Storage	Spent Fuel Handling and Storage System	
535	Vented Fuel Facility	Primary Heat Transport	Circuit A & B Heat Exchangers, Particulate Removal Circuit Cooler
536	Loop Transmitter Room	Fast Neutron Loops	
537	1L5 Loop	Fast Neutron Loop	Surge Tanks, Educator Tank
538	1L4 Loop	Fast Neutron Loop	Reflux Boiler, Boiler
539	1L2 Loop	WR-1L2 Experimental Loop Fast Neutron Loop	Surge Tank, Loop Degas, Steam/Water Separator Vessel, Heat Exchangers and Coolers Boiler
540	1L2 Sample Station & Transmitter	WR-1L2 Experimental Loop	Chemical Addition Tank
541	1L2 Auxiliary	WR-1L2 Experimental Loop	Surge Tank, Heat Exchangers and Coolers
543			
601	Caged Area		
601	50 Ton Crane		
602	A & B Primary Pumps	Primary Heat Transport	Circuit A, B, & C Main Heat Exchangers

Table 3: Summary of the radiological hazard survey results for B100 rooms

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm²)	
103	Drain Tank Room	Primary Heat Transport Spent Fuel Handling and Storage System Active Drainage System	Circuit A and B Drain Tank Active Dry Storage/Fuel Wash Down System - Washing Cell Drain Filter Active drainage sump A - general drainage, Organic drainage sump A	Low	6 PAD results from room characterization for Asbestos Abatement: Average dose rate: 3.46 Max average dose rate: 4.95 Max dose rate: 56.0 Average max dose rate: 36.40	200	Minimal	300/cos 9 of 10 swipes COS (Asbestos characterization)	NAD	3 swipes, max values: Cs-137: 0.03 Asbestos characterization campaign: 10 swipes, max values: Cs-137: 0.004	Feb 2015 Chain of Custody: Room 104-103-302-WR1-Scoping-2014-01-20-CB-001 COC WR1-2015-06-29
103	Corridor to Room 111			Minimal	0.05-0.2	16 mrem/h nc, 8 mrem/h @ 30 cm, 2 mrem/h @ 1 m, opening to drain tank room	Likely Clean	COS/COS	-	Organic sludge from APO PM2 Cs-137: 169 Bq/sample Am-241: 15 Bq/sample Co-60: 0.42 Bq/sample Eu-154: 1.8 Bq/sample	Feb 2016 (dose rate) Chain of Custody: B100-Rm 103-WR-1 Scoping-2014-01-14-CB-001 B100-Rm 103-WR-1 Scoping-2014-01-14-CB-003 Lowest Cs-137: Am-241 Ratio: 11:1
104	Degassing	Primary Heat Transport Heating and Cooling System	Circuit A &B Degassing tanks; Circuit A&B Volatiles/Water Collection Tanks; Circuit A&B Seal Drain Tanks COW Tanks, Condensers	Low	4-9 PAD results from room characterization for Asbestos Abatement: Average dose rate: 3.46 Max average dose rate: 4.95	40: AD0-P1 and AD0-P2 motor 600: D0-TK1 bottom north 60: AD0-P1 pump 40: B standpipe 300: AD0-ST1	Low	2k/cos	NAD	9 swipes, max values: Cs-137: 0.6 Am-241: 0.02 Eu-154: 0.003 1 organic sludge sample: 10 Bq/sample	Feb 2015 Chain of Custody: Room 104-103-302-WR1-Scoping-2014-01-20-CB-001 WR1-2015-06-29-AW001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm²)	
					Max dose rate: 56.0 Average max dose rate: 36.40	25: BD0-ST2 20: BD0-P2 55: AD0-P2 pump 25: BD0-P1 motor 150: AD0-ST2 80: between A/BDO-TK1 at floor level 15: BD0-P2 motor 1000: AD0-TK1 bottom south 60: BD0-ST1 200: AD0-TK1 underneath at 1 m height 80: A standpipe 50: BD0-TK1 bottom 30: BD0-P2 pump				Activity balance on some swipes indicates a pure beta emitter present Asbestos characterization campaign: 27 swipes, max values: Cs-137: 0.03 Am-241: 0.001	Max Cs-137: Am-241 ratio: 30:1
107	Moderator	Heavy Water System Helium System AO&GS Active Drainage System	Dump tank, Helium Accumulator tank, Moderator Heat Exchanger, Moderator Demineralizer System; Heavy Water Priming Flow Cooler Helium Condenser, Organic Supply System – Dechlorination Hold Tank	Minimal	0.05-0.2	10 mrem/h nc, 0.3 mrem/h @ 30 cm check valve at MH V6 13 mrem/h nc, 3 mrem/h @ 30 cm, 0.06 @ 1 m, at waist height bottom of valve, above and right of drip tray 2 mrem/h nc, 0.2 mrem/h @ 30	Minimal	200 cpm β bench/cos 15 swipes COS (Asbestos characterization)	-		Jan 2015 Feb 2016 (dose rate) Chain of Custody: WR1-107-Tritium-2015-05-28-TB-02 WR1-107-Tritium-2015-06-01-TB-001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
			Active drainage sump B - heavy water drainage			cm, 0.1 @ 1 m, between two tanks 1.3 mrem/h nc, 1.1 mrem/h @ 30 cm, 0..05 mrem/h @ 1m, far right upon entry, just before stairs.					WR1-107-Tritium-2015-06-02-TB-001 WR1-107-Tritium-2015-06-03-TB-001 WR1-107-Tritium-2015-06-04-TB-001 WR1-107-Tritium-2015-06-05-TB-001 (2) WR1-107-Tritium-2015-06-10-TB 02 WR1-107-Tritium-2015-06-10-TB-03 WR1-107-Tritium-2015-06-15-TB 01 WR1-107-Tritium-2015-06-16-TB-01 WR1-107-Tritium-2015-06-16-TB-02 WR1-107-tritium-2015-06-17-TB-002 WR1-107-Tritium-2015-06-18-TB-001 WR1-107-Tritium-2015-06-18-TB-002

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
											WR1-107-tritium-2015-06-22-TB-002 WR1-107-tritium-2015-06-22-tb-003 WR1-107-Tritium-2015-06-24-TB-001 WR1-107-Tritium-2015-07-06-TB-001 WR1-107-Tritium-2015-07-06-TB-002 WR1-Tritium-2015-06-17-TB-01 WR-107-tritium-2015-05-28-TB-001 WR-107-tritium-2015-05-28-TB-003
108	Boron Addition Room	Heavy Water and Helium System	Boron Addition	Minimal	0.05-0.2	30 mrem/h nc, 10 mrem/h @ 30 cm, 4 @ 1m, process drain	Likely Clean	COS/COS 6 swipes COS (Asbestos characterization)	-		Jan 2015 Feb 2016 (dose rate)
109	Fuel Wash Down Drain Tank Room	Spent Fuel Handling and Storage System Active Dry Storage/Fuel Wash Down System	Fuel Wash down Systems Heaters	Minimal	0.5	30 mrem/h nc, 10 mrem/h @ 30 cm, 1.5 @ 1m, process drain and under TK-3	Likely Clean	COS/COS 24 swipes COS (Asbestos characterization)	-		Jan 2015 Feb 2016 (dose rate)

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
110	Service Pipe Trench			Low	0.5-2.0	30 mrem/h nc, 10 mrem/h @ 30 cm, 1.5 @ 1m, process drain	Likely Clean	1k/COS 26 swipes COS (Asbestos characterization)	-	1 Swipe: Cs-137: 0.23	Jan 2015 Feb 2016 (dose rate) Chain of Custody: B100-Rm 110-WR-1 Scoping-2014-01-14-CB-002
111	Active Solvent	Spent Fuel Handling and Storage System Active Dry Storage/Fuel Wash Down System Heating and Cooling System	Fuel Wash down Systems Dump Tank, Storage Tank COW Tanks, Condenser	Minimal	0.2	5 mrem/h nc, 0.5 mrem/h @ 30 cm, 0.25 mrem/h @ 1m along piping, from AD NV19 overhead north to AD-V5 west (active drain)	Minimal	500 cpm β 7 swipes COS (Asbestos characterization)	-		Jan 2015 Feb 2016 (dose rate)
113	Strainer Room				0.05	0.05	Likely Clean	COS/COS 4 swipes COS (Asbestos characterization)	-		Jan 2015
114	Fuel Wash Down Drain Tank (Upper Extension)	Spent Fuel Handling and Storage System	Fuel Wash down Systems	Low	1	6 mrem/h nc, 2 mrem/h @ 30 cm washdown tank	Likely Clean	COS/COS 5 swipes COS (Asbestos characterization)	-		Jan 2015
115	Access Shaft (NW Corner of RM 601 to NE corner of RM 302 and 104)			--	--	Unable to enter	--	Unable to enter	-		No ACM Inaccessible area
116	Access Shaft (SW corner of Rm 601 to SW			--	--	Unable to enter	--	Unable to enter	-		No ACM Inaccessible area

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
	corner of Rm 107)										
201	Lower Access			Moderate	25 PAD results from room characterization for Asbestos Abatement: Average dose rate: 2.10 Max average dose rate: 2.40 Max dose rate: 9.0 Average max dose rate: 7.77	100	Low	5k/COS Asbestos characterization: 3 swipes with up to 20 cpm alpha and 1kcpm beta/gamma 7 swipes COS	500 cpm on rubbers through 2 layers of plastic bags	Asbestos characterization campaign: COC WR1-2015-07-14-AW-001: 1000 cm^2, 3 swipes with 0.008 to 0.04 Bq/cm^2 Cs-137, 3 swipes with 0.002 to 0.004 Bq/cm^2 Co-60, 1 swipe with 0.002 Bq/cm^2 Am-241	Nov 2011
202	CO2 Sample Room – Lower Level			Minimal	0.05	11 8 mrem/h nc, 0.5 mrem/h @ 30 cm, 0.15 @ 1 m, G8, copper pipe near door to Rm 201 43 mrem/h nc,25 mrem/h @ 30 cm, 5 @ 1 m, at 301 door, upper floor	Likely Clean	COS/COS Asbestos Characterization: Lower: Direct: 500 to 1k cpm Loose: COS (5 swipes) Upper: Direct: NA Loose: COS (5 swipes on pipes and elbows)	NAD	21 swipes, max values: Cs-137: 0.15 Am-241: 0.005 Co-60: 0.07 Activity balance on some swipes indicates a pure beta emitter present	Jan 2015 No ACM Feb 2016 (dose rate) Chain of Custody: B100-WR-1-Scoping-2014-02-11-CB-001 Lowest Cs-137: Am-241 Ratio: 30:1
203	Bay Clean-Up	Spent Fuel Bay Circulation System	Change Volume Tank, Bay Water Heat Exchanger, Sand Bed Filter, Ion Exchange Column	Minimal	0.5	40 mrem/h nc, 3 mrem/h @ 30 cm, 0.2 @ 1 m, BYW TK1 pipe at bottom left 1.5 mrem/h nc, 0.6 mrem/h @ 30 cm, 0.5 @ 1 m, Process Drain	Likely Clean	450/COS Asbestos Characterization: Direct: NAD/NA Loose: COS (18 swipes)	NAD (upper and lower level)	1 swipe: Cs-137: 0.06	Jan 2015 Feb 2016 (dose rate) Chain of Custody: B100-2014-01-15-TR-001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
						line (PD), far right upon entry					
301	Flask Maintenance – Upper Level			Low	3	3	Minimal	500/COS	-	6 swipes, max values: Cs-137: 19.1 Am-241: 0.92 Eu-154: 0.21 Activity balance on some swipes indicates a pure beta emitter present	Jan 2014 No ACM Chain of Custody: B100-WR-1-Scoping-2014-01-31-CB-001 Lowest Cs-137: Am-241 Ratio: 20:1
301	Flask Maintenance – Lower Level			High	100	5000: stored radioactive source	Minimal	500/COS	-		July 2011
302	Degassing	Primary Heat Transport	Circuit A&B Degassing Tanks; Circuit A&B Primary and Secondary Condensers; Circuit A&B Low Boiler Condensers	Low	3-8 PAD results from room characterization for Asbestos Abatement: Average dose rate: 1.86 Max average dose rate: 2.13 Max dose rate: 12.6 Average max dose rate: 10.85	1000	Minimal	150/COS Asbestos Characterization: Direct: NA Loose: COS (23 swipes)	NAD through 2 layers of plastic bags	9 swipes, max values: Cs-137: 0.02	Feb 2015 Chain of Custody: Room 104-103-302-WR1-Scoping-2014-01-20-CB-001
303	Spent Fuel Bay South										Inaccessible area
304	Spent Fuel Bay North										Inaccessible area
305	Spent Fuel Work Bay										Inaccessible area
306	Fuel Storage Block									10 swipes, max values: Cs-137:5.72	Jan 2014 Chain of Custody:

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
										Am-241:0.05 Eu-154: 0.002 Co-60: 0.006 Activity balance on some swipes indicates a pure beta emitter present	B100-WR-1-Scoping-2014-01-31-CB-001 Lowest Cs-137: Am-241 Ratio: 25:1
402	Ion Chamber	Neutron Power System		Minimal	0.05-0.2	6 mrem/h nc, 2 mrem/h @ 30 cm, 0.4 mrem/h @ 1 m, 6 feet up wall in front of door	Minimal	Up to 60k β on lip of hatch Asbestos Characterization: Direct: NAD (north wall), NA (south wall) Loose: COS (10 swipes)	-	3 swipes taken, max values: Cs-137: 80.5 Am-241: 1.9 Pa-234: 0.05 Eu-154: 0.04 Co-60: 0.006	Jan 2015 Feb 2016 (dose rate) Chain of Custody: B100-2014-01-09-JL-001 B100-2014-01-10-JL-001 Lowest Cs-137:Am-241 ratio 42:1
403	Service Corridor			Minimal	0.1	0.1	Moderate	25k (on swiffer) /cos Asbestos Characterization: Direct: NAD Loose: COS (1 swipe)	NAD on low rubbers	5 swipes taken Cs-137: 3.68	Feb 2016 Chain of Custody: B100-WR-1-SCOPING-2014-02-20-CB-001 B100-RM 403-2014-05-16-TB-001
404	Activity Monitoring			Low	10 PAD results from room characterization for Asbestos Abatement: Average dose rate: 2.35 Max average dose rate: 4.40 Max dose rate: 17.5	150: Activity monitoring lines – east side 250: Header overhead 40: Activity monitoring lines – west side	Likely Clean	COS/COS Asbestos Characterization: Direct: NA Loose: COS (10 swipes)	NAD	10 swipes, max values: Cs-137: 0.001 Co-60: 0.0007	July 2012 Chain of Custody: RM 538-404-410-WR1-Scoping-2014-03-10-CB-001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
					Average max dose rate: 10.30						
406	Transmitter Room			Minimal	0.1	8	Likely Clean	COS/COS Asbestos Characterization: Direct: NA Loose: COS (11 swipes on ACM, 1 swipe of floor drain)	NAD on low rubbers		Feb 2015
407	Vault Ventilation			Minimal	0.1	0.5	Likely Clean	COS/COS Asbestos Characterization: COS (1 swipes on pipe and 2 on surrounding ducting)	NAD on rubbers		Feb 2015 Chain of Custody: B100-RM 407-WR-1-Scoping-2014-02-07-CB-001
408	Reactor Building Ventilation	Heating and Cooling System	Condenser	Minimal	0.2-0.5	60 mrem/h nc, 7 mrem/h @ 30 cm, 2 mrem/h at 1 m, process drain line (PD) far back left upon entry. 2-5 mrem/h nc, 1.2 mrem/h @ 30 cm, 0.3 mrem/h at 1 m, PD lie, east/west overhead upon entry, entire length of PD line	Likely Clean	COS/COS Asbestos Characterization: Direct: NAD/ NA Loose: COS (15 swipes)	-		Jan 2015 Feb 2016 (dose rate)
409	Surge Tank & Pipe Shaft	Primary Heat Transport	Circuit A and B Surge Tank	Low	5	40	Low	Asbestos Characterization: Direct: NA	NAD		July 2012

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm²)	
								Loose: COS (11 swipes)			
410	1L1 Loop	WR-1L1 Experimental Loop	Make-Up Tank, Loop Cooler	Low	1 PAD results from room characterization for Asbestos Abatement: Average dose rate: 2.35 Max average dose rate: 4.40 Max dose rate: 17.5 Average max dose rate: 10.30	2	Likely Clean	COS/COS Asbestos Characterization: Direct: NA Loose: COS (number of swipes not specified)	NAD	10 swipes, max values: Cs-137: 0.001 Co-60: 0.0007	Feb 2015 Chain of Custody: RM 538-404-410-WR1-Scoping-2014-03-10-CB-001
412	Active Dry Storage Pit										Inaccessible area
413	Fuel Wash-down Pit										Inaccessible area
414/415	Crawl space	WR-1L2 Experimental Loop Active Drainage System	Loop Catch Tank WR-1 extension sump - general drainage	Minimal	0.05	14 mrem/h nc, 5 mrem/h @ 30 cm, 0.2 mrem/h at 1 m, right of stairwell upon entry	Low	2k β	NAD on rubbers	Asbestos characterization campaign: 100 Swipes taken in 414, max values: Cs-137: 0.306 46 swipes taken in 415: COS	Jan 2015 Feb 2016 (dose rate) Removable contamination on lid of Sump 415 pit and cables Chain of Custody: COC: WR1-2015-07-06-JG-001
501	Upper Access			Low	5	200	Low	1.5k/20	NAD	15 swipes, max values: Cs-137: 0.35 Am-241: 0.02 Nb-94: 0.02 Eu-154: 0.002	Feb 2015 Chain of Custody: B100-WR-1-Scoping-2014-01-31-CB-001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
										Co-60: 0.01 Activity balance on some swipes indicates a pure beta emitter present Asbestos characterization campaign: 10 swipes, max values: Cs-137: 0.04 Am-241: 0.002 Co-60: 0.08 Nb-94: 0.002	WR1-2015-07-09-AW001 WR1-2015-07-09-AW001 COC Lowest Cs-137: Am-241 Ratio: 20:1
502	Moderator Ion Exchange Pit	Heavy Water System	Ion Exchange Columns	--	--	Sealed – unable to enter	--	--	-		No ACM Inaccessible area
503	Organic Purification Pit	Primary Heat Transport	Circuit A&B Absorption Columns; Circuit A&B Filters	--	--	Sealed – unable to enter	--	--	-		No ACM Inaccessible area
504	Auxiliaries	Thermal Shield Cooling System	Head Tank, Chemical Addition Tank, Head Tank, Chemical Addition Tank, Heat Exchanger	Low	1-4 PAD results from room characterization for Asbestos Abatement: Average dose rate: 0.34 Max average dose rate: 0.34 Max dose rate: 3.5 Average max dose rate: 2.63	20	Low	2k β Asbestos Characterization: Direct: N/A Loose: COS (10 swipes)	500 cpm beta/gamma on low rubbers through 2 layers of plastic bags		Nov 2011
506	Header	Primary Heat Transport	Circuit A and B Inlet Header; Circuit A and B Outlet Header	Low	5 PAD results from room characterization for Asbestos Abatement: Average dose rate: 1.23	15 300: AP0-V25 300: BP0-V25 200: A Header return 4000: AP0-V5	Low	2k β Asbestos Abatement Upper Header: Direct: NA Loose: COS (8 swipes)	NAD	Asbestos characterization campaign: 15 swipes taken in Lower Header, max value: Cs-137: 0.01 8 Swipes of Upper Header: COS	Feb 2015 Chain of Custody: B100-2014-01-13-JH-001 B100-2014-01-14-JH-001

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
					Max average dose rate: 1.62 Max dose rate: 15.8 Average max dose rate: 8.26			Asbestos Abatement Lower Header: Direct: N/A Loose: 1 biased swipe with 10 cpm alpha, 400 cpm beta/gamma 14 biased swipes COS			WR1-2015-07-14-AW-001
508	Spent Fuel Storage	Spent Fuel Handling and Storage System		Low	0.02-0.05	2 mrem/h nc active drain, 1 mrem/h nc B200 pump out line	Likely Clean	10k/200 Asbestos Abatement: Direct: NAD/NA Loose: COS (15 swipes)	-	8 swipes taken: Cs-137: 2.47 Am-241: 0.01 Co-60: 0.001 Activity balance on some swipes indicates a pure beta emitter present	Jan 2015 Chain of Custody: B100-2014-01-13-JH-001 B100-2014-01-14-JH-001 Lowest Cs-137: Am-241 Ratio: 7:1
535	Vented Fuel Facility	Primary Heat Transport	Circuit A&B Heat Exchangers - Particulate Removal Circuit Cooler	Minimal	0.1	2	Minimal	500 β Asbestos Abatement: Direct: NAD Loose: COS (10 swipes)	NAD	3 swipes, max values: Cs-137: 2.9 Co-60: 0.002	Feb 2015 Chain of Custody: B100-WR-1-Scoping-2014-02-12-CB-001
536	Loop Transmitter Room	Fast Neutron Loops		Minimal	0.2	0.4	Likely Clean	COS/COS Asbestos Abatement: Direct: NAD Loose: COS (10 swipes)	NAD		Feb 2015
537	1L5 Loop	Fast Neutron Loops	Surge Tanks, Educator Tank	Low	1-5 PAD results from room characterization for Asbestos Abatement:	400: pipe elbow HMV6-1L5	Minimal	200 β Asbestos Abatement: Direct: NA	NAD	11 swipes, max values: Cs-137: 051 Co-60: 0.95	Feb 2015 Chain of Custody:

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
					Average dose rate: 0.31 Max average dose rate: 0.40 Max dose rate: 2.2 Average max dose rate: 2.08			Loose: COS (10 swipes)			B100-WR-1-Scoping-2014-01-31-CB-001
538	1L4 Loop	Fast Neutron Loops	Reflux Boiler, Boiler	Low	1 PAD results from room characterization for Asbestos Abatement: Average dose rate: 5.38 Max average dose rate: 0.51 Max dose rate: 4.8 Average max dose rate: 4.80	5	Minimal	500 β	500 on low rubbers through 2 layers of plastic bags	10 swipes, max values: Cs-137: 0.3 Co-60: 0.0007 Asbestos characterization campaign: 20 Swipes, Max values: Cs-137: 0.03 Am-241: 0.0001	Feb 2015 Chain of Custody: RM 538-404-410-WR1-Scoping-2014-03-10-CB-001 WR1-Asbestos Abatement-2015-07-07-AW-001 Lowest Cs-137: Am-241 ratio: 300:1
539	1L2 Loop	WR-1L2 Experimental Loop Fast Neutron Loops	Surge Tank, Loop Degas, Steam/Water Separator Vessel, Heat Exchangers and Coolers Boiler	Moderate	10 PAD results from room characterization for Asbestos Abatement: Average dose rate: 0.61 Max average dose rate: 0.70 Max dose rate: 4.8 Average max dose rate: 3.65	60	Moderate	80k/400	20k cpm on low rubbers through 2 layers of plastic bags. 1.5 kcpm on waste through 3 layers of plastic bags	12 swipes, max values: Cs-137: 10.3 Am-241: 0.16 Pa-234: 0.05 Eu-154: 0.04 Co-60: 0.006 Asbestos characterization campaign: 8 swipes, max values: Cs-137:0.9 Am-241: 0.005 Co-60: 0.2	Jan 2014 Chain of Custody: B100-WR-1-Scoping-2014-01-31-CB-001 WR1-Asbestos Abatement-2015-07-07-AW-001 Lowest Cs-137: Am-241 Ratio: 50:1

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
540	1L2 Sample Station & Transmitter	WR-1L2 Experimental Loop	Chemical Addition Tank	Minimal	0.2	8: fumehood drain	Minimal	25k/cos	NAD	6 swipes taken, max values: Cs-137: 18.5 Co-60: 0.06 Activity balance on some swipes indicates a pure beta emitter present Asbestos characterization campaign: 10 swipes, max values: Cs-137:0.4	Feb 2015 Chain of Custody: B100-2014-01-17-JH-001 WR1-2015-07-14-AW-001
541	1L2 Auxiliary	WR-1L2 Experimental Loop	Surge Tank, Heat Exchangers and Coolers	Minimal	0.05	1.5	Minimal	200 β	NAD	6 swipes taken, max values: Cs-137:1.42 Asbestos characterization campaign: 8 swipes, max value: Cs-137:0.03	Feb 2015 Chain of Custody: B100-2014-01-17-JH-001 WR1-2015-07-14-AW001
543										1 swipe taken Activity balance on some swipes indicates a pure beta emitter present	Feb 2014 Chain of Custody: B100-WR-1-Scoping-2014-02-12-CB-001
601	Caged Area			Minimal	0.02-0.5	80 mrem/h nc, 3 mrem/h @ 30 cm, 0.5 mrem/h at 1 m, west end of flask (shielded) 5 mrem/h nc, 0.8 mrem/h @ 30 cm, 0.5 mrem/h at 1 m, east end of flask	Likely Clean	COS/COS Asbestos Abatement: Direct: NA Loose: COS (4 swipes)		Asbestos characterization campaign: 5 swipes taken, max values: Cs-137: 0.04 Am-241: 0.001	Feb 2016 (dose rate) Chain of Custody: WR1-2015-07-06-JG-002 Lowest Cs-137: Am-241 ratio: 40:1

Room	Description	System	Components	Gamma Radiation Hazards			Contamination Hazards				Last Survey Date/ Comments
				Hazard Grade	GA Dose Rates (mrem/h)	Maximum/Elevated Dose Rates (mrem/h near contact)	Hazard Grade	Removable Surface Contamination (cpm on-swipe) Beta/Alpha	Contamination found on PPE after room entry (Ludlum 44-9 unless otherwise noted)	Gamma Analysis Results (Bq/cm^2)	
601	50 Ton Crane			Minimal	0.02	0.2	Likely Clean	COS/COS	-		Feb 2015
602	A & B Primary Pumps	Primary Heat Transport	Circuit A, B, C main heat exchangers	Low	5-20 PAD results from room characterization for Asbestos Abatement: Average dose rate: 2.30 Max average dose rate: 3.94 Max dose rate: 121.0 Average max dose rate: 33.01	10 rem/h	Minimal	200 β	NAD	18 swipes, max values: Cs-137: 0.016	Feb 2015 Chain of Custody: RM 602-WR1-Scoping-2014-03-12-CB-001 Should remove garbage can in south west corner prior to doing any work in 602 or the upper area of 506, as most dose rates in this room are coming from there.

Table 4: Component details for the degassing systems

Component		Location	Purpose	Dimensions	Material
Name	**Identification**				
Degassifier Tank	A/B/CDO-TKI	104/302 (528/647)	To degas the PHT system coolant and provide a coolant reservoir	6 ft. O.D. x 12 ft. (7 ft. O.D. x 10.82 ft.)	C.S. SA-285 Gr C (C.S. AS15 Gr 70)
Volatiles & Water Collection Tanks	A/BDO-TK16 (CDO-TK 6) A/B/CDO-TKJ.7	104 (528)	To collect volatiles/water condensed by the low boiler condenser	16 in. O.D. x 40 in. (24 in. O.D. x 48 in.)	C.S. Al06 Gr B
Adsorption Columns	A/B/CDO-TK2/3	503 (528)	To remove particulate fouling material from the coolant	20 in. O.D. x 60 in.	C.S. SA285 Gr C (C.S. A515 Gr 70)
Inventory Storage Tank	CDO-TK18	528	To contain the surplus coolant from CDO-TKI during fueling operations on C circuit	54 in. O.D. x 76 in.	C.S. A515 Gr 70
Seal Drain Tank	DO-TK2	104	To collect seal drainings from A/B circuit degassing pumps	6 in. dia. By 17 ft.	C.S. ASTM AI06 GR B pipe
Drain Tank	CDO-TK4	528	To collect particulate removal system and other returnable drainings from C circuit	2 ft. O.D. by 4 ft.	C.S. ASTM A515 Gr 70
Waste Drain Tank	CDO-TKS	528	To collect seal drainings from C circuit and 1L6 loop	2 ft. O.D. by 5 ft.	C.S. ASTM A515 Gr 70
Main Condenser	A/B/CDO-CD1	302 (647)	First stage condenser for vapours from degassifier tank	8.625 in. O.D. x 81 in.	C.S. Al06 Gr B
Secondary Condenser	A/B/CDO-CD2	302 (647)	Second stage condenser for vapours from degassifier tank	6.625 in. O.D. x 81.625 in.	C.S. SA285 Gr C
Low Boiler Condenser	A/BDO-CD3 (CDO-HXl)	302 (647)	To condense low boilers and water vapour from the degassifier tank exhaust	9 (24) in. dia. By 53 (48) in.	C.S. ASTM Al06 Gr B (A5l5 Gr 70)
Heat Exchanger Standby Cooler	A!B/CDO-HX2	104 (528)	To provide standby cooling for the PHT system (To cool the particulate removal circuit flow and to supply standby cooling for the PHI system)	10.75 in. 0.0. x 96 in. (10.75 in. 0.0. x 80 in.)	C.S. Al06 Gr B
Heat Exchanger Particulate Removal Circuit Cooler	A/BDO-HX3	535	To cool the particulate removal flow to 300°C	12 in. O.D. x 26.8 in.	C.S. Sal06 Gr B
Back-Up Filters	A/B/CDO-FR CDO-FR2	503 (532)	To remove fouling particulates > 1 μm in size from the coolant	13.25 in. O.D. x 62 in.	C.S. Al06 Gr B

Note: Based on Reference [5]. The C circuit information is shown in brackets where different from the A and B circuits.

Table 5: Characteristics of potential radiological contaminants of concern

Activation Products

^{3}H	This radionuclide can be produced in a reactor by several mechanisms. Neutron capture in deuterium in D_2O moderators is a major source of production in reactors using D_2O. The concrete bioshield is also a source of production from the $^{6}Li(n, a)^{3}H$ reaction with a 953 barns (b) cross-section. Tritium decays (half-life: 12.33 y) by β – emission (maximum energy: 19.0 keV), and is a pure β emitter. Tritium in the form of water vapour is extremely mobile in nature and readily exchanges with water in human tissue. The low β energy and the lack of γ emission adds to the difficulty in measuring and assessing levels of tritium in air and other materials.
^{14}C	This radionuclide is mainly produced by the activation of trace nitrogen by the $^{14}N(n, p)^{14}C$ reaction with a cross-section of 1.81 b. Additional minor routes are via $^{13}C(n, \gamma)^{14}C$ from 1.1% abundant ^{13}C with a cross-section of 0.9 millibarns (mb) and ^{12}C (98.89%, 3.4 mb) indirectly via ^{13}C. ^{14}C decays (half-life: 5,730 y) by β– emission (maximum energy: 156 keV), and is a pure β emitter. Nitrogen is present in air and in most reactor construction materials and, through activation, generates a significant contribution to the overall radioactive inventory, particularly in concretes and graphite.
^{36}Cl	This radionuclide is principally produced by neutron capture from the reaction $^{35}Cl(n, \gamma)^{36}Cl$ with a cross-section of 0.04 mb. Another method of production is by the $^{39}K(n, \alpha)^{36}Cl$ reaction with a cross-section of 2 b. ^{36}Cl is also produced indirectly via ^{34}S. ^{36}Cl decays (half-life: 3.01×10^{5} y) principally by β– emission (maximum energy: 709 keV). ^{36}Cl also decays by electron capture with the emission of some weak X-rays. ^{36}Cl is present via activation of trace chlorine in most reactor construction materials, e.g. in stainless steel and aluminium reactor components, and is important from the viewpoint of disposal, because of its long half-life, the solubility of chloride salts, low retardation in the geosphere, and potential pathways to humans from a waste repository.
^{55}Fe	This radionuclide is produced by the $^{54}Fe(n, \gamma)^{55}Fe$ reaction in the 5.9% abundant stable iron isotope ^{54}Fe with a cross-section of 2.25 b. ^{55}Fe is a hard-to-measure radionuclide that decays (half-life: 2.73 y) by electron capture (weak X-ray emissions) to ^{55}Mn. After production of ^{55}Fe in a reactor core, translocation of this and other radionuclides from the reactor vessel through the coolant system will be a function of corrosion and deposition rates. Since under typical reactor conditions carbon steels are more susceptible to corrosion than stainless steel or nickel alloys (e.g. Inconel), ^{55}Fe should be more abundant in the translocated inventory in reactors employing larger relative amounts of carbon steel that typically contain more ^{55}Fe in the corrosion films. The abundance of translocated ^{55}Fe will also be affected by the chemical controls maintained in the coolant loops, e.g. pH and oxygen levels, since these influence the corrosion rates.
^{59}Ni	This radionuclide is produced by the $^{58}Ni(n, \gamma)^{59}Ni$ reaction in the 68.3% abundant isotope ^{58}Ni with a cross-section of 4.6 b. ^{59}Ni is a hard-to-measure radionuclide that decays (half-life: 76,000 y) by electron capture (continuous X spectrum from inner bremsstrahlung to 1.07 MeV) to ^{59}Co. Because of a rather substantial production rate of ^{59}Ni in all parts of the inner containment, this isotope is thus one of the limiting radionuclides for dose considerations after the decay of other short lived residual radionuclides has occurred. ^{59}Ni is considered an important radionuclide for waste disposal.
^{63}Ni	This radionuclide is produced by the $^{62}Ni(n, \gamma)^{63}Ni$ reaction in the 3.6% abundant isotope ^{62}Ni with a cross-section of 14.2 b. ^{63}Ni. This is a hard-to-measure radionuclide that decays (half-life: 100.1 y) by β– emission (maximum energy: 67 keV) to the ground state of ^{63}Cu. It is considered an important radionuclide for waste disposal. Nickel alloys, Monel and copper–nickel were used in heat exchangers in some early reactors. Inconel (60–80% Ni) has been used extensively in reactor systems, both for reactor internals and heat

	exchanger surfaces; sometimes, for these surfaces, Monel (67% Ni) and copper–nickel (30% Ni) are applied.
^{60}Co	This radionuclide is produced by the ^{59}Co(n, γ)^{60}Co reaction in the 100% abundant stable cobalt isotope ^{59}Co with a cross-section of 18.7 b. ^{60}Co decays (half-life: 5.27 y) by β– emission (maximum energy: 318 keV) to excited levels of ^{60}Ni and produces two major γ-rays: 1.17 MeV and 1.33 MeV. Cobalt is a trace constituent in both carbon and stainless steels (ranging from 80 to 150, and 230 to 2600 ppm, respectively). Cobalt is also present in Inconel and Monel. This isotope is the dominant dose producing radionuclide in the reactor interior on a 10 to some 50 year timescale. The production rate of ^{60}Co is sufficiently high in the high flux region near the core that a substantial portion of the stable cobalt (up to one third) may be transmuted over the life of the reactor. After production in the reactor core, translocation and deposition of ^{60}Co throughout the reactor systems will be a function of (1) corrosion controls, (2) the effectiveness of the reactor coolant cleanup system, and (3) the radioactive waste management practices.
^{94}Nb	This radionuclide is produced by the ^{93}Nb(n, γ)^{94}Nb reaction in the 100% abundant stable isotope, which has a cross-section of 1.15 b. ^{94}Nb decays (half-life: 20,300 y) by β– emission (maximum energy: 472 keV) to a single level of ^{94}Mo at 1.574 MeV, which goes to the ground state via a 871 and 703 keV γ cascade. The presence of relatively high levels of Nb in stainless steel (5–300 ppm), Inconel (400–50,000 ppm), pressure tubes, or fuel channels would lead to the production of significant amounts of the very long lived ^{94}Nb in reactor core materials. For long deferral intervals before decommissioning, ^{94}Nb may in fact represent the principal contributor to the personnel exposure during the dismantling of the reactor pressure vessel. However, the extreme insolubility of Nb does not permit significant translocation from the pressure vessel and deposition in other plant systems. Therefore, ^{94}Nb has been a very minor constituent of the residual radionuclide contamination of the plant circuits.
^{133}Ba	Barium-133 (half-life: 10.7 y) is produced by a combination of epithermal neutron capture reactions on 0.1% abundant Ba-132. Barium-132 has a neutron capture cross-section of only 0.76 barns and a resonance integral of eight barns. Normally this isotope should be of negligible importance, however, significant amounts of Ba-133 can be produced in concrete containing barite as a high density aggregate. ^{133}Ba emits a series of relatively low energy gammas, at 276, 302, 356, and 382 KeV.
^{152}Eu, ^{154}Eu	The two europium isotopes are the dominant activation products in bioshield concrete on a time scale of 10 to 20 years or longer. ^{152}Eu decays (half-life: 13.5 y) by β– emission (maximum energy: 1.477 MeV) to ^{152}Gd, which decays (half-life: 1.1×10^{14} y) by α emission. ^{154}Eu decays (half-life: 8.6 y) by β– emission (maximum energy: 1.85 MeV) to stable ^{154}Gd. Both have very large neutron capture cross-sections; ^{152}Eu is produced primarily by thermal neutrons, whereas ^{154}Eu also has a substantial resonance integral. The activities of Eu isotopes require consideration, owing to the presence of trace quantities of rare earth elements in source materials used in bioshield concretes.

Fission Products

^{90}Sr	^{90}Sr is principally produced by fission and is one of the most abundant fission products. ^{90}Sr decays (half-life: 28.7 y) by β– emission (maximum energy: 546 keV) to ^{90}Y. ^{90}Sr is a pure β emitter but is generally found in equilibrium with its daughter ^{90}Y, which decays (half-life: 64 h) by β– emission (maximum energy: 2.27 MeV). ^{90}Y is also a pure β emitter. As one of the major fission products, there is a potential for large contamination inventories of this radionuclide.
^{99}Tc	^{99}Tc is a member of a fission product mass chain with quite large cumulative fission yields. It is also produced by neutron capture and subsequent β– decay from ^{98}Mo. ^{99}Tc decays (half-life: 211,100 y) by β– emission (maximum energy: 294 keV) to ^{99}Ru. ^{99}Tc can be an important ingestion hazard in safety

	assessments of the final disposal of contaminated components and activated steel components with a high (1,000–5,000 ppm) initial Mo concentration.
^{129}I	^{129}I is produced by fission (decay product of ^{129}Te). ^{129}I decays (half-life: 1.6×10^7 y) by β– emission (maximum energy: 154 keV) to an excited state of ^{129}Xe. Because of the decay of the ^{129}I precursor, the quantity of ^{129}I in a contamination field will increase slowly after irradiation and reach a peak only after several months. The long life of this radionuclide and its nature as a volatile β emitter are considered very important for waste disposal.
^{137}Cs	This radionuclide is produced by fission and is one of the most abundant fission products. ^{137}Cs decays (half-life: 30 y) by β– emission (maximum energy: 1.17 MeV) to ^{137}Ba. Approximately 85% of the β decays are through ^{137m}Ba and thus are accompanied by the emission of its 662 keV photons. Barium X-rays and conversion electrons are also emitted. Because of its high water solubility, ^{137}Cs is easily transported is also nearly always present in the radionuclide mixture.

Actinides

Pu and Am	The Pu and Am isotopes are mainly found in the crud layers as contaminants due to fuel pin leakages. The Pu isotopes are the main α emitters found in corrosion layers. The presence of α emitters in crud may require increased radiation protection measures against contamination during the dismantling operations and also determine the waste disposal criteria.
^{238}Pu	^{238}Pu is produced by decay of ^{238}Np (half-live: 2.1 d; maximum energy: 1.2 MeV) and by α decay of ^{242}Cm (half-life: 163 d; maximum energy: 6.1 MeV). These parent isotopes are produced by the decay of radionuclides produced by multiple neutron capture in ^{235}U and ^{238}U. ^{238}Pu decays (half-life: 87.7 y) by α emission (maximum energy: 5.499 MeV) to ^{234}U.
^{239}Pu	^{239}Pu is produced by β decay (maximum energy: 438 keV) of ^{239}Np, which is the daughter of another β emitter, ^{239}U. ^{239}Pu decays (half-life: 24,110 y) by α emission (maximum energy: 5.157 MeV) to ^{235}U and is considered a member of the actinium series.
^{241}Pu	^{241}Pu is produced by multiple neutron capture in ^{238}U, ^{239}Pu, and related isotopes. It decays (half-life: 14.4 y) primarily by α emission (maximum energy: 21 keV) to ^{241}Am and is considered a member of the neptunium series.
^{241}Am	The ^{241}Am isotope is produced by the β– decay of ^{241}Pu. It decays (half-life: 432 y) by α emission (maximum energy: 5.5 MeV) to ^{237}Np and is a member of the neptunium decay series. ^{241}Am also emits a low energy gamma ray (maximum energy: 60 keV).
^{242}Cm	A member of the uranium decay series, ^{242}Cm is produced by β– decay of ^{242}Am, which is produced by neutron capture from ^{241}Am. ^{242}Cm decays (half-life: 163 d) by α emission (maximum energy: 6.1 MeV) to the excited state or the fundamental state of ^{238}Pu.
^{244}Cm	A member of the thorium decay series, ^{244}Cm is produced by multiple neutron captures. ^{244}Cm decays (half-life: 18.1 y) by α emission (maximum energy: 5.8 MeV) to the excited state or the fundamental state of ^{240}Pu.

Table 6: B100 accessible asbestos-containing materials [7]

Room	Room Name	Material	Quantity (m^3)	Hazard[a]
N/A	Exhaust Tunnel	Parging cement - fitting	0.04	Asbestos, friable
509	Room	Parging cement - fitting	0.06	Asbestos, friable
512	Distribution Room	Parging cement - fitting	0.06	Asbestos, friable
513	Distribution Room	Parging cement - fitting	0.17	Asbestos, friable
514	Batteries	Parging cement - fitting	0.00	Asbestos, friable
516	Heating and Air Conditioning Room	Parging cement - fitting	0.86	Asbestos, friable
522	Faculty Room	Vinyl tile	0.02	Asbestos, non-friable
522	Offices	Plaster	0.37	Asbestos, non-friable
524	Conference Room	Vinyl tile	0.40	Asbestos, non-friable
526	Corridor	Parging cement - fitting	1.52	Asbestos, friable
526	Corridor	Vinyl tile	0.56	Asbestos, non-friable
530	Heating and Ventilation Room - Ducts	Parging cement - fitting	0.40	Asbestos, friable
534	Offices	Parging cement - fitting	0.04	Asbestos, friable
534	Offices	Vinyl tile	0.34	Asbestos, non-friable
556	Offices	Parging cement - fitting	0.01	Asbestos, friable
556	Offices	Vinyl tile	0.09	Asbestos, non-friable
557	Offices	Vinyl tile	0.33	Asbestos, non-friable
558	Offices	Vinyl tile	0.07	Asbestos, non-friable
560	Offices	Vinyl tile	0.07	Asbestos, non-friable
566	Offices	Vinyl tile	0.11	Asbestos, non-friable
567	Offices	Vinyl tile	0.11	Asbestos, non-friable
568	Office and Storage	Vinyl tile	0.21	Asbestos, non-friable
568	Ducts	Parging cement - fitting	0.06	Asbestos, non-friable

Room	Room Name	Material	Quantity (m^3)	Hazard[a]
582	Offices	Vinyl tile	0.08	Asbestos, non-friable
583	Offices	Parging cement - fitting	0.15	Asbestos, friable
583	Offices	Vinyl tile	0.12	Asbestos, non-friable
Miscellaneous on Level 500	Stairwells/Other	Vinyl tile	0.14	Asbestos, non-friable
Miscellaneous on Level 500	Stairwells/Other	Parging cement - fitting	0.05	Asbestos, non-friable
6005	Offices	Vinyl tile	0.10	Asbestos, non-friable
6005	Offices	Plaster	0.19	Asbestos, non-friable
6006	Offices	Vinyl tile	0.09	Asbestos, non-friable
6006	Offices	Plaster	0.19	Asbestos, non-friable
6007	Offices	Vinyl tile	0.04	Asbestos, non-friable
6007	Offices	Parging cement - fitting	0.00	Asbestos, non-friable
6007	Offices	Plaster	0.19	Asbestos, non-friable
6008	Offices	Vinyl tile	0.10	Asbestos, non-friable
6008	Offices	Parging cement - fitting	0.01	Asbestos, non-friable
6008	Offices	Plaster	0.19	Asbestos, non-friable
6010	Offices	Plaster	0.19	Asbestos, non-friable
6013	Offices	Plaster	0.19	Asbestos, non-friable
613	Offices	Vinyl tile	1.58	Asbestos, non-friable
614	Offices	Vinyl tile	0.17	Asbestos, non-friable
614	Offices	Plaster	0.19	Asbestos, non-friable
615	Offices	Vinyl tile	0.26	Asbestos, non-friable
615	Offices	Plaster	0.19	Asbestos, non-friable
616	Control Room	Vinyl tile	0.33	Asbestos, non-friable

Room	Room Name	Material	Quantity (m^3)	Hazard[a]
616	Control Room	Plaster	0.19	Asbestos, non-friable
623	Offices	Vinyl tile	0.10	Asbestos, non-friable
623	Offices	Plaster	0.19	Asbestos, non-friable
624	Offices	Vinyl tile	0.10	Asbestos, non-friable
624	Offices	Plaster	0.19	Asbestos, non-friable
628	Storage	Vinyl tile	0.06	Asbestos, non-friable
628	Storage	Parging cement - fitting	0.00	Asbestos, non-friable
628	Storage	Plaster	0.19	Asbestos, non-friable
634	Offices	Plaster	0.19	Asbestos, non-friable
646	Corridor	Vinyl tile	0.37	Asbestos, non-friable
646	Corridor	Parging cement - fitting	0.00	Asbestos, non-friable
648	Corridor	Vinyl tile	1.21	Asbestos, non-friable
648	Corridor	Parging cement - fitting	0.01	Asbestos, non-friable
649	Offices	Vinyl tile	0.45	Asbestos, non-friable
651	Offices	Vinyl tile	0.11	Asbestos, non-friable
652	Air Handling Unit	Parging cement - fitting	0.36	Asbestos, non-friable
653	Driveway	Parging cement - fitting	0.01	Asbestos, non-friable
655	Offices and Storage	Vinyl tile	0.20	Asbestos, non-friable
661	Offices	Vinyl tile	0.09	Asbestos, non-friable
662	Offices	Vinyl tile	0.19	Asbestos, non-friable
663	Offices	Vinyl tile	0.09	Asbestos, non-friable
664	Offices	Vinyl tile	0.09	Asbestos, non-friable
665	Offices	Vinyl tile	0.09	Asbestos, non-friable
666	Offices	Vinyl tile	0.09	Asbestos, non-friable

Room	Room Name	Material	Quantity (m^3)	Hazard[a]
666	Offices	Parging cement - fitting	0.03	Asbestos, non-friable
667	Corridor	Vinyl tile	0.19	Asbestos, non-friable
667	Corridor	Parging cement - fitting	0.02	Asbestos, non-friable
667	Corridor	Plaster	0.19	Asbestos, non-friable
687	Offices	Vinyl tile	0.11	Asbestos, non-friable
688	Corridors	Vinyl tile	0.24	Asbestos, non-friable
688	Corridors	Parging cement - fitting	0.01	Asbestos, non-friable
689	Control Room	Vinyl tile	0.14	Asbestos, non-friable
690	Slowpoke Reactor Hall	Vinyl tile	1.14	Asbestos, non-friable
691	Offices	Vinyl tile	0.10	Asbestos, non-friable
692	Offices	Vinyl tile	0.10	Asbestos, non-friable
693	Offices	Vinyl tile	0.10	Asbestos, non-friable
694	Offices	Vinyl tile	0.11	Asbestos, non-friable
695	Offices	Vinyl tile	0.11	Asbestos, non-friable
695	Offices	Parging cement - fitting	0.00	Asbestos, non-friable
696	Offices	Vinyl tile	0.11	Asbestos, non-friable
697	Offices	Vinyl tile	0.11	Asbestos, non-friable
698	Offices	Vinyl tile	0.11	Asbestos, non-friable
N/A	Vestibule	Vinyl tile	0.03	Asbestos, non-friable
N/A	Vestibule	Parging cement - fitting	0.00	Asbestos, non-friable
Miscellaneous on Level 600	Rooms, Washrooms, and Corridors	Vinyl tile	5.75	Asbestos, non-friable
Miscellaneous on Level 600	Rooms, Washrooms, and Corridors	Parging cement - fitting	0.15	Asbestos, non-friable
Miscellaneous on Level 600	Rooms, Washrooms and Corridors	Plaster and tenite	2.57	Asbestos, non-friable

Room	Room Name	Material	Quantity (m^3)	Hazard[a]
702	Lab	Vinyl tile	0.19	Asbestos, non-friable
702	Lab	Parging cement - fitting	0.01	Asbestos, non-friable
702	Lab	Transite	0.74	Asbestos, non-friable
703	Balance Room	Vinyl tile	0.06	Asbestos, non-friable
703	Balance Room	Parging cement - fitting	0.00	Asbestos, non-friable
704	Lab	Vinyl tile	0.09	Asbestos, non-friable
704	Lab	Parging cement - fitting	0.01	Asbestos, non-friable
704	Lab	Transite	0.74	Asbestos, non-friable
705	Corridors	Vinyl tile	0.37	Asbestos, non-friable
707	Offices	Vinyl tile	0.11	Asbestos, non-friable
708	Offices	Vinyl tile	0.13	Asbestos, non-friable
709	Offices	Vinyl tile	0.09	Asbestos, non-friable
710	Offices	Vinyl tile	0.09	Asbestos, non-friable
711	Offices	Vinyl tile	0.13	Asbestos, non-friable
713	Offices	Vinyl tile	0.11	Asbestos, non-friable
714	Offices	Vinyl tile	0.11	Asbestos, non-friable
718	Offices	Vinyl tile	0.09	Asbestos, non-friable
719	Offices	Vinyl tile	0.09	Asbestos, non-friable
720	Offices	Vinyl tile	0.12	Asbestos, non-friable
721	Offices	Vinyl tile	0.12	Asbestos, non-friable
Miscellaneous on Level 700	Stairwells/Other	Vinyl tile	0.22	Asbestos, non-friable
Miscellaneous on Level 700	Stairwells/Other	Parging cement - fitting	0.11	Asbestos, non-friable
Total			30.36	

[a] A friable substance can be reduced to fibres or finer particles by the action of comparatively little pressure.

Table 7: Results of Asbestos Characterization Samples [8]

Sample Location Description	Sample Description	Laboratory Results	
		% Asbestos Fibres	Asbestos Type
Room 530	TSI – CBW elbow	15	Chrysotile
Mezzanine of Room 601	TSI – HHW north	15	Amosite
		5	Chrysotile
Mezzanine of Room 601	TSI – intake duct	45	Amosite
Mezzanine of Room 601	TSI – HBS	20	Amosite
		15	Chrysotile
Room 535	TSI – PRO south	30	Chrysotile
Room 535	TSI – PRO west	15	Amosite
		5	Chrysotile
Room 104, east	TSI	15	Chrysotile
Room 104, west	TSI	45	Amosite
Room 302, west	TSI	50	Amosite
Room 302, east	TSI	15	Chrysotile
Room 538	TSI – Loop pipe	15	Amosite
		5	Chrysotile
Room 410	TSI – Loop pipe	15	Amosite
		5	Chrysotile
Room 506	TSI – Loop pipe	30	Chrysotile
Room 602	TSI – Loop pipe	15	Chrysotile
Room 107, heat exchanger	Mastic, black	5	Chrysotile
Room 415	TSI – pipe insulation	5	Chrysotile
Room 516, Chiller #2	Insulation	40	Chrysotile
	Mastic, black	10	Chrysotile
Room 609, along N wall	Vinyl floor tile, dark brown	3	Chrysotile
Room 648, east wall	TSI	30	Chrysotile
Room 601	Vermiculite, bag	<1	Tremolite
Room 601	Vinyl floor tile, beige/brown	5	Chrysotile
Room 601	Vinyl floor tile, beige/tan	3	Chrysotile
Room 614	Vinyl floor tile, beige/brown	5	Chrysotile
Room 614	Vinyl floor tile, beige/brown	5	Chrysotile
	Mastic, black	No asbestos fibres detected	
Room 617, west hallway relag room	Vinyl floor tile, beige/brown	5	Chrysotile
Room 620, storage	Vinyl floor tile, beige/brown	5	Chrysotile
Room 622	Vinyl floor tile, beige	3	Chrysotile
	Mastic, yellow	No asbestos fibres detected	
Room 528	Vinyl floor tile, grey	5	Chrysotile
	Mastic, yellow	No asbestos fibres detected	
Hallway - Room 642/654	Mastic / tar	7	Chrysotile
	Duct insulation, fibreglass	No asbestos fibres detected	
Hallway – Room 645	Vinyl floor tile, beige	2	Chrysotile
Room 650	Laboratory counter top	3	Amosite
		12	Chrysotile
Room 662, office	Vinyl floor tile, beige	3	Chrysotile
	Mastic, yellow	No asbestos fibres detected	
Room 695	Tar paper, insulation	20	Chrysotile
Room 708	Vinyl floor tiles, beige	10	Chrysotile
	Mastic, yellow/black	<1	Chrysotile
Room 702, lab	Floor tile, beige	3	Chrysotile
Room 715	Mastic, black	8	Chrysotile
	Insulation, fibreglass	No asbestos fibres detected	
Room 531	TSI – tar, black	5	Chrysotile
Hallway 517	Vinyl floor tile, beige	3	Chrysotile
	Mastic, beige	No asbestos fibres detected	
Room 522	Vinyl floor tile, beige/gray	5	Chrysotile
	Mastic, black	No asbestos fibres detected	

Table 8: Locations of lead-containing paint within B100 [7]

Room #	Paint Colour	Description
648 & 665	Yellow	Stripes on floor
690	Yellow	On hoist
516	Yellow	Caution stripes on floors
513	Yellow	Caution stripes on floors
513	Yellow	Hand railing

Table 9: Chemicals potentially present in B100 [7]

Chemical/Material	Expected Locations in WR-1	Description
Organic Coolant	PHT System; Auxiliary Organic Systems; Organic Supply System; Fuel Wash Down System. Rooms and surfaces containing or adjacent to the above systems.	OS-84 (previously labeled HB-40) hydrogenated terphenyls used as reactor coolant.
Xylene	Spent Fuel Handling and Storage System - Wash Tube Facility	Used as a cleaning solvent. Irradiated fuel was washed with xylene prior to transfer to the fuel storage block for inspection. The wash tube facility consisted of a xylene-filled storage tube.
Boron	Heavy Water System; Auxiliary Systems including Boron Addition System	Added as boric acid to heavy water to control reactor reactivity. Boric acid was used as an additive in the WR-1 heavy water moderator. Trace amounts of boron should be expected in the boron addition system and the equipment associated with the reactor heavy water systems.
Palladium	Organic Supply System	Palladium bed absorption columns used in system (columns have been removed, but other equipment may be contaminated).
Palladium	Helium System	5% palladium on pelletized alumina used in a recombiner.
Potassium Hydroxide	Chemical Addition Tank in the Concrete Cooling System	Used to control the pH of the water used to cool the concrete shielding around WR-1.
Platinum	Flux Detectors	Used as wire in the detector.
Magnesium Oxide	Flux Detectors	Used as an insulator.
Gadolinium Nitrate	SDR Liquid Absorber Safety System	Gadolinium nitrate solution flowed into the pool to shut the reactor down.
Ozone Depleting Substances	Multiple Systems	Air conditioning and refrigeration systems.
Multiple Ion Exchange Columns	Heavy Water System; Distilled Water System; Spent Fuel Bay Circulation System; Concrete Cooling System; Boron Addition System; SDR Auxiliary Systems; WR-1L2; Fast Neutron Loop	Numerous ion exchange columns are incorporated in the systems of WR-1.

Chemical/Material	Expected Locations in WR-1	Description
Mercury	Batteries, manometers, thermometers, electrical and electronic switches, in fluorescent lamps, and light switches	
Mould	On piping, insulation, and walls	
Beryllium		Anecdotal
Hydraulic oils		Anecdotal
Glycol		Anecdotal
Chromium/cadmium		Anecdotal
Hydrazine		Anecdotal

Table 10: List of potential radionuclides of concern

Radionuclide of Concern[a]	Origin	Emissions	$T_{1/2}$ (Years)
^{3}H	Activation Product	β- (0.006 MeV[b])	12.3
^{14}C	Activation Product	β- (0.049 MeV)	5.7×10^{3}
^{36}Cl	Activation Product	β- (0.251 MeV)	3.01×10^{5}
^{60}Co	Activation Product	β- (0.096 MeV)/γ	5.3
^{59}Ni	Activation Product	EC[c]/X-rays	7.6×10^{4}
^{63}Ni	Activation Product	β- (0.0174 MeV)	101.2
$^{90}Sr/^{90}Y$	Fission Product	β- (0.196 /0.934 MeV)	28.8
^{94}Nb	Activation Product	β-(0.145 MeV)/γ	2.03×10^{4}
^{99}Tc	Fission Product	β- (0.0846 MeV)	2.12×10^{5}
^{108m}Ag	Activation Product	EC[c]/X-rays	438
^{137}Cs	Fission Product	β- (0.174 MeV)/γ	30.1
^{152}Eu	Activation Product	β- (0.298 MeV)/γ	13.5
^{235}U	Fuel	α	7.0×10^{7}
^{238}U	Fuel	α	4.5×10^{9}
^{241}Am	Transuranic	α	432.6
^{238}Pu	Transuranic	α	87.7
^{239}Pu	Transuranic	α	2.41×10^{4}
^{241}Pu	Transuranic	β- (0.007 MeV)	14.3
^{244}Cm	Transuranic	α	18.1

[a] Radionuclides with $T_{1/2}$ of less than 3 years are not included.

[b] Average beta (β-) energy

[c] EC = electron capture

Table 11: Calculated total radioactive inventory in the remaining WR-1 core components [10]

Component	Total Radioactivity After Shutdown – Bq (Ci) [% of activity]				
	10 y	**50 y**	**100 y**	**1,000 y**	**10,000 y**
Calandria Tubes	8.6E+12 (232) [0.14%]	2.4E+11 (6) [0.03%]	1.4E+11 (4) [0.02%]	1.5E+09 (0.04) [0.01%]	1.15E+09 (0.03) [0.01%]
Steel Channels	2.1E+15 (~57,000) [35.6%]	4.0E+14 (~10,800) [45.3%]	2.8E+14 (~7,600) [45.3%]	4.7E+12 (~125) [31.2%]	3.1E+12 (~84) [29.2%]
Ozhennite Channels	4.5E+13 (~1,200) [0.73%]	8.5E+12 (230) [0.94%]	6.0E+12 (162) [0.95%]	4.1E+11 (~10) [2.8%]	2.3E+11 (~6) [2.2%]
Zr-2.5%Nb Channels	2.7E+13 (730) [0.46%]	3.5E+12 (95) [0.39%]	3.3E+12 (89) [0.51%]	2.8E+12 (77) [18.9%]	2.0E+12 (55) [19.1%]
Calandria Vessel	2.4E+15 (~65,000) [40.5%]	4.7E+14 (~12,700) [53.3%]	3.3E+14 (~8.900) [53.3%]	7.1E+12 (~190) [47.2%]	5.2E+12 (~142) [49.5%]
Thermal Shield	1.4E+15 (~38,000) [22.6%]	5.1E+10 (1.4) [0.01%]	5.9E+07 (0.002) [0.00001%]	4.7E+07 (0.001) [0.0003%]	1.6E+07 (0.001) [0.0001%]
Total	6.0E+15 (~162,000)	8.8E+14 (~24,800)	6.1E+14 (~16,500)	1.6E+13 (~422)	1.06E+13 (~)

Table 12: Calculated activation radionuclide activity in the WR-1 core [10]

Radionuclide	Total Radioactivity After Shutdown – Bq (Ci) [% of activity]				
	10 y	**50 y**	**100 y**	**1,000 y**	**10,000 y**
C-14 ($T_{1/2}$ =5730 y)	3.0E+12 (~80) [0.05%]	3.0E+12 (~80) [0.34%]	3.0E+12 (~80) [0.47%]	2.7E+12 (~73 Ci) [17.8%]	8.9E+11 (~24 Ci) [8.5%]
Fe-55 ($T_{1/2}$ =2.7 y)	3.0E+15 (~81,000) [49.5%]	1.0E+11 (~2.7) [0.01%]	2.7E+05 (0.00001 [0.000%]	0	0
Co-60 ($T_{1/2}$ =5.3 y)	1.9E+15 (~51,300) [31.1%]	9.7E+12 (262) [1.1%]	1.4E+10 (0.4) [0.002%]	0	0
Ni-59 ($T_{1/2}$ =76,000 y)	8.3E+12 (224) [0.14%]	8.3E+12 (224) [0.9%]	8.3E+12 (224) [1.3%]	8.2E+12 (~222 Ci) [54.7%]	7.6E+12 (~205 Ci) [71.7%]
Ni-63 ($T_{1/2}$ =100 y)	1.1E+15 (~29,700) [19.1%]	8.6E+14 (~23,200) [97.3%]	6.0E+14 (~16,200) [97.7%]	1.8E+12 (~32 Ci) [8.5%]	0
Nb-94 ($T_{1/2}$ =20,300 y)	3.0E+12 (81) [0.05%]	3.0E+12 (81) [0.33%]	3.0E+12 (81) [0.5%]	2.9E+12 (~79) [19.0%]	2.1E+12 (~57) [19.8%]
Total	6.0E+15 (~162,000)	8.8E+14 (~24,800)	6.1E+14 (~16,500)	1.6E+13 (~422)	1.06E+13 (~286)

Table 13: Summary of the reactor core Ag-108m radioactivity inventory estimate

Core Component	10 Years Post Shutdown [Bq]		50 Years Post Shutdown [Bq]	
	Ratio Co:Ag [9]	Ag-108m [10]	Ratio Co:Ag [9]	Ag-108m [10]
Calandria Tubes	5.61E-07	3.97E+06	8.70E-05	3.20E+06
Fuel Channel (Stainless Steel)	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fuel Channel (Ozhenite)	4.19E-08	1.29E+06	6.49E-06	1.04E+06
Fuel Channel (Zr-2.5%Nb)	4.29E-08	9.22E+05	6.64E-06	7.37E+05
Calandria	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Thermal Shield	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total		6.18E+06		4.98E+06

Table 14: Comparison of total estimated WR-1 core activation

Nuclide	Origin-S (Bq)	Modified by Sample Result (Bq)
^{14}C	3.02×10^{12}	8.11×10^{8}
^{55}Fe	1.03×10^{11}	3.33×10^{11}
^{60}Co	9.65×10^{12}	4.60×10^{12}
^{59}Ni	8.30×10^{12}	3.80×10^{12}
^{63}Ni	8.55×10^{14}	4.68×10^{14}
^{94}Nb	2.95×10^{12}	8.56×10^{11}
Total	8.79×10^{14}	4.78×10^{14}

Table 15: Estimated 90% radionuclide inventories at mid-core elevation of WR-1 biological shield [14]

Radionuclide	Bq/cm^2	Total Estimated Shield Radioactivity After Shutdown Bq (mCi) [% of activity]				
		10	50 y	100 y	1,000 y	10,000 y
Cl-36 ($T_{1/2}$ =3.0 x 10^5 y)	0.003	4.2E+03 (1.1 x 10^{-4}) [0.000007%]	4.2E+03 (1.1 x 10^{-4}) [0.0003%]	4.2E+03 (1.1 x 10^{-4}) [0.001%]	4.2E+03 (1.1 x 10^{-4}) [0.002%]	4.1E+03 (1.1 x 10^{-4}) [0.003%]
Ca-41 ($T_{1/2}$ =1.0 x 10^5 y)	100	1.4E+08 (3.8) [0.24%]	1.4E+08 (3.8) [11.2%]	1.4E+08 (3.8) [24.0%]	1.4E+08 (3.8) [71.5%]	1.3E+08 (3.5) [87.7%]
Ni-63 ($T_{1/2}$ =100 y)	480	6.7E+08 (18) [1.2%)	5.1E+08 (14) [41.0%]	3.6E+08 (9.8) [62.2%]	7.6E+05 (0.020) [0.39%]	0
C-14 ($T_{1/2}$ =5730 y)	44	6.2E+07 (1.7) [0.11%]	6.2E+07 (1.7) [4.9%]	6.1E+07 (1.7) [10.5%]	5.5E+07 (1.5) [28.1%]	1.8E+07 (0.5) [12.3%]
Co-60 ($T_{1/2}$ =5.3 y)	39,000	5.5E+10 (1500) [95.1%]	2.8E+08 (7.7) [22.7%]	4.0E+05 (0.011) [0.07%]	0	0
Eu-152 ($T_{1/2}$ =13.5 y)	1,400	2.0E+09 (53) [3.4%]	2.5E+08 (6.8) [20.2%]	1.9E+07 (0.52) [3.3%]	0	0
Total		5.8E+10 (1600)	1.3E+09 (34)	5.8E+08 (15.8)	2.0E+08 (5.3)	1.5E+08 (4.0)

Table 16: Summary of WR-1 fuel failures

Year	No. Failures	Fuel Assemblies
1966	2	WS-4 , EXP-WR1-906
1967	2	EXP-WR1-912, EXP-WR1-906B
1968	8	WN31, EXP-WR1-951, EXP-WR1-911B, WN5, WN25, EXP-WR1-911A, EXP-WR1-902E,DA11C
1969	2	LC033, EXP-WR1-954
1970	5	EXP-WR1-920A, XP-WR1-922A, EXP-WR1-919D, 917B and 922D
1971	20	EXP-WR1-373B, EXP-WR1-933B, EXP-WR1-204, EXP-WR1-206, EXP-WR1-FNF02, EXP-WR1-936B, EXP-WR1-930,EXP-WR1-933B, EXP-WR1-373B, EXP-WR1-933B, EXP-WR1-204, EXP-WR1-206, EXP-WR1-FNF02, EXP-WR1-936B, WZ3, EXP-WR1-945, EXP-WR1-203B, EXP-WR1-FNF05
1972	27	EXP-WR1-936A, EXP-WR1-947, EXP-WR1-929, EXP-WR1-928C, EXP-WR1-937A, EXP-WR1-936B/937B, EXP-WR1-930B, EXP-WR1-934A, EXP-WR1-925A, EXP-WR1-945B, EXP-WR1-934C, EXP-WR1-932, WN67, WN78, EXP-WR1-928C, WN63, EXP-WR1-945C, EXP-WR1-942D, EXP-WR1-942F, WN104, WR1-EXP-9258
1973	11	WR1-EXP-935, EXP-WR1-942G, EXP-WR1-925C, EXP-WRl-925A, WNI02, EXP-WRl-979B, EXP-WRl-974B, WNI51, EXP-WRl-FNF9l6,EXP-WRI-FNF08, EXP-WRI-FNF07
1974	17	EXP-WR1-FNF01, WN078, FNF014, EXP-WR1-988A, EXP-WRl-979G, EXP-WR1-2L, WN154M, WN071, EXP-WR1-969B, WN086M, WN052
1975	13	WN086M, WN052, EXP-WRl-947A, WNl14, WNl16, WN120, EXP-WRl-937G, EXP-WRl-217A, WT13, WN163, WN170, WN131, WT16, EXP-WRl-979C, WN152M,

Year	No. Failures	Fuel Assemblies
1976	9	WN147, FN004A, WN202, WN208, CN044, WN198, CN068, WN133, WN158
1977	3	EXP-WRl-603, EXP-WRl-1001A, EXP-WRl-221
1978	7	CN013A, EXP-WR1-1007D, EXP-WR1-1007E, EXP-WR1-1007B, FN028, EXP-WR1-1007B, FN028
1979	9	EXP-WR1-1007J, EXP-WR1-1007D, EXP-WRl-1007A, EXP-WR1-1007H, EXP-WRl-223, EXP-WRl-1007J, EXP-WR1-1007E, EXP-WRl-1007F, EXP-WRl-1007G
1980	8	EXP-WRl-606, DNOOIA, EXP-WR1-1008, CN104A, CN031A, EXP-WR1-225, EXP-WR1-606,
1982	1	CN138
1983	7	CN140, CN151, CN118A, CN146, CN148, CN139, CN124A
TOTAL	151	

Table 17: Estimate of fission product and actinide radioactivity released into the PHT

Radioactivity (Bq) for Release of 1 kg uranium for average burn-up of 685 GJ/kg (Based on burn-up tables in Reference [22])									
Nuclide	**Radioactive Decay Period**								
	Discharge	**5 y**	**10 y**	**30 y**	**40 y**	**50 y**	**100 y**	**1,000 y**	**10,000 y**
Sr-90	6.50E+11	5.80E+11	5.10E+11	3.10E+11	2.50E+11	1.90E+11	5.80E+10	1.90E+01	
Cs-137	9.60E+11	8.60E+11	7.60E+11	4.80E+11	3.80E+11	3.00E+11	9.70E+10	1.00E+02	
Eu-154	4.20E+10	2.80E+10	1.90E+10	3.90E+09	1.80E+09	8.10E+08	1.60E+07	0.00E+00	
Eu-155	3.20E+10	1.60E+10	7.90E+09	4.80E+08	1.20E+08	2.90E+07	2.70E+04	0.00E+00	
Tc-99	1.30E+08	1.30E+08	1.30E+08	1.30E+08	1.30E+08	1.30E+08	1.30E+08	1.30E+08	1.20E+08
I-129	2.80E+05	2.80E+05	2.80E+05	2.80E+05	2.80E+05	2.80E+05	2.80E+05	2.80E+05	2.80E+05
U-235	5.76E+05	5.76E+05	5.76E+05	5.76E+05	5.76E+05	5.76E+05	5.76E+05	5.76E+05	5.76E+05
U-238	1.24E+07	1.24E+07	1.24E+07	1.24E+07	1.24E+07	1.24E+07	1.24E+07	1.24E+07	1.24E+07
Np-237	9.30E+05	9.30E+05	9.40E+05	1.10E+06	1.10E+06	1.20E+06	1.60E+06	5.50E+06	6.70E+06
Np-239	9.90E+14	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.80E+07	1.60E+07
Pu-238	2.90E+09	3.00E+09	2.90E+09	2.50E+09	2.30E+09	2.10E+09	1.40E+09	1.20E+06	0.00E+00
Pu-239	6.10E+09	6.10E+09	6.10E+09	6.10E+09	6.10E+09	6.10E+09	6.10E+09	5.90E+09	4.60E+09
Pu-240	8.70E+09	8.70E+09	8.70E+09	8.70E+09	8.70E+09	8.70E+09	8.60E+09	7.80E+09	3.00E+09
Pu-241	8.60E+11	6.70E+11	5.30E+11	2.00E+11	1.30E+11	7.70E+10	7.00E+09	0.00E+00	
Am-241	2.20E+08	6.30E+09	1.10E+10	2.10E+10	2.40E+10	2.50E+10	2.50E+10	6.00E+09	3.20E+03
Am-243	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.90E+07	1.80E+07	1.60E+07
Cm-244	6.50E+08	5.40E+08	4.40E+08	2.10E+08	1.40E+08	9.60E+07	1.40E+07	0.00E+00	
Total	9.93E+14	2.18E+12	1.86E+12	1.03E+12	8.03E+11	6.10E+11	2.03E+11	1.99E+10	7.77E+09
Cs:Am	4.50E+03	1.40E+02	6.90E+01	2.30E+01	1.60E+01	1.20E+01	3.80E+00	1.70E-08	

Table 18: Percentage of estimated fission product and actinide radioactivity released into the PHT

Percentage (%) of Radionuclide Activity									
Nuclide	**Radioactive Decay Period**								
	Discharge	**5 y**	**10 y**	**30 y**	**40 y**	**50 y**	**100 y**	**1,000 y**	**10,000 y**
Sr-90	0.07%	26.62%	27.48%	30.01%	31.12%	31.15%	28.53%	0.00%	0.00%
Cs-137	0.10%	39.47%	40.94%	46.46%	47.30%	49.18%	47.71%	0.00%	0.00%
Eu-154	0.00%	1.29%	1.02%	0.38%	0.22%	0.13%	0.01%	0.00%	0.00%
Eu-155	0.00%	0.73%	0.43%	0.05%	0.01%	0.00%	0.00%	0.00%	0.00%
Tc-99	0.00%	0.01%	0.01%	0.01%	0.02%	0.02%	0.06%	0.65%	1.54%
I-129	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%
U-235	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%
U-238	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.06%	0.16%
Np-237	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.03%	0.09%
Np-239	99.74%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.09%	0.21%
Pu-238	0.00%	0.14%	0.16%	0.24%	0.29%	0.34%	0.69%	0.01%	
Pu-239	0.00%	0.28%	0.33%	0.59%	0.76%	1.00%	3.00%	29.67%	59.19%
Pu-240	0.00%	0.40%	0.47%	0.84%	1.08%	1.43%	4.23%	39.22%	38.60%
Pu-241	0.09%	30.75%	28.55%	19.36%	16.18%	12.62%	3.44%	0.00%	
Am-241	0.00%	0.29%	0.59%	2.03%	2.99%	4.10%	12.30%	30.17%	0.00%
Am-243	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.09%	0.21%
Cm-244	0.00%	0.02%	0.02%	0.02%	0.02%	0.02%	0.01%	0.00%	
Total	100%	100%	100%	100%	100%	100%	100%	100%	100%

Table 19: 1994 primary heat transport closed system measured exposure rates and estimated radionuclide inventory

Room	Description	Highest Near Contact Exposure Rate (mR/h)	GBq			
			^{137}Cs	^{90}Sr	^{60}Co	Other[a]
103	Drain Tanks	70	5.9	2.1	0.17	0.25
104	Degassing	1295	109	39.8	3.1	4.7
302	Degassing	1055	89	32.4	2.5	3.8
306	Fuel Storage Block	280	23.5	8.6	0.67	1.0
404	Activity Monitoring	320	26.9	9.8	0.77	1.2
506	Header	1130	95.2	34.7	2.7	4.1
602	Primary Pumps	983	82.6	30.1	2.4	3.5
TOTAL			432 (11.7 Ci)	158	12.3	18.5

[a] Other included ^{94}Nb, ^{95}Zr, ^{125}Eu, ^{226}Ra, and ^{241}Am which represented less than 2.1% of the total activity from the surface swipes analysis.

Table 20: Primary heat transport and auxiliary organic and gas systems

Nuclide	No. of Samples	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 90% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90% UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Ag-108m	39	1	-2.78E-02	1.00E+00	5.99E-02	2.84E-02	15,457,445	1.14E+06	1.23E+06	1.62E+06	Chebyshev
Am-241	39	25	-9.91E-03	9.05E+02	3.20E+01	2.43E+01		6.75E+08	7.56E+08	1.09E+09	Chebyshev
C-14	39	35	4.58E-03	1.18E+02	3.31E+00	3.03E+00		7.37E+07	8.39E+07	1.26E+08	Chebyshev
Cm-243/244	3/1[a]	3	9.60E-04	1.34E+01	4.38E-01	5.58E-02		7.87E+06	8.19E+06	8.78E+06	$\bar{x} + z\bar{\sigma}$
Co-60	39	26	-1.21E-01	5.07E+01	1.53E+00	1.30E+00		3.32E+07	3.76E+07	5.55E+07	Chebyshev
Cs-137	39	39	2.14E-02	2.12E+04	9.73E+02	5.90E+02		1.94E+10	2.14E+10	2.96E+10	Chebyshev
Fe-55	3/1	2	1.03E+00	4.83E+01	2.82E+00	2.88E-01		4.94E+07	5.10E+07	5.40E+07	$\bar{x} + z\bar{\sigma}$
H-3	39	31	4.53E-02	2.24E+02	1.33E+01	6.44E+00		2.54E+08	2.75E+08	3.65E+08	Chebyshev
Nb-94	39	13	-4.09E-01	7.88E+00	2.19E-01	2.02E-01		4.88E+06	5.56E+06	8.36E+06	Chebyshev
Ni-59	3/1	0	-8.30E-01	2.92E+00	-8.59E-02	7.96E-01		1.44E+07	1.89E+07	2.73E+07	$\bar{x} + z\bar{\sigma}$
Ni-63	3/1	2	6.17E-01	3.00E+02	8.72E+00	8.60E-01		1.52E+08	1.57E+08	1.66E+08	$\bar{x} + z\bar{\sigma}$
Pu-238	3/1	4	3.23E-01	2.30E+02	8.75E+00	8.38E-01		1.52E+08	1.57E+08	1.65E+08	$\bar{x} + z\bar{\sigma}$
Pu-239/240	3/1	4	7.14E-01	5.72E+02	1.70E+01	1.27E+00		2.88E+08	2.95E+08	3.09E+08	$\bar{x} + z\bar{\sigma}$
Pu-241	3/1	4	9.24E+00	4.39E+03	1.60E+02	1.48E+01		2.76E+09	2.85E+09	3.00E+09	$\bar{x} + z\bar{\sigma}$
Sr-90	3/1	4	4.11E+01	1.81E+04	5.98E+02	9.47E+01		1.11E+10	1.17E+10	1.26E+10	$\bar{x} + z\bar{\sigma}$
U-235	39	6	-2.30E+00	3.55E+00	7.77E-02	1.41E-01		2.25E+06	2.72E+06	4.67E+06	Chebyshev
U-238	3/1	4	6.40E-04	9.51E-01	2.64E-02	4.66E-01		1.31E+07	1.58E+07	2.07E+07	$\bar{x} + z\bar{\sigma}$
							Total Inventory	3.49E+10	3.75E+10	4.70E+10	

[a] #/# refers to # of individual and # of composite samples analyzed.

Table 21: Comparison of individual measurement results

Sample ID	^{137}Cs (Bq/cm^2)	^{90}Sr (Bq/cm^2)	^{137}Cs: ^{90}Sr Ratio
PHT 1-3-2	9,430	3,490	2.7:1
PHT 5-1-3	405	293	1.4:1
PHT 11-2-3	21,200	18,100	1.2:1
Composite	--	40. 5	--
Comparison of N = 3 Means and Full ^{137}Cs Sample Population and ^{90}Sr Weighted Means			
N = 3 Arithmetic Mean	10,345	7,294[a]	1.8:1
N = 39 ^{137}Cs Mean:^{90}Sr Weighted Mean[b]	973	598	1.6:1

[a] Composite result is not included.

[b] Weighted mean = (40.5 Bq/cm^2 x 0.9231) + (7,294 Bq/cm^2 x 0.0769), where the composite represents 36 of 39 locations (92.31%) and the mean of the three individual samples represents 3 of 39 locations (7.69%).

Table 22: Process drain system characterization inventory summary

Nuclide	No. of Samples	# >MDA	Min (Bq/cm²)	Max (Bq/cm²)	Mean @ 80% Confidence (Bq/cm²)	SD (Bq/cm²)	System Surface Area (cm²)	90% UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Ag-108m	8	0	-1.08E-03	1.48E-03	9.53E-05	8.01E-04	5,248,610	2.60E+03	3.32E+03	4.95E+03	Student's t
Am-241	8	1	-1.30E-02	2.02E+00	2.61E-01	7.11E-01		5.33E+06	7.12E+06	1.45E+07	Chebyshev
C-14	1/0 [a]	0	3.34E-02	3.34E-02	3.34E-02	4.43E-03		2.05E+05	2.14E+05	2.29E+05	x+zσ
Cm-243/244	1/1	1	5.92E-02	5.92E-02	5.92E-02	1.18E-02		3.90E+05	4.13E+05	4.55E+05	x+zσ
Co-60	8	1	-9.20E-03	3.84E-02	5.05E-03	1.41E-02		1.05E+05	1.41E+05	2.88E+05	Chebyshev
Cs-137	8	3	3.06E-05	6.15E+01	7.76E+00	2.17E+01		1.62E+08	2.16E+08	4.42E+08	Chebyshev
Fe-55	1/0	1	6.23E+00	6.23E+00	6.23E+00	1.42E+00		7.46E+06	4.23E+07	4.50E+07	x+zσ
H-3	1/0	1	1.23E+00	1.23E+00	1.23E+00	1.48E-01		7.45E+06	7.73E+06	8.26E+06	x+zσ
Nb-94	8	1	-4.27E-03	2.69E-01	3.32E-02	9.53E-02		7.03E+05	9.45E+05	1.93E+06	Chebyshev
Ni-59	1/1	1	-1.18E-01	3.43E-01	2.86E-01	5.54E-01		5.23E+06	6.29E+06	8.27E+06	$\bar{x} + z\bar{\sigma}$
Ni-63	1/0	0	3.55E-02	NA	Not Detected			–	–	1.86E+05	–
Pu-238	1/1	0	1.89E-01	1.89E-01	1.89E-01	2.84E-02		1.18E+06	1.24E+06	1.34E+06	x+zσ
Pu-239/240	1/1	1	8.48E-01	8.48E-01	8.48E-01	7.82E-02		4.98E+06	5.13E+06	5.41E+06	x+zσ
Sr-90	1/1	1	1.36E-02	3.74E+00	4.80E-01	9.77E-02		3.17E+06	3.36E+06	3.71E+06	$\bar{x} + z\bar{\sigma}$
Tc-99	1/0	0	4.74E-04	NA	– [b]	–		–	–	2.49E+03 [c]	–
U-235	8	0	-1.18E-01	7.73E-03	-1.27E-02	4.27E-02		1.71E+05	2.79E+05	7.24E+05	Chebyshev
U-238	1/1	1	2.26E-04	8.53E-03	1.26E-03	1.51E-04		7.65E+03	7.94E+03	8.48E+03	Chebyshev
Total Inventory								1.98E+08	2.91E+08	5.32E+08	

[a] #/# refers to # of individual and # of composite samples analyzed.

[b] Measurement uncertainty not reported.

[c] Inventory is not a UCL, based on single sample Bq/cm² result multiplied by the system total surface area.

Table 23: Active drain system characterization inventory summary

Nuclide	Sample #	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 80% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90% UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Ag-108m	8	0	-3.77E-03	1.60E-01	2.04E-02	5.65E-02	1,360,250	1.09E+05	1.46E+05	2.98E+05	Chebyshev
Am-241	8	2	-7.59E-02	1.05E-01	2.72E-02	5.90E-02		7.71E+04	9.08E+04	1.22E+05	Student's t
C-14	1/0 [a]	1	7.66E-02	7.66E-02	7.66E-02	7.66E-03		1.18E+05	1.21E+05	1.28E+05	x+zσ
Cm-242/244[b]	1/1	0	-1.31E-02	2.32E-03	Not detected			–	–	–	–
Co-60	8	2	-8.02E-03	3.07E+01	3.84E+00	1.09E+01		2.09E+07	2.80E+07	5.72E+07	Chebyshev
Cs-137	8	7	2.02E-02	2.70E+03	3.38E+02	9.54E+02		1.84E+09	2.46E+09	5.03E+09	Chebyshev
Fe-55	1/1	2	7.84E+00	2.27E+01	1.52E+01	2.44E+00		2.50E+07	2.62E+07	2.85E+07	$\bar{x} + z\bar{\sigma}$
H-3	1/0	1	1.51E+00	1.51E+00	1.51E+00	1.49E-01		2.31E+06	2.39E+06	2.53E+06	x+zσ
Nb-94	8	0	-1.22E-01	7.40E-03	-1.30E-02	4.41E-02		4.60E+04	7.48E+04	1.93E+05	Chebyshev
Ni-59	1 /1	0	-2.49E-01	6.29E-01	5.19E-01	1.36E+00		3.09E+06	3.76E+06	5.02E+06	$\bar{x} + z\bar{\sigma}$
Ni-63	1/1	1	4.67E+01	4.67E+01	4.67E+01	1.8		6.67E+07	6.76E+07	6.92E+07	x+zσ
Pu-238	1/1	1	1.98E-02	1.98E-02	1.98E-02	2.83E-03		3.19E+04	3.33E+04	3.59E+04	x+zσ
Pu-239/240	1/1	1	6.55E-03	6.51E-02	1.39E-02	5.83E-03		2.90E+04	3.19E+04	3.73E+04	$\bar{x} + z\bar{\sigma}$
Sr-90	1 /1	1	7.20E-01	5.15E+00	1.27E+00	1.71E-01		2.03E+06	2.12E+06	2.28E+06	$\bar{x} + z\bar{\sigma}$
Tc-99	8	0	<1.8E-04	<9.9E-04	Not detected			–	–	–	–
U-235	8	1	-1.63E-02	1.53E+00	1.93E-01	5.40E-01		1.04E+06	1.40E+06	2.85E+06	Chebyshev
U-238	1/1	2	2.62E-04	7.93E-04	3.28E-04	3.72E-05		5.11E+02	5.30E+02	5.64E+02	$\bar{x} + z\bar{\sigma}$
TOTAL INVENTORY								1.96E+09	2.59E+09	5.19E+09	

[a] #/# refers to # of individual and # of composite samples analyzed.

[b] Undetected, no uncertainty provided with result.

Table 24: Helium gas system characterization inventory summary

Nuclide	Sample #	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 90% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90 % UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Am-241	9	0	-3.78E-02	1.59E-02	-5.98E-03	3.29E-03	693,985	-3.09E+03	-2.73E+03	-1.94E+03	Student's t
C-14	9	9	4.12E-01	7.75E+01	1.36E+01	8.64E+00		1.55E+07	1.82E+07	2.94E+07	Chebyshev
Co-60	9	8	5.72E-03	1.33E+00	1.84E-01	1.45E-01		2.29E+05	2.74E+05	4.62E+05	Chebyshev
Cs-137	9	8	9.42E-03	1.31E-01	4.48E-02	1.17E-02		3.48E+04	3.61E+04	3.89E+04	Student's t
Fe-55	9	9	1.26E+00	6.05E+00	3.20E+00	7.20E-01		2.45E+06	2.53E+06	2.70E+06	Student's t
H-3	9	6	-2.86E+00	1.04E+03	1.90E+02	1.22E+02		2.16E+08	2.55E+08	4.12E+08	Chebyshev
Nb-94	9	0	-1.50E-03	3.05E-03	4.39E-04	2.12E-04		3.73E+02	3.96E+02	4.47E+02	Student's t
Ni-63	9	9	2.97E+00	1.43E+01	7.57E+00	1.71E+00		5.81E+06	5.99E+06	6.40E+06	Student's t
Sr-90	9	0	4.57E-03	2.20E-02	1.17E-02	2.62E-03		8.93E+03	9.21E+03	9.84E+03	Student's t
U-235[a]	9	1	-5.34E-03	4.99E-03	2.23E-04	1.46E-03		6.28E+02	7.85E+02	1.14E+03	Student's t
U-238[a]	9	2	5.46E-02	1.95E+00	7.02E-01	2.73E-01		5.76E+05	6.05E+05	6.70E+05	Student's t
							Total Inventory (Bq)	2.41E+08	2.82E+08	4.52E+08	

[a] Uranium concentrations and inventories are based on gamma spectroscopy results. ^{238}U activity was determined via ^{234}Th photopeaks and ^{235}U as reported.

Table 25: Heavy water system characterization inventory summary

Nuclide	Sample #	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 90% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90% UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Am-241	12	4	-3.87E-02	1.09E+00	1.16E-01	8.83E-02	3,962,313	5.96E+05	6.40E+05	7.33E+05	Student's t
C-14	12	12	2.78E-02	2.87E+01	9.82E+00	3.36E+00		5.04E+07	5.57E+07	7.72E+07	Chebyshev
Co-60	12	12	2.50E-02	8.70E-01	1.90E-01	7.24E-02		1.00E+06	1.12E+06	1.58E+06	Chebyshev
Cs-137	12	10	0.00E+00	4.77E-01	5.19E-02	3.90E-02		3.39E+05	4.00E+05	6.49E+05	Chebyshev
Fe-55	12	12	2.57E+00	9.36E+00	5.32E+00	4.60E-01		2.18E+07	2.20E+07	2.25E+07	Student's t
H-3	12	12	2.41E+01	5.45E+02	2.36E+02	5.51E+01		1.02E+09	1.05E+09	1.11E+09	Student's t
Nb-94	12	3	-4.32E-03	1.70E-02	2.62E-03	1.76E-03		1.31E+04	1.40E+04	1.59E+04	Student's t
Ni-63	12	12	6.07E+00	2.21E+01	1.26E+01	1.09E+00		5.16E+07	5.21E+07	5.32E+07	Student's t
Sr-90	12	0	9.34E-03	3.40E-02	1.94E-02	1.67E-03		7.94E+04	8.02E+04	8.20E+04	Student's t
U-235[a]	12	4	-2.74E-03	3.31E-02	3.97E-03	2.80E-03		2.01E+04	2.15E+04	2.45E+04	Student's t
U-238[a]	12	1	-5.72E-01	4.00E-01	2.60E-02	7.17E-02		3.49E+05	4.61E+05	9.19E+05	Chebyshev
							Total Inventory (Bq)	1.15E+09	1.18E+09	1.26E+09	

[a] Uranium concentrations and inventories are based on gamma spectroscopy results. ^{238}U activity was determined via ^{234}Th photopeaks and ^{235}U as reported.

Table 26: Tritium activity level obtained

Method	Activity (Bq)
1% Remaining Estimate	1.27×10^{14}
Absorbed Tritium Analysis	1.11×10^{9}
Amount of ^{3}H Released (2015)	2.47×10^{15}
Amount of ^{3}H Released (2017)	3.80×10^{14}
Rate of ^{3}H Release (2015)	2.26×10^{15}
Rate of ^{3}H Release (2017)	3.14×10^{14}
H Solubility Limit in SS	5.31×10^{12}

Table 27: Experimental loops characterization inventory summary

Nuclide	No. of Samples	# >MDA	Min (Bq/cm²)	Max (Bq/cm²)	Mean @ 90% Confidence (Bq/cm²)	SD (Bq/cm²)	System Surface Area (cm²)	90% UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Ag-108m	27	3	-7.23E-02	1.83E-01	1.51E-02	8.12E-03		8.56E+04	9.48E+04	1.33E+05	Chebyshev
Am-241	27	13	-4.18E-01	3.38E+01	1.88E+00	1.25E+00		1.12E+07	1.27E+07	1.85E+07	Chebyshev
C-14	3/0[a]	3	3.48E-02	2.86E-01	1.67E-01	1.12E-02		7.82E+05	8.00E+05	8.33E+05	$\bar{x} + z\bar{\sigma}$
Cm-243/244	0/2	2	-6.30E-03	7.07E-02	5.14E-02	1.04E-02		2.80E+05	2.96E+05	3.27E+05	$\bar{x} + z\bar{\sigma}$
Co-60	27	20	-9.71E-02	3.53E+02	4.21E+01	2.10E+01		2.35E+08	2.58E+08	3.56E+08	Chebyshev
Cs-137	27	27	7.34E-02	7.88E+02	1.76E+02	4.22E+01		8.67E+08	9.15E+08	1.11E+09	Chebyshev
H-3	3/0	1	2.57E-01	4.49E+00	1.70E+00	1.56E-01		8.21E+06	8.45E+06	8.91E+06	$\bar{x} + z\bar{\sigma}$
Nb-94	27	15	-1.16E+00	8.41E+00	5.64E-01	3.76E-01	4,326,315	3.38E+06	3.80E+06	5.55E+06	Chebyshev
Ni-59	0/3	0	-1.04E+00	2.70E+00	4.34E-01	5.06E+00		2.99E+07	3.79E+07	5.28E+07	$\bar{x} + z\bar{\sigma}$
Ni-63	0/3	2	3.57E-01	1.07E+02	1.52E+01	1.55E+00		7.43E+07	7.68E+07	8.13E+07	$\bar{x} + z\bar{\sigma}$
Pu-238	0/3	2	2.11E-01	4.06E-01	2.60E-01	3.03E-02		1.29E+06	1.34E+06	1.43E+06	$\bar{x} + z\bar{\sigma}$
Pu-239/240	0/3	3	2.97E-02	1.22E+00	3.15E-01	2.28E-02		1.49E+06	1.53E+06	1.59E+06	$\bar{x} + z\bar{\sigma}$
Sr-90	0/3	3	1.25E+00	4.51E+01	1.23E+01	1.66E+00		6.23E+07	6.49E+07	6.98E+07	$\bar{x} + z\bar{\sigma}$
U-235	27	3	-4.66E-01	3.49E-01	2.04E-02	2.87E-02		1.60E+05	1.92E+05	3.26E+05	$\bar{x} + z\bar{\sigma}$
U-238	0/3	3	5.05E-05	2.68E-04	1.01E-04	8.99E-06		4.87E+02	5.01E+02	5.27E+02	$\bar{x} + z\bar{\sigma}$
							Total Inventory (Bq)	1.30E+09	1.38E+09	1.71E+09	

[a] Refers to the # of individual and # of composite samples analyzed.

Table 28: Fuel transfer system characterization inventory summary

Nuclide	Sample #	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 90% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90 % UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Ag-108m	1/1[a]	0	8.60E-02	1.19E+00	2.09E-01	2.81E-01	8,319,476	4.63E+06	5.45E+06	7.01E+06	$\bar{x} + z\bar{\sigma}$
Am-241	9	3	-2.25E-02	1.12E+02	1.25E+01	1.24E+01		2.07E+08	2.54E+08	4.47E+08	Chebyshev
C-14	1/0	1	1.98E+01	1.98E+01	1.98E+01	1.79E+00		1.84E+08	1.89E+08	1.99E+08	x+zσ
Cm-243/244	1/1	0	1.29E-03	5.98E-01	6.76E-02	1.50E-02		7.05E+05	7.49E+05	8.32E+05	$\bar{x} + z\bar{\sigma}$
Co-60	9	2	-2.50E-03	1.83E+01	2.04E+00	2.03E+00		3.39E+07	4.15E+07	7.30E+07	Chebyshev
Cs-137	9	9	4.98E-02	6.79E+03	7.69E+02	7.49E+02		1.26E+10	1.55E+10	2.71E+10	Chebyshev
Fe-55	9	1	-3.49E-01	7.23E+01	7.88E+00	9.07E-01		7.35E+07	7.62E+07	8.12E+07	Chebyshev
H-3	1/0	1	1.20E+02	1.20E+02	1.20E+02	7.74E+00		1.08E+09	1.10E+09	1.15E+09	x+zσ
Nb-94	9	0	-3.69E+00	-1.87E-01	-9.15E-01	3.56E-01		-4.65E+06	-3.31E+06	2.21E+06	Chebyshev
Ni-59	1/1	0	-3.69E+00	-5.59E-01	-9.07E-01	4.27E+00		3.71E+07	4.98E+07	7.34E+07	$\bar{x} + z\bar{\sigma}$
Ni-63	1/1	1	7.41E-01	1.17E+02	1.37E+01	1.59E+00		1.28E+08	1.33E+08	1.41E+08	$\bar{x} + z\bar{\sigma}$
Pu-238	1/1	2	7.17E-03	8.25E+00	9.23E-01	1.27E-01		8.82E+06	9.20E+06	9.90E+06	$\bar{x} + z\bar{\sigma}$
Pu-239/240	1/1	1	1.74E-03	4.39E+01	4.88E+00	4.25E-01		4.41E+07	4.54E+07	4.77E+07	$\bar{x} + z\bar{\sigma}$

Nuclide	Sample #	# >MDA	Min (Bq/cm^2)	Max (Bq/cm^2)	Mean @ 90% Confidence (Bq/cm^2)	SD (Bq/cm^2)	System Surface Area (cm^2)	90 % UCL (Bq)	95% UCL (Bq)	99% UCL (Bq)	UCL Type
Sr-90	1/1	2	5.26E-01	2.03E+03	2.26E+02	4.53E+01		2.31E+09	2.45E+09	2.70E+09	$\bar{x} + z\bar{\sigma}$
Tc-99	9	0	<6.8E-04	<8.3E-04	Not detected			–	–	–	–
U-235	9	1	-5.62E-02	2.45E+00	2.74E-01	2.73E-01		4.55E+06	5.58E+06	9.80E+06	Chebyshev
U-238	1/1	2	4.78E-05	1.24E-01	1.38E-02	2.08E-03		1.34E+05	1.40E+05	1.52E+05	$\bar{x} + z\bar{\sigma}$
Total Inventory (Bq/cm^2)								1.67E+10	1.98E+10	3.20E+10	

[a] Refers to the # of individual and # of composite samples analyzed.

Table 29: Summary of non-radiological contaminants in WR-1

Contaminant	Quantity (kg)	Uncertainty Factor Range
Potassium Hydroxide	0.01	0.1-10
Boron	0.0009	0.1-10
Lead	40,800	1-3
Xylene	1.9	0.1-10
Palladium	15.5	0.1-10
Chromium	148	1-10
Cadmium	91.4	1-10
HB-40 (OS-84)	87,700	0.5-2
Mercury	0.74	1-10

Table 30: Overall summary of historical WR-1 radionuclide inventory estimates

Reactor System	10 y		30 y		40 y		50 y		100 y		1,000 y		10,000 y	
	Bq (Ci)	%	Bq (Ci)	%	Bq (Ci)	%	Bq (Ci)	%	Bq (Ci)	%	Bq (Ci)	%	Bq (Ci)	%
Reactor Core	5.99E+15 (161,872)	99.97%	1.16E+15 (31,432)	99.91%	9.81E+14 (26,524)	99.92 %	8.92E+14 (24,107)	99.93%	6.30E+14 (17,029)	99.97%	1.51E+13 (407)	99.86%	3.43E+12 (93)	99.77%
Biological Shield	5.76E+10 (1.6)	0.001%					1.25E+09 (0.034)	0.00014 %	5.85E+08 (0.016)	0.0001 %	5.85E+08 (0.016)	0.004%	1.50E+08 (0.0040)	0.004%
PHT system	1.86E+12 (50)	0.03%	1.04E+12 (28)	0.09%	7.99E+11 (22)	0.08%	6.19E+11 (17)	0.07%	2.03E+11 (5.5)	0.03%	1.99E+10 (0.5)	0.13%	7.75E+09 (0.21)	0.23%
TOTAL	5.99E+15 (161,923)		1.16E+15 (31,460)		9.82E+14 (26,546)		8.93E+14 (24,124)		6.30E+14 (17,035)		1.51E+13 (408)		3.44e12 (95.5)	

Table 31: Summary of characterization changes

System	Pre-2018 (Bq)	Post-2018 (Bq)	Bounding Value (Bq)	Where to Find (Pre-2018, Post-2018)
Bioshield	4.1E+09	N/A	4.1E+09	Interpolated from Table 15
Core	1.1E+15	4.77E+14	1.1E+15	Interpolated from Table 12, Table 14
Out of Core	1.1E+12	8.45E+10	1.1E12	Table 17
Total H-3 Out of Core	1.27E+14	2.47E+15	2.47E+15	Table 26
Totals	1.18E+15	2.95E+15	3.53E+15	

Table 32: WR-1 reference documents [34]

File Number	Title	Date
153-121120-TSW	Isotopic Determination of Actinides and Strontium in Swipe samples for Workplace Hazards	2012 January
153-121130-COG-008	Darlington Sample Characterization: Results of Analyses for Actinides and Other Hard-to-Detect Radionuclides	2010 October
153-124000-REPT-011	Nuclear Forensics- Literature Review of Isotopic Attribution Signatures from Major Actinides	2013 May
153-127600-IAD-001	Recent Progress in Actinide Chemistry Research	2010 March
21-005.03_Vol1	WL Building 100 Administrative Level For Tritium	1993 November
21-030.02_Vol1	Building 100 Temporary Operated Instruction 93-01	1993 January
361101-01613-REPT-001	Preliminary Estimates Of AECL's Current And Future Low- And Intermediate-Level Radioactive Wastes	
AECL-11213	Radiological Assessment of ^{36}Cl in the Disposal of Used CANDU Fuel	
AECL-2552	The Commissioning of WR-1	1966 June
AECL-2553	Fuel, Materials and Coolant for WR-1	
AECL-2810	Reactor Organic Coolants: I. Characteristics of Irradiated Hydrogenated Terphenyls	1966 May
AECL-4756	A toxicological assessment of organic reactor coolants HB40	1974 June
AECL-4763	The WR-1 Reactor a General Description	1974 November
AECL-5962	A Review of the Potential For Actinide Redistribution In CANDU Thorium Cycle Fuels	1978 February
AECL-6317	Organic Coolant in Winnipeg Riverbed Sediments	1979 March
AECL-6317	Determination of the Radioactive Inventory of The Remaining Core Components for the WR-1 Reactor	
AECL-839	Detection of Gaseous Fission Products in Water- A Method of Monitoring Fuel Sheathing Failures	1959 May
AECL-8405	Organic Coolants and their Applications to Fusion Reactors	1987 August
AECL-8975	Fission Product Release to the Primary Coolant of a Reactor	1985 June
AECL-9239	Behavior of Depositing Fission Products Released from Defective Fuel	1986
AECL-CONF-792	Characterization of Corrosion Deposits and the Assessment of Fission Products Released From Used CANDU Fuel	1986 September
COG-91-324 / RC-726	Final Report on Zr-2.5nb Pressure Tube Deformation Program In WR-1	
COG-92-025	Characterization of the Actinide Content of Irradiated Fuel Using Gamma Spectrometry of the Fission Products	1992 April
COG-94-458	Scoping Estimates for Ontario Hydro Power Reactor Process and Decommissioning Waste: Prediction and Characterization of ^{36}CI Waste Inventories	
COG-95-250	Preliminary Assessment of Low- and Intermediate-Level Waste Disposal in the Michigan Basin: Release of C-14 from a Cementitious Repository in Devonian Carbonate Rock	

File Number	Title	Date
COG-96-360	Scoping Estimates for AECL Reactor Process and Decommissioning Waste: Preliminary Screening Studies for Intermediate- and Low-Level Wastes	1997 December
COG-96-361	Scoping Estimates for AECL Reactor Process and Decommissioning Waste: Characterization and Immobilization of Aluminum Wastes	
COG-96-369	Preliminary Assessment of Bedrock Disposal of LOW- and Intermediate-Level Waste at the Chalk River Laboratories Site: Performance Assessment	
COG-97-17 /RC-1900	WR-1 Bio Shield Core Analysis	1997
COG-98-328	Modelling of Fission Product Behaviour in the Primary Heat Transport System of CANDU Reactors Under Accident Conditions	1998 October
DM-138	Why Maximum CANDU Potential Lies with Organic Coolant and A thorium Fuel Cycle	1973 January
EXP-WR1-20701	A proposal to decontamination the WR-1L2 loop	1971 January
EXP-WR1-91302	Examination of a Partially-Fuelled Bundle Containing Uranium-Impregnated Graphite, Irradiated in WR-1	1969 January
WNRE-188	WR-1 Reactor Handbook	
RC-1290 R1	The WR-1 Reactor Phase 1 Decommissioning Interim End state Report	
RC-1291-R1	The Monitoring and Surveillance Plan for the WR-1 Deferment Period	1996 March
RC-1554 Rev 1	Technical Basis for Tritium Dosimetry in Korean CANDU Reactors	
RC-172	A Report to the Nuclear Safety Advisory Committee On the Monitoring and Maintenance of the WR-1 Facility During 1988	1989 March
RC-2548	Using Gamma-Ray Spectrometry to Identify and Quantify Sources Of Zr, Nb, Sn, Sb And Fission Products In CANDU Reactors	2001 March
RC-388	A Report to the Nuclear Safety Advisory Committee On The Monitoring And Maintenance of the WR-1 Facility During 1989	1990 March
RC-586	A Report to the Nuclear Safety Advisory Committee on the Monitoring and Maintenance of The WR-1 Facility During 1990	1991 March
RC-593	Project Plan For Decommissioning the WR-1 Reactor: Phase 1	
RC-594	The Overall Decommissioning Plan for the WR-1 Reactor	1991 December
RC-654	WL Source Term	1998 December
RC-779	Annual Safety Review - 1991	1992
SAB-TN-443	Total Radioactive Inventory of the Remaining Core Components for the WR-1 Reactor	1992 October
TN-03-3012	Ingestion and Inhalation Doses from Actinides in CANDU Fuel	2004 May
WDI-112	WR-1L4 and WR-1L5 Final hazards Report and Addendum III WR-1 Final Hazards Report	1972 December
WDI-112	WR-1L4 and WR-1L5 Final hazards Report and Addendum III WR-1 Final Hazards Report	

File Number	Title	Date
WDI-126	Particulates in WR-1 Organic Coolant	1969 January
WDI-129	Some Physical Properties of the HB-40 Coolant Used in WR-1	1969 January
WDI-33	WR-1 Loop Design and Installation Specifications	1965 April
WDI-88	WR-1 Boiling-Water Loop Hazards Evaluation - Addendum II to the WR-1 Final Hazards Report	1968 April
WDI-99	Effect of the Exp-WR1-912 Uranium Graphite Fuel Failure in the WR-1 Reactor	1968 February
WNRE-???	Effect of the Exp-WR1-912 Uranium Graphite Fuel Failure in the WR-1 Reactor	
WNRE-117	WR-1L2 Water Cooled Loop Final Hazards Report for the Modified Loop (Addendum VI to the WR-1 Final Hazards Report)	1973 November
WNRE-118	A Summary of Significant Events and Major Changes that Occurred During Installation of a Third Primary Heat Transport Circuit in WR-1	1973-February
WNRE-144	Significant Changes to WR-1 Operating Systems 1965 to 1976 Addendum VII to the WR-1 Final Hazards Report	1978 April
WNRE-16	WR-1 C Circuit Final Hazards Report Addendum V Final Hazards Report	1972 November
WNRE-16	WR-1 'C' Circuit Final Hazards Report Addendum V	1972 November
WNRE-175	A Compendium of Radiation Dosimetry and Neutron Flux Densities in WR-1 and OCR-500	1974 July
WNRE-210	The Behaviour of Various Designs of (TH, Pu)02 Fuel Elements Irradiated in the 1L2 Look in WR-1	1992 May
WNRE-32	Operating Experience with the WR-1 Organic Cooled Research Reactor	1970 October
WNRE-32	Operating Experience with the WR-1 Organic Cooled Research Reactor	1970 October
WNRE-349	The Conversion of the WR-1 Reactor to (TH, Pu)02 fuel, The (TH, Pu)02 Reference Core for WR-1	1977 March
WNRE-356	WR-1 Uprating Program Addendum VII to the WR-1 Final Hazards Report	1980 September
WNRE-363	The Detection of Organic Coolant Leakage From the WR-1 Reactor Heat Exchangers	1977 April
WNRE-392	Fission Product Release Following a Loss-of-Coolant/Loss-of-Emergency-Coolant Accident in WR-1 With UC or Thoria Fuel	1978 May
WNRE-404	A Report to the Nuclear Safety Advisory Committee on the Operation of WR-1 During 177	1978 May
WNRE-51	A report to the Nuclear Safety Advisory Committee on the Operation of WR-1 and the Performance of its Regulating and Protective Systems During 1970	1971 March
WNRE-590	Failures in WR-1 Fuel Assemblies During 1983	1984 June
WNRE-71	WR-1L6 Organic Cooled Loop Final Hazards Report, Addendum iv WR-1 Final Hazards Report	1972 December

File Number	Title	Date
WNRE-799	Report to the Nuclear Safety Advisory Committee on the Monitoring and Maintenance of the WR-1 Facility during 1987	1988
WNRE-82	Nuclear Safety Advisory Committee on the Operation of WR-1 During 1971	1972 March
WR-1 77-137	Implementation of Administrative Level and Mandatory Action Indices for Organic Coolant Leaks	1977 July
WR1-73-2	WR-1 Unusual Occurrence Report: Failure of Shear Pin in Hanger Rod of Fuel Rod EXP-WR1-601	1973 April 29
WR1-74-3	WR-1 UNUSUAL OCCURRENCE REPORT - FAILURE OF 21 TON TRANSFER FLASK HOIST	1974 December
WR1-76-1	WR-1 Unusual Occurrence Report Failure of an 8 in. Globe 'A' Circuit Flow Throttling Valve	1976 March
WR1-76-1_ ADDENDUM	WR-1 Unusual Occurrence Report Failure of an 8 in. Globe 'A' Circuit Flow Throttling Valve	1976 June
WR1-76-2	WR-1 Unusual Occurrence Report Spill of Hot Organic Coolant From 1l4 Loop	1976
WR-1-76-2	WR-1 Unusual Occurrence Report: Spill of Hot Organic Coolant From 1L4 Loop	1976 December 24
WR1-77-1	WR-1 Unusual Occurrence Report: Freezing of the Moderator System Heat Exchanger	1977 January 13
WR1-77-2	WR-1 Unusual Occurrence Report: Loss of Organic Coolant to the Winnipeg River	1977 March 22 to April 18
WR1-78-1	WR-1 Unusual Occurrence Report Broken 5/16 in. Diameter Cap Screw In the WR-1l6 Loop Inlet Feeder	1977 April
WR1-79-1	WR-1 Unusual Occurrence Report Fission Product Release from The Pneumatic Carrier Facility	1979 August
WR-1-79-1	WR-1 Unusual Occurrence Report: Fission Product Release From the Pneumatic Carrier Facility	1979 May 17
WR1-79-2	WR-1 Unusual Occurrence Report Auto Ignition of Oil Soaked Insulation	1979 November
WR1-80-1	WR-1 Unusual Occurrence Report Stuck Capsule #441c in WR-1 Pneumatic Carrier Facility Site 3	1980 February
WR1-82-1	WR-1 Unusual Occurrence Report Fire in Organic Soaked Lagging	1982 November 02
00216-REP-03902-00003-R003	Reference Low and Intermediate Level Waste Inventory for the Deep Geologic Repository OPG: 00216-REP-03902-00003-R003	

Figures

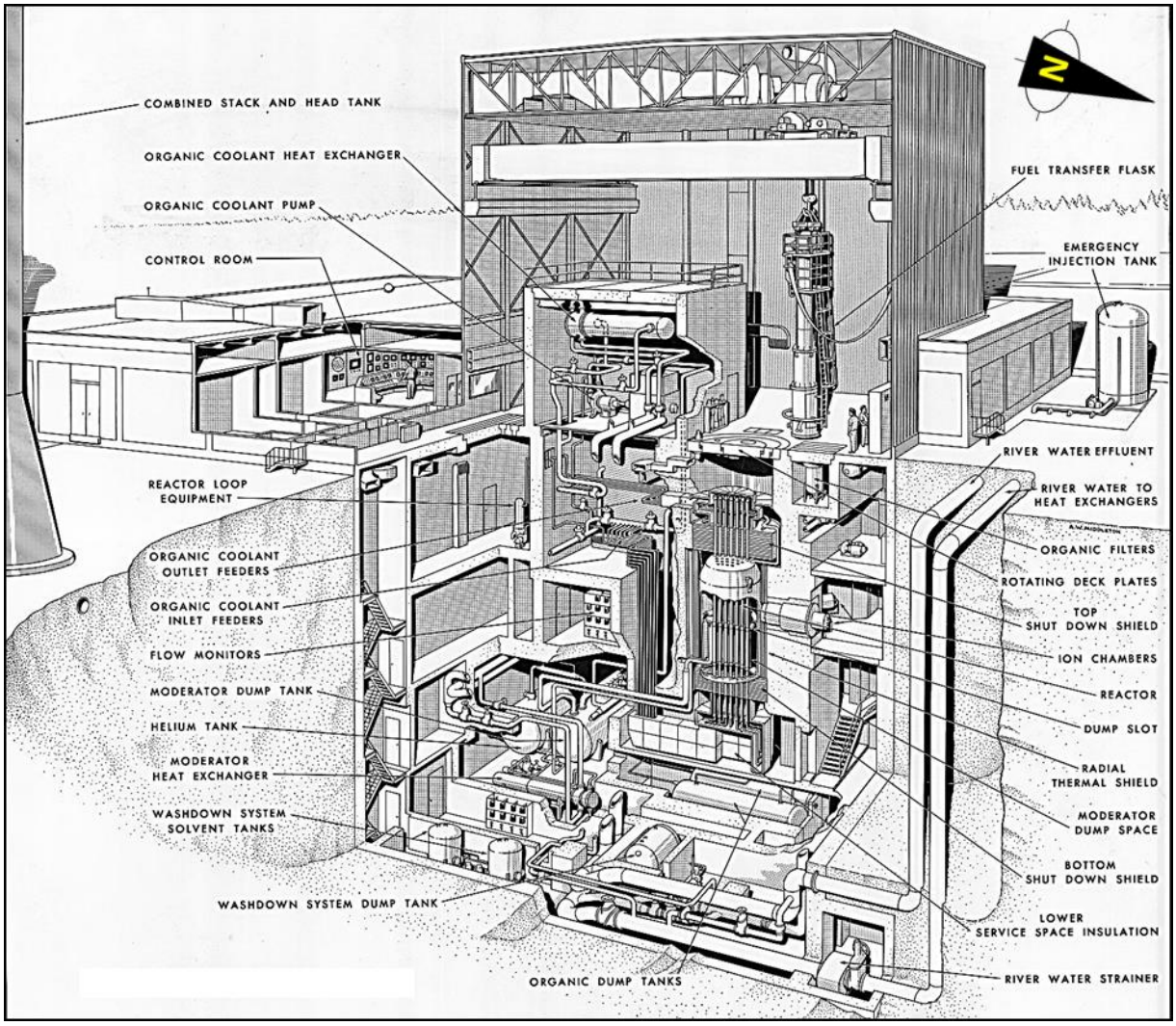


Figure 1: Cross-section of B100 and WR-1

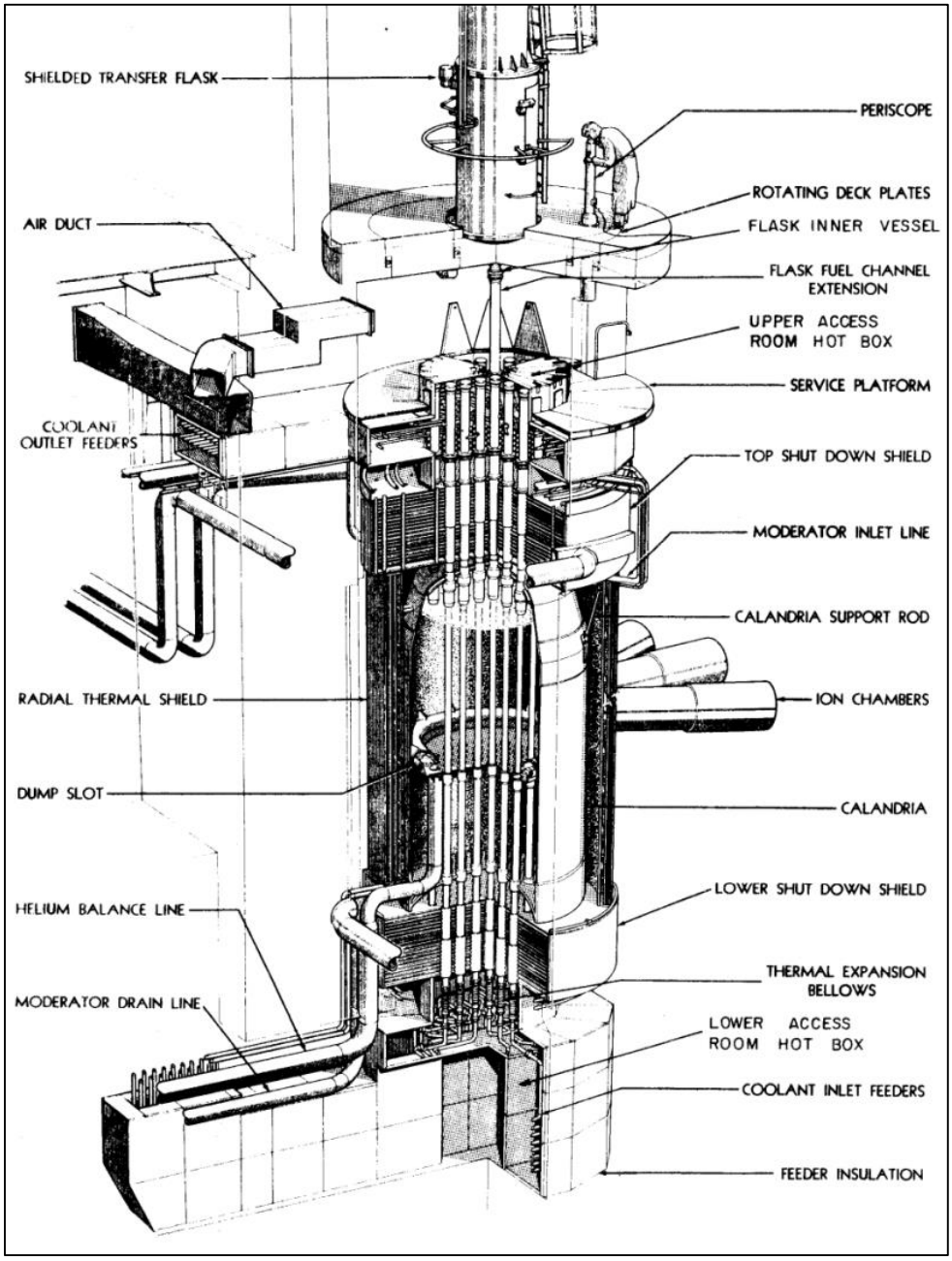


Figure 2: WR-1 reactor layout

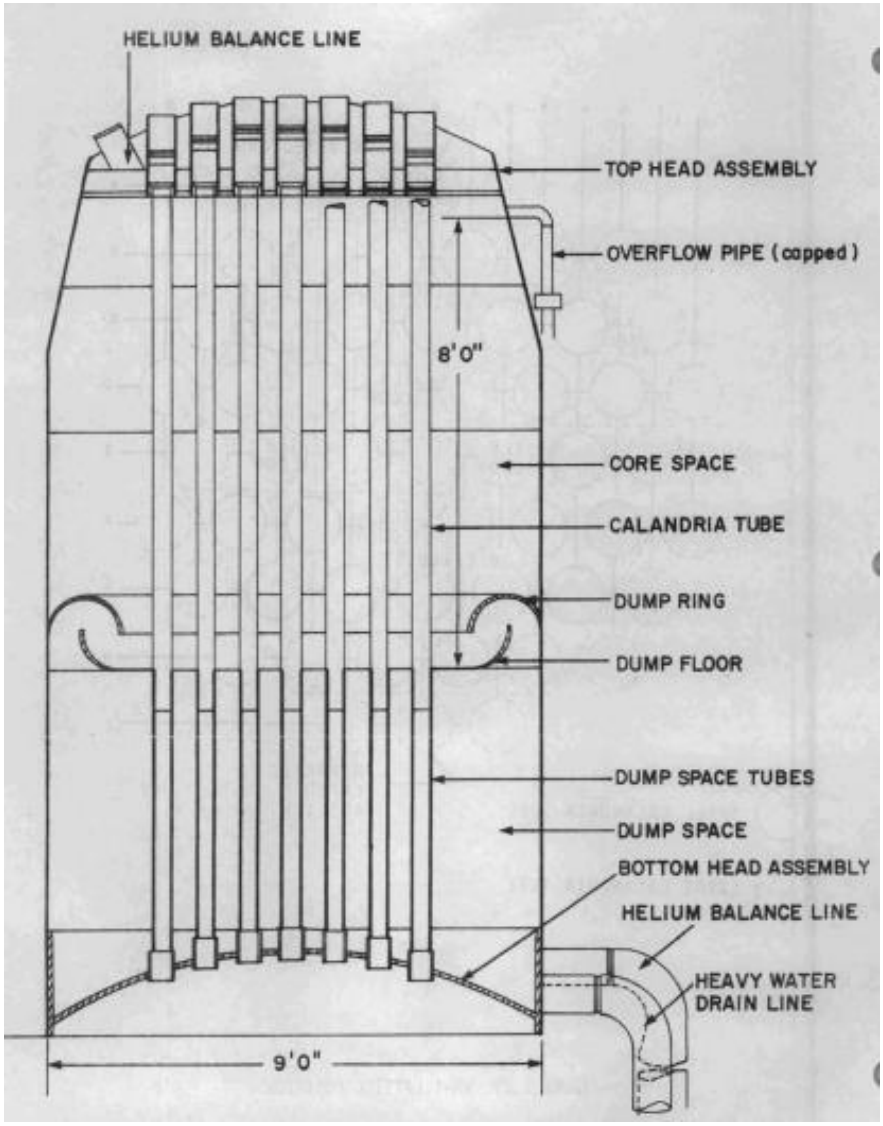


Figure 3: Calandria vessel cross-section diagram

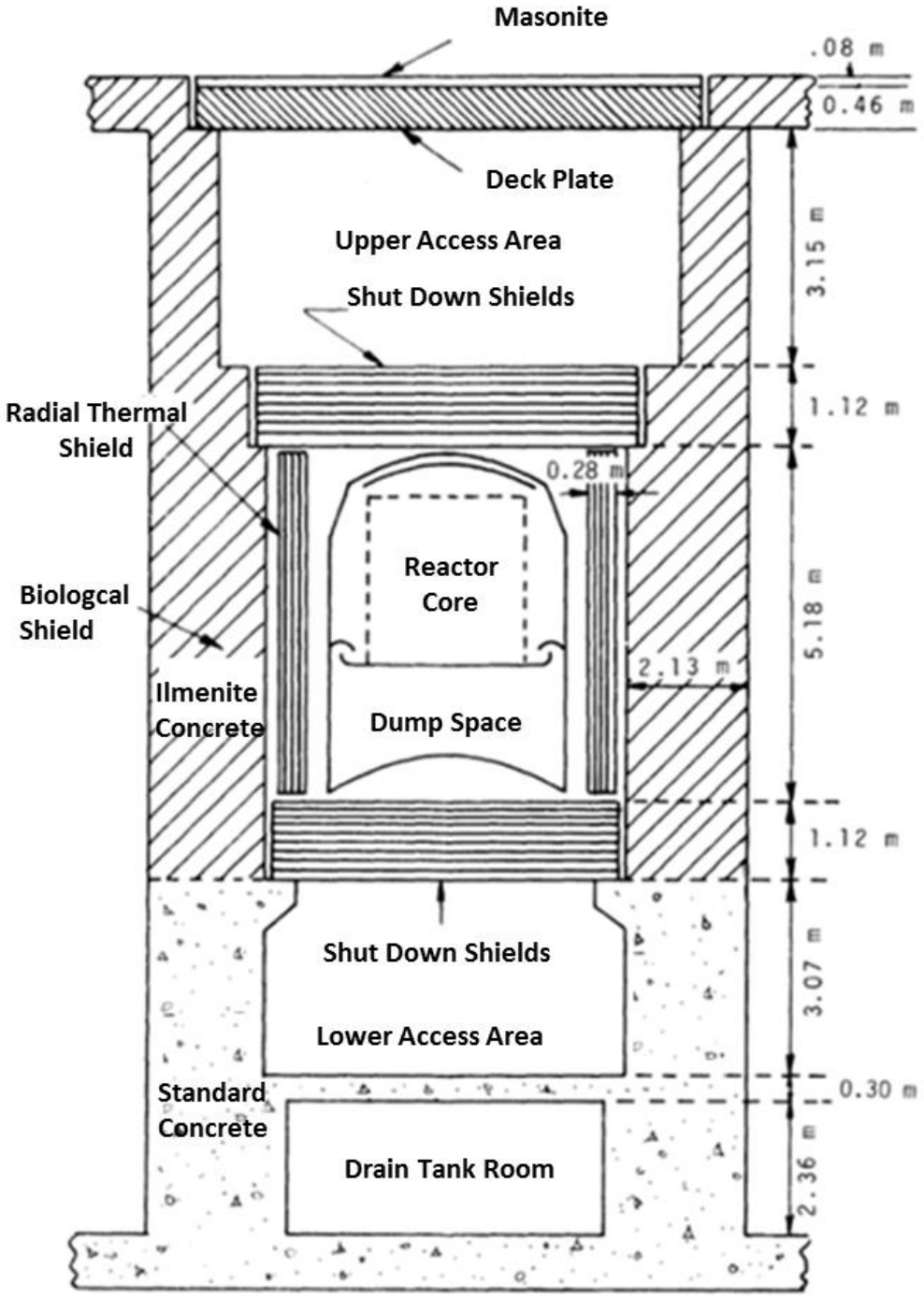


Figure 4: Diagram of the WR-1 main shield system

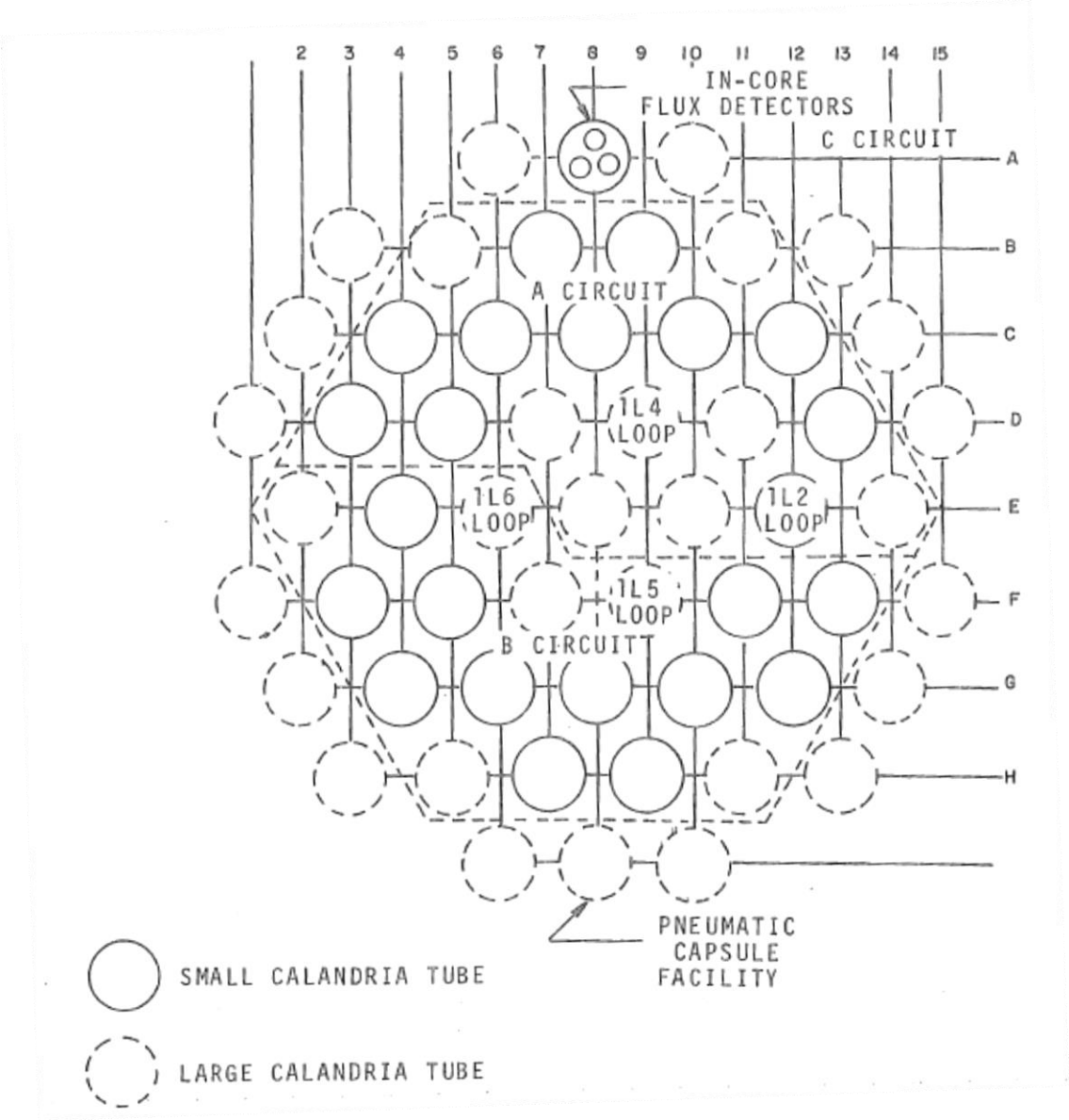


Figure 5: WR-1 calandria tube lattice

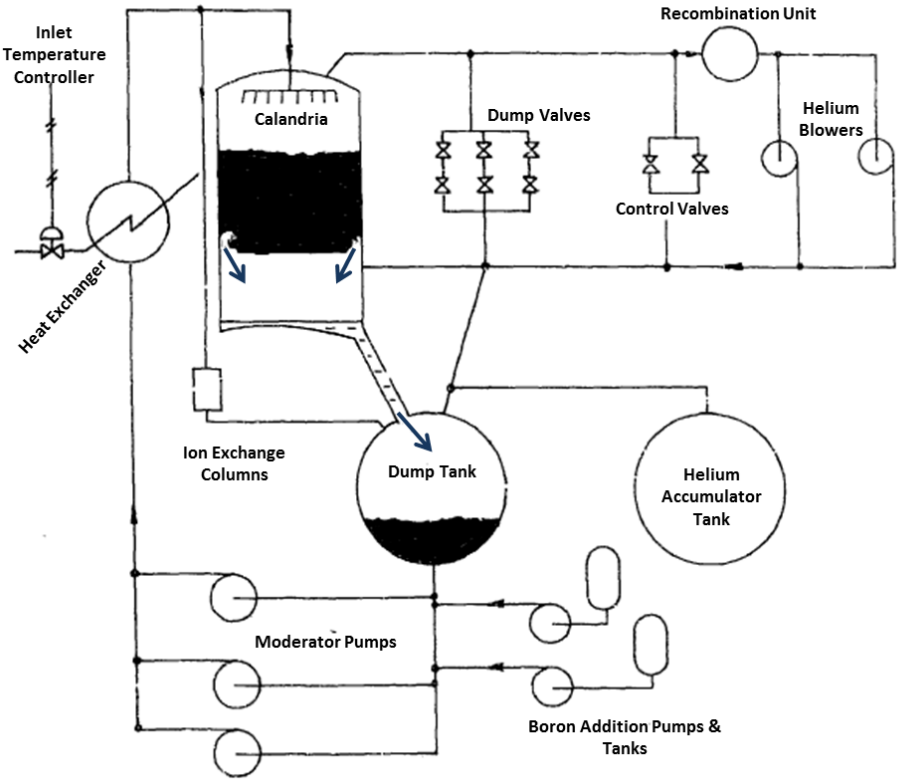


Figure 6: Heavy water and helium systems with associated auxiliary system components

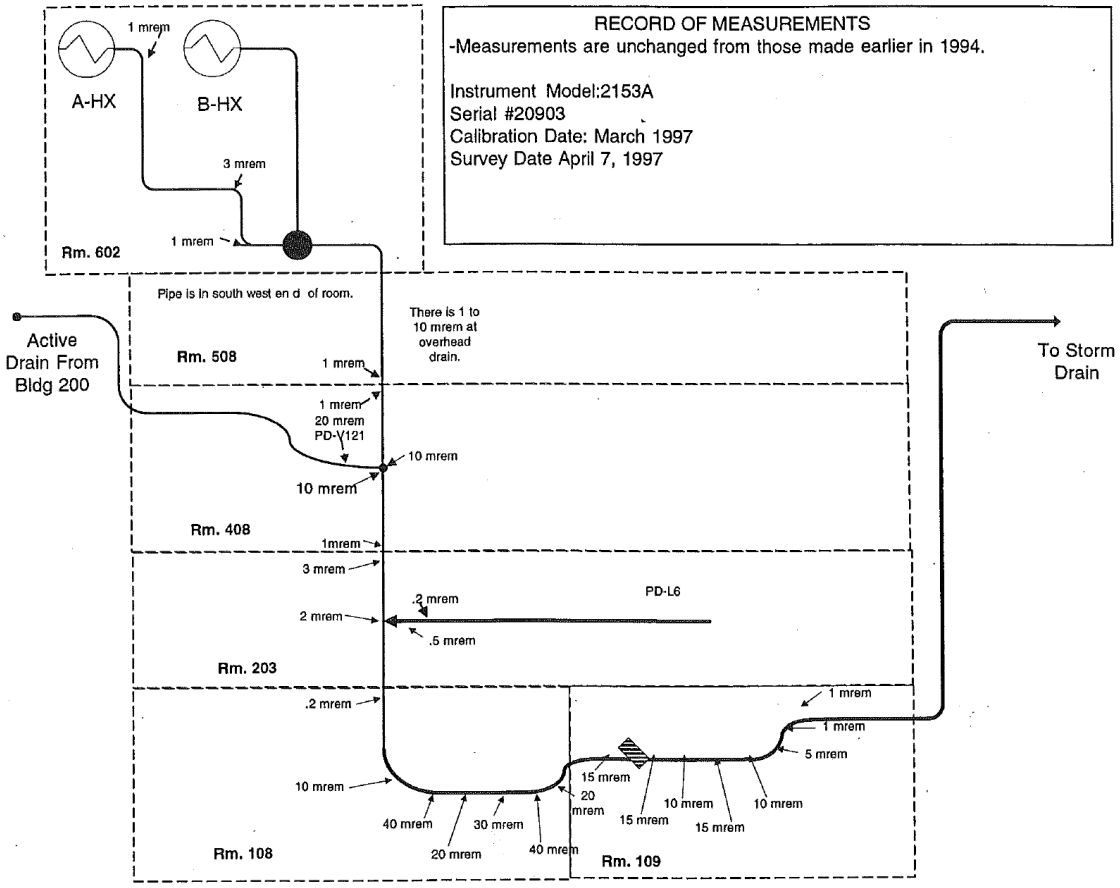
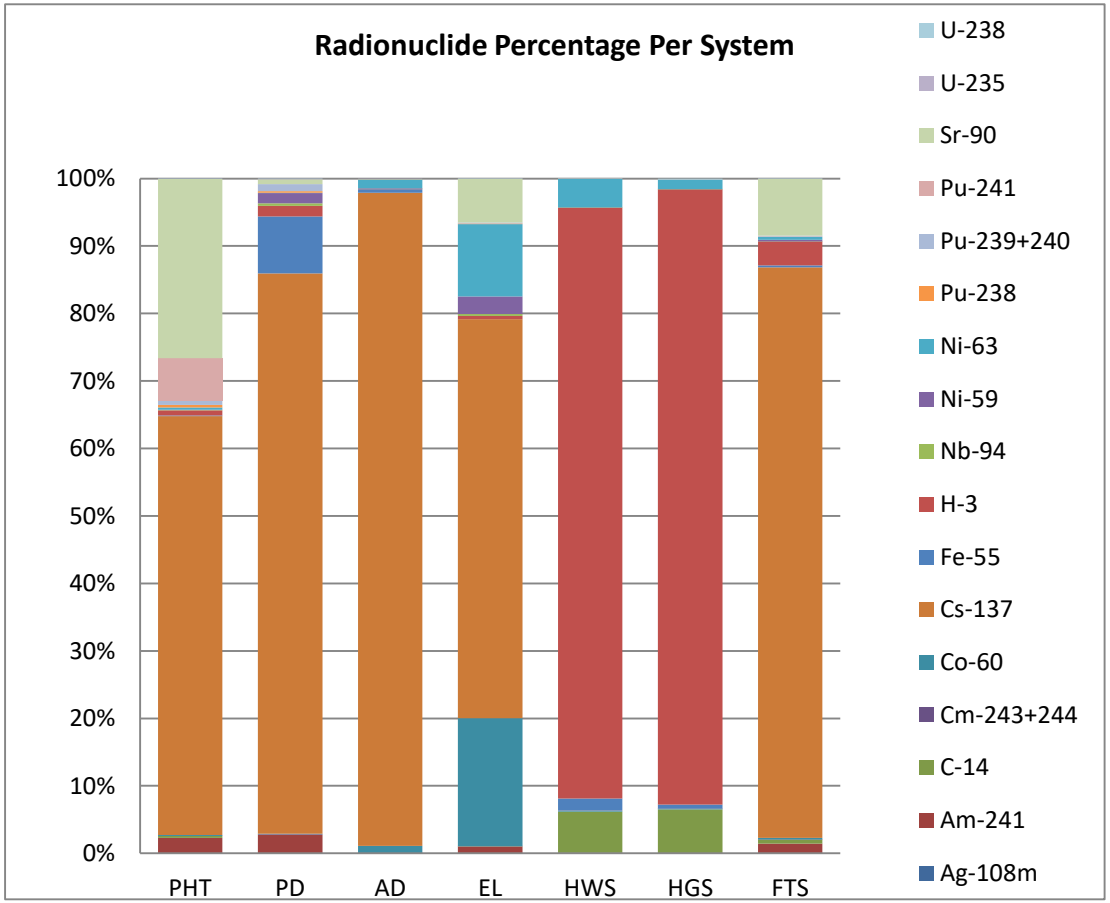
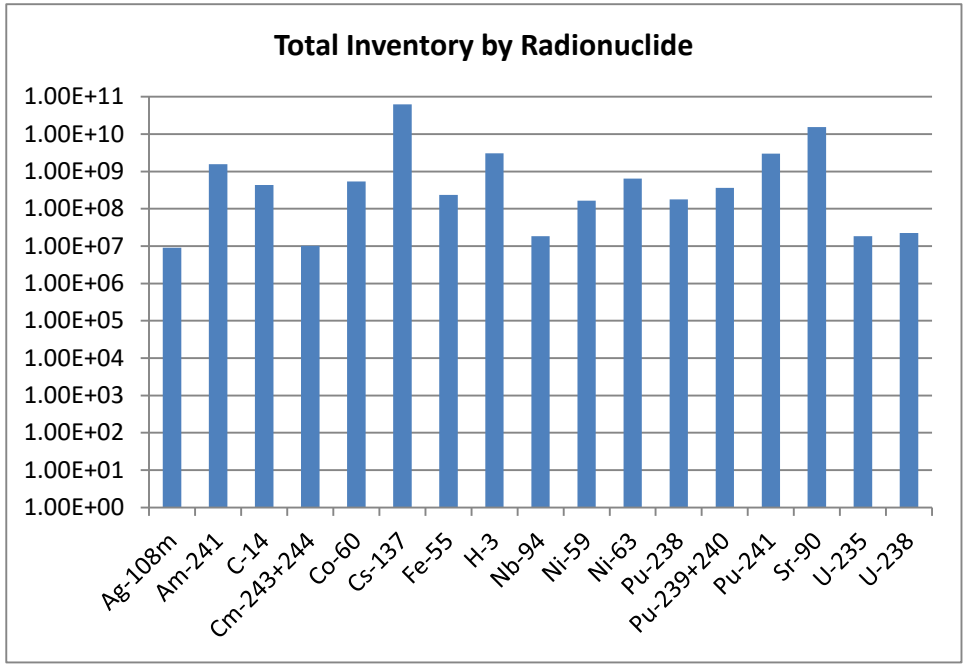


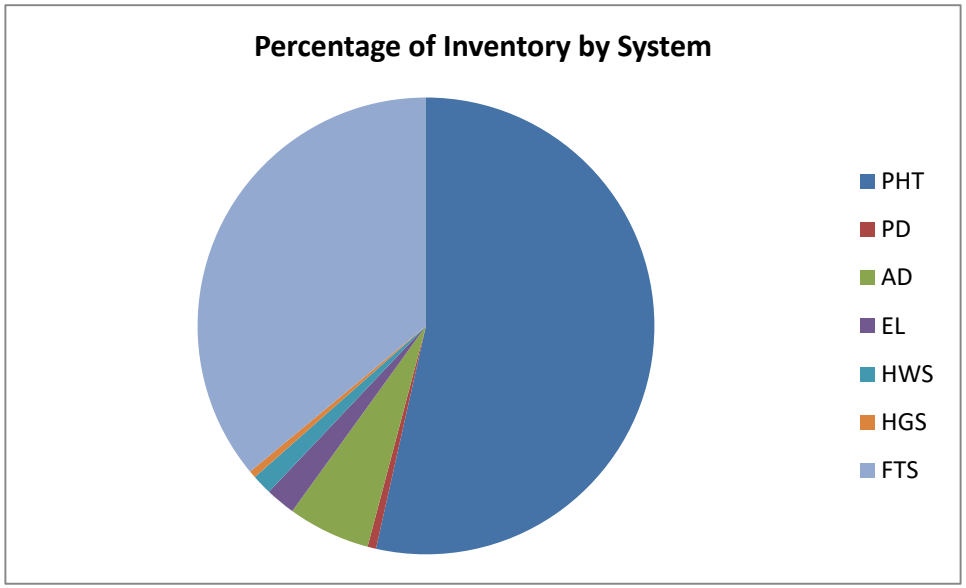
Figure 7: Radiation dose rate surveys of the internally contaminated process drain line



Radionuclide	Total Bq	Percent of Inventory
U-238	2.24E+07	0.03%
U-235	1.84E+07	0.02%
Sr-90	1.55E+10	17.39%
Pu-241	3.00E+09	3.37%
Pu-239/240	3.65E+08	0.41%
Pu-238	1.78E+08	0.20%
Ni-63	6.38E+08	0.72%
Ni-59	1.63E+08	0.18%
Nb-94	1.83E+07	0.02%
H-3	3.05E+09	3.43%
Fe-55	2.34E+08	0.26%
Cs-137	6.33E+10	71.08%
Co-60	5.44E+08	0.61%
Cm-243/244	1.03E+07	0.01%
C-14	4.32E+08	0.49%
Am-241	1.57E+09	1.77%
Ag-108m	9.07E+06	0.01%
Total	**8.90E+10**	**100%**

Figure 8: System ROC percent mixture and total inventory by ROC





System:	Primary Heat Transport	Process Drains	Active Drains	Experimental Loops	Heavy Water	Helium Gas	Fuel Transfer
Percent of Inventory:	53.5%	0.6%	5.8%	2.1%	1.4%	0.5%	36.0%

Figure 9: Total ROC activity and distribution per system

APPENDIX A REVISION HISTORY

Rev. No.	Date	Details of Rev.	Prepared By	Reviewed By	Approved By
0	2016/08/22	Issued as “Approved for Use”.	L. Rasmussen	J. Miller	B. Barrios D. Howlett for B. Wilcox
1	2019/03/22	Issued as “Approved for Use”. Incorporated results of 2017/18 reactor system characterization plan, non-radiological inventory estimates and titium estimations.	L. Rasmussen J. Gordon	J. Miller	B. Wilcox
2	2020/10/28	Issued as “Approved for Use”. Incorporated additional estimates of AG-108m in core components and Tritium in thermal shield water.	J. Miller	J. Gordon L. Rasmussen	B. Wilcox