Replacement Research Reactor Project

SAR CHAPTER 12
OPERATIONAL RADIOLOGICAL SAFETY

Prepared By

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12.1 INTRODUCTION

The objectives for this Chapter of the SAR are as follows:

a) To describe requirements applicable to radiological safety, including the requirements and design bases applicable to the waste management systems.

b) To provide a summary description of the approach to radiological safety adopted.

c) To describe features that contribute to ensuring the radiological safety of operating personnel, members of the public and the environment.

d) To evaluate the approach to radiological safety so as to demonstrate that it meets the identified requirements and, where appropriate, the identified safety design bases.

The areas discussed in this chapter are as follows:

a) The proposed radiation protection plan adopted for the operating plant including the ANSTO policy; organisation, staffing and responsibilities; facilities, equipment and instrumentation; procedures and training.

b) The sources of radiation within the facility.

c) The design aspects relevant to minimising the doses to operating personnel including access control and zoning, the provision of shielding, ventilation systems and radiation monitoring.

d) The approach to radioactive waste management including minimisation of wastes generated and the controlled management of any such wastes that are produced.

e) The dose assessment for both operating personnel and members of the public arising during normal operation.

12.1.1 Radiation Protection Policies

In accordance with national and international recommendations, a system of radiation protection has been adopted for the Replacement Research Reactor Facility (Reactor Facility) to limit doses to facility workers and the public during normal operations and also to reduce intervention doses in the event of a radiological accident. The Radiation Protection Policies applicable to Reactor Facility are based on ANSTO endorsement of the IAEA and ICRP recommendations and are in compliance with the ARPANS Act and Regulations and National Codes of Practice.

It is ANSTO policy (APOL 2.1) that all activities undertaken by ANSTO are conducted in a manner that:

a) Places the protection of human health and safety and the environment as its highest priority;

b) Promotes a positive safety culture and environmental awareness; and

c) Strives for continual improvement in safe working and prevention of pollution.

ANSTO is committed (ANSTO Safety Directive 5.1 Radiation Safety Principles) to maintaining standards of radiation safety recommended by the Australian Radiation Protection and Nuclear Safety Agency, (through the ARPANS Act and Regulations), and the International Atomic Energy Agency (through the Basic Safety Standards). The following radiation safety principles are applied:

a) Justification

Activities, which may give an increase in radiation exposure to individuals, shall be justified and approved by the relevant ANSTO safety assessment process prior to their commencement.

b) Dose limitation

Individual doses due to the combination of exposures from all ANSTO activities shall not exceed the specified annual effective dose limits and equivalent dose limits recommended by the ARPANSA Radiation Protection Series 1 and incorporated in the ARPANS Regulations 1999 and the NOHSC Standard.

c) Optimisation of protection and safety

The magnitude of individual doses, the number of people who are exposed, and the likelihood of incurring exposures to radiation, shall be as low as reasonably achievable, after taking into account economic and social factors (ALARA), (ANSTO Safety Directive 5.2 ANSTO Policy on As Low As Reasonably Achievable). Dose constraints are used for the control of exposure from practices.

d) Defence in depth

Defence in depth approach shall be used in the design and operating procedures of the facility to minimise the potential for failures in protection or safety measures.

e) Safety culture

A positive safety culture shall be encouraged to govern the actions and interactions of all individuals and organisations engaged in the design, construction and operation of the facility.

12.1.1.1 Risks from Radiation

Some of the energy in the radiation emitted by radioactive substances is absorbed by any material that the radiation encounters. This absorption takes place in both inanimate material and in the cells of a living being. Radiation dose reflects the amount of energy absorbed per unit mass of material. This can result in damage to living cells and, if this damage is not repaired, may lead to the development of fatal cancers, non-fatal cancers or genetic effects. There are many other causes of cell damage, such as chemicals in the food we eat and pollutants in the air we breathe, which can cause much higher rates of damage than background radiation.

The body has developed very effective repair mechanisms to deal with the damage caused, so that there is only a very small probability of damage leading to cancer.

At levels of exposure to radiation up to about 100 times that due to natural background radiation, there are no immediate effects. Any effect, were it to occur, would arise some years after the exposure to the radiation.

The absorbed energy itself does not give a direct indication of possible biological effects. Different tissues of the body react differently to different forms of radiation. The sievert, often denoted (Sv), is a measure of the radiation dose to biological tissues which takes these differences into account, and represents both the amount of radiation energy absorbed by living tissue and the extent to which the particular energy transfers can affect biological processes. One thousandth of a sievert is a millisievert and one millionth is a microsievert.

Another unit of dose is the person-sievert, this is the measure of the radiation dose to a group of people. This group may be a specific selection of individuals, a local population, or even the global population. The person-sievert is effectively the sum of the radiation doses to the individuals in a population.

The most widely used model of the effect of radiation in cells is the linear risk model, according to which a specific percentage radiation dose increase always increases the risk by the same percentage amount. The linear risk model is believed to overestimate the risk for most types of late (long after the exposure) health effects in the low dose region because of the effectiveness of the repair mechanism of the human body. An alternative to the linear model is the linear-quadratic risk model. This model suggests that the risk of adverse health effects does not have a linear relationship to radiation dose. At small doses, the curve is gentle but as the dose increases, the curve becomes steeper. Both practical and theoretical tests have been found to support this model.

Another risk model for which there is some physical supporting evidence is the hormesis model. According to the hormesis model, some radiation is beneficial to health, a phenomenon that holds true for many substances. Supporters of this model cite the fact that people who live in mountainous areas, at high altitudes with high radiation levels, suffer less cancer than others do. Although true, this is not necessarily the correct conclusion, as cancer may arise from a number of different causes. However, all of the things around us contain radioactivity naturally. Humankind has evolved in a radiation environment and has sub-cellular processes, which act to repair the types of damage that radiation can cause.

For the purposes of ensuring the safety of people who may be exposed to radiation as a result of the use of radioactive materials, dose limits have been recommended for radiation workers and members of the public by the International Commission on Radiological Protection. The Commission, established in 1928, conducts substantial research, and liaises with the many bodies including the World Health Organisation, International Labour Organisation and the International Atomic Energy Agency. The recommendations of the Commission are based on many considerations. Their recommendations embody the linear model, no matter how much less than background levels the dose of radiation may be. Some experts regard this as a very cautious basis for radiation control.

12.1.1.2 Radiation Dose Limitation

ANSTO are committed to applying national and international recommendations for limiting exposure to ionising radiation and so incorporates ARPANSA RPS 1 into its safety management system through safety directives. (ANSTO SD 5.1, 5.2)

As stated in ANSTO Safety Directive 5.1, the following legal dose limits for both occupationally exposed personnel and the general public are complied with:

a) Limits for occupational exposure

   The effective whole body dose shall not exceed 20 mSv annually, averaged over five consecutive years, and shall not exceed 50 mSv in any single year.

   The equivalent annual dose to specific parts of the body shall not exceed:

   (i) 150 mSv to the lens of the eye

(ii) 500 mSv to the hands and feet

(iii) 500 mSv to the skin (average dose received by any one square centimetre of skin)

As part of meeting this safety criterion, the effective dose from normal operation of the Reactor Facility to any occupationally exposed person is constrained to be less than 15 mSv per year.

b) Limits for public exposure

The effective dose shall not exceed 1 mSv annually

The equivalent dose shall not exceed:

(i) 15 mSv to the lens of the eye

(ii) 50 mSv to the skin (average dose received by any one square centimetre of skin)

Natural background radiation and personal exposure for medical purposes is not considered.

12.1.2 The ALARA Principle

ALARA is an acronym for "As Low As Reasonably Achievable". ALARA is a concept meaning that the design and use of sources, and the practices associated therewith, including their disposal, should be such as to ensure that exposures are kept as low as reasonably achievable, economic and social factors being taken into account.

Reduction of radiation dose and emissions shall be achieved by the application of the ALARA principle as stated in ANSTO Safety Directive SD 5.2 "ANSTO Policy on As Low as Reasonably Achievable".

In the case of practices involving radiation exposure, it is necessary to determine whether the doses are ALARA or not. The steps for that procedure are shown in Figure 12.1/1. In addition, non-radiological and non-quantifiable factors can be considered.

The safety management system at ANSTO includes the application of international and national standards for radiation safety through the Safety Directives 5.1 to 5.7. Together with training programs, safety approval and monitoring systems and an event response system, it can be demonstrated that there is a framework for commitment to good radiation protection.

12.1.2.1 Operational Reference Levels for Doses

Individual radiation doses are minimised in practice through the implementation of the ALARA principle. In addition to dose limits, local control levels are imposed. Annual Limits on Intake, Derived Air Concentrations and Derived Surface Contamination Limits shown in Tables 12.1/1 and 12.1/2, respectively, apply (see ANSTO Safety Directive S.D. 5.3).

Operational reference level is a generic term used for practical measurements of activity, surface and airborne contamination and dose and dose rates and is presented as such in this chapter. Where specific reference levels are used, their specific nomenclature is used, e.g. Derived Air Concentration, Notification level, etc.

The following operational reference levels for doses apply, as defined by IAEA Safety Series No 115 International Basic Safety Standards for the Protection against Ionizing Radiation and for the Safety of Sources:

a) The effective whole body dose to any occupationally exposed person during the normal conduct of experiments, processes and operations is constrained to be less than 15 mSvy^{-1}.

b) Investigation levels are set at 1 mSv effective dose or 40 mSv to the skin or extremity per month for occupationally exposed people. Exposures above these levels require documented investigations and follow up actions to ascertain the source of exposure, determine whether the dose was actually received by the worker and, where practicable, reduce future radiological exposure.

c) The potential effective dose to any member of the public off-site as a consequence of the normal conduct of experiments, processes and operations is constrained to be less than 0.3 mSvy^{-1} for the site as a whole and 0.1 mSvy^{-1} for the Reactor Facility.

d) Where it is shown that the annual radiation dose during the normal conduct of experiments, processes and operations would be less than 2 mSv to any occupationally exposed person, and 0.02 mSv to any member of the public off-site, an ALARA assessment is not required, however other non-radiological factors may be considered.

For the Reactor Facility, ANSTO has committed to an ALARA objective of 0.01 mSvy^{-1} for members of the public with respect to airborne emissions from the LHSTC Site as a whole. This translates into a design objective of less than 0.005 mSvy^{-1} for the Reactor Facility. The following table summarises these values:

Reference Level (Safety Criterion) Descriptor	Occupationally Exposed Worker	Member of the public / non-occupationally exposed worker
Dose limit	20 mSv y^{-1}	1 mSvy^{-1}
Dose Constraint	15 mSv y^{-1}	0.1 mSvy^{-1}Reactor operation (0.3 mSvy^{-1} LH ANSTO)
ALARA objective	2 mSv y^{-1}	0.01 mSvy^{-1} Reactor operation (0.02 mSvy^{-1} LH ANSTO)
Reactor Facility Design objective	<2 mSv y^{-1}	<0.005 mSvy^{-1} (Reactor Facility)
Individual investigation levels	Effective dose ≥ 1mSv per month	N/A

The Reactor Facility design objective value of 0.005 mSvy^{-1} for the most exposed member of the public represents a half percent of the public dose limit and five percent of the public dose constraint.

For airborne releases, dose calculations are shown in Section 12.5.7. The source term used in these calculations (explained in Section 12.2.1 and shown in Table 12.2/5) can be considered as the maximum postulated annual release since it was determined using the worst case upper values between theoretical models and measured values extrapolated from similar facilities. The actual airborne emissions from the Reactor Facility are determined through monitoring during operation of the facility.

The Reactor Facility has been designed to ensure annual doses are below the objective values. Doses may be further optimised using the ALARA principles. Dose estimates for Reactor Facility staff are given in Section 12.5.

12.1.2.2 Operational Reference Levels for Emissions

The Operational Reference Levels for Emissions[1], were established prior to Reactor Facility operation to fulfil the dose constraints and ensure that good engineering and operating practices are implemented and maintained. The operational reference levels for airborne discharges from ANSTO are set out in the May 2001 Discharge Authorisation for ANSTO from ARPANSA. In this document and an amendment in October 2002, Notification Levels are used as values for airborne emissions from stacks to trigger reporting to ARPANSA as soon as these levels are reached. The levels set in the discharge authorisation ensure the achievement of the ALARA objective of 0.02 mSv per year. These operational reference levels are set initially in conjunction with ARPANSA and will be reassessed as feedback from operational experience is acquired.

All releases in excess of the Air Effluents Monitoring Intelligent (AEMi) and Tritium Monitoring systems minimum detection limits are recorded and reported to ARPANSA on a 4 weekly, quarterly and annual basis and when notification levels are reached. These measurements may be performed automatically by the system controlling units, which can be accessed from different supervision units of Reactor Control and Monitoring System (RCMS).

In the unlikely event of a release in the containment air in excess of preset alarm levels, containment isolation will occur automatically to prevent further discharge to the atmosphere (see Chapter 7, Section 7.8.2.3.2).

12.1.3 Radiation Protection Plan

A Radiation Protection Plan (RPP) is provided for the Reactor Facility. This Plan is compatible with the ARPANSA Regulatory Guideline on Review of Plans and Arrangements RP-STD-15-03, August 2003, Version 0. The RPP contains information to demonstrate that arrangements are in place to ensure the radiation safety requirements are satisfied.

The purpose of this Radiation Protection plan is to describe the organisational arrangements for the control of exposures to ionising radiation during all activities involved with the operation of the Reactor Facility. The plan outlines the systems & processes that ensure compliance with standards and regulatory requirements on radiation protection, and the application of optimisation of protection, which contribute to the development of a safety culture in Reactor Facility at ANSTO.

This plan applies for all activities associated with the normal operation of the Reactor Facility and the radiation protection of all personnel. This includes the reactor groups, those who utilise the reactor facilities, support services and personnel within ANSTO and external to ANSTO, including the following activities:

a) Reactor Operation & Maintenance,

b) Radioisotope Irradiation,

c) Neutron Activation Analysis (NAA),

d) Neutron Beams Utilisation,

[1] IAEA Safety Series No 115 International Basic Safety Standards for the Protection against Ionising Radiation and for the Safety Of Radiation Sources

e) Management of radioactive waste, and

f) transport of radioactive materials within and to/ from the facility.

12.1.3.1 Organisation Staffing and Responsibilities for Radiation Protection

The RPP describes the Radiation Safety organisation, including roles and responsibilities. As appropriate, the functional responsibilities of the health physics group in areas such as radiation protection advice, support, training, monitoring, dosimetry and measurement services are included.

The Reactor Manager has responsibility and authority by ANSTO for the safe operation of the facility. The functions of a Radiation Safety Committee are described in the RPP. The Reactor Facility Radiation Protection Adviser (RPA) is an experienced ANSTO Health Physicist appointed to advise and implement radiation safety programs in the Reactor Facility.

All plant personnel have individual responsibility to comply with the Reactor Facility radiation protection measures and controls. Special attention is given to the training of all plant personnel to ensure their awareness of radiological risks, available protection measures, radiation safety practices, procedures and local safety rules.

The Commonwealth Occupational Health and Safety Act requires employees to observe local safety instructions and to use the safety equipment provided. These criteria also apply to contractors, students and research workers who could be deemed occupationally exposed and legally entitled to be within the facility (reference the Safety Management Plan for further detail).
In particular, persons working within the Reactor Facility do not expose themselves or others to radiation to an extent greater than is absolutely necessary for the purposes of their work, and ensure that any dose received does not exceed the given limits.

12.1.3.2 Radiation Protection Facilities, Equipment, Instrumentation, Personal Protective Equipment

The equipment and instruments required to perform a full radiological survey of all foreseeable activities in the Reactor Facility are provided.

The plant fixed area instrumentation measure and register external gamma dose rates in relevant areas (see Figures 12.3/17 to 12.3/21). This instrumentation is part of the Area Radiation Monitoring (ARM), a subsystem of the Radiation Monitoring System (RMS). The ARM provides real time data to the Reactor Control and Monitoring System (RCMS) through intelligent digital, software operated detector arrays. This data include logging the level, status and alarms from all the area radiation monitors (see Section 12.3.5.1). These radiation monitors have local displays and trigger visual and audible alarms if dose rates are higher than pre-determined values that can be set up independently for every location. An independent set of analogue ARM units is Post Accident Monitoring System (PAM) and Reactor Protection Systems (RPS) conformed. These units, installed at various locations inside the Reactor Facility, send their signals directly to the PAM and RPS for processing. In addition, neutron detectors have been located in the reactor beam hall to determine possible neutron fields originating from experimental devices associated with beams and to which personnel could potentially be exposed. These detectors are part of the Neutron Monitoring System (NMS), a sub-system of the RMS (see Section 12.3.5.1). These fixed area instruments also trigger visual and audible alarms if dose rates are higher than pre-determined values that can be set up independently for every location.

Personnel doses are measured using Thermoluminescent Dosimeter (TLD) badges and in special cases, direct reading dosimeters. Techniques for detecting and measuring potential internal contamination are available.

Portable monitors are used to complement/validate the information provided by fixed detectors and to survey specific operations or procedures. Portable instruments are also used to measure objects leaving areas where contamination is expected as well as to release material to users outside the Reactor Facility. Potentially contaminated surfaces are surveyed using portable instruments or applying smearing techniques. Portable air samplers are also provided.

In order to reduce the possibility of contamination spread, measurement of potential external contamination of persons are controlled routinely as close as possible to the contamination source by means of portable equipment. Walk through monitors (portals) are also provided at the main entrances to the Reactor Facility. These 2 (two) portals are part of the Personnel Contamination Monitoring (PCM), a subsystem of the RMS dedicated to personnel monitoring.

Releases through the stack are continuously monitored by several detectors, which in conjunction allow measuring and registering the activity of aerosols, iodine, tritium and inert gases contained in the release. The RMS through the Air Effluents Monitoring (AEM) dedicated to the aerosols, iodine and inert gases surveillance and the Tritium Monitoring System (TRM) performs this on line monitoring.

The AEM comprises two set of detectors. One set is connected to the Reactor Control and Monitoring System (RCMS). The second set of detectors is conformed to PAM and RPS. Their signals are used by the RPS to trigger alarms and initiate containment isolation actions (see Chapter 7, Section 7.8.2.3.2). They also track the emissions in eventual post-accident conditions allowing a post accident follow up through the PAM.

Potentially contaminated liquid wastes and low level waste effluents streams are measured on-line and independently by dedicated Waste Streams Monitor (WASMO) gamma detectors at the inlet to their corresponding waste storage tanks. WASMO also forms part of the RMS. If liquids that can potentially exceed the allowed limits are detected, they can be managed by the basin valves manifold and then handled off-line. Additionally, dedicated Liquid Effluent Monitor (LEM) detectors measure the activity of the liquids discharge from the two waste storage tanks through lines LHSTC B and C (low level radioactive liquid waste and normally non-radioactive liquid waste respectively), towards ANSTO Effluents Processing Plant. The LEM is part of the RMS and supports the waste discharge commands of the Radioactive Liquid Waste Management System (RLWMS), thereby allowing waste accountability. LEM triggers local and remote (RCMS) alarms signals in case that allowed limits are being approaching or exceeding. It also triggers the closure of the waste connection pipelines if values are higher than the maximum specified limit.

The monitoring of primary and secondary coolant fluids and for potential failure in fuel element cladding and rigs complete the radiation surveillance tasks performed by the RMS. The Active Liquid Monitor (ALMO) monitors continuously both the gamma activity of the Reactor and Service Pools Cooling System (RSPCS) and of the Primary Cooling System (PCS). The Secondary Water Activity Monitor (SAMO) monitors continuously the non-active water from the secondary circuit to detect possible leaks of primary coolant on the heat exchangers that could lead to contamination spread. The Failed Fuel Elements Monitor (FFEM) accomplishes additional detection of possible fuel element and molybdenum rigs failures by monitoring continuously the delayed neutron levels in the PCS and RSPCS water. These monitors trigger local as well as remote (to RCMS) indications and alarm signals.

12.1.3.3 Written Operating Procedures and Training

Written procedures are implemented and maintained covering activities that have radiological safety significance. The procedures include sufficient information to meet the requirements of all the higher-level documents.

12.1.3.4 Radiological Monitoring Program

Routine, non-routine, task-related and special monitoring programs are implemented in all radiological classified areas of the plant. These programs include the determination of the following:

a) external beta, gamma and neutron radiation fields

b) contamination of objects, surfaces and persons

c) air contaminants

d) activity of effluents and wastes as necessary

The frequency of such measurements depends on the work carried out in each area. The monitoring programs are described in Section 9 of the RPP.

Radiological monitoring and control regarding plant and personnel protection includes individual and area monitoring. These controls represent an essential part of the RPP that allow evaluation of compliance with the established limits and provide information related to changes in exposure levels that may indicate the need to adopt corrective measures.

When the results indicate the need to implement corrections in plant operation and maintenance, they are immediately reported to the Reactor Manager or relevant individuals/groups so that the corresponding corrective measures are applied.

Research projects or special operations (non-routine) that require the handling of radioactive sources, contaminated materials, etc. are included in a monitoring program after assessment.

12.1.3.5 Effluent/Emission Monitoring

As mentioned in section 12.1.3.2, liquid effluents originating at the Reactor Facility are monitored prior to their transfer to the LHSTC waste management system. On-line monitoring of gaseous effluents is carried out in order to detect and record all discharges and monitor changes in levels or abnormal trends. Reporting of the levels of discharge is performed within ANSTO and then to ARPANSA.

The following are the basic goals of effluent measuring systems:

a) Verify that activity releases are lower than the notification levels (see Section 12.1.2.1).

b) Provide data so that population exposure may be determined by means of an appropriate environmental model as agreed with ARPANSA.

c) Provide information about the proper functioning of the plant and of its effluent treatment system.

d) Quickly detect and identify the nature and extent of unplanned releases.

e) Activate emergency warnings and the response system.

f) Provide information leading to a better knowledge of effluent behaviour and of radioactive material dispersed in certain surrounding areas.

g) Provide online release records.

h) Provide capability to determine the activity - total and by radionuclide – being released.

The online monitoring units and registers for liquid (LEM) and gaseous radioactive effluents (AEM and TRM) are part of the Radiation Monitoring System and are connected to the Reactor Control and Monitoring System (RCMS). Should reference values be exceeded, both systems trigger actions to prevent further releases. A description of these monitoring systems, their specific functions and requirements are presented in Section 12.3.5.

Information on measured environmental radiation at the LHSTC site and its vicinity is reported in the ANSTO annual environmental reports (E Reports). The first of such surveys covering the period 1960-64 was published in 1966 (Giles 1966), and the most recent are given in Hoffmann et al 2002-2003. These reports provide results of measured radioactivity and radiation levels for airborne emissions, liquid effluent, and potential radiation exposures.

12.1.3.6 Fluid System Monitoring

The following liquids are monitored for radioactivity within the Reactor Facility:

a) Primary Cooling System (PCS), and Reactor and Service Pools Cooling System (RSPCS): both circuits are measured online using gamma detectors that belong to the Active Liquid Monitor (ALMO) system (see Section 12.3.5.8). These fluids are also monitored on line using delayed neutron detectors that belong to the Failed Fuel Element Monitor (FFEM). Failure of a Fuel Assembly or molybdenum rig, while unlikely, would also be detected by this monitoring.

b) Secondary Cooling System: Secondary Water Activity Monitor (SAMO) equipment is used to measure secondary cooling water downstream of the heat exchangers (see Section 12.3.5.6).

c) Liquids collected by the Active and Trade Waste Collection Network: are measured on entry to their collection tanks by online detectors that belong to the WASMO system.

Waste from laboratories with concentrations higher than allowed for B line (low level radioactive liquid waste) disposal are managed off-line through dedicated manifolds attached to the basins.

12.1.3.7 Classification of Areas, Persons and Tasks

Areas are classified with regard to radiological significance (reference section 12.3.1.1 for classification/ zoning of areas). Radiological training of personnel is commensurate with the type of work they perform.

12.1.4 On Site Movement of Irradiated Materials

12.1.4.1 Introduction

A complex of hot cells and shielded transportation tubes is in place to minimise personal exposure during the irradiation and post-irradiation processing of samples. Detailed information on these systems is provided in Chapter 11.

The Long Residence Time General Purpose Irradiation Facilities are used to irradiate samples transferred through pneumatic means from the cells in the Above Pool Hot Cell Complex (APHCC) to their respective rigs, located in the irradiation tubes of the Reflector Vessel within the reactor pool.

There are two hot cells dedicated to this facility, namely A and B. The transfer tubes pass through the floor of the cells and are arranged in the technical floor. Along this path, the transfer tubes are shielded.

The samples are loaded into cans (metal capsules of approximately cylindrical shape) for irradiation. Should the samples be powder, two cans are used, one inside the other, to provide more secure containment of the contents.

Once the cans are in the loading position of the Long Residence Time (LRT) terminal, the desired irradiation position is selected and the sample sent automatically. After the irradiation period, which may range from one minute to the whole reactor operating cycle – the process reverses and the LRT terminal collects the irradiated can and deposit it in an unloading position.

The irradiation target is received in the decay-station contained in the loading/unloading device in the PHCs, where it remains during the prescribed time interval. Regarding the activity of the sample, two possibilities exist:

a) The activity is low enough to allow sending the can automatically through the IPTS, from the PHC to the Radioisotope Production Plant in Building 23.

b) The activity is over the prescribed limit, with the following possibilities:

   i) the can is returned to the decay station

   ii) the can is sent to a loading/unloading station module

12.1.4.2 Hot Cells

The cells are part of the Radioisotope Handling System (RHS).

The following are the cell functions:

Reception of samples for irradiation.

Reception of the irradiated material.

Sample handling to measure activity (manual probe).

Transfer of cans to the LHC.

Transfer of cans to and from the Radioisotope Production Plant.

The cells are not used for the opening or closing of cans, so the chances of contamination are related only to the potential for can failure.

Different hot cells are available in the Reactor Facility. These are the Loading radioisotope Hot Cell (LHC) and the Above Pool Hot Cells Complex (APHCC), which consists of the Transfer Hot Cell (THC) and two Pneumatic conveyor Hot Cells (PHCs). .

Some cells are just used as an intermediate station for samples, before they are sent to another station, and other cells may contain a source for a certain amount of time.

Pressure within the cells is kept at a level below that of the surrounding environment by a dedicated ventilation system with absolute and charcoal filters located into a dedicated room.

12.1.4.3 Small Inter-building Shielded Transfer Container

This container is used to transfer samples to other Buildings within the LHSTC. The Small Inter-building Shielded Transfer Container (SIC) is mounted on a cart for its movement.

The SIC is transferred by means of the heavy-duty service lift, then by means of the cask loading monorail to the truck bay for waiting or directly to the truck for transport. The container is loaded from its top and through a tube joining the inside of the cell with a station to which the former connects.

12.1.4.4 Bottom Loaded 6 and 10-tonne Transfer Casks

These containers are used to transfer higher activity samples than the Small Inter-building Shielded Transfer Containers. The Casks are loaded through the bottom and have a check valve and an internal hoist system that allows the sample to be raised from the working tray within the LHC, to the inside of the container. The Cask shield materials are lead and steel.

The loading station comprises a conduit through the cell ceiling, blocked by a movable sealed door that opens once the container is positioned for loading. Transfer from the Above Pool Hot Cells Complex to the Loading Radioisotope Hot Cell

This is carried out through Transfer Pipes and elevators. The inter-hot cell elevator (ICE) is intended to load only one target at a time. The elevator has enough space to carry a sealed cartridge which can contain a leaking target can.

There are two ducts for the ICE, one in operation, the other for backup, which are operated from the APHCC. These ducts follow the most favourable path to remain within the reactor concrete block, observing the specified shielding thickness. The ducts are designed with a minimum number of curves of the highest possible radius and enter the cell through the ceiling against the back wall.

The unloading stations at the low end of the ICE ducts have an isolation system which, together with a similar system on the upper stations makes a double valve system. This prevents communication between the ventilation of the LHC and that of the APHCC and maintains containment. The stations are within normal window viewing range and are easily accessed by means of the Master Slave Manipulators.

12.1.4.5 Transfer Pipes

There are four ducts for this purpose coming from PHCs A and B. These ducts follow the most favourable path to remain within the reactor concrete block, observing the specified shielding thickness. They are designed with a minimum number of curves of the highest possible radius and enter the cell through the ceiling against the back wall. Standard stainless steel pipe, is the material chosen for the TPs.

12.1.4.6 Transfer between Pneumatic Conveyor Hot Cell and Irradiation Positions

This is carried out by means of a Long Residence Time (LRT) pneumatic transfer system using nitrogen-gas as the fluid to drive the cans to and from their irradiation positions. There is one tube per irradiation position that carries the can to and from such irradiation position. Pipes follow a path that allows the specified shielding thickness to be maintained. Pipes are designed with a minimum number of curves of the highest possible radius. The pipe material is stainless steel into the cells, stainless steel into the Technical Floor area and aluminium into the reactor pool. The pipes exit the cells through their floor by means of a tubing flange.

Cans are loaded into and unloaded from the system, by means of the terminal station. These units take the irradiation cans and send them to their programmed irradiation positions. The terminal station also receives cans and carries them to an unloading position on the cell working tray. The terminal stations are within window normal viewing range and are easily accessed by means of the Master Slave Manipulators.

12.1.4.7 Transfer between Cells and the Radioisotope Production Plant Building

This is carried out by means of an IPTS using air as fluid and driving special cartridges (carriers) that contain the samples. The material is stainless steel inside and outside the reactor block. Shielding along the pipes path is equivalent to 600 mm of heavy concrete.

There are two ducts connecting each cell with the Radioisotope Production Plant Building 23. These pipes exit the cells through their floor. Cans are loaded into and unloaded from the system introducing them with the Master Slave Manipulators to a special cartridge (carrier) that is later placed in a station.

The maximum allowed activity for the samples to be sent through the IPTS is for a radiation level equivalent of 40 GBq of cobalt-60. The station is within window normal viewing range.

12.1.4.8 Transfer between Service Pool and the Transfer Hot Cell

The activated samples stored in the Service Pool can be loaded into the Transfer Hot Cell (part of the APHCC) using the Service Pool Elevator (SPE). This elevator has an interlock system that prevents the movement of molybdenum targets without enough decay.

The SPE shielding is provided by the reactor concrete block.

12.1.4.9 Spent Fuel Storage and Handling

All spent fuel storage and handling, from the time spent fuel assemblies are withdrawn from the reactor core until spent fuel is loaded into a transport cask and shipped off-site, can occur within the reactor building. The design of the reactor facility has eliminated the need for the routine movement of spent fuel around the Lucas Heights Science and Technology Centre.

The Fuel Storage and Handling System is described in Chapter 10.

12.1.5 Audit and Review Programs

12.1.5.1 Performance Indicators

12.1.5.1.1 Individual Monitoring

Dose results from dosimetry programs are periodically reviewed and compared to investigation levels and dose constraints. An indicator of the Radiation Protection Plan & specifically the monitoring programs, dose minimisation and limitation may be the number of investigations and the relevant actions following each investigation.

Collective effective dose values (man-mSv per year or month) and average individual effective dose values (mSv per year or month) for various work groups may be identified over a specified time period as performance indicators by the Reactor facility management and RSC, following initial exposure data collection.

12.1.5.2 ALARA & Event Reports

Results of ALARA assessments and the number of assessment reports required for radiation workers in the Reactor Facility may be an indicator of the effectiveness of optimising the radiation protection techniques adopted.

Events that are subsequently assessed as incidents or accidents will be reported following safety response and assistance under the procedure in SD 4.1 ANSTO Event Response System. The outcomes of radiological incident or accident events and the findings and recommendations may be considered as an indicator of radiation safety in Reactor Facility.

12.1.5.2.1 Periodic Review of the Plan

A periodic review of this plan is undertaken at a frequency in line with the Quality Management System for Reactor Facility.

End of Section

Table 12.1/1 Most Restrictive Annual Limits on Intake and Derived Air Concentrations for Some Radionuclides (Assuming 5 µm Activity Median Aerodynamic Diameter for the Inhalation Pathway)

Radionuclide	Annual Limits on Intake (ingestion) Bq	Annual Limits on Intake (inhalation) Bq	Derived Air Concentrations Bq m^{-3}
H-3 gas		1.11×10^{13}	4.63×10^{9}
HTO	1.11×10^{9}	1.11×10^{9}	4.63×10^{5}
C-11 Vapour		6.25×10^{9}	2.60×10^{6}
C-11 Dioxide		9.09×10^{9}	3.79×10^{6}
C-11 Monoxide		1.67×10^{10}	6.94×10^{6}
F-18		2.15×10^{8}	8.96×10^{4}
Na-22	6.25×10^{6}	1.00×10^{7}	4.17×10^{3}
P-32	8.33×10^{6}	6.90×10^{6}	2.87×10^{3}
S-35 inorganic	1.05×10^{8}	1.82×10^{7}	7.58×10^{3}
Cl-36	2.15×10^{7}	3.92×10^{6}	1.63×10^{3}
Ar-41			1.51×10^{4}
Sc-48	1.18×10^{7}	1.25×10^{7}	5.21×10^{3}
Cr-51	5.26×10^{8}	5.56×10^{8}	2.31×10^{5}
Mn-54	2.82×10^{7}	1.67×10^{7}	6.94×10^{3}
Mn-56	8.00×10^{7}	1.00×10^{8}	4.17×10^{4}
Co-57	9.52×10^{7}	3.33×10^{7}	1.39×10^{4}
Co-58	$2.70 x10^{7}$	1.18×10^{7}	4.90×10^{3}
Fe-59	1.11×10^{7}	6.25×10^{6}	2.60×10^{3}
Co-60	5.88×10^{6}	1.18×10^{6}	4.90×10^{2}
Ni-63	1.33×10^{8}	3.85×10^{7}	1.60×10^{4}
Zn-65	5.13×10^{6}	7.14×10^{6}	2.98×10^{3}
Ga-67	1.05×10^{8}	7.14×10^{7}	2.98×10^{4}
Br-82	3.70×10^{6}	2.27×10^{7}	9.47×10^{3}
Kr-85			3.64×10^{6}
Sr-90	7.14×10^{5}	2.60×10^{5}	1.08×10^{2}
Y-90	7.41×10^{6}	1.18×10^{7}	4.90×10^{3}
Mo-99	1.67×10^{7}	1.82×10^{7}	7.58×10^{3}
Tc-99m	9.09×10^{8}	6.90×10^{8}	2.87×10^{5}
Tc-99	2.56×10^{7}	6.25×10^{6}	2.60×10^{3}
Cd-109	1.00×10^{7}	2.08×10^{6}	8.68×10^{2}

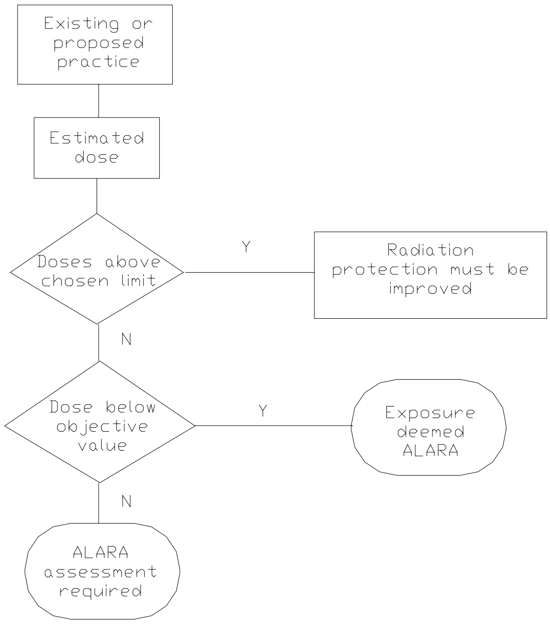
Radionuclide	Annual Limits on Intake (ingestion) Bq	Annual Limits on Intake (inhalation) Bq	Derived Air Concentrations Bq m^{-3}
In-111	6.90×10^{7}	6.45×10^{7}	2.69×10^{4}
I-123	9.52×10^{7}	1.82×10^{8}	7.58×10^{4}
I-125	1.33×10^{6}	2.74×10^{6}	1.14×10^{3}
I-131	9.09×10^{5}	1.82×10^{6}	7.58×10^{2}
Xe-133			6.67×10^{5}
Cs-134	1.05×10^{6}	2.08×10^{6}	8.68×10^{2}
Cs-137	1.54×10^{6}	2.99×10^{6}	1.24×10^{3}
U-nat (measured as U-238)	1.3×10^{4}	2.5×10^{2}	1.0×10^{-1}

Table 12.1/2 Derived Surface Contamination Limits

10 Bq cm^{-2}	100 Bq cm^{-2}		1000 Bq cm^{-2}		
Sr-90	Na-22	Ag-110m	H-3	Br-77	Ce-139
Ra-223	P-32	Cd-109	C-14	Rb-81	Ce-141
Ra-224	Co-56	Cd-115m	Na-24	Sr-85	Nd-147
	Co-60	In-113m	S-35	Sr-87m	Gd-153
	Cu-64	Sb-124	Cl-36	Y-87	Tb-160
	Cu-67	I-125	Ca-45	Y-88	Eb-169
	Zn-65	I-131	Ca-47	Mo-99	Tm-170
	Ga-68	Cs-134	Sc-46	Tc-99m	Yb-169
	Se-75	Cs-137	Sc-47	Tc-99	Lu-177
	Rb-86	La-140	Cr-51	Ru-103	Hf-181
	Sr-89	Pm-147	Mn-54	Ag-111	W-185
	Y-90	Eu-152	Fe-55	In-111	Re-186
	Ru-106	Eu-154	Fe-59	Sn-113	Ir-192
		Bi-210	Co-57	Sb-125	Au-198
	All other non-alpha emitting nuclides not referred to elsewhere in this table		Co-58	I-123	Hg-197
			Ni-63	Cs-129	Hg-203
			Ga-67	Cs-131	Tl-201
			Ge-68	Ba-133	Tl-204

End of Tables

Figure 12.1/1 Steps in ALARA Assessment



End of Figures

12.2 REACTOR FACILITY RADIATION SOURCES

All potential sources of radiation produced by normal reactor operation are identified and summarised in Table 12.2/1.

12.2.1 Contained Sources

12.2.1.1 Reactor Fuel

The major source of contained radionuclides in the Reactor Facility is the reactor core, where fission products accumulate in the Fuel Assemblies (FAs) during the reactor operation. These radionuclides constitute the most important contained source both when the fuel is in the core or in the spent fuel storage racks (in the Service Pool). Using Defence in Depth methodology, several safety barriers have been utilised to prevent fission product migration to the environment, namely, the fuel matrix and cladding, reactor pool water inventory, and reactor containment.

The fuel cladding is designed to withstand normal operational and storage conditions with a safety margin. Nevertheless in the case of a failure (known as FA failure) the failed FA are contained and stored underwater minimising the possible spread of contamination. A means of containing a failed FA is available. Both the Failed Fuel Element Monitor (FFEM) and the Active Liquid Monitor (ALMO) are available to detect failed fuel.

Table 12.2/2 shows the relevant radioisotopes contained in the equilibrium core one minute after a reactor shutdown. Calculated data using ORIGEN 2 (ORIGEN2 “Oak Ridge Isotope Generation and Depletion Code”, ORNL, 1980) with Reactor Facility characteristics are compared with the inventory presented in the EIS calculated using AUS neutronics system with a ENDFB6 cross section library (Robinson, 1993, 1998). Both are similar and in agreement with the typical inventory postulated by the IAEA (IAEA-TECDOC-400, 1986). Since the reactor core calculated inventory is obtained from the specific fuel assembly design, it is taken as the applicable inventory for calculations associated with normal Reactor Facility operations.

12.2.1.2 Targets for Radioisotope Production

Other relevant contained sources are the activated samples that are irradiated in special canisters designed to withstand core conditions (neutron flux, gamma heating, etc.) as well as transportation requirements (i.e. pneumatic transport system accelerations). For certain samples, double canisters are used to increase performance. The Reactor Facility has adopted the can design used at HIFAR which has not registered any failures.

12.2.1.3 Process Systems Sources

Different components contain activated liquids and can be considered as radiation sources.

Table 12.2/3 shows these components and their location.

12.2.1.4 Other Sources

Activated components removed from the core (such as spent absorbing plates) are stored in the service pool and managed as solid waste.

Other minor contained sources are the calibration sources employed in the plant and the neutron source used mainly in the first reactor start-up. They have leak-proof characteristics designed and tested by their suppliers.

12.2.2 Airborne and Liquid Sources for Environmental Considerations

There are several sources able to be dispersed in pool water and later in air, such as:

a) activation of coolant

b) corrosion products

c) fission products from fuel cladding surface contamination

The concentration of these sources in pool water has been calculated by modelling the production and transport and has been compared with measured values in similar installations. The two data sets were in good agreement and where differences occurred, conservative (i.e. greatest) values have been chosen. The estimated concentration in water is shown in Table 12.2/4 and the postulated values for releases through the stack are shown in Table 12.2/5. Table 12.2/4 contains the nuclides relevant for the release by evaporation from the reactor pool surface and does not contain other nuclides such as Na24 or Ni16. Table 12.2/5 is obtained assuming that for noble gases a surface exchange coefficient of 0.0057 mms^{-1} takes place in the pool while for aerosols a transportation mechanism due to water evaporation occurs.

The main radiological issue for nitrogen-16 is related to direct radiation since it is a very short lived isotope and not released in airborne emissions. The calculations performed were intended to obtain the concentration in several points of the Primary Cooling System and Reactor and Service Pools Cooling System circuits (see Section 12.3.2.5.) to size the decay tanks as well as determine the room shielding. Additional calculations were performed to determine the concentration in the heavy water system components.

Also of relevance is argon-41 produced by neutron activation of air. This isotope is minimised by preventing the presence of air in areas with relevant neutron fluxes, such as the pneumatic systems that are driven by nitrogen gas. Table 12.2/5 shows the maximum credible release assumed for this nuclide. This value came, as in previous cases, from detailed engineering modelling of production and transport processes and subsequent comparison with measurements in similar facilities assuming conservative values.

Tritium produced in the Reflector Vessel by deuterium activation could be released to the heavy water room through water losses during maintenance activities. The room characteristics and the system design and construction features assure that airborne tritium releases are negligible. A target of 3.7 10^{10} Bq per year has been set as an airborne emission target as shown in Table 12.2/5.

Other liquid or gaseous sources that may disperse radioactive products into the environment are controlled by means of drains and ventilation systems.

A comparison between airborne discharges from HIFAR and those predicted from the replacement Reactor Facility is given in Table 12.2/6. The table shows that for the major nuclides of concern, Argon-41 and tritium, the reductions are approximately 97.4% and 99% respectively. The table also shows that the total reduction for all nuclides is approximately 97.5%

End of Section

Table 12.2/1 Potential Sources of Radiation Produced by Normal Reactor Operation

Component/System	Contributors	Isotopes	Activity	Location and comments
Core	FA Coolant Control rods Structural components	Radiation fields from fission process in fuel elements. Activated products Fission products from irradiated fuel elements	During reactor shutdown, fission products in FA are the main component During operation, neutrons and gamma from fission process (see Section 12.3.2.2.1.)	Reactor pool. Structural components (e.g. fuel end pieces) are minimised in the fuel design to reduce solid waste production.
Reflector Vessel	Samples being irradiated Heavy water Structural components	Isotopes being produced. Tritium Nitrogen-16	Variable. It is a function of the isotope production activities. Tritium concentration build-up in time (rate in the order of 96 GBq per kg per year)	Reactor pool. No Air is used to cool rigs to limit Ar-41 production. Structural components have been chosen taking into account neutronic, hydraulic and decommissioning considerations.
Spent fuel storage racks in reactor pool	Spent fuel assemblies waiting to be moved to service pool	Fission products from irradiated fuel assemblies	FA are the main component (see Table 12.2/2.	Reactor pool. FA remain here until they are transported to the service pool
Transfer channel	FA and irradiated samples being transferred.	Samples produced Irradiated fuel assemblies	Different arrangements of FA (from one to three) with at least a decay period of 33 days. Irradiation rigs See section 12.3.2.3. item d.	Channel between reactor pool and service pool. Sources are moved with the Operation Bridge through the channel. That means that all sources are “in-transit”.

Component/System	Contributors	Isotopes	Activity	Location and comments
Spent fuel assembly storage (service pool)	Spent fuel assemblies	Fission products	FA activity decrease with time see Table 12.2/2.	Service pool. Capacity for 10 years.
Pneumatic transport system	Irradiated samples		Equivalent to 100 GBq of Na-24, see Section 12.3.2.6. item 1(PHC) (Al-28 from can activation is also considered in some calculations)	It connects the reactor pool and the cells. No Air is used to move samples to limit Ar41 production.
Hot cells complex (APHCC)	Irradiated samples	Isotopes being produced Activated Al canisters	Molybdenum production. Te and Ir sources. See Section 12.3.2.6. item 1	Located in Pooltop Level A large proportion of the annual radionuclide production remains in these cells for a certain time until delivered to IPTS or LHC.
Loading cell (LHC)	Irradiated samples	Isotopes being produced	See Section 12.3.2.6. item 1	.
Inter hot cells elevator (ICE)	Irradiated samples	Isotopes being produced	See Section 12.3.2.6. item 1	Connect hot cells complex with loading cell Sources are “in-transit” but shields have been calculated considering possible stuck sources.
PCS (pipes, pumps and decay tank)	Water activation. Corrosion products Traces of fission products	N-16 Na-24 Mg-27 Ar-41 Cr-51 Mn-56	$<\approx 10^6$ Bqcm^{-3} $<\approx 10^4$ Bqcm^{-3} $\approx 10^3$ Bqcm^{-3} $\approx 2 \times 10^3$ Bqcm^{-3} $\approx 10^2$ Bqcm^{-3} $\approx 10^1$ Bqcm^{-3}	These are maximum concentration at core position neglecting purification system. Primary pump rooms Decay tank room. Some fission products may come from uranium deposited externally in fuel claddings during manufacture.

Component/System	Contributors	Isotopes	Activity	Location and comments
D_2O treatment system	Water activation. Corrosion products	N-16 H-3	3×10^6 $Bqcm^{-3}$ 96 $GBqkg^{-1}y^{-1}$	Concentration of H-3 increases with time
RSPCS	Same as in the PCS	Same as in the Core Cooling System	N-16 $\approx 2 \times 10^5$ Bq/cm^3	Nuclides from PCS came through the interconnection flow.
Water treatment system (resins)	Miscellaneous	Various	Variable	see Section 12.4.4.10.
Liquid waste storage tanks and collection network	Miscellaneous	Various	Variable	see Section 12.4.7.
Hot water layer purification units	Miscellaneous	Various	Equiv. 10^9 Bq of Cs^{137}	see Section 12.4.6.1.
Inter-building pneumatic transfer system (IPTS)	Samples been transferred	Various	Equiv. 40 GBq of Co^{60}	Special trench between buildings. An interlock prevents transfers of targets emitting radiation higher than 40 GBq of Co-60

Table 12.2/2 Relevant Equilibrium Core (after 33 full power days) Inventory One Minute after Shutdown

Isotope	Half-life	EIS reported Inventory [TBq]	Reactor Facility Calculated Inventory [TBq]
Xe-131m	11.9 day	1.98×10^2	1.90×10^2
Xe-133m	52.6 hr	1.21×10^3	1.10×10^3
Xe-133	125.8 hr	4.09×10^4	3.62×10^4
Xe-135m	15.3 min	7.21×10^3	6.46×10^3
Xe-135	548.4 min	4.07×10^3	2.83×10^3
Xe-138	14.1 min	3.55×10^4	3.35×10^4
Kr-83m	109.8 min	3.15×10^3	2.82×10^3
Kr-85m	268.8 min	7.46×10^3	6.54×10^3
Kr-85	10.7 yr	4.50×10^1	4.77×10^1
Kr-87	76.3 min	1.48×10^4	1.31×10^4
Kr-88	170.4 min	2.01×10^4	1.85×10^4
I-130	12.4 hr	1.40×10^2	3.56×10^2
I-131	193.0 hr	1.82×10^4	1.69×10^4
I-132	137.0 min	2.68×10^4	2.45×10^4
I-133	20.8 hr	4.07×10^4	3.70×10^4
I-134	52.6 min	4.71×10^4	4.16×10^4
I-135	394.2 min	3.79×10^4	3.45×10^4
Te-125m	58.0 day	2.48×10^0	4.40×10^0
Te-127m	109.0 day	9.01×10^1	1.01×10^2
Te-127	561.0 min	8.78×10^2	1.09×10^3
Te-129m	33.6 day	8.26×10^2	6.29×10^2
Te-129	69.6 min	4.46×10^3	4.22×10^3
Te-131m	30.0 hr	2.69×10^3	2.16×10^3
Te-131	25.0 min	1.57×10^4	1.47×10^4
Te-132	78.2 hr	2.66×10^4	2.43×10^4
Te-133m	55.4 min	1.94×10^4	1.55×10^4
Te-133	12.5 min	2.25×10^4	2.15×10^4
Te-134	41.8 min	4.06×10^4	3.55×10^4
Cs-134m	174.6 min	1.16×10^4	6.73×10^2
Cs-134	753.1 day	2.11×10^2	4.63×10^2
Cs-136	13.2 day	1.44×10^2	2.50×10^2
Cs-137	30.0 yr	3.42×10^2	4.01×10^2

Isotope	Half-life	EIS reported Inventory [TBq]	Reactor Facility Calculated Inventory [TBq]
Cs-138	32.2 min	3.98×10^{4}	3.63×10^{4}
Rb-86	18.6 day	9.49×10^{0}	2.53×10^{1}
Rb-88	17.8 min	2.04×10^{4}	1.87×10^{4}
Rb-89	15.2 min	2.71×10^{4}	2.43×10^{4}
Ru-103	39.3 day	1.83×10^{4}	1.89×10^{4}
Ru-105	4.44 hr	7.92×10^{3}	8.48×10^{3}
Ru-106	368 day	7.99×10^{2}	1.08×10^{3}

Table 12.2/3 Components Considered as Radiation Sources

Component	Location	Shape	Comments
Waste collection tank “C”		Cylinder	Variable water content with nuclide concentrations lower than allowed values for discharges to the LHSTC C line. No shields are needed.
Waste collection tank “B”		Cylinder	Variable water content with nuclide concentrations lower than allowed values for discharges to the LHSTC B line. No shields are needed.
Heavy Water Storage Tank		Cylinder	From zero to whole heavy water system inventory. Tritium concentration variable with time Short half-life N-16 content when SSS is triggered. Is located below room level to provide shielding.
PCS heat exchangers 1, 2 and 3	Basement	Box	Filled with PCS water. The shielding provided is explained in Section 12.3.2.4.
PCS pipes	Basement (various rooms)	Pipes	Filled with PCS water.
HWLS heaters 1 and 2	Basement	Cylinder	Filled with water from HWLS. No shields are needed
Reactor Water Purification System filters 1 and 2	Basement (behind shielded walls)	Cylinder	Filled with RSPCS water. Retain gross particles from circuit. They are located together with resins beds.
Reactor Water Purification System resin beds 1 and 2	Basement (behind shielded walls)	Cylinder	Filled with RSPCS water. Retain activation, corrosion and fission products from circuit. The shielding provided is explained in Section 12.3.2.8
Reactor Coolant spent resin storage tanks 1 and 2	Basement (behind shielded walls)	Cylinder	Temporary storage of spent resins. Similar to resin bed shields is applied.
RSPCS heat exchanger	Basement	Box	Filled with RSPCS water
PCS decay tank	Basement	Cylinder	Filled with PCS water. Short half-life N-16 present. . The shielding provided is explained in Section 12.3.2.5.

Component	Location	Shape	Comments
RSPCS decay tank	Basement	Cylinder	Filled with RSPCS water. Short half-life N-16 present. The shielding provided is explained in Section 12.3.2.5.
Heavy water purification system resin beds 1 and 2	Basement (behind shielded walls)	Cylinder	Filled with heavy water. Activation and corrosion products present. Shielding similar to other resin bed is applied.
Heavy water system heat exchanger	Basement	Box	Filled with heavy water. The shielding provided is explained in Section 12.3.2.9.
Reactor ventilation system absolute filters (four)	Pooltop	Cube	Aerosols from Reactor Containment.
Reactor ventilation system activate charcoal filters (four)	Pooltop	Cube	Negligible activity in normal operation.
Hot Cell Ventilation system absolute filters (four)	Pooltop	Box	Aerosols from APHCC.
Hot Cell Ventilation system activate charcoal filters (two)	Pooltop	Box	Negligible activity in normal conditions.

Table 12.2/4 Nuclide Concentration Estimated in Reactor Pool Water at 20 MW

Nuclide	Concentration [Bqm^{-3}]
Ar-41	1.5×10^{8}
Cr-51	1.2×10^{6}
Xe-135	2.0×10^{6}
Xe-133	1.8×10^{7}
Kr-88	2.0×10^{6}
Kr-87	7.4×10^{5}
Kr-85m	7.4×10^{5}
I-133	1.0×10^{6}
Ba-140	4.4×10^{4}
La-140	7.0×10^{4}
Cs-137	6.0×10^{3}
I-131	3.5×10^{4}

Table 12.2/5 Postulated Annual Releases of Relevant Nuclides through the Reactor Facility Stack in Normal Operation Conditions

Nuclide	Annual release [Bq]
Ar-41	4.23×10^{12}
Ba-140	1.36×10^{5}
Cr-51	3.63×10^{6}
Cs-137	1.82×10^{4}
I-131	1.06×10^{7}
I-133	3.48×10^{8}
Kr-85m	2.12×10^{10}
Kr-87	2.12×10^{10}
Kr-88	5.65×10^{10}
La-140	2.12×10^{5}
Xe-133	$4.94 \text{ X } 10^{11}$
Xe-135	5.65×10^{10}
H-3	3.70×10^{10} *
Sr-90	1.54×10^{6}

The ventilation systems handling tritium in the facility are designed to constrain tritium levels to below 155 GBqy^{-1}. The value quoted here is an initial estimate of likely releases based upon projected operation and maintenance schedules of the heavy water systems. The actual value will be determined on the basis of operational experience (see Section 12.4.6.4)

Table 12.2/6 Annual Stack Discharge Estimates – Comparison of Significant Nuclides with HIFAR and EIS

	HIFAR	EIS	Reactor Facility
Argon 41 TBq	164	82	4.2
Tritium TBq	4.32	4.32	0.037
Iodine 131 MBq	14	14	10
Strontium 90 MBq	2.03	2.03	1.5

End of Tables

12.3 REACTOR FACILITY DESIGN FOR RADIOLOGICAL SAFETY

12.3.1 Reactor Facility Design Features

In agreement with the basic principle of radiological protection, the Reactor Facility has been designed according to the following objectives:

a) Exposure of individuals to radiation must not exceed the applicable established limits.

b) Exposure of individuals must be reduced to the lowest value that can be reasonably reached, taking into account economical and social factors (ALARA criteria).

c) The radiological protection objectives are achieved through the application of a radiation protection program designed to complement the radiation safety aspects of the Reactor Facility design.

The following sections describe how the design minimises the exposure of personnel, minimises the undesirable production and release of radioactive material and reduces the time spent for maintenance and operational activities with potential radiation exposure hazards, to as low as reasonably achievable.

12.3.1.1 Zoning and Access Control

The layout of the facility provides for the isolation of radioactive material from the facility personnel and from the general public. The layout includes zoning that classifies the facility areas according to their potential for radioactive contamination and/or radiation exposure.

The zoning objectives are:

a) To minimise the transport of contamination to other areas.

b) To reduce to the minimum the annual dose received by personnel during normal operation of the plant.

Non restricted areas have been classified according to the three levels (red, blue, white) established by ANSTO Safety Directive 5.4 for external radiation and for contamination, in agreement with the Table 12.3/1.

A Derived Surface Contamination Limit is specified for every radionuclide in ANSTO Safety Directive 5.4 Appendix 2 (page 105/13). It represents values of surface contamination that would result in 20 mSv/year if exposure occurred for an average of 2,000 hours/year.

Consistent with ANSTO SD 5.4, modifying factors such as occupancy factors can be applied when assessing potential individual exposure levels for area classification. Estimated doses can be divided by values ranging from 1 (full occupancy) to 0.3 (lesser occupancy areas), down to 0.1 (low occupancy), to give representative exposure levels for individuals.

Other area types are defined to protect personnel from radiation hazards and plant critical areas. Two concepts have been introduced in relation to area classification:

Forbidden areas: are locations where potential dose rates or contamination levels may fluctuate to values higher than those expected in radiation/ contamination red areas. Personnel Access to Forbidden areas is prevented during specific reactor states or plant operating conditions through the use of physical barriers and administrative controls.

Restricted areas: are locations where personnel occupancy is controlled and restricted for certain activities. In these areas radiological conditions may change as a result of activities outside the immediate area. Access to a restricted area is permitted when specific administrative controls and procedures for the classified area are carried out. Special area and individual monitoring requirements (e.g. direct reading dosimeters) or specific plant operating conditions (e.g. reactor in shutdown state or pneumatic transfer system disabled) may be required. Planning and assessment may be to a greater extent than for blue and red area classifications.

During commissioning, the Radiation Protection Advisor in conjunction with the Reactor Manager or relevant responsible officers are responsible for setting the Area Classification. This may be modified as appropriate following operational experience.

Certain operations in classified areas may raise or lower the hazard and potential radiation exposure for workers. A temporary upgrade or downgrade in classification may be considered. Warning notices and protective measures must be accordingly modified to reflect this upgrade or downgrade and proper notifications must be made. A typical example is the Reactor Beam Hall during operations for the replacement of neutron beams in-pile assemblies.

The use of the Reactor Facility by researchers and operations personnel is also taken into consideration. Routine procedures are available to confirm that the assigned classification of each area remains valid to ensure that access and working requirements are consistent with safety requirements.

Areas have been designed with the criteria of preventing the spread of contamination. Access to areas of higher radiological hazards is to be from areas of lower classification.

A double circulation concept has been applied: one set of stairs (and corresponding lift) connects white zones with each other, and the other set of stairs (and corresponding lift) connects blue zones with each other. Emergency exits allow communication between both in case that evacuation is required.

Showers and basins are distributed in the building and an area for preliminary decontamination of tools and objects was established with appropriate basins and instruments. Further decontamination process could be continued, if necessary, in other LHSTC site facilities.

Particular requirements for different areas of the facility are clearly described in lists of classifications held within the facility.

General characteristics and main applicable concepts are presented below.

12.3.1.1.1 Basis for the Assignment of Areas

Active vents and drains are designed so that they do not affect areas other than those they are servicing.

The Reactor Beam Hall is monitored after experiments are assembled and disassembled so it can be assessed for correct area classification.

12.3.1.1.2 Requirements for Accessing Areas

All the rules and procedures relating to access/egress and work in different areas are referred to in Reactor Facility document lists.

General conditions are:

a) White Radiation Area: Individual dose monitoring is not required in white radiation areas.

b) Blue Radiation Area: Individual dose and area monitoring is required.

c) Red Radiation Area: Individual dose and area monitoring is required.

d) White Contamination Area: Individual dose monitoring is not required in white contamination areas. Health physics surveys may be programmed to confirm maintenance of White Area status.

e) Blue Contamination Area: Individual dose monitoring and area monitoring are required in blue contamination areas. Routine contamination and radiation monitoring as per area monitoring program.

f) Red Contamination Area: Individual dose monitoring and regular contamination and radiation monitoring are required in red contamination areas.

g) Restricted Area: Access to a restricted area is permitted when specific administrative controls and procedures for the classified area are carried out. Special area and individual monitoring requirements (e.g. direct reading dosimeters) or specific plant operating conditions (e.g. reactor in shutdown state or pneumatic transfer system disabled) may be required. Planning and assessment may be to a greater extent than for blue and red area classifications.

h) Forbidden Area: Personnel Access to Forbidden areas is prevented during specific reactor states or plant operating conditions through the use of physical barriers and administrative controls.

12.3.1.1.3 Circulation Inside the Reactor Building

12.3.1.1.3.1 Access to the Reactor Building

The Reactor Building has three accesses:

a) Through the main entrance building and office area on the eastern approaches to the facility.

b) Through the neutron guide hall or west ramp only to the experimental physics area of the reactor beam hall.

c) Through the southern vehicle air lock and dock to the isotope loading areas. This access is just for the delivering of isotopes and the ingress of components (spares), personnel access is not allowed through it.

Each of these access points has physical security entrance and exit controls that register which individuals are present inside the Reactor Building.

12.3.1.1.3.2 Use of Dressing Rooms

Change areas are used for personnel of the facility, to change street clothes for working clothes. Clean blue/red area clothes can be picked up from these rooms. They do not contain contaminated clothes. A contamination checkpoint gate is provided for this purpose.

12.3.1.2 Component Layout

Access requirements related to operation, calibration, inspection, maintenance, repairs and substitution of equipment have been evaluated to ensure that operation, inspection and maintenance activities are properly considered. The layout of the plant has been optimised in order to minimise staff exposures to radiation and the potential for contamination spread.

The following features have been considered:

a) Separation of zones

b) Radiation shielding

c) Adequate provisions to allow appropriate ventilation

d) Devices for equipment manipulation

e) Techniques for remote control

f) Provisions for dressing rooms

g) Access control

h) Facilities for decontamination

The length of the path to be followed by staff through radiation and contamination controlled areas is shortened to minimise transit times, and to limit potential radiological exposures by avoiding excessive grouping of sources.

To minimise personal radiation doses and prevent propagation of contamination, the layout of areas is such that the personnel need not go through high radiation zones to arrive at others of lower radiation risk, nor go through zones of high contamination potential to arrive at others with less contamination potential.

The time required for maintenance, testing and repairs in radiation and contamination zones is based on the ALARA principle.

The following rules have been followed as regards spatial distribution of the plant:

a) Provision of clear transit ways with appropriate dimensions for an easy access to different parts of the reactor building.

b) Allocation of enough space around equipment so repair work and inspections can be performed easily. This rule has been employed in:

   (i) Primary Cooling System pump rooms, which are big enough to host components with plenty of space surrounding them.

   (ii) Cell operation room has adequate space to deal with large tools such as Master-slave manipulators.

c) Provision of clear transit ways of adequate dimensions to facilitate the transportation of elements to a decontamination and repair workshop or to evacuate them. The routes are the shortest possible so that the potential for contamination spread is minimised.

d) Mounting of components at a convenient height to facilitate work on components of frequent usage. Several examples can be mentioned such as the location of valves and local indicators and the waste storage tanks that have been located preserving space under them to facilitate cleaning and inspection activities.

e) Provision of stairs, access platforms and permanent bridge cranes close to areas where they are needed for element maintenance or transportation, including the Operations Bridge, which provides access to the reactor and service pools and the provision of adequate lifting devices for components heavier than 50 kg.

f) Provision of appropriate installations so that the dismantling of shields and isolation devices is quick and easy when such operations are necessary for the purpose of carrying out maintenance or routine inspections.

g) Provision of contamination control points to preclude unnoticed spread of contamination. To this end, several points inside the designated contamination blue areas are provided with portable contamination detectors to allow an effective control. The background radiation levels at those points have been estimated and are considered low enough to allow these measurements.

In the design and distribution of systems, several characteristics were incorporated, aimed at reducing radiological exposure, in accordance with the ALARA principles:

a) In areas of high radiation levels the working space surrounding components (pumps, valves, etc.) that require periodical maintenance, are shielded against radiation emitted by adjacent components of others systems.

b) Indicator panels, auxiliary units, moving units, control equipment and other non-radioactive components that do not need to be set up with radioactive components are installed outside the high radiation sub-zones.

c) Adequate elements of shielding are located among duplicated radioactive systems, allowing maintenance or repair operations to be carried out in one of them while the other is operating:

d) Sampling devices for radioactive liquids are located away from radiation fields. Remotely operated tongs are used to reduce personnel exposure to the minimum.

e) When withdrawing shielding in zones of high radiation becomes necessary, auxiliary lifting mechanisms are deployed to allow a quick and simple withdrawal.

f) Sedimentation of radioactive mud is minimised when possible. Components where it can accumulate are provided with means to remove sediments or to simplify maintenance operations.

g) In piping design, adequate measures to reduce radiation hazards have been taken, providing them with appropriate shielding or locating them as far as possible from occupied areas in corridors. Active pipes passing through offices or non-controlled areas are avoided.

h) Adequate facilities are provided to allow decontamination of components and persons. Several basins and showers are distributed over the plant and an area has been designed for decontamination of objects.

i) Adequate circulation of air is provided to prevent build up of active isotopes in air or movements of contaminated air to areas with lower concentrations.

12.3.1.3 Other Health Physics Related Design Features

Other Reactor Facility design features related with Health Physics are:

a) Provisions for draining, flushing and decontamination of components where active liquids circulate.

b) Coupling and disconnection of components are designed to be as fast as possible compatible with safety considerations in order to reduce exposure times.

c) Components are separated in groups according to their radiation fields.

d) Delay stages are introduced to allow isotope decay when needed. Both decay tanks respond to this characteristic allowing nitrogen-16 activity to decay several orders of magnitude. The same concept is applied to the downward Primary Cooling System flow through reactor chimney produced by the interconnection-line that prevent the buoyancy of active water from reaching upper layers of the pool. Additionally, the hot water layer provides a delay and cleaning stage in the transport of isotopes to reactor hall.

e) Reduction of the potential heavy water leakage to the environment.

f) Adequate lighting levels are provided to allow quick and efficient surveillance and maintenance operations reducing in this way the radiation exposure.

12.3.1.4 Material Characteristics from the Radiological Viewpoint

The choice of structural materials for the reactor components is discussed in Chapter 5.

From the radiological point of view material selection criteria take into consideration corrosion behaviour, activation properties and surface contamination.

Corrosion characteristics are included in the materials discussion of Chapter 5.

Regarding activation properties the design has minimised those materials that, when activated, produce radionuclides of long half-life, such as alloys with high cobalt content.

When surface contamination is possible a layer of protective coating (epoxy-type) is provided).

Shielding materials have been selected to ensure they remain effective over the planned lifetime of the facility and taking into account any radiation-induced damage. This issue is particularly important for beam shutters in relation to accumulated radiation dose and gamma heating phenomena.

See Chapter 5, Section 5.9 for further analysis of reactor materials.

12.3.2 Shielding

12.3.2.1 General Design Criteria

In order to design shielding for a radiation source the ambient equivalent dose as a function of design parameter (e.g. thickness) must be calculated and the contribution from other sources must be taken into consideration, ensuring that design dose values are not exceeded.

Where an improved protection can be achieved shielding thickness is optimised.

Transitory radioactive sources and activity accumulation due to radionuclides of long half-lives during the life span of the plant are taken into account in the calculations.

After evaluation of source strength, the process continues with calculation of the main block, followed afterwards by calculation of penetrations. Penetrations may create pathways along which radiation channelling can be produced (mainly of neutrons and gamma rays), which might cause unacceptable dose rates outside the shielding. Minimisation of penetration effects is an objective of the design, by means of the following basic method:

a) Minimise the solid angle that contains substances of low density through which radiation can be transmitted.

b) Interpose as much material as possible in the way of radiation.

c) Scatter radiation towards the surrounding material, increasing in this way the probability of absorption.

This is achieved by means of:

a) Minimising the area and the number of direct paths that contain substances of low density.

b) Designing shielding of bigger size than penetrations, in order to cover boundary zones.

c) Using, where feasible, broken lines or curves, so that there is shielding in every straight direction.

d) Filling the gaps with plastic or any other material of compensatory shielding.

The access to a zone of intense radiation constitutes a particular case of penetration of shielding. In general their dimensions are bigger than the thickness of shielding.

The configuration to be adopted for the shielding of an access depends on the intensity of the radiation field and the dose rate requirements outside of the shielded source.

Material selection takes into consideration the characteristics of the source to be shielded and the structural requirements. Properties of commonly used materials are included in Section 12.3.2.11.

The main shielding structures and calculation techniques are presented below.

12.3.2.2 Reactor Pool Shielding

Two different situations are taken in consideration:

a) reactor at Power Operation state

b) reactor at Shutdown state

12.3.2.2.1 Power Operation State

Main radioactive sources considered for shielding design are:

a) Reactor core: The following contributions are considered: fission neutrons, fission gamma rays, gamma rays from (n, gamma) reactions and gamma rays from decay of fission products, actinides and activation products.

b) Gamma source from nitrogen-16: This source is due to neutron intrinsic water activation, by the O-16(n,p)N-16 reaction (also Nitrogen-17 and Oxygen-19).

Factors used to calculate the neutron source from the core are:

Power MW	MW/fission	Neutrons/fission	Neutrons/sec	Neutrons/sec/cm^3
20	3.204×10^{-17}	2.418	1.5094×10^{18}	2.3818×10^{13}

These neutrons have an energy distribution given by the Watt equation

$\chi(E) = 0.4395 \exp(-1.0122\ E) \sinh(\sqrt{(2.249\ E)})$ [E] = MeV

Factors used to calculate the gamma source from the core are:

Power MW	MW/fission	Gamma/fission	Gamma/sec	Gamma/sec/cm3
20	3.204×10^{-17}	19	1.1860×10^{19}	1.8716×10^{14}

The gamma spectrum for prompt gamma rays is:

$N_p(E) = 26.8 \exp(-2.3E)$ for E<1 MeV

$N_p(E) = 8.0 \exp(-1.1E)$ for E>1 MeV

The gamma spectrum for fission product gamma rays is:

$N_{fp}(E) = 7.4 \exp(-1.1E)$

The gamma sources due to nitrogen-16, main element of activation of light water (with a concentration of 7 x10^5 Bqcm^{-3}) are the following:

Parent Isotope & Relative Abundance	Nuclear reaction & neutron energy	Half-life & activation cross section	type, energy & yield of radiation emitted
O-16 (100%)	O-16(n,p)N-16 E ≥ 11.6 Mev	7.35 sec 46 mbarn	γ, 6.13 Mev (76%) γ, 7.10 Mev (6%)

12.3.2.2.1.1 Radial Shielding

Heavy concrete is chosen as structural shielding material for calculation purposes. Shielding calculations are performed with the , DORT[2] and MCNP[3] codes, using appropriate models to represent the geometry of the reactor from the core centre up to the main reactor pool side-wall (reactor face).

12.3.2.2.1.2 Upper Axial Shielding

The water column above the reactor core provides the axial shielding. For this reason it is necessary to determine the minimum required height for the water column, in agreement with the dose rate constraint. To calculate this value, the DORT code is used with two-dimensional geometry.

Assuming a maximum contact dose rate with water on the pool vertical axis of 1 µSvh^{-1}, a minimum value for the water height above the core is obtained. This value takes into account the shielding necessary to deal with radiation resulting from both, full power reactor operation and nitrogen-16 due to core coolant along the chimney up to Primary Cooling System loop exit.

To provide shielding against gamma rays from the activation of water and impurities at the reactor pool surface, a hot water layer system is provided. This layer has a purification subsystem to reduce the radionuclides concentration.

12.3.2.2.1.3 Lower Axial Shielding

Entrance to the Control Rod Drive room by plant personnel with reactor at Power Operation state is restricted.

The CRD roof shield is constructed of heavy concrete and the dose rate distribution in ceiling has been calculated for neutron and gamma radiation.

An additional gamma source is originated in a pipeline of Reflector Cooling and Purification System located in this room. A dedicated shielding was provided for this pipeline in order to reduce the dose rate in this room.

[2]/ DORT: A Two and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System. ORNL-RSIC-CCC 650 (1996)

[3]/ MCNP - A General Monte Carlo Code for Neutron and Photon Transport, Briesmeister J.F., LA7396-M, Rev.2, Los Alamos National Laboratory (1986)

12.3.2.2.1.4 Reactor Shutdown State

Calculations for the source term of the core were made considering average burn-up and constant power (axial and temporal) for all Fuel Assemblies.

The photon source terms for the core as a function of time after shutdown (activation products, actinides and daughters, fission products) are:

Photons/sec						
Energy group	Shutdown	1.0 MIN	15.0 MIN	1.0 HR	4.0 HR	1.0 DY
5.75×10^{-1}	9.13×10^{17}	7.44 xX 10^{17}	5.32×10^{17}	3.91×10^{17}	2.86×10^{17}	1.79×10^{17}
8.50×10^{-1}	4.70×10^{17}	4.18×10^{17}	3.13×10^{17}	2.31×10^{17}	1.41×10^{17}	9.22×10^{16}
1.25×10^{0}	2.90×10^{17}	2.52×10^{17}	1.73×10^{17}	1.07×10^{17}	5.02×10^{16}	1.12×10^{16}
1.75×10^{0}	1.30×10^{17}	1.12×10^{17}	7.32×10^{16}	5.66×10^{16}	4.11×10^{16}	3.03×10^{16}
2.25×10^{0}	5.75×10^{16}	5.26×10^{16}	3.81×10^{16}	2.27×10^{16}	7.90×10^{15}	8.96×10^{14}
2.75×10^{0}	2.50×10^{16}	2.14×10^{16}	1.48×10^{16}	8.95×10^{15}	2.97×10^{15}	1.19×10^{15}
3.50×10^{0}	1.47×10^{16}	1.12×10^{16}	3.94×10^{15}	2.16×10^{15}	5.03×10^{14}	9.59×10^{12}
5.00×10^{0}	7.78×10^{15}	4.65×10^{15}	1.81×10^{14}	3.07×10^{13}	1.18×10^{13}	8.86×10^{10}
7.00×10^{0}	6.32×10^{13}	1.46×10^{10}	4.34×10^{04}	1.21×10^{03}	1.22×10^{03}	1.23×10^{03}

12.3.2.2.1.5 Radial Shielding

Since the gamma and neutron fields decrease several orders of magnitude, the calculated shield is enough for shutdown conditions.

12.3.2.2.1.6 Upper Axial Shielding

Once the reactor is shutdown a minimum water column height, above the core, should be maintained to provide shielding. A complete 3-D model is adopted to represent the core, the reflector, and the water column above them. The MERCURE IV[/4/] code, has been used for this calculation.

The water column needed to allow the removal of a Fuel Assembly from the core to the reactor pool storage rack was measured from the top of the Fuel Assembly. The Fuel Assembly is assumed to have decayed for 15 minutes before the removal. The dose rate assumed for this operation is 10 µSvh^{-1}.

12.3.2.2.1.7 Lower Axial Shielding

When the reactor is shutdown, personnel access to the control rod drive room is allowed. The shielding designed to meet the dose rate requirement during Reactor Power Operation state satisfies limits during shutdown. Secondary sources such as activation of materials can be neglected due to very low neutron flux during reactor operation and the correct material selection.

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[4]/ MERCURE-IV: Un programme de Monte Carlo a trois dimensions pour l'integration de noyaux d'attenuation en ligne droite. C. Dupont et J. Nimal - Raport SERMA/T/N° 436 (1980).

12.3.2.3 Service Pool Shielding

The radioactive sources considered are the irradiated Fuel Assemblies stored in the Service Pool.

The Fuel Assemblies are placed at the service pool, following the refuelling strategy, at least 35 days after they are removed from the core. During the first stage, the fuel element remain at the temporary storage grid placed in the reactor pool allowing decay.

Therefore, the most unfavourable Fuel Assembly array from the shielding point of view, is when the full storage capacity (10 years of continuous operation plus a core loading) is reached with elements decayed 1, 2, ...,n reactor operation cycles.

Fuel Assemblies are stored in square containers located on the same level.

Shielding calculations cover four main topics:

a) Lateral or sidewall shielding.

b) Upper axial shielding.

c) Lower axial shielding.

d) Transfer channel.

Calculations using MERCURE code with the most unfavourable array of Fuel Assemblies described above show the following results.

a) Lateral shielding: Heavy concrete thickness was derived from the Table 12.3/3 and from INVAP experience in previous design.

b) Upper axial shielding: This is provided by the water column over the irradiated Fuel Assemblies.

c) Lower axial shielding: It is assumed that at Reactor Shutdown state, access to the decay tank room is necessary. To obtain in this room dose levels which do not restrict its access during shutdown, it is necessary to have heavy concrete at the reactor pool floor, obtained from Table 12.3/5 and INVAP experience in previous design.

d) Transfer channel: the transfer channel connects the reactor pool with the service pool. During the transfer operation, the water column above the transferred components provides the required biological shield. Calculation of the minimum water column required was performed assuming three different situations:

   (i) One irradiated Fuel Assembly with 33 days of decay

   (ii) Three irradiated Fuel Assemblies (one of every refuelling chain) with 33 days of decay

   (iii) Rigs irradiated (Mo target) with 1 hour decay

Table 12.3/6, Table 12.3/7 and Table 12.3/8 shows the dose rate variation with water height. Assuming that components of different size are moved through this channel, a depth to accommodate those components has been adopted.

12.3.2.4 Primary Cooling System Pump Room Shielding

The reactor has three Primary Cooling System pump rooms. They are physically isolated and, in order to allow personnel access during reactor operation the system is provided with a decay tank. At this condition, the dose rate at a point inside this room has the main contribution of sodium-24 concentration of 820 $Bq.cm^{-3}$ in the Primary Cooling System coolant.

Nitrogen-16 contribution is neglected due to the decay produced at the decay tank.

The walls that divide the pump rooms from the corridor and the piping connection room provide enough shielding to these areas for the source considered, allowing work with minimum radiation hazard..

12.3.2.5 Decay Tanks

Shielding for two decay tanks (Primary Cooling System and Reactor and Service Pool Cooling System tanks) has been calculated.

The source is the water volume with a concentration of nitrogen-16 estimated on the basis of: irradiation time, neutronic flux and transit times. The nitrogen-16 concentration changes inside the decay tanks and, for the calculations the following concentrations in the inlet and outlet of every tank have been assumed:

	Primary Cooling System Decay Tank [$Bqcm^{-3}$]	Reactor and Service Pool Decay Tank [$Bqcm^{-3}$]
Inlet	6.9×10^{5}	$2. \times 10^{5}$
Outlet	8.4	2.8

The main shielding is provided by the heavy concrete walls that constitute the reactor pool and service pool shielding. MERCURE Code was used to calculate wall thickness.

Labyrinth access is calculated on the basis of contained radiation source and space needs. The code MERCURE-4 and MCNP-4C were used for calculations, with a complete model considering both decay tanks and associated piping.

12.3.2.6 Hot Cells

Different hot cells are available in the Reactor Facility. They include the Loading Hot Cell (LHC) and the Above Pool Hot Cells Complex (APHCC) comprises the Transfer Hot Cell (THC) and two Pool Hot Cells (PHCs).

A typical calculation for a hot cell shielding is presented here. Some cells are to be used as an intermediate station for samples, before they are sent to another station, and other cells may contain a source for a certain amount of time.

In order to determine the minimum shielding thickness, the maximum source to be contained must be determined.

1. Radiation Sources

   Among the different radiation sources to be considered are:

   a) Aluminium canisters used to transport samples in the pneumatic conveyor system for PHCs.

   b) Five Mo-99 targets loaded in a rig in the irradiation position. The rig has been irradiated for 7 days and has decayed for several hours before it is transferred into the THC and LHC.

   c) 100 GBq of Na-24 (plus the cans material activity) for PHCs.

   d) Item (b) but just one target for ICE (Inter hot cells elevator).

2. Shielding Materials

   Iron and lead were selected as shielding door material and supplements.

3. Design Ambient Dose Equivalent Rate

   A maximum contact dose rate value (e.g. 10 μSvh^{-1}) was used.

4. Calculation Method

   The MERCURE-4 and MCNP-4C Code were used. The source is located in a position that can be considered typical. Provisions are taken to ensure that, in any possible location, doses in occupied areas are below acceptable values.

5. Wall, roof and floor thickness

   The calculated thicknesses are sufficient to attenuate the radiation levels to ensure that doses to operating personnel are within safe limits.

6. Shielding Door Thickness

   The calculated door lead thickness is sufficient to attenuate the radiation levels to ensure that doses to operating personnel are within safe limits.

12.3.2.7 Neutron Guide Bunkers

Neutron guide bunkers are provided with shielding designed to protect the staff and researchers.

The main shielding structures are calculated as in previous cases. A plug door has been provided for maintenance activities.

12.3.2.8 Resin Beds

Resin beds are located behind shielding that protects the area from where system valves are controlled. Labyrinth access is provided to allow in service inspection.

Shielding has been designed so access to a resin bed train is possible while neighbouring units are in operation.

The main shielding structure is calculated as in previous cases. Special care is taken with penetrations and layout of pipelines to ensure protection during resin bed transport.

The source used in the shielding calculation corresponds to the maximum load admissible before resin beds are replaced by fresh ones.

12.3.2.9 Heavy Water System

The heavy water room has been defined as a forbidden area during reactor operation. Nevertheless shielded walls have been located surrounding pumps and heat exchangers in order to reduce doses originated by a concentration of 3×10^6 $Bqcm^{-3}$ of nitrogen-16 produced in the D_2O..

The resin beds are located behind an additional shield to allow maintenance activities in the pumps and other components located inside the room during shutdowns.

12.3.2.10 Optimisation of the Main Shielding Structures

12.3.2.10.1 General Considerations

One of the main radiation protection objectives is that radiation exposure levels of site personnel and public should remain below prescribed limits and kept ALARA.

Further optimisation was performed when required during the detailed engineering stage and during the implementation of radiological practices.

The conceptual approach to optimisation of radiation protection is a cost-benefit analysis involving the cost of radiation protection and the cost of detriment. It implies that improvements of radiation protection are correlated to reductions in the cost of detriment and that efforts in improving radiation protection can be quantified by their cost.

The International Commission on Radiological Protection (ICRP) introduced the concept of radiation detriment to identify and quantify all deleterious effects due to exposure to ionising radiation. The collective dose is an extensive quantity useful as an indicator of the health detriment, because the individual health detriment is proportional to the effective dose equivalent. In the case of doses produced by external irradiation shielding optimisation does not require consideration of the collective dose to represent the corresponding detriment.

Optimisation of the level of radiation protection can be formally presented by a system of mathematical expressions that must be solved simultaneously, as follows:

$$U = X_{(\omega)} + Y_{(\omega)} = \text{minimum} \quad [1]$$

where:

X is the cost of radiation protection

Y is the cost of radiation detriment

ω is a parameter representing the level of radiation protection

U expresses the final result of the optimisation procedure and is usually called the objective function.

In our case ω represents the shielding thickness.

The main shielding structures for which the optimum thickness have been calculated are:

a) Reactor pool (radial and axial)

b) Service pool (lateral, axial)

c) Decay tanks (lateral wall)

d) Secondary and auxiliary shields

12.3.2.10.2 Calculation Methodology

In all cases, the radioactive sources used in the calculations are those considered to cause the worst demand on the shielding.

From Eq. [1], and using as optimisation parameter the additional thickness (*ath*):

$$\frac{\partial X_{(ath)}}{\partial ath} = \frac{-\partial Y_{(ath)}}{\partial ath} \quad [2]$$

The detriment cost can be expressed as a function of the same parameter:

$$Y_{(ath)} = \alpha . N . R . \rho . T . \dot{E}_{(ath)} \quad [3]$$

where:

α : is the dimensional constant expressing the cost assigned to the collective dose unit (α= A10^5 per person Sievert –ANSTO SD 5.2).

N :is the number of exposed individuals

R : is the occupancy time factor

ρ : is the constant ratio between average and maximum dose rates

T : expected life time of the installation

$\dot{E}_{(ath)}$: equivalent dose rate corresponding to the level of protection considered and can be expressed for a flat shield as:

$$\dot{E}_{(ath)} = \dot{E}_0 . e^{-(\mu . ath)} \qquad [4]$$

$\dot{E}_0$: dose rate corresponding to the minimum level of protection required to fulfil specified limits.

μ : constant representing the attenuation of the radiation in the shielding material

The cost of protection *X* can be divided into two terms:

$$X_{(ath)} = C_h . V_{h(ath)} + C_o . V_{o(ath)} \qquad [5]$$

where:

C_h: is the cost of installed heavy concrete per unit volume

$V_{h(ath)}$: is the volume of heavy concrete corresponding to the additional thickness *ath*

C_o: is the cost of installed ordinary concrete per unit volume

$V_{o(ath)}$: is the increment of ordinary concrete volume resulting from the increase of the inner HC wall external radius

As the shielding being considered are either a slab, cylinder (axial) or cylinder (radial), specific expression for the cost of protection (*X*) and of its derivative $\partial X_{(ath)} / \partial (ath)$ for each case have been formulated.

The value used to represent the cost of detriment is chosen equal to A$100,000 per person Sievert (ANSTO Safety Directive 5.2).

In addition to the previous formalism, objective values are defined so that, if dose levels fall below them, no further optimisation is required. See Section 12.1.2.

The objective value for radiation workers is 2 $mSvy^{-1}$ (ANSTO SD 5.2).

12.3.2.11 Materials

Main characteristics of commonly used shielding materials.

12.3.2.11.1 Normal Weight Concrete

Normal concrete design values are shown in Table 12.3/9.

For shielding purposes a normal concrete density of 2,200 kgm^{-3} has been conservatively assumed. The composition for this case is presented in Table 12.3/10.

12.3.2.11.2 Heavy Weight Concrete

The heavy weight concrete combines radiation absorption properties with good mechanical characteristics and durability.

The heavy concrete composition is presented in Table 12.3/11.

12.3.2.11.3 Steel

The design values are shown in Table 12.3/12.

12.3.2.11.4 Lead

The lead density design values is 11.3 $gcm^{-}3$.

12.3.3 Minimisation of Exposures above the Reactor Pool

In order to keep to a minimum the dose rate for areas above the pool that may be occupied, the design incorporates the following provisions.

a) The Primary Cooling System is provided with a by-pass flow that flows through what is called the “interconnection branch” from the Primary Cooling System to the Reactor and Service Pool Cooling System and then to the reactor pool. Therefore the flow is forced to go in a downwards direction through the reactor chimney thus preventing the primary coolant flow (and consequently nitrogen-16, sodium-24, argon-41 and/or other activated material transported by the coolant) leaving the core from reaching the reactor pool surface.

b) A hot water layer is provided. The hot water layer is a layer of water on the reactor. The hot water layer objective is to minimise the mixing between the bulk of the reactor pool water that contains radioisotopes such as sodium-24, argon-41, with the top layer thus reducing the dose rate at the reactor pool surface.

c) The hot water layer is also provided with its own water purification system with filters and ion exchange resins that are effective for the removal of Na^{24} ions. In this way high purity is ensured at the water layer extending on the reactor pool surface.

d) Attention is given in the design of the hot water layer and any other streams feeding into the upper zone of the reactor pool to provide appropriate flow diffusers to protect the hot water layer from perturbing flows. A diffuser is also placed on the returning line of the Reactor and Service Pool Cooling System in order to minimise the mixing currents within the reactor pool.

e) The Operations Bridge that runs above the reactor pool is the natural place to carry out operations over the reactor pool. The bridge floor is located above the water surface and distance provides an extra reduction in the dose rate in areas possible to be occupied.

f) Complementing the purification system of the hot water layer, there is the Reactor Coolant Purification System provided with ion exchange resins that helps to remove Na^{24} and other particles from the reactor pool water. This system keeps the water quality by reducing the corrosion phenomena and preventing, at the same time, the activation of impurities in the water that would otherwise occur.

12.3.4 Ventilation

12.3.4.1 System Description

The Containment Ventilation System encompass the Containment Air Supply, the Containment Air Exhaust, the Hot Cells Ventilation, the Heavy Water Room Ventilation and the Control Rod Drives Conditioning (See Chapter 10). The air circulation and conditioning functions are supplemented by the Containment Energy Removal System (See Chapter 7).

12.3.4.2 Required Air Change Rate

The following methodology was used to determine the required air change rate within the Containment. The basic air change rate to achieve recommended concentrations was first determined. An ALARA assessment was then undertaken to determine whether further reduction in the concentrations was cost effective.

12.3.4.2.1 Basic Air Renewal Rate

The balance equation for the concentration in the reactor hall for an isotope "i" is:

$$\frac{dC_i}{dt} = \frac{R_i}{V} - \frac{Q.C_i}{V} - \tau_i.C_i \qquad [6]$$

where:

C_i : Concentration of isotope "i" (Bq/m^3)

V : Reactor hall volume

R_i : Release rate of isotope "i" from the pool top to the hall

Q : Renewal flow (m^{3}h^{-1})

τ_i : Decay constant of isotope "i" (1/h)

The release from the reactor pool top to the reactor hall area has been determined assuming that:

a) The concentration in water corresponds to the values listed in Table 12.2/4.

b) Evaporation from the reactor and service pools occurs due to the hot water layer system.

c) For aerosols the activity contained in the evaporated water is assumed to be dispersed in the reactor hall.

d) For noble gases an interchange velocity of 0.057 mms^{-1} has been measured in a similar facility.

In equation [6], the first term on the right side is the source due to releases from the pools, the second is the activity removed by the air renewal and the third one is the removal by radioactive decay.

Considering a steady state regime, the time derivative cancels:

$$C_i = \frac{R_i}{Q + \tau_i.V} \qquad [7]$$

Considering all the relevant isotopes and the respective dose conversion factors:

$$\sum_{i=1}^{isot} \frac{R_i . f_{di}}{Q + \tau_i . V} \leq \text{dose rate target} \quad [8]$$

where:

f_{di}: dosimetric factor (taken from IAEA SS-115 and corrected to account for the Hall geometry)

The minimum Q satisfying equation [8] will comply with the design target.

12.3.4.2.2 ALARA Assessment

The optimisation requirement of radiation protection is that the doses should be kept "as low as reasonably achievable" (ALARA). This requirement consists in limiting the detriment from a given practice or radiation source to a value such that further reductions are considered less significant than the additional efforts required to achieve such reductions.

In a general way a cost benefit analysis is performed:

Cost = Y(Q) + X(Q) [9]

where:

Y(Q) : the cost of human detriment

X(Q) : the protection cost

The optimum condition is obtained by calculating the minimum of expression [9].

The cost of human detriment is calculated as:

$$Y(Q) = \alpha . \underset{E}{\overset{c}{S}} \quad [10]$$

where:

α : cost assigned to the unit collective dose (100,000 A$ per person Sievert)

S^{C}_{E}: Collective effective dose that can be computed as:

$$\underset{E}{\overset{c}{S}} = N . f_{occ} . T_{life} \sum_{i=1}^{isot} f_{di} . C_i \quad [11]$$

where

f_{occ}: Occupancy factor of the reactor hall

N: number of personnel working in the reactor hall

T_{life}: lifetime of the ventilation system (40 years)

In order to compute the protection cost X(Q) only the dependence of the operation cost on the power have been considered. This is a conservative assumption as no credit is given to the associated increase on the capital cost of the system on the ventilation flow.

$$X(Q) = a \ b \ Ftt \ T_{life} \ Q \quad [12]$$

where:

a: cost of the electric energy ($ US/kWh)

b: required electric power per flow

Ftt: Percentage of time the system is in operation

Taking the derivative of [12] with respect to Q and searching for the extreme condition (=0):

$$0 = a.b.Ftt.T_{life} - \alpha.N.f_{occ}.T_{life}\sum^{i_{sot}}\frac{f_{di}.R_i}{(Q+\tau_i.V)^2} \qquad [13]$$

Solving [13] for Q the minimum value of Q is obtained. Nevertheless, based on the objective value declared in Section 12.1.2, a renewal flow calculated using the Equation 8 and assuming a dose rate target equal to this objective is used.

This optimal value for Q is compared with the value previously obtained and the maximum value adopted.

12.3.5 Radiation and Contamination Monitoring Instrumentation

This Section presents the design requirements, standards and description of systems and related instrumentation used for radiation exposure and contamination monitoring inside the Reactor Facility.

This description comprises the Radiation Monitoring System (RMS) of the Reactor Facility, which performs all on line radiation surveillance tasks related to the operation of the reactor. The surveillance tasks include both exposure and contamination monitoring during normal operation and potential accident conditions. Portable detectors, used inside the Reactor Facility for surface contamination detection on individuals, tools and clothes, portable air samplers as well as personnel dosimeters are also available.

The RMS accomplishes the following functions:

Monitoring of Area

Monitoring of Personnel

Monitoring of Gases

Monitoring of Liquids

The RMS is a subsystem of the Reactor Control and Monitoring System (RCMS). The Information brought by the RMS is sent to the RCMS and –partially- to the Reactor Protection systems (RPS) and the Post-Accident Monitoring System (PAM) consoles. See Chapter 8, Section 8.2, Section 8.6 and Section 8.7 for RPS, PAM and RCMS details.

The general design requirements of the RMS are similar to those of the RCMS (see Chapter 8, Section 8.7.2). These requirements are imposed to all subsystems of the RMS and can be described as follows:

a) Provision of multiple inputs to RMS

b) Provision of video display units (VDUs) to display real time values, automatic alarming of abnormal releases or contamination events, and automatic compiling, logging and integration of releases.

c) Provision of individual hard-wired displays isolated from the computer-based system when post accident monitoring capabilities are required. Post-accident radiation monitoring signals are also inputs into the RCMS by qualified isolation devices.

The specific design requirements for each subsystem integrating the RMS are presented in the corresponding Section of this Chapter.

The RMS includes sensors, cable system, intelligent processing monitors, own controlling software and communication protocols. For RPS and PAM contributions, the system generates isolated hard-wired signals, which provide adequate and reliable separation between process and engineering safety functions.

The Area Radiation Monitoring System (ARM) and the Neutron Monitoring System (NMS) perform the RMS area radiation monitoring function (gamma and neutrons) inside the Reactor Facility.

Walk-Through (Portal) monitors are located at the exit from areas of the Facility where personnel finally exit the controlled areas.

The RMS performs the on line monitoring of gaseous effluents through the Air Effluents Monitoring (AEM) and the Tritium Monitoring (TRM). The AEM monitors continuously the radioactive stack emissions of aerosols, iodine and inert gases. The TRM performs the on line monitoring of tritium releases through the stack, using a fixed monitoring intelligent unit, capable of discriminating tritium from inert gases.

Additionally, the TRM monitors tritium at other specific locations. The RMS monitors the liquids inside the Reactor Facility through different subsystems.

One of these is the Liquid Effluent Monitor (LEM), which integrates the Radioactive Liquid Waste Management System (RLWMS) and supports the waste discharge commands.

Other surveillance functions are performed by three different monitoring stations: the Waste Streams Activity Monitor (WASMO), the Active Liquids Monitor (ALMO) and the Secondary Water Activity Monitor (SAMO).

WASMO performs on-line monitoring of all potentially radioactive liquid waste, collected from routine plants and floor cleaning operations, emergency showers drainage, standard equipment drainage, etc., that are delivered to dedicate Waste Tanks.

The ALMO monitors continuously the gamma activity of the Primary Cooling System (PCS) and the Reactor and Service Pools Cooling system (RSPCS). These monitors also complement the FFEM, as a diverse monitoring function.

The SAMO monitors continuously water from the Secondary Cooling System (SCS) to detect possible leaks through the heat exchangers that could lead to contamination of the normally non-active secondary cooling water.

The Failed Fuel Elements Monitor (FFEM) controls fuel elements as well as rigs conditions supervising continuously the PCS and the RSPCS and providing early warnings.

The RMS reports to the RCMS and has inputs from all the radiation monitoring sub-systems. In agreement with Design Requirements, VDU's (video display units) provides for the radiation monitor displays to display real time values, automatic alarming of abnormal releases or high dose-rate events, and automatic compiling, logging and integration of releases.

Redundant detection units were implemented for those subsystems related to RPS / PAM functions. Their monitoring signals are also supplied to the RCMS system by qualified isolation devices.

The following RMS parameters that feed the PAM System are displayed in both the MCR and the ECC:

Reactor Pool Area Gamma Dose Rate.

Reactor Hall Gamma Dose Rate.

Basement Gamma Dose Rate.

Reactor Pool Open End (RPO Top) Gamma Dose Rate.

Stack aerosols activity.

Stack Iodine activity.

Stack Noble Gases activity.

Uninterruptible back-up power supplies (UPS back-up) are provided for radiation monitors from the corresponding RCMS or RPS/PAM redundancies UPS's. UPS do not feed sampling pumps.

Design of RMS units was performed according the following standards, where applicable:

IEEE 323 (1983-R1996) Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Station

IEEE 344 (1987-R1993) for Recommended Practice for Seismic Qualification of Class 1E Equipment

IEEE 830 (1998) for Recommended Practice for Software Requirements

IEEE 352 (1987 – R 1999) for General Principles of Reliability Analysis

Military Handbook 217F for Reliability Prediction of electronic equipment

IAEA TECDOC 478 – Components Reliability Database

IEEE 1023-1988 for the Application of Human Factors Engineering to Systems, Equipment, and Facilities

IEEE 379 – 1994 for the Application of the Single Failure Criterion

IEEE 384-1992 (R1998) for Independence of Class 1E Equipment and Circuits

Additionally, all measured data presented at the monitoring units described below, are displayed in units according Norm AS 1000 (The International System of Units). All monitoring units integrating RMS comply with the guidelines concerning Radiation Protection Criteria described at Sections 5.4 of Regulatory Guideline RG-5 from Australia's Nuclear Safety Bureau (1998), and 4.24 of RB – STD 43.00 from ARPANSA Regulatory Branch (Dec.2000).

Although there is diversity in the processes monitored, there exists a similarity between these monitors from the point of view of their functional behaviour.

This description is applicable to monitors of: LEM, FFEM, SAMO, WASMO, ALMO and AEM.

The monitor architecture is composed of logical units and the functions of the two softwares, the <u>Embedded Software (ES)</u> and the <u>Configuration and Maintenance Software (CMS)</u>.

Figure 12.3/22 illustrates the system context of the radiation monitor and the placement of ES and CMS within the operation of the monitor. Following, there is a brief description of the blocks in the figure:

a) <u>Sampling process:</u> consists in all the process necessary for taking samples from the monitored medium to an adequate recipient where the detector is housed, for the corresponding activity measuring. In this process are involved electrical and manual valves, pumps, flow rate meters, pressure meters, water-level (magnetic gauges) switches, other position microswitches. This process is managed by a “Hard logic Unit”.

b) Counting process: the process of acquiring the radiation activity of the samples, taken in the sampling process, by means of the spectrometry chain, which consists in a radiation detector and all the electronic signal conditioning instrumentation.

c) Hardware status: it refers to the current status of the hardware components of the equipment. It includes the state of voltage sources, spectrometry chain and other electrical components.

d) Operation console: is the local user interface of the monitor. This console is wired to a “Hard Logic Unit” which is responsible of executing the requested actions, verifying if the necessary conditions are met, initiating and synchronizing the operations. ES: Comprises the internal software of the monitor. It functions acquiring data coming from the counting and sampling processes, perform statistical calculations with the count-rates measured, check the hardware status for possible failures that could invalidate the measuring, and communicate with the RCMS.

e) CMS: The configuration and maintenance software that runs on a portable computer (a PC notebook), for the configuration, calibration, and low level inspection of the monitor.

f) VDU: are the display units at the SU for remote monitoring.

g) Remote communications: Each monitor sends response messages to queries coming from a device that actuates as the Modbus Master unit. The ES, in charge of the intelligent unit of the monitor, is responsible of managing the communications through the serial ports on the computer board. This field communication is redundant.

h) Local communication: locally, ES also handle a serial communication link with a portable notebook console running the CMS. Through this link, ES modify its internal variables with new parameters entered from the notebook.

12.3.5.1 Fixed Area Monitoring System

The Reactor Facility has a system of fixed radiation detectors in those areas where continuous supervision of personnel is considered necessary.

The fixed area monitoring system comprehends two RMS subsystems:

Area Radiation Monitoring (ARM).

Neutron Monitoring System (NMS).

The specific design requirements imposed to the ARM are the following:

1. To provide sufficient fixed-area monitors to cover the area around the reactor pool and any area where radiation measurement is relevant.
2. To provide a major set of area radiation monitors with local displays and alarms to give adequate warning to personnel of rising or excessive radiation levels.
3. To apply the following criteria to determine the distribution of radiation detectors in each area :
   a) Redundancy in places where it is necessary to have a reliable reading available under different operation conditions.

b) In those places where a significant variation in the exposure rate can be potentially produced, an additional detector of higher range is provided.

c) In areas where dose distribution is expected not to be homogeneous additional detectors are provided.

4. To provide sufficient radiation monitors in the Reactor Hall, which verify conditions of post-accident monitoring (PAM) instruments
5. To provide sufficient radiation monitors which verify conditions of RPS instruments.

All the ARM units are based on Geiger-Muller detectors of two types:

Low and medium dose rate detectors

High dose rate detectors

Some gamma Area Radiation Monitoring units are "Post Accident Monitoring (PAM) – conformed". For those units the area monitoring modules avoid the use of embedded "local intelligence", i.e. consisting only of the sensing head connected directly, via the Conditioning Units (CDU's) and Isolation Units (IU's), respectively, to the Post Accident Monitoring consoles, following the PAM Architecture of the Radiation Monitoring System (RMS).

Also, three analogue ARM units are foreseen to be "RPS- conformed". As in the case of “PAM-conformed” units, they are only dose rate “sensing” units supplying the information in form of an electric current. As they are not “intelligent” units, they neither perform any comparison with a pre-set alarm level nor present any alarm signal. These functions are performed within the RPS to which the (current) signals are being sent.

Another type of detector arrays included in ARM are ”intelligent” digital detectors, software operated.

The intelligent digital units are designed for radiological surveillance of areas, providing adequate warnings to the personnel potentially exposed to abnormal radiation levels, by means of visual and acoustic alarms.

Each (micro-controller based) “intelligent” module contains a Geiger detector, signal conditioning electronics, high and low voltage power supply and serial communication hardware.

The “intelligent modules” are designed in order to act as “stand – alone” radiation level detectors as well as intelligent remote units, at the same time. As remote units they have several differences with respect to analogue ones. In first place they can be interchanged without any need of system re-calibration. In second place they perform self - check of their functions.

The whole ARM system, including digital and analogue units, is fed by an un-interruptible 24 V line.

The Neutron Monitoring System (NMS) includes fixed monitors, able to measure neutron dose rate. The following specific design requirements were imposed to NMS:

1. To provide a fixed set of radiation monitors to measure neutron dose-rates at appropriate locations.
2. To provide with local displays and alarms all NMS units, in order to give adequate warning to personnel.

In order to accomplish these requirements, area monitors capable of determining the ambient effective dose rate produced by neutrons are located in the Reactor Beam Hall. The monitors have local and remote readings as well as alarms similar to gamma area monitors.

12.3.5.2 Continuous Noble Gases, Aerosols, Iodine and Tritium Effluent Monitoring

The following design requirements were imposed to the Air Effluent Monitoring System (AEM):

1. To monitor continuously the radioactive stack emissions in normal and post accident conditions.
2. To generate real-time data (allowing isotope identification) and stack emissions accountability, under normal operation conditions. To monitor for radioactive aerosols, iodine and noble gases.
3. To provide a different and independent (analogue) assembly for aerosols, iodine and noble gases effluents monitoring, conforming the requirements for Post Accident Monitoring (PAM) and Reactor Protection System (RPS) equipment (constructed as a Safety Category 1 ESF, meeting Seismic Category 1 design requirements).
4. To provide alarms and initiation signals for containment isolation (AEM only).
5. To provide information about stack releases to be used for plume plotting.
6. To take into consideration iodine plate-out in sample lines for all iodine monitors.
7. To sample stack effluents coming from containment and hot cells to provide signals to RPS

The Air Effluents Monitor-Intelligent supervises –through continuous, “on line” sampling - the stack emissions of gaseous effluents (Aerosols, Iodine, Noble Gases) from the Reactor Facility, verifying that discharge notification levels are not exceeded under normal operation, and tracking the emissions in post accident conditions. On-line (non-PAM) tritium monitoring for stack releases, is performed (separately) using equipment described below.

The Air Effluents Monitor consists of 2 (two) subsystems : one is “analogue/hardwired” Air Effluent Monitoring (AEM), conformed to PAM (Post Accident Monitoring) and RPS (Reactor Protection System). Analogue AEM assemblies are hardwired to RPS and PAM consoles and no intelligent units are designed for this links. The other one is Air Effluent Monitoring-Intelligent Stack Effluent Spectrometric Channel (AEMi), connected to RCMS (Reactor Control and Monitoring System) for alarms presentation.

Signals generated by the analogue AEM are used by the RPS in order to trigger alarms and initiate containment isolation actions.
Requirements on RPS assume triple redundancy for analogue AEM assemblies, among them units I and II being shared with PAM I and PAM II redundancies. The AEMi unit connected to the RCMS adds gross-beta detection heads –and corresponding electronics- for Aerosols and Inert Gases.

For each assembly, the monitor’s air sampling pump takes a fraction of the effluent gases first through a glass-fibre filter, and then through an activated charcoal filter, the first one to retain aerosols/particulates and the second one to retain iodine. After flowing through these filters, the air sample is driven into a delay chamber where noble gases concentration is measured. Pressure switches installed at each detection container enables to monitor the differential pressure at the air sampling circuit, allowing supervision of filter condition and shielding enclosures.

In the case of analogue monitors, air samples are taken (by iso-kinetic nozzles) from containment's ventilation duct exhaust (including air from reactor's hall and hot cells), while for the AEMi the samples are pumped directly from reactor's stack (including air sampled from reactor's containment and from reactor building non-conventional areas).

The sampling lines, both from stack up to monitor's inlet, as well as inside the monitor itself connecting the different detecting units are designed following recommendations in order to minimize Iodine plate-out and Aerosols/particulate deposition effects. The purpose of the stack monitor (AEMi) is to provide information of what and how much is being released –as a whole- through the stack.

The air effluents monitoring assemblies are supported by SPS diesel generators.

In agreement to design requirements established, hardwired AEM subsystem is classified as Safety Category 1, being designed accordingly to Standards IEEE Class1E. The AEMi is classified as Safety Category 2.

A seismic evaluation of analogue AEM was carried out to demonstrate structural integrity and ability of this equipment to perform its required safety functions under SL-2 conditions. This verification covered the tubing and hoses, the supports and the mounting of the electric and electronic components. The basic input data of equipment, components and structures used in this analysis correspond to the detailed engineering stage and when appropriate provided by the manufacturers. This included all relevant parameters pertinent to the analysis. Material characteristics were extracted from appropriate codes and standards and/or provided by the manufacturers. The Seismic Spectra for Seismic Class SL-2 for the pooltop level was considered. The analysis showed that the structures are quite rigid and robust, with wide margin for the stress imposed by a SL-2 earthquake.

With respect to Tritium effluent, the following specific design requirements were imposed to TRM:

1. To monitor Tritium effluent at the Gaseous Effluents stack with a unit completely independent from AEM.
2. To ensure appropriate discrimination to distinguish between tritium and other radioactive species such as (^{14}C and ^{85}Kr, ^{133}Xe, etc.).
3. To sample stack effluents coming from containment, hot cells and labs.

In agreement to these requirements, a fixed Tritium Monitoring Unit with Tritium / Noble Gases discrimination capability is installed in the Gaseous Effluents stack. This is directly connected to the stack to monitor continuously air effluents to measure Tritium concentration released to the environment, separating this contribution to those coming from discharges of other radioactive species such as ^{14}C and ^{85}Kr, ^{133}Xe or ^{41}Ar.

Air samples taken from the stack, come from Containment and Hot Cells, as well as from reactor building non-conventional areas; the whole samples are monitored continuously by the Tritium Monitoring unit with Noble Gases discrimination capability.

On-line tritium monitoring for stack releases is non–PAM and has back up facilities for bubbler sampling.

12.3.5.3 Walk Through (Portal) Monitors

Walk through (portal) monitors designed to detect and measure surface contamination on people who leave the Reactor Facility building are provided.

These monitors are located to ensure that all personnel who may have potentially entered the containment are monitored for contamination prior to leaving the facility.

12.3.5.4 Continuous Tritium Detectors

Tritium Monitoring System (TRM) monitors tritium at specific locations. The following design requirements were imposed to this system:

1. To sample Tritium at the Heavy Water Room (with actions on the local ventilation), and at the Ventilation Room adjacent to the Heavy Water Room (administrative control only).
2. To sample Tritium at the air treatment circuit corresponding to the Tritiated Water - processing Glove Box in the Chemistry Blue Lab
3. To sample stack effluents coming from containment, hot cells and labs.

In order to achieve the TRM functions at both rooms described in 1 (above), a “Fixed Tritium in Air” Monitoring Unit, for continuous monitoring of tritium in air, is provided in the heavy water process room. This system has command of the heavy water process room ventilation) and can initiate tritium scrubbing if tritium is detected.

This function is achieved sampling the ventilation duct with a Tritium / Noble Gases monitoring Unit similar to that used to monitor air stack effluents.

Monitoring functions described at 2. above are performed using a movable Tritium in Air Monitor which consists of a gas flow Tritium ionisation chamber. This monitor performs not only surveillance at the Tritiated Water Glove Box but also local monitoring throughout the Chemistry Blue Lab and the Heavy Water Ventilation Room

External communication is performed by means of PC’s serial communication link, thus enabling remote communication and data transmission to the Supervision Units at the Main Control Room, Emergency Control Centre and Health Physics Room. The movable Tritium monitoring unit is linked to the network by means of an isolation device.

12.3.5.5 Monitoring of Liquid Effluents and Waste Streams

There are two systems monitoring possible activated liquids. One, the Waste Streams Monitor (WASMO), is dedicated to measure the activity of effluents at the collection tanks and the other, Liquid Effluent Monitor (LEM), measures the activities being released to the B and C lines of LHSTC.

Both systems are sodium iodine (NaI) detector based, each one with the corresponding electronics to acquire data and, in the case of LEM, trigger isolation valves.

Due to the fact that requirements to discharge effluents to B and C lines are not only related with radiological parameters (for example, chemical restrictions apply), homogenisation, sampling and further lab analyses procedures are necessary before allowing the release.

The following specific design requirements were imposed to WASMO:

1. To provide continuous on-line gamma monitoring for all potentially radioactive liquid waste.
2. To supply on-line gamma monitoring for all low activity level waste streams.

WASMO performs on-line monitoring of all potentially radioactive liquid waste (collected from routine plants and floor cleaning operations, emergency showers drainage, standard equipment drainage, etc.) delivered to one of the Waste Tanks. On the other hand, it performs monitoring of low-level waste effluents streams (collected from labs, decontamination equipment, eventual pool leaks drainages and pool water from process equipment drainages), delivered to the other Waste Tank.

The WASMO detectors trigger alarms when normal values are exceeded indicating that an abnormal operation is being performed in the labs or areas connected to the collection network.

Local Communication Interface consists on a Notebook PC, as portable Console, providing a local display for visual Interface, equipment calibration and adjustment of parameters and alarms limits.

Remote Communication Interface is achieved through a (duplicated) serial communication channel, linked to the network.

With respect to the LEM the following specific design requirements were imposed:

1. To generate alarm signals to warn of radioactive liquid discharges approaching or exceeding the allowed limits.
2. To perform “off-line” (at the Labs) sampling and analysis of liquid effluents, prior to make a decision about discharge of liquid effluent.
3. To provide automatic stopping of liquid discharge when the gamma activity levels sampled on the current discharge exceeds the set-up.
4. To monitor effluents discharge lines “B” (low-level liquid waste) and “C” (potentially active liquid waste) during liquid waste releases.
5. To acquire signals from flow totalising indicators (for each one of the discharging lines), in order to quantify total activities being released.
6. To act the effluents discharging valves (corresponding respectively to lines “C” and “B) in order to enable / stop liquid effluents discharges, according the gamma total activity levels being measured and the reference limits set by laboratory analysis.

The LEM forms part of the Radioactive Liquid Waste Management System (RLWMS) and supports the waste discharge commands.

This monitor is classified as Safety Category 2.

The main task of the LEM is to monitor the gamma activity levels of the liquid effluents stored in the Waste Tanks. Monitoring is performed during homogenisation of the waste, prior to sampling, and then during the discharge through lines C (normally non-radioactive liquid waste) and B (low-level radioactive liquid waste), towards ANSTO’s Effluents Processing Plant, in order to decide whether this discharge can proceed or if it shall be suspended (following radiation protection criteria and activity release standards). In case authorised discharged limit established for discharge lines B and C is surpassed, the LEM “commands” the discharge valves, in order to cut off the discharge.

The LEM trigger local - as well as remote (at RCMS)- alarm signals in case that a liquid release approaching (or exceeding) the allowed limits for total activity, and time rate activity concentration is detected.

Local Communication Interface consists on a Notebook PC, as portable Console, providing a local display for visual Interface and equipment calibration and adjustment of parameters. Remote Communication Interface is achieved through a (duplicated) serial communication channel, linked to the network.

12.3.5.6 Secondary Cooling System Water Monitoring

This monitor is designed to detect coolant leakage between the primary and secondary circuits due to a failure in heat exchangers.

The specific design requirement imposed to this system is to supervise the gamma activity levels of Secondary Cooling System.

If a leakage appears in the heat exchangers, part of the water of the primary circuit will flow to the Secondary Cooling System (SCS). As the gamma activity in the primary circuit is mainly due to ^{24}Na and ^{41}Ar, if a leakage occurs these isotopes would appear in the SCS. Secondary water is continuously sampled and measured by the Secondary Water Activity Monitor (SAMO). A high value over the background signal triggers an alarm signal.

A fraction of the flow of the Secondary Cooling System is directed to a detection chamber where a NaI scintillator is located. The detector provides contamination alarm signals in case of detection of abnormal activity concentrations.

Water samples are driven into the monitor by direct convection, from the Secondary Cooling System pumped by the Secondary System Main Pumps during reactor normal operation, and by the Residual Heat Pumps during reactor programmed shutdown, respectively. This water samples are continuously monitored by the SAMO.

This monitor is classified as Safety Category 2.

Local Communication Interface, consists on a Notebook PC, as portable Console, providing a local display for visual Interface and equipment calibration and adjustment of parameters and alarms limits. Remote Communication Interface is achieved through a (duplicated) serial communication channel, linked to the network.

12.3.5.7 Failed Fuel Elements Monitor

The Failed Fuel Elements Monitor (FFEM) is used to detect fuel plate cladding failures at early stages.

The following specific design requirements were imposed to this system:

1. To enable early warning of eventual fuel failures.
2. To enable early warning of eventual molybdenum rigs failures.

The FFEM continuously monitors water samples taken from the PCS continuous flow.

It supervises the Primary Cooling Circuit (FFEM # PCS), as well as the Reactor and Service Pools Cooling System (FFEM # RSPCS), in order to detect possible Fuel Element and molybdenum rigs failures, and to provide early warnings through the RCMS.

The FFEM # PCS monitors fission products generated and released to the Primary Cooling water in case of Fuel Elements failure. The FFEM # RSPCS, at its turn, detects fission products generated and released to the Reactor and Service Pools Cooling System in case of a rig failure.

These monitors are classified as Safety Category 2.

Samples of both Primary Cooling System and Reactor and Service Pool Cooling System are taken before entering the corresponding Decay Tank, both being returned downstream to the same lines after being monitored.

Two independent sampling pumps lead water samples taken separately (at constant flow rates) from Primary Cooling System and Reactor and Service Pool Cooling System, into the respective measuring units.

The sampling pumps are installed at their corresponding Pump Assemblies separated from FFEM's corresponding racks, being commanded from each monitor's Intelligent Unit IU.

Local Communication Interface, consists on a Notebook PC, as portable Console, providing a local display for visual Interface and equipment calibration and adjustment of parameters.

Remote Communication Interface is achieved through a (duplicated) serial communication channel, linked to the network.

12.3.5.8 Active Liquids Monitor (ALMO)

The specific design requirements imposed to this system is to supervise the gamma activity levels of Primary Cooling System and Reactor and Service Pools Cooling System.

The Active Liquids Monitor (ALMO), monitors continuously both the gamma activity of the Reactor and Service Pools Cooling System (ALMO # RSPCS) and that of the Primary Cooling System (ALMO # PCS), respectively.

This monitoring function is achieved by two separate gamma Dose Rate detecting units, both respectively connected in this case to independent Intelligent Units (processing modules. The first corresponding to the Primary Cooling System monitoring station, while the second is dedicated to the Reactor and Service Pools Cooling System monitor. These monitors also complement FFEM, as a diverse monitoring function. They are classified as Safety Category 2.

The location of the detection unit was selected, in the case of the Reactor and Service Pools Cooling System monitoring in order to enable measurement of the water coming from pool cooling systems and rigs as well. This unit was installed close to the line that connects the RSPCS outlet pipeline with the pumps suction lines. In case of Primary Cooling System, monitoring unit is installed close to the Decay Tank outlet line after block isolation valve.

Detection units are placed, in each case, in front of the pipes being scanned. No direct contact between detection units and scanned flow exists.

Each gamma activity sensing unit is connected to its own processing module, installed nearby inside the locations described above.

12.3.6 Decontamination System

The facility design aims at prevention of contamination, easy identification of its origin in case it occurs, and rapid correction of the problem.

Areas where spills or leaks of active liquids can occur are designed so that spills can be controlled and the area decontaminated. Examples are:

a) edges between walls and floors are rounded

b) floors are covered with a special paint to prevent absorption of liquids

c) slopes are directed towards local hot drainage

d) the “hot drains” are connected with the system of radioactive wastes collection

Areas of possible contamination are provided with necessary equipment for contamination control. Measuring instruments are protected to avoid their contamination and the way instruments must be used is clearly indicated in the proximity of the equipment.

12.3.7 Post Accident Access Requirements

Re-entry to the facility after an accident is supported by the information provided by the PAM instrumentation. This instrumentation provides information through hardwired panels located in the MCR and ECC. As the habitability of the ECC is ensured through its dedicated ventilation system, the information to plan the re-entry to those areas relevant to control an abnormal scenario is always available.

The ARM PAM units located in the Reactor Hall are distributed in such a way that satisfy the re-entry needs since they cover from the RPO water surface up to the MCR window to the Reactor Hall.

End of Section

Table 12.3/1 Classification Levels for Radiation and Contamination Areas

Radiological area colour code	Potential radiation exposure level (individual, effective) $mSvy^{-1}$	Removable surface contamination levels (averaged over 2000 hy^{-1})	Potential airborne contamination levels (averaged over 2000 hy^{-1})
RED	6 to 20	0.3 to 1 Derived Surface Contamination Limit (at times potential levels up to hundreds of Derived Surface Contamination Limits in localised areas).	0.3 to 1 Derived Air Concentration (at times potential levels up to hundreds of Derived Air Concentrations–when respiratory protection is indicated-)
BLUE	1 to 6	0.05 to 0.3 Derived Surface Contamination Limit (at times potential levels up to tens of Derived Surface Contamination Limits in localised areas)	0.05 to 0.3 Derived Air Concentration (at times potential levels up to tens of Derived Air Concentrations)
WHITE	<1	<0.05 Derived Surface Contamination Limit	<0.05 Derived Air Concentration

Table 12.3/3 Service Pool Radial Dose Rates for Different Thickness of Heavy Concrete

Lateral	
Heavy concrete thickness (cm)	Dose rate (μSvh^{-1})
90	1629.0
105	175.8
120	16.0
135	1.6
150	0.1

Table 12.3/5 Service Pool Bottom Shielding Dose Rates for Different Thickness of Heavy Concrete

Down axial	
Heavy concrete thickness (cm)	Dose rate (μSvh^{-1})
75	5159.0
100	132.2
125	2.6
130	1

Table 12.3/6 Surface Dose Rate Variation with Water Height and Decay Time. Case I (One Irradiated Fuel Assembly)

Water Height over top of Fuel Assembly (cm)	Doses Rate (μSvh^{-1}) for a range of Decay Times					
	15 minutes	1 hour	1 day	5 days	33 days	68 days
238.5	17750.0	10510.0	1506.0	1165.0	266.2	52.1
258.5	7055.0	4097.0	529.5	410.8	93.1	17.3
278.5	2847.0	1619.0	189.7	146.8	32.7	6.0
298.5	1167.0	657.1	69.1	53.8	12.1	2.2
318.5	491.3	269.3	25.5	19.7	4.5	0.8
338.5	209.7	112.5	9.6	7.4	1.7	0.3
358.5	90.7	47.4	3.4	2.6	0.6	0.1
378.5	38.9	19.7	1.1	0.8	0.2	
398.5	17.4	8.5	0.4	0.3		
418.5	7.6	3.6	0.2	0.1		
438.5	3.3	1.5				
458.5	1.5	0.6				
478.5	0.7	0.3				
498.5	0.3					
518.5	0.1					

Table 12.3/7 Surface Dose Rate Variation with Water Height and Decay Time. Case II (Three Irradiated Fuel Assemblies)

Water Height over top of Fuel Assemblies (cm)	Doses Rate (μSvh^{-1}) for a range of Decay Times					
	15 minutes	1 hour	1 day	5 days	33 days	68 days
238.5	49140.0	29210.0	4224.0	3249.0	744.9	151.3
258.5	19470.0	11390.0	1492.0	1147.0	262.6	49.1
278.5	7863.0	4485.0	540.7	414.2	93.5	17.0
298.5	3256.0	1831.0	196.1	149.6	33.5	6.3
318.5	1357.0	746.2	71.9	56.4	12.6	2.3
338.5	581.9	310.5	26.7	20.4	4.7	0.9
358.5	252.5	131.2	9.8	7.4	1.7	0.3
378.5	108.4	55.1	3.2	2.4	0.5	0.1
398.5	48.6	23.8	1.2	0.9	0.2	
418.5	21.1	10.1	0.5	0.4		
438.5	9.1	4.1	0.2	0.1		
458.5	4.1	1.8				
478.5	1.8	0.7				
498.5	0.8	0.3				
518.5	0.4	0.1				
538.5	0.2					

Table 12.3/8 Surface Dose Rate Variation with Water Height and Decay Time. Case III (Rigs irradiated)

Water Height over top of Fuel Assemblies (cm)	Doses Rate (µSvh^{-1}) for a range of Decay Times			
	15 minutes	1 hour	5 hours	1 day
250	639.8	380.7	124.7	36.4
265	323.7	191.4	61.4	17.4
280	167.3	95.3	30.6	8.6
295	84.3	49.0	15.1	4.3
310	44.3	25.6	7.8	2.1
325	23.5	12.9	3.9	1.1
340	12.1	6.9	2.1	0.5
355	6.5	3.6	1.0	0.3
370	3.5	1.9	0.6	
385	1.8	1.0	0.3	
400	1.0	0.5		
415	0.6	0.3		
430	0.3			
445	0.2			

Table 12.3/9 Normal Concrete Design Values

Parameter	Value
Density	2,200 kgm^{-3}
f_{ck} (Characteristic cylinder compressive strength of the concrete at 28 days)	20.0 MPa
f_{cd} (design compressive concrete strength)	13.3 MPa
C^c (coefficient for thermal expansion)	1×10^{-5}°C^{-1}
ε_{c1} (strain at peak stress)	0.0020
ε_{cu} (ultimate concrete strain)	0.0035
∩ (Poisson's ratio) for elastic strains	0.2
j_c (partial safety factor for concrete)	1.5
Ecm (mean value of elasticity modulus)	29,000 MPa
Ecd (design elasticity modulus)	19,300 MPa
f_{ctm} (mean value of tensile concrete strength)	2.2 MPa
f_{ctd} (design tensile concrete strength)	1.46 MPa
f_b (bond stress)	3.45 MPa
f_{bd} (design bond stress)	2.30 MPa

Table 12.3/10 Normal Concrete Composition

Element	Atomic density (at/b.cm)
H	1.320×10^{-2}
O	4.380×10^{-2}
Na	9.220×10^{-4}
Ca	1.460×10^{-3}
Fe	3.320×10^{-4}
Si	1.590×10^{-2}
Al	1.670×10^{-3}
K	4.410×10^{-4}
C	1.100×10^{-4}

Table 12.3/11 Heavy Weight Concrete Composition

Heavy Concrete Constituent	Fraction weight
H	1.58E-03
C	7.94E-04
O	3.25E-01
Mg	2.28E-03
Al	1.64E-02
Si	4.22E-02
S	2.74E-03
Cl	1.11E-04
Ca	8.00E-02
Ti	7.63E-04
Mn	1.56E-03
Fe	0.51254
Cu	0.001103

Element	Atomic density (at/b.cm)
H	9.893×10^{-3}
O	4.564×10^{-2}
Mg	6.242×10^{-5}
Ca	1.178×10^{-3}
Fe	1.448×10^{-2}
Si	1.643×10^{-3}
Al	4.438×10^{-4}
S	2.002×10^{-5}
Ti	7.717×10^{-3}
C	9.280×10^{-6}

Table 12.3/12 Steel Design Values

Parameter	Value
Steel Grade	36/52 (ES-262-1974)
Surface configuration	Ribbed steel bars
Density	7,850 kgm^{-3}
f_{yk} (characteristic value of yield stress)	360 MPa
f_{tk} (characteristic value of tensile strength)	520 MPa
ε_u (elongation of reinforcement at maximum load)	> 0.14
j_S (partial safety factor for steel)	1.15
f_{yd} (Yield stress for design)	313 MPa
f_{td} (Tensile strength for design)	452 MPa
E_S (Modulus of elasticity)	200,000 MPa
C_S (Steel coefficient of thermal expansion)	$1x10^{-6}$°C^{-1}
Coefficient of thermal expansion	1.2 x 10^{-5}°C^{-1}

End of Tables

Figure 12.3/7 Reactor Pool Radial Dose Rate Variation as a Function of Distance from Core Axis

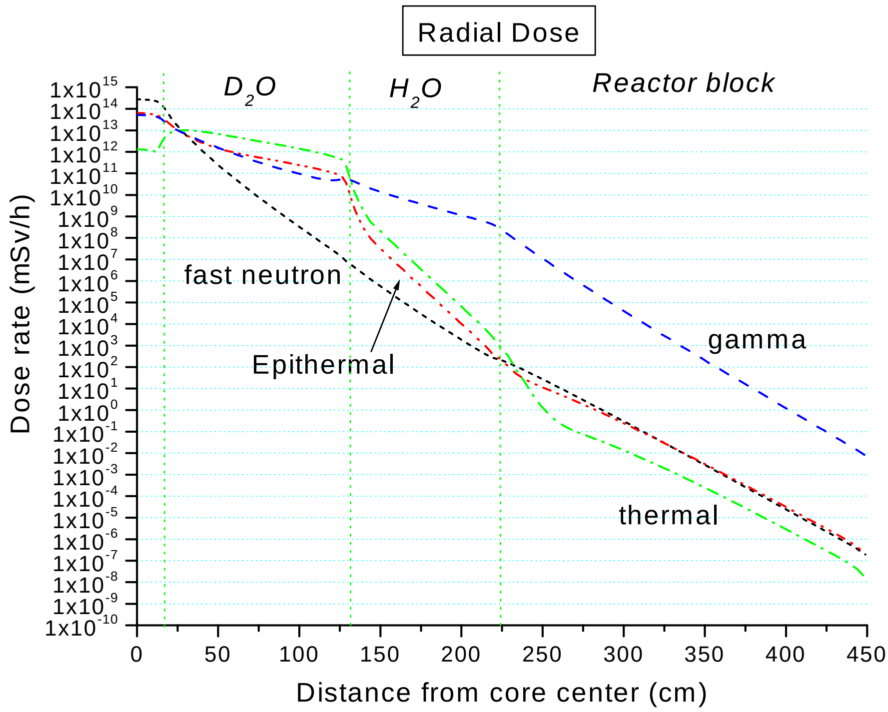
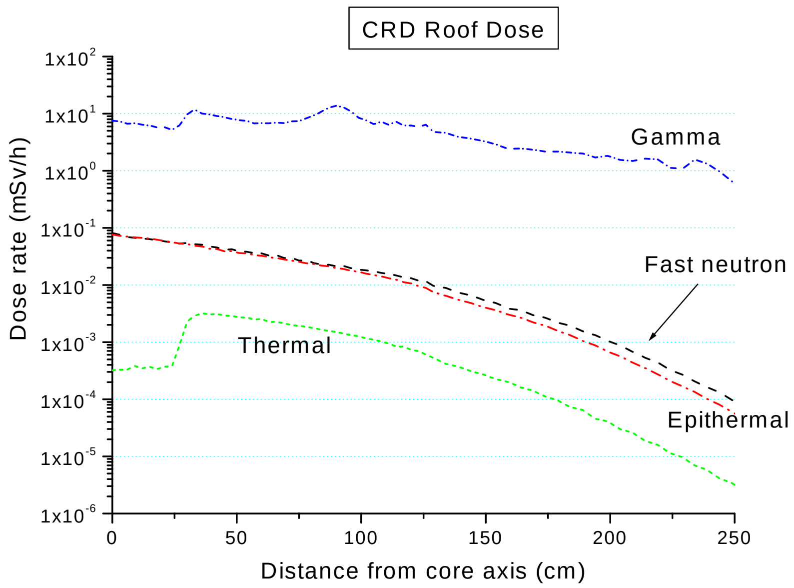


Figure 12.3/8 Dose Rate Versus Radial Position Near to the Ceiling Control Rod Drive Room



12.4 WASTE MANAGEMENT SYSTEM

12.4.1 Waste Management Principles

The waste management system for the Reactor Facility has been designed to guarantee the safety of the personnel involved in the activity and that of the general public and to minimise potential environmental impacts.

12.4.2 Monitoring, Control, Segregation and Classification

The Reactor Facility has the following means for monitoring, control, segregation and classification of radioactive and non-radioactive waste.

a) For monitoring:

(i) Two Liquid Effluent Monitors (LEM)

(ii) Two Waste Stream Monitor (WASMO)

(iii) One Secondary Water Activity Monitor (SAMO)

(iv) Two Active Liquid Monitors (ALMO)

(v) Two Failed Fuel Element Monitors (FFEM)

(vi) Chemical Control Unit

(vii) Portable monitors for beta and gamma radiation

(viii) Portable contamination monitors

(ix) Movable Tritium Monitor

(x) Environmental activity monitors

(xi) Monitoring of gases (AEM, TRM)

b) For sampling in:

(i) Laboratory active waste collector basins

(ii) General collector tanks

(iii) LOCA and Refill pools

(iv) Effluent collection pool of the demineralised water plant

(v) Process systems

c) For control:

(i) Procedures for administrative controls

(ii) Database for inventory control and tracking

(iii) Alarms and interlocks for the monitoring systems

d) For segregation and classification:

i. Separation of areas in accordance with the potential contamination hazard
ii. Drains, networks and vessels for liquid segregation
iii. Dedicated ventilation or removal systems for those areas where gases are produced
iv. Vessels and procedures for solid waste segregation
v. Areas dedicated to solid segregation and classification of wastes

The segregation and classification of the waste is performed in accordance with established procedures as described in ANSTO Safety Directive 5.7 Safe Management of Radioactive Waste and in the appropriate Waste Operations Quality documents.

12.4.3 Waste Generation

12.4.3.1 Reduction and Minimisation of the Generated Waste

Waste generation has been considered from the design stage through the appropriate selection of materials, taking into account all waste generation pathways and providing for waste management systems with all the necessary facilities.

Technical and administrative controls are utilised to reduce the volume and activity of the waste generated.

The start-up, operation and maintenance of all facility systems are in accordance with waste reduction principles, setting strict controls and procedures for operations within the reactor hall and minimising the generation of contaminated waste water.

Waste reduction principles are applied. These principles include segregation of wastes at source, limitation of areas where operations with radioactive materials are carried out, limitation of the contamination in each area, and recovery of items for reuse wherever practical.

For maintenance operations, the water from the pipes of the Primary Cooling System, pools and related system are stored for reuse.

Environmental monitors in those areas where radioactive gases are generated are provided. High-activity alarms on these monitors will initiate actions to minimise emissions.

12.4.3.2 Types of Generated Waste

12.4.3.2.1 Solid Waste

The possible sources of solid radioactive waste are:

a) irradiated target cans
b) irradiation rigs and used reactor components
c) spent control rods
d) ventilation system waste (filters, High Efficiency Particle Air (HEPA) filters and molecular sieve beds)
e) laboratory samples
f) cleaning materials and used personal protection items
g) spent ion exchange resins
h) contaminated processing items
i) cooling tower dry sludge (normally non-radioactive)
j) general (non-radioactive) waste

12.4.3.2.2 Liquid Waste

The possible sources of liquid radioactive waste are:

a) cooling water blowdown
b) secondary system drains
c) liquid waste from the demineralised water plant
d) ventilation water system drain
e) demineralised waste water recovered from the drainage of large equipment in maintenance operations
f) washbasin and shower liquids
g) floor drain liquids
h) non-active liquids from laboratories
i) LOCA liquids in the unlikely event of an accident
j) active liquids from laboratories and other areas

12.4.3.2.3 Gaseous Waste

Gaseous waste comprises airborne effluent from the Reactor Facility. These gases are produced by different sources:

a) gaseous radioactive elements or compounds from the pools, coolant systems, irradiation facilities and experimental facilities.
b) airborne radioactive elements produced in support facilities including fume cupboards in blue labs and decontamination areas.

12.4.3.3 Expected Generation of Waste

12.4.3.3.1 Waste Generated during Pre-commissioning and Commissioning

The estimated generation of solid and liquid waste and airborne emissions during pre-commissioning and commissioning has been provided below.

a) Solid waste
   (i) Low-level compacted solid waste: Approximately 6.5 m^3 (papers towels, tissue paper, rags, mops, plastic gloves, clothing, vials, pipettes, plastic tubing, etc).
   (ii) Low-level non-compactable solid waste: Approximately 1.5 m^3 (activated aluminium components and cans, activated solid samples, contaminated items from laboratories and reactor).
   (iii) Intermediate level non-compactable solid waste: Approximately one 72 litre container (activated metal components).
   (iv) Ion exchange resins: 8 litres (from the reflector cooling and purification system). The pool purification and hot water layer system produces an estimated 1100 kg of spent ion exchange resins during the commissioning.
b) Liquid waste
   (i) Trade (C-line): Approximately 27,800 m^3 (from trade drainage in controlled areas, beam hall building, auxiliary building and cooling tower blowdown, 25000 m^3 of waste water corresponding to this point)

(ii) Low-level liquid waste (B-line): Approximately 10 m^3 (from active liquid waste drainage).

(iii) Others: 400 litres of waste oil, less than 1 litre of tritiated water samples.

c) Airborne waste

(i) Argon-41: expected to be less than 1 TBq

(ii) Xenon and krypton: less than 0.2 TBq

(iii) Tritium: less than 4 GBq

12.4.3.3.2 Waste Generated in Operation

The estimated generation of solid and liquid waste and airborne emissions for routine operation has been provided below. The values are based on estimated on the anticipated utilisation of the facility. Generation estimated per year of operation:

a) Solid waste

(i) Low-level compactable solid waste: Approximately 5 m^3 prior to compaction (papers towels, tissue paper, rags, mops, plastic gloves, clothing, vials, pipettes, plastic tubing, etc).

(ii) Low-level non-compactable solid waste: Approximately 1 m^3 (activated aluminium components and cans, activated solid samples, contaminated items from laboratories and reactor).

(iii) Intermediate level non-compactable solid waste: Approximately one 72 litre aluminium bin (activated metal components).

(iv) Filters: Approximately 88 HEPA filters, 3 charcoal filters, and 4 molecular sieve beds per year, equivalent to about 10 m^3 of low-level waste

(v) Ion exchange resins from heavy water reflector cooling and purification system: 8 litres.

(vi) The ion exchange resin bed from the Reactor Water Purification and the Hot Water Layer Purification: approx 1100 kg. No spent resins will be produced during the first two years, resins being provisionally stored at the Process Room

b) Liquid waste

(i) Trade (C-line): Approximately 52800 m^3 (from trade drainage in controlled areas, beam hall building, auxiliary building and cooling tower blowdown, 50000 m^3 of waste water corresponding to this point).

(ii) Low-level liquid waste (B-line): Approximately 150 m^3 (from active liquid waste drainage).

(iii) Others: 400 litres of waste oil, tritiated samples below 1 litre.

c) Airborne waste

(i) Argon-41: approx 4 TBq per year

(ii) Xenon and krypton: 0.6 TBq per year

(iii) Tritium: 37 GBq per year

For more details of the postulated airborne emissions from operation of the Reactor Facility see Table 12.2/5.

12.4.3.4 Tracking Policy for Generated Wastes

All radioactive waste generated, treated, stored and transferred both within and outside the reactor building are tracked. All radioactive waste are identified by an associated record, which identifies the source of the waste, its type, volume and main radionuclides.

Wastes shipped from the Reactor Facility to other parts of ANSTO are in accordance with the ANSTO Safety Directive on Safe Movement and Transport of Radioactive Materials (SD 5.6).

Records are kept of both liquid waste arising in all tanks in the Reactor Facility and all transfers to ANSTO Waste Operations Liquid Waste Treatment Facility.

12.4.4 Solid Waste Management

12.4.4.1 General Description

Figure 12.4/1 provides a simplified diagram of the solid waste management system.

Solid waste is segregated at source and all necessary procedures and waste reduction measures are closely followed to minimise such waste.

Solid waste is segregated into:

a) Non-radioactive Solid

b) Low Level Solid Waste

c) Intermediate Level Solid Waste

Non-radioactive solids are processed through ANSTO's waste clearance system before being released off-site in accordance with ANSTO's waste management procedures.

Low activity solid wastes are classified and managed in accordance with ANSTO's waste management procedures. Low level solid waste is stored until collection by ANSTO's Waste Operations Section.

Consumable solid waste is sealed inside plastic lined fibreboard drums that are labelled to provide identification of the waste origin, the dose rate and the radioactive content before being transported to ANSTO approved storage areas. Compaction takes place in the Waste Management Section. All compacted low-level solid waste is stored in 200 litre drums at the LHSTC in a designated storage facility.

Intermediate level solid wastes is stored in the service pool where a shearing facility is available to cut large items into smaller sizes for more efficient storage. Long lived intermediate level solids are transferred to ANSTO Waste Operations in a shielded container for storage.

12.4.4.2 Specific Solid Waste Items

12.4.4.2.1 Irradiated Target Cans

Cans of different materials are irradiated in the pneumatic irradiation facilities at the Reactor Facility, namely: high-purity aluminium, high-purity aluminium with cadmium lining, high-density polythene and titanium. Titanium and cadmium lining cans are 100% recyclable and it is estimated that there will be 100 throughout the reactor's lifetime. Polythene cans are not reusable and become low-activity solid waste. Aluminium cans are reusable for a limited number of times, and undergo decontamination prior to reuse.

When necessary, irradiated cans are opened and decontaminated at the radioisotope production plant.

12.4.4.2.2 Discarded Irradiation Rigs and Components

There are 17 irradiation rigs.

The discarded thermocouples and other materials are stored in the Service Pool until they are removed in 72 litre aluminium bins using a shielded transfer flask.

In the case of damaged irradiation rigs, these are disassembled and transferred under water to the Service Pool. After decay, they are cut with the shearing tool and removed in the same container.

The Rig Safety Cap, Bottom Restriction Nose and Can Supporting Branch are expected to be replaced during the reactor lifetime. They are stored in the Service Pool until they are cut and removed in a 72 litre aluminium bin using the shielded flask.

12.4.4.2.3 Neutron Beam Guide Tubes

Neutron beam guide tubes are replaced every ten years and are expected to be intermediate level solid waste.

There is one heavy concrete storage area where neutron beam guide tubes are stored throughout the reactor lifetime. Neutron beam guide tubes are removed and transferred to the storage area by means of a specially designed mobile driven transfer cask.

12.4.4.2.4 Cold Neutron source

The “Vacuum Containment” and the “In-Pile Assembly” constitute the CNS (Cold Neutron Source) Thimble constructed with Aluminium & Stainless Steal (SS).

Aluminium is degraded by the radiation, so the in-pile should be replaced every 10 years approximately. Once removed, the spent components can be stored in the SPO for decay. Two sets of can be stored there allowing a decay time of 20 years before final disposal.

The aluminium section can be cut underwater (in the reduction volume reduction facility) and separated from the stainless steel part. These parts can be cut later into smaller sections to fit into 72 liters bin normally used in ANSTO for solid waste management.

Could be some Co-60 in the Al and SS component so it is expected to be intermediate level solid waste after 10 years decay.

The charcoal filters for purification of oil in the CNS circuits should be replaced every 3-4 years. No activation is anticipated and it is expected to be exempt level waste.

Mass expected 400 kg

12.4.4.3 Waste Facilities in the Service Pool

The Service Pool has installations that allow storing the following:

a) Spent fuel elements for 10 years (Chapter 10, Section 10.1)

b) Damaged rigs and replaced rigs for radioactive decay for one year.

c) Facilities for cutting and size reduction.

d) Sufficient room to store at least one 72 litre aluminium bin for the storage and removal of intermediate level solid waste using a shielded flask.

e) Removable platform for one fuel transport cask.

The Service Pool has a shearing tool for cutting and reducing the size of metal elements under water. The tool can be manually handled from platforms located above the Service Pool water surface.

Intermediate level solid waste are disposed into 72 litre aluminium bins that can be transported in a shielded flask from the Service Pool storage to ANSTO's retrievable waste storage facility. The 72 litre aluminium bins are currently used for storage of similar wastes from HIFAR operations.

The amount of intermediate level waste generated in normal operation during one year is expected to fit in one 72 litre bin.

12.4.4.4 Spent Control Rods

All control rods (1 central and 4 lateral control rods) are replaced approximately every 10 years. The control rods contain mainly hafnium as component element, with some trace impurities.

When the rods are spent, they are removed and stored in the Reactor Pool in positions designed to contain them.

When activity levels have decreased to acceptable levels for the transport flask, the control rods are cut in the Service Pool with the shearing tool and placed in the 72 litre aluminium bin for transfer to ANSTO's retrievable waste storage facility in Building 27.

12.4.4.5 Hot Cell

No waste production is expected in the hot cells in normal operation because they are only used for transferring cans and not for opening them.

Nevertheless, a table is provided with an opening through which low level solid waste can be placed into a plastic container, which can be removed into a shielded transfer vessel.

It is possible to transfer materials of certain dimensions from any cell to the Loading cell, either through the Inter Hot Cell Elevator or through the Transfer Pipe. The loading cell allows access for a low-shielded container or the bottom loading flask (10 Tonne), should it be required to remove intermediate level solid waste.

12.4.4.6 Ventilation System Waste

Spent High Efficiency Particle Air (HEPA) filters are used to minimise the emission of airborne radioactivity. The filters are routinely replaced and the spent filters contain low levels of radioactivity and are classified as low-level waste.

Spent active charcoal filters are mainly produced in radioisotope production plants. The activated charcoal filters for iodine retention consist of frames of the same dimensions as HEPA filters, built in stainless steel which can be reused.
Table 12.4/1 lists the annual quantities of solid waste generated from the ventilation system.

12.4.4.7 Solids and Sludge from Ion Exchange Filters

Small quantities of fine solids and sludge accumulate on the ion exchange beds. A very low production of ion exchanger resins back-washing sludge is expected in normal reactor operation.

The largest production that may be expected is during the first reactor start-up. The ion exchange resins used to purify the pool water and the hot water layer tend to accumulate solid waste during the first hydraulic tests. After a short period, the resins are back-washed and the accumulated solid waste is separated. This waste is not active because the first hydraulic test is carried out before the first start-up of the reactor.

In normal reactor operation, the sludge produced in ion exchange resins in the cooling circuit are partly discharged as solids in suspension into lines B or C depending on radioactive content. The solids are periodically removed during scheduled reactor shutdowns. This sludge is low-level solid waste.

12.4.4.8 Laboratory Waste

Waste receptacles (e.g. for paper filters, gloves, tissue paper, disposable pipettes, test tubes and similar materials) are provided in the laboratories. Radioactive waste is segregated in different categories and managed in accordance with ANSTO Radioactive Waste Management Policy, Safety Directives and established procedures.

12.4.4.9 Cleaning Materials and Used Personal Protective Items

The quantities of these materials are expected to be similar to those generated by HIFAR. This is low-activity waste that is put in labelled receptacles and transferred to ANSTO Waste Operations for management.

12.4.4.10 Spent Resins

Ion Exchange Resins are used to maintain water quality in the reactor systems. Once the resins have exhausted their exchange capacity, they must be replaced.

There are two tanks for storage of spent resin from the Reactor Water Purification system and the Reactor Hot Water Layer. These tanks allow the radioactivity in the resins to decay before being transferred from the Reactor Facility. The availability of two tanks allows the segregation of generated waste as the operations require. The tanks are biologically shielded by concrete walls and ceiling.

The unloading of the columns is performed through water circulation from whichever storage tank is available. Water flowing from the Spent Resin Handling and Storage System conveys the spent resins to the tank. This circulation is constantly carried out by a centrifugal pump until the entire load of spent resin, together with the sludge, is retained in the tank to decay. For this purpose, the control logic sets the actuated valves to allow flow from the desired column to the selected storage tank. All actuated valves are provided with limit switches and alarms to indicate their proper operating position.

The same description also applies for discharging the resins after decay. Spent resins are transferred to movable shielded tanks) for shipment to the Waste Operations Section for storage and conditioning.

The spent resin discharge area is normally white only, for contamination and radiation. It is estimated that the discharge operation will be carried out once a year at the most, the area being temporarily reclassified as blue for the transfer operation.

The accounting of spent resin production is carried out by means of a register of tank discharges.

It is expected that the resins will contain the nuclides shown in Table 12.4/2, similar to those already produced at HIFAR. The short-lived radionuclides will decay to negligible levels in the spent resin storage tanks.

The Reactor Facility is also provided with resin-handling systems that handle spent resins coming from heavy water purification and cooling, core cooling and hot water layer systems.

The resin beds of the Reflector Cooling and Purification system are removable containers. The system has two beds, so that, when replacement is required, one of the beds is in operation while the other is left to decay. After adequate time for decay, the bed can be removed and taken to the dry storage room for another year.

12.4.5 Liquid Waste Management

12.4.5.1 Introduction

The Reactor Facility is designed to minimise the production of liquid waste. Collection systems allow for the segregation of liquid waste according to the radioactivity level and for monitoring, and temporary storage of liquid wastes.

Most radioactive liquid waste generated at the Reactor Facility is low-level waste acceptable for discharge through the low-activity line (B-line) for treatment by the ANSTO Waste Operations Section.

The general management strategies for liquid waste are presented in Figure 12.4/4.

Sources of liquid waste from the Reactor Facility are as follows:

a) cooling water blow down

b) secondary system drains

c) liquid waste from the de-mineralised water plant

d) ventilation water system drain

e) de-mineralised water recovered from cleaning operations

f) wash basin and shower liquids

g) floor drain liquids

h) non-radioactive liquids from laboratories

i) LOCA liquids in the unlikely event of an accident

j) radioactive liquids from laboratories and other areas

These liquids, that may contain radioactive or chemical contamination, are managed through LHSTC B or C lines.

Heavy water leaks are unlikely in view of the design of the heavy water system. Effluents containing heavy water, which are likely to contain tritium, are collected separately from other liquids, stored and separately transferred to ANSTO Waste Operations. The expected volume during normal operation is small.

A waste stream monitor (WASMO) is installed in each of the temporary liquid waste storage tanks (see Figure 12.4/5). This system monitors the activity of the liquid arriving at the tank. This recording is automatic and the signal is transmitted to a centralised data system along with the date, time and activity. The system allows detection of any discharge exceeding the admissible limits of that line (see Section 12.4.5.3.7) and allow the operator to take prompt corrective action.

12.4.5.2 Radioactive Liquid Waste Quantification

The total trade liquid waste discharge is in the order of 53,000 m^3/year. This volume include low level radioactive liquid waste (B-line), non-radioactive liquid waste (C-line)_ and secondary cooling tower blowdown. The largest volume is secondary cooling tower blowdown (~50000 m^3/year), which is not expected to contain any radioactivity.

Secondary cooling water blowdown is delivered directly to the ANSTO Liquid Waste Treatment Plant Holding tanks via a new dedicated delivery line. Additionally, a backup system utilising a new collection tank and the existing C line is also used.

12.4.5.3 Description of the Collection Network System

The Radioactive Liquid Waste Management System manages all active or potentially-active liquid effluents, drainage water, LOCA drainage water or make-up water generated in the pools or anywhere else within the facility.

The basic concept serving the design of the system is the separation of liquid effluents to be disposed of into three streams, namely:

a) Potentially-radioactive liquid waste (C-line)

b) Low-level liquid waste (B-line)

c) Sewage (to municipal waste collection network)

Additionally, there are two collection and liquid retaining networks:

a) Demineralised water recovery from the drainage of large equipment in maintenance operations.

b) Spilled water in the event of a LOCA.

The system was designed so that, under normal operation conditions, the two networks operate independently from each other.

Figure 12.4/5 shows the main collection-decay tanks for both lines, the basic interconnection and the monitoring, detection and sampling system.

12.4.5.3.1 Normally non-radioactive Liquid Waste (C-line)

The liquid waste from these drains is usually non-radioactive. However, since they are within controlled areas, these liquids may contain some radioactivity. Hence, they are drained through an individual network and collected at the Trade Liquid Waste Collection Tank.

This line collects effluents with activities less than or equal to those allowed by the C-line activity criteria (see Section 12.4.5.3.7) for delivery to the LHSTC liquid treatment plant.

Floor drains have stainless steel inlet grates without a liquid trap and are connected to the collection network. Drains located within process areas are usually closed and may be opened when required.

This network collects the following drains:

a) drains from equipment with water quality compatible with C-line admissible values

b) drainage from fan coils

c) drainage from floor (manually opened when required)

d) drainage from hand washing

e) emergency showers drainage

12.4.5.3.2 Collection of Low Level Liquid Waste (B-line)

This line collects liquids with activities greater than those collected by the C-line and with activities less or equal than the limits for the B-line at the LHSTC. These include:

a) decontamination equipment drains
b) laboratories drains
c) back washing filters discharge
d) pools water collected in process equipment drains
e) collection of leaks from pools

Equipment for low level liquid waste includes:

a) two stainless steel centrifugal pumps with magnetic coupling (one on stand-by)
b) pipelines and valves for homogenisation, sampling and discharge
c) level transmitter with high and very high level indication and alarm
d) automatic low-level pump switch off due to low level
e) flow integrator to measure volumetric discharge to the B line.

Both collecting tanks (B and C) have an on-line gamma activity detector WASMO (Waste Stream Monitor) for the inlet streams, keeping a record of time and discharged activity.

Another on-line gamma activity detector LEM (Liquid Effluent Monitor) is located in both tanks recycling line, sensing, recording and sending a control signal for the opening or closing of the discharge valves to Lucas Heights, depending on the permissible activity levels for each line (see Figure 12.4/5).

12.4.5.3.3 Collection of Reactor Pool Water

During maintenance periods, coolant water may be temporarily discharged into the Refilling Pool. The surface of this pool is coated with Epoxy resin and is kept clean at all times so as not to degrade the quality of the water to be returned to the system.

The network is provided with two pumps (one on stand-by), each capable of handling 100% of the design flow, to return demineralised water from the pool to the Reactor Coolant Purification System. These pumps also feed the Emergency Make-up Water System.

Since this water is recycled to the Reactor and Service Pool Cooling System, it is not liquid waste.

12.4.5.3.4 Loss of Coolant Accident Drainage

The emergency system for collection of water due to a LOCA is located inside the containment.

The system has been designed on the basis of minimising the effects caused by flooding of the area. Calculations have been made so that the worst case accident, causing a loss of water inventory up to the siphon-breaker level of the Primary Cooling System, provokes a maximum flooding of 100 mm measured at floor level in the process room and a maximum of 150 mm in the pump area.

The process area is divided by plinths of at least 100 mm in order to avoid spilled water emerging from minor leaks to flood the entire process area.

Water is conducted to a LOCA Pool. In case this volume is not enough, an overflow to the Refilling Pool increases the liquid holding capacity. The normal practice is to keep the pools at their lowest possible level. The water collected in a LOCA emergency are sent to RSPCPS for recycling. This water is not liquid waste since it is not discharged.

Liquid waste generated through leaks from those systems with potentially active water is minimised through a detection system.

The pools structures and external walls provide sufficient shielding. Both pools are vented directly to the process area ventilation system.

12.4.5.3.5 Floor Drainage System

Drainage is classified by area and type according to Table 12.4/3.

12.4.5.3.6 Heavy Water Room

The heavy water room system is designed to contain any leak of heavy water.

The floor of the area is designed as a pool, collecting the entire volume of heavy water in case of an accidental liquid spill.

Floor drains connected to two drainage pipelines lead the spilled liquid towards the heavy water storage tank.

During operation, the Reflector Vessel is kept at such a pressure that potential leaks will lead to entry of light water into the reflector system.

Sampling of heavy water is carried out with a specially designed system that prevents leaks in either liquid or gaseous form. This system has a vacuum pump capable of condensing and freezing the heavy water contained in the gaseous phase, returning it to the storage tank.

12.4.5.3.7 Criteria for Discharge to ANSTO's Liquid Waste Treatment Plant

The criteria for discharge to the B and C lines are specified in the Safe Management of Radioactive Waste Safety Directive. Under this directive there are guidelines for the limits of radioactivity that can be discharged to the B and C lines, and the provision for higher levels of activity to be discharged with written approval. The current guidelines in the Safety Directive for the release of liquid waste to the B-line are:

Beta activity (including tritium) 1 MBq per day

Alpha activity 50 kBq per day

The quantities of radioactivity that can be discharged via the C-line are limited to one-tenth of those listed above for the B-line.

Normally non-radioactive liquid wastes (for discharge to C-Line) and low level liquid waste (for discharge to B-line) effluents streams are measured on-line and independently by dedicated the Waste Streams Monitor (WASMO) gamma detectors at the inlet to their corresponding waste storage tanks. If liquids that can potentially exceed the allowed limits are detected, they can be managed by the system valves manifold and then handled off-line.

Additionally, the dedicated Liquid Effluent Monitor (LEM) detectors measure the activity of the liquids discharged from the two waste storage tanks to the LHSTC B and C lines (low level radioactive liquid waste and normally non-radioactive liquid waste respectively), towards ANSTO liquid waste treatment facility. The LEM is part of the RMS and supports the waste discharge commands of the Radioactive Liquid Waste Management System (RLWMS), thereby allowing waste accountability. The LEM triggers local and remote (RCMS) alarms signals in case that allowed limits are being approached or exceeded. It also triggers the closure of the waste connection pipelines if values are higher than the maximum specified limit.

The B and C lines go to ANSTO's liquid waste treatment facility operated by Waste Operations and Technology Development. After treatment, the water is held in holding tanks and analysed for radioactivity and other contaminants.

After meeting the discharge criteria set out in the Trade Waste Agreement with Sydney Water, the water is discharged to the sewer.

12.4.5.4 Cold Neutron Source

It is expected that, after 10 years of continuous operation, 7.4×10^{11} Bq of tritium is present in the deuterium circuit. To handle this deuterium, a connection with the heavy water recombiner has been provided. The involved masses are such that, after recombination, this activity will be contained in around 80 litres of heavy water.

12.4.6 Gaseous Waste Management

12.4.6.1 Introduction

Areas where gaseous radioactive waste is generated during normal operation have absolute filtration and where applicable activated charcoal absorption filters in the ventilation extract from them.

A simplified diagram of the gaseous waste collection and discharge system is shown in Figure 12.4/8.

Gases are mainly collected in three areas, namely:

a) Reactor Hall and Process Room (containment)

b) Heavy Water Room

c) Other controlled areas

The Heavy Water Room and the Above Pool Hot Cells Complex have their own monitoring and ventilation systems, which also vent into the stack.

Other controlled areas are separated into two sectors: the glove box with its own tritium monitoring and all other controlled areas outside the confinement.

The Air Effluent Monitoring System allows online measurement of aerosols, tritium, noble gases and iodine concentrations in the Reactor Facility stack, as well as gamma spectrometry of the effluents. If limiting concentrations in the containment air are exceeded the containment isolation is triggered by the First Reactor Protection System. This system is Post Accident Monitoring-conformed.

Area and stack monitors provide follow-up monitoring of the generated airborne waste. Figure 12.4/9 is a schematic of the gaseous effluent monitoring system. The TRM monitors the Stack, heavy water room and chemistry lab.

12.4.6.2 Gaseous Waste Quantification

The following is the airborne waste produced by the reactor:

a) Argon-41 and other noble gases (krypton and xenon isotopes)

b) Tritium and tritiated water vapour

c) Airborne particles (aerosols

d) Iodine

Table 12.2/5 provides an estimate of the annual release from the Reactor Facility Stack.

Argon41 is produced from dissolved Argon 40 in the pool water, which is activated according to the following reaction:

$$^{40}Ar\ (n,\gamma)\ ^{41}Ar$$

Small quantities of radioactive iodine and noble gases are produced as fission products in the Reactor Pool Water due to small amounts of uranium surface contamination on the cladding of the fuel plates.

Tritium is generated through the activation of the deuterium present in the heavy water, both in the reflector and in the cold source. The activation reaction is:

$$^{2}H\ (n,\gamma)\ ^{3}H$$

Tritium generation from the Reactor Facility is lower than HIFAR because the reflector heavy water system remains sealed during all normal reactor states.

12.4.6.3 Gaseous Waste System Description

The reactor is divided into several zones according to their potential for contaminated air hazards. Those rooms with potential for gaseous waste generation have provisions for air renewal and purification through filters. The airflow is directed from lower contamination risk areas to higher contamination risk areas.

In rooms where the access of personnel is not necessary during operation, the air is confined or the rooms are submitted to independent localised treatment, e.g. heavy water room.

The ventilation systems of those controlled areas outside the confinement (e.g. blue laboratories) have HEPA filters and charcoal beds or demisters prior to discharge through the stack.

The exhaust gas removal system collects exhaust gases from the vacuum pumps and fume cupboards and directs them to the stack.

Air ducts are airtight, welded or with flanged joints, built in epoxy-painted carbon steel or stainless steel sheet metal, or stainless steel commercial pipes.

Airborne radioactivity produced in areas within the containment is extracted by the ventilation system by means of the exhaust fans. The exhaust air passes through HEPA filters that retain particles larger than 0.3 microns with a 99.95% efficiency, and is released through the stack. If the Air Effluents Monitor (AEM) system, detects the presence of radioactive iodine in the stack, the gas is treated after circulating through the filters with a charcoal adsorber impregnated with potassium iodide to adsorb the iodine.

Area radiation monitors measuring gamma radiation are located in radiological representative points of the Reactor Facility. Some of the detectors are Post Accident Monitoring-conformed. The system provides a local alarm if the set point is exceeded.

There is a tritium monitoring with fixed detector to constantly measure the tritium concentration in the Heavy Water Room and trigger alarms if limits are exceeded. The system is complemented with portable equipment.

In order to reduce the production of Argon 41, the target cooling and transfer system is a closed system that uses nitrogen both for pneumatic transport as well as for target cooling. Moreover, target loading and unloading devices have nitrogen sweeping to reduce air entry to the system

The Argon 41 produced in the reactor arises mainly from the activation of the argon dissolved in the water circulating through the core. This is segregated from contact with the containment atmosphere by the hot water layer.

It is estimated that the emission of Argon 41 will not exceed 4.23 TBq per year. The air effluent monitoring system has high-activity alarms and on-line registration of emitted activity.

12.4.6.4 Tritium and Heavy Water Room

The facilities for the Reflector Cooling Systems and the tritium-handling glove box, including the area, the equipment and the operating conditions, are consistent with the application of the ALARA principle.

During operation, the Reflector Vessel is kept at a lower pressure so that any leaks would flow as light water into the reflector system rather than heavy water into the reactor pool.

The Reflector Cooling and Purification circuit has welded joints and was subjected to helium leak tests prior to commissioning. It is cooled through a closed intermediate circuit, which in turn is cooled by the Secondary Cooling System.

It has a vacuum pump and cold trap to remove water and dry the tank and pipelines prior to repair work. The water gathered in the cold traps is transferred to the storage tank.

The deuterium generated through radiolysis is reduced through gas cover re-circulation by a catalytic re-combiner, turning the deuterium into heavy water to be returned to the system.

The heavy water area is airtight, operates at a pressure below that of the surrounding areas and has an area tritium monitor.

The Cold Neutron Source moderator system is designed under high-gas-tightness requirements. A casing or double barrier with helium and nitrogen blanketing allow detection of deuterium leaks.

The test glove box has a dedicated system for gas treatment connected to the building general ventilation system.

The area dedicated to the Reflector Cooling and Purification System, designed to avoid tritium release into the surroundings, has a dedicated ventilation system that retains the tritium and the air from the ventilation system of the containment.

Under normal operation the area vents towards the stack only the equivalent to the air mass flow leak of the Heavy Water Room, while tritium presence and activity in the air within the area (Heavy Water Room) is constantly monitored through a Tritium Monitor, which reports to the Reactor Control and Monitoring System.

The ventilation circuit is periodically closed, forming a closed circuit that circulates the air to retain tritium present in the air. Tritium as water is partly retained in the condensation system and in the rest of the filters. Tritium concentration in the Heavy Water Room varies in saw-tooth form after a year, while its release through the stack is limited by the number of purification cycles at closed circuit in one year.

Spent molecular sieves are treated as low-level short-lived solid waste (in accordance with the IAEA-SS-111-G-1.1 classification) and sent to ANSTO's Waste Operations Section.

The highest specific activity by tritium expected on the depleted molecular sieves is of 3.7 MBq/g when the specific activity of the heavy water is of 370 $GBqkg^{-1}$.

The ventilation systems handling tritium in the facility are designed to constrain the tritium levels to below 155 $GBqy^{-1}$. The estimated annual emission of tritium of 37 $GBqy^{-1}$ is based on projected operation and maintenance schedules of the heavy water systems. The actual value will be determined once operational experience has been gained over a number of years.

These levels are much lower than current tritium emissions from HIFAR (see ANSTO/E740).

12.4.7 Waste Management References

1. ANSTO/Sydney Water Trade Waste Agreement, 1998
2. Replacement Nuclear Research Reactor- Draft Environmental Impact Statement- ANSTO & PPK.
3. APOL2.2 – ANSTO's Radioactive Waste Management Policy

Safety Directive 5.7 – Safe Management of Radioactive Waste

4. Safety Directive 5.2 – ANSTO Policy on As Low As Reasonably Achievable
5. Safety Directive 5.6 – Safe Management and Transportation of Radioactive Materials
6. ANSTO/LA-01 Rev2 Application to ARPANSA for Facility Licence – Section 4B – Waste Operations and Technology Development, September 2000.
7. ANSTO/E740 Hoffman E L., Loosz T and Mokhber-Shahin L, Environmental and Effluent Monitoring at ANSTO Sites, 1999.

End of Section

Table 12.4/1 Expected Amount of Solid Waste Production from Spent Ventilation Filters per year

Filter type	System	Average Quantity	Expected frequency
HEPA	Ventilation system	86	43 every 6 months
HEPA	Glove box gaseous treatment system	2	(*)
Charcoal bed	Ventilation system	3.5	7 every 2 years
Molecular Sieve Bed	Heavy water room ventilation system	2	> 1 year
Molecular Sieve Bed Charcoal Absorber	Inner Neutron Guide Helium Cooling System	1	> 1 year
Molecular Sieve Bed	Glove box gaseous treatment system	1	(*)

(*)- The replacement frequency for the HEPA filter and the molecular sieve beds depends on operations.

Table 12.4/2 Radionuclides Expected on Ion Exchange Resins

Nuclide	T ½	Gamma Emission (E Mev/Yield)
Probable isotopes		
Na^{24}	15.1 hours	2.75 (100), 1.36 (100)
Mg^{27}	9.45 minutes	1.01 (28), 0.84 (72)
Tc^{99m}	6.04 hours	0.14 (89)
Cr^{51}	27.8 days	0.32 (10)
Mo99	66.0 hours	0.14 (4), 0.74 (13), 0.36 (1.4)
Mn^{56}	2.58 hours	2.66 (0.7), 2.52 (1), 2.11 (14.3), 1.81 (27), 0.84 (99)
Less probable isotopes		
Sb^{122}	2.7 days	0.69 (3.7), 1.14 (0.8), 0.55 (71), 1.28 (0.8)
Sb^{124}	60.2 days	1.69 (49), 0.60 (98)
Fe^{59}	44.6 days	1.29 (43). 1.10 (56), 0.19 (3)
Mn^{54}	312.2 days	0.83 (100)
Ce^{141}	32.5 days	0.15 (69)
Ce^{144}	284.2 days	0.13 (19), 0.08 (4.5)
Cs^{134}	2.06 years	1.37 (3), 0.80 (85.4), 0.60 (98), 0.57 (15)
Zn^{65}	243.8 days	1.12 (51)
Ni^{65}	2.52 hours	1.48 (28), 1.12 (9.8)
Zr^{95}	64.4 days	0.76 (54.7). 0.72 (44.3)
Co^{60}	5.27 years	1.33 (100), 1.17 (100)
Cs^{137}	30.1 years	0.66 (90)
Nb^{95}	35.5 minutes	0.77 (100)
Cs^{138}	32.2 minutes	2.64 (8.1), 1.44 (76), 1.01 (30),
I^{131}	8.04 days	0.36 (83)

Table 12.4/3 Drainage Classification by Area and Type

Area	Drainage						
	Floor Drain	Fan Coil Drain	Wash-basin Drain	Shower Drain	Equip-ment Drain	Active Drain	LOCA Collect-ing Drain Hole
Emergency Make-up Water System	1				1		
Cell Filter Room	1						
Plant Room	1						
Rig Maintenance Workshop	1		1			1	
APHCC Area	1		1	1			
Reactor Hall	4				1		
Personnel Decontamination			1	1			
Pneumatic Cell						3	
Mechanical Plant Room					1		
Instrument Room	1						
Valve Room	3						
Mechanical Plant Room	1						
NAA Office	1		1			1	
NAA Irradiation Lab	1			1		2	
Plant Room		1					
Silicon Neutron Transmutation Doped	1		1			1	
Chemistry Lab White	1		2			1	
Chemistry Lab Blue	1		1	1		1	
Waste Delivery	1		1				
Object Decontamination	2					1	
Shower	1		1	1			
Chemistry Ventilation		1					
Contamination Control Area	1		2				
Chemistry Store	1		1			1	
Loading/Unloading	1		1				
Cleaner	1		1				
Hot Cell Support	1		1				
Active Workshop	1		1			1	
Loading Hot Cell						1	
Cell Operation	1						

Area	Drainage						
	Floor Drain	Fan Coil Drain	Wash-basin Drain	Shower Drain	Equip-ment Drain	Active Drain	LOCA Collect-ing Drain Hole
Process Room	2						2
Decay Tank Room	1				1		
Manifold: Hot Water Layer Purification System 2220 Reactor Coolant Water and Purification System 2000 Spent Resin Handling and Storage System 5210 (Process Room)	1				2		1
Manifold Reactor Pool Hot Water Layer System 2210 (Process Room)	1				1		2
Reactor & Service Pool Cooling	1				1		1
Piping Connection System 1110	1				1		1
Core Cooling Pump Room	1				3		3
Fuel Cladding Failure Detector	1					1	1
Electrical Switchboards	1						
Control Rod Drives Room					1		
Pump Maintenance	2		1			1	
Lobby	1		1	1			
Irradiation Facilities Service	1						
Airlock	1		1	1			
Resins	2						
Active Equipment Handling Storage	1						

End of Tables

Figure 12.4/1 Diagram of the Solid Waste Management Strategy

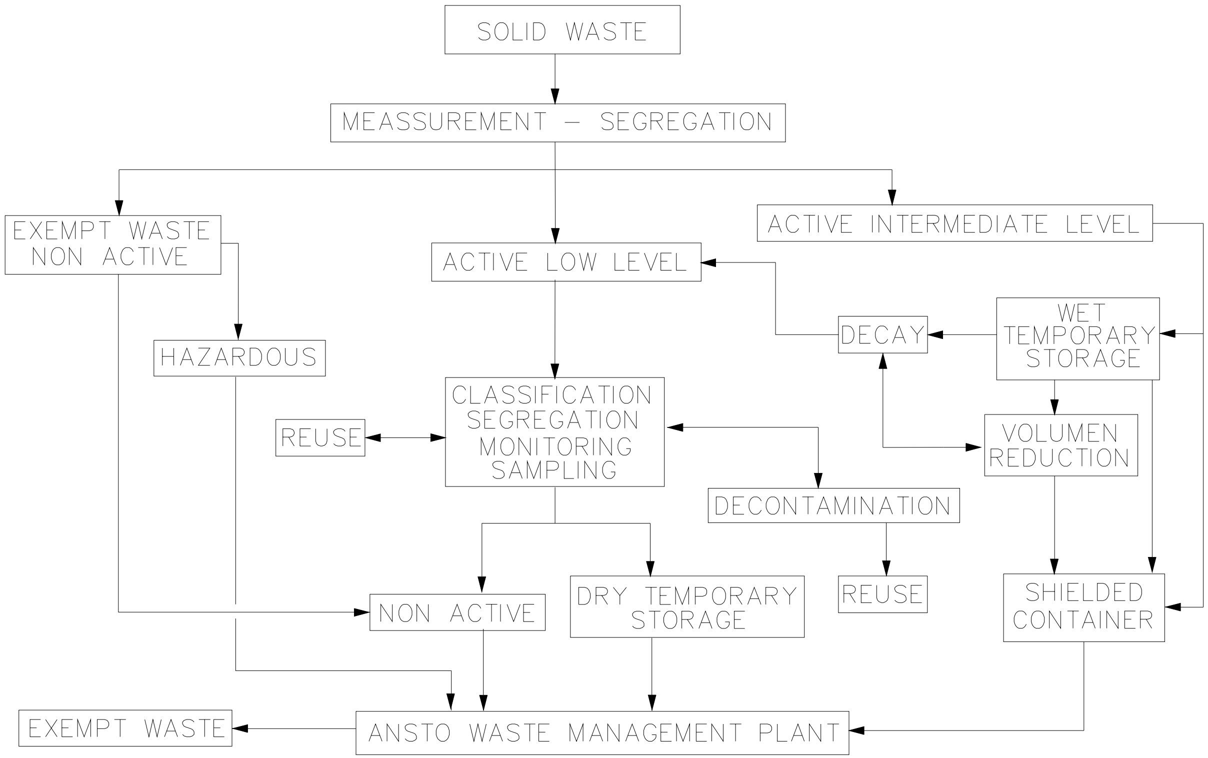


Figure 12.4/4 General Management Strategies for Liquid Waste

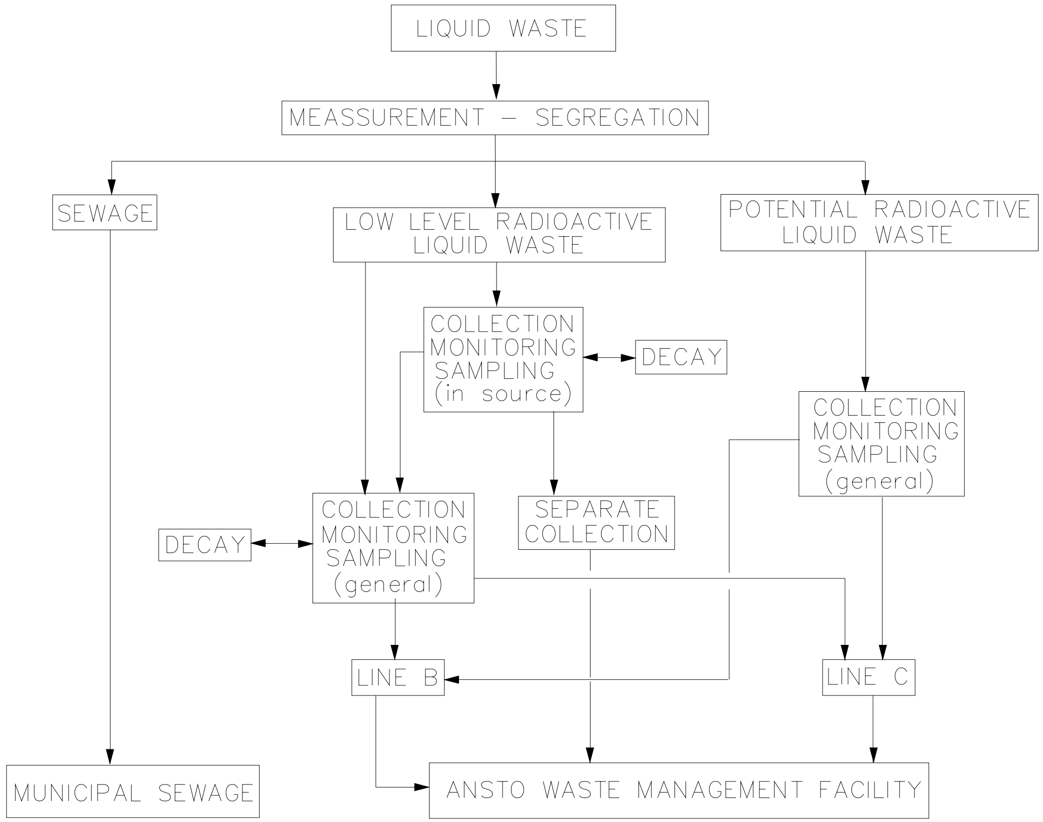


Figure 12.4/5 Radioactive Liquid Waste Simplified Diagram

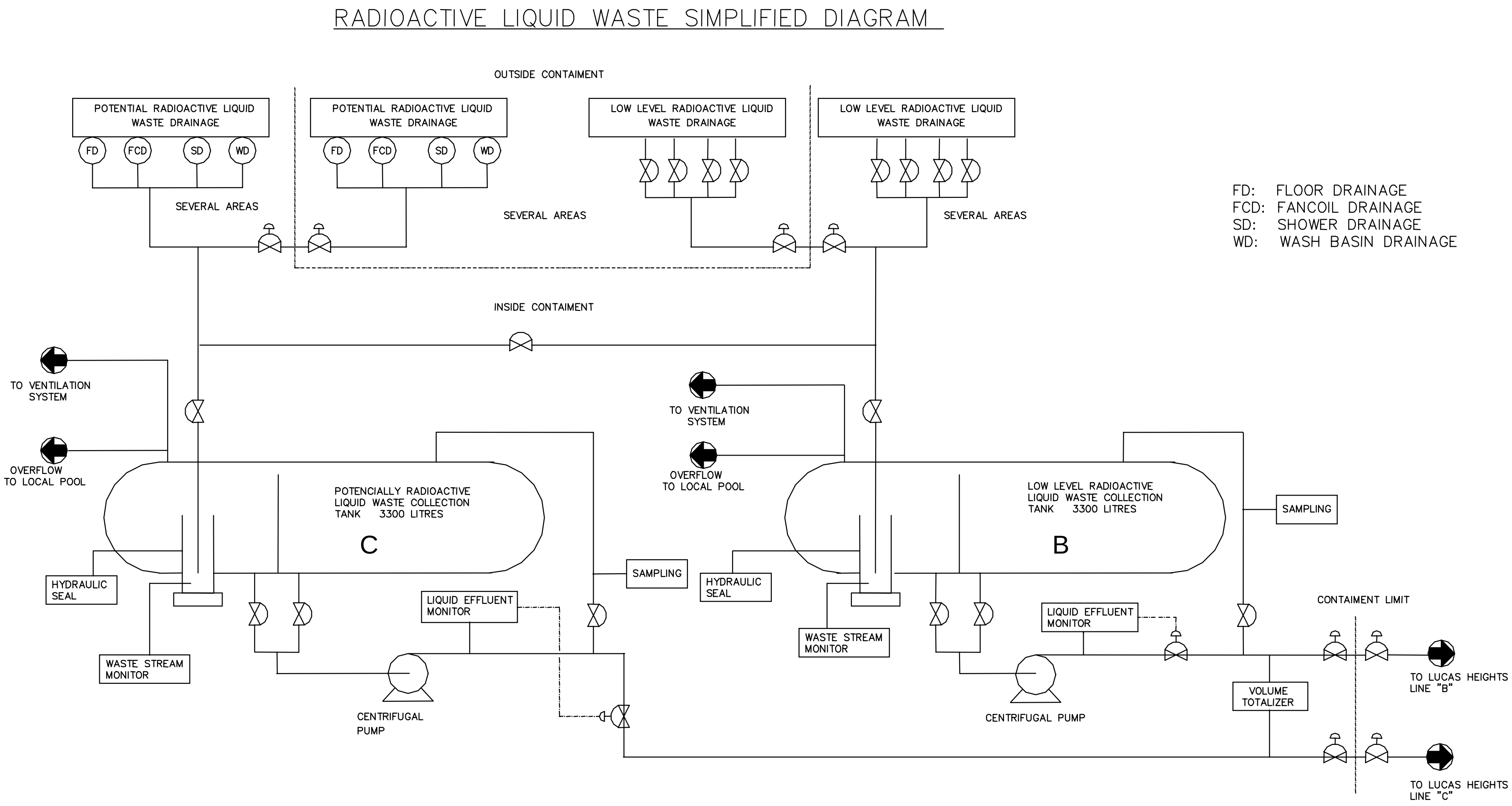


Figure 12.4/8 Airborne Waste Diagram

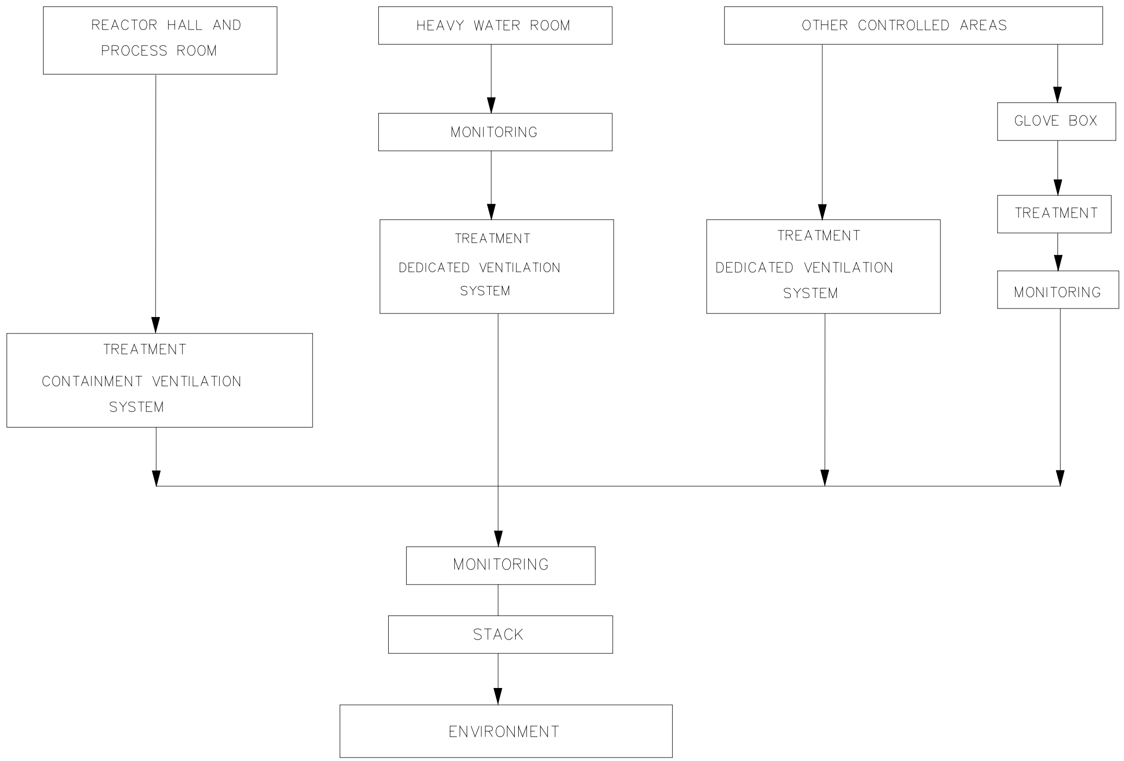
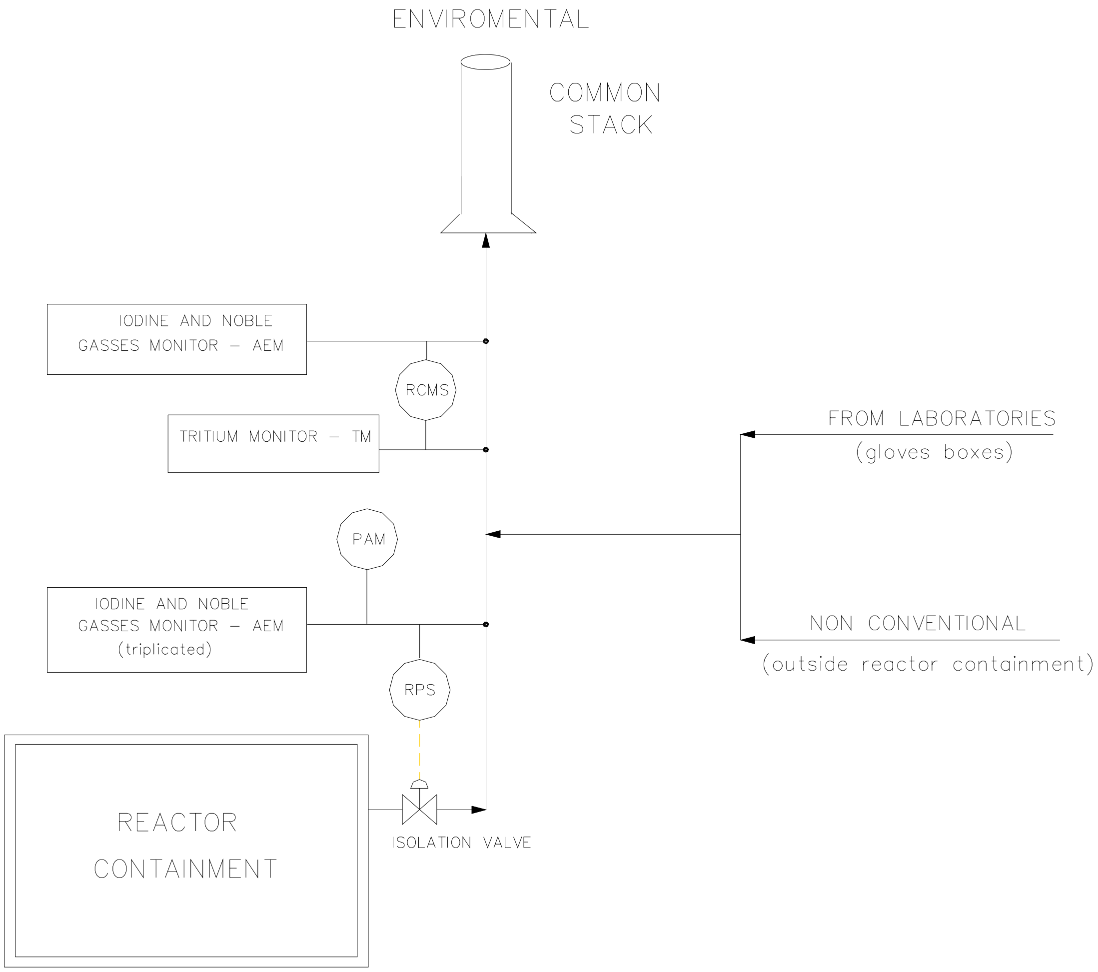


Figure 12.4/9 Concept of Gaseous Effluent Monitoring through Stack



End of Figures

12.5 DOSE ESTIMATES FOR NORMAL OPERATION

In the PSAR, the estimated annual doses for the staff were evaluated with the information available at that stage of the design. Therefore, representative dose rates for the different areas were assigned and the number of staff members and the time spent in those areas were preliminarily estimates. The estimations of doses to staff for this version of the SAR have used more detailed dose maps that have been modelled using the available data combined with a more detailed knowledge of the tasks to be performed by different worker groups in the plant.

Although this is an improvement in the evaluation of the estimated annual collective dose and individual average doses, they will not be precisely determined until operational experience is obtained. The actual doses received by staff as a result of the Reactor Facility operations will not be accurately known until data has been collected from:

- Radiological surveys throughout the facility
- Area Radiation Monitoring
- Personal dosimetry (TLD's and EPD's)
- Utilisation and frequency of tasks performed, including the actual occupancy of areas

Therefore, until the above operational data is available, the estimated doses received during the routine operation of the Reactor Facility shall be evaluated using preliminary dose rate maps and the estimated occupancy factors / utilisation frequency derived from the proposed operational activities.

The estimated doses received from radioisotope production and NTD silicon operations have been assessed using estimates, based on the estimated utilisation of these production tasks. The actual doses will be dependent on production demand, measured dose rates in operational areas and the occupancy of those areas.

The estimated doses produced by the activities associated with Research Activities (eg beam utilisation) fall outside the scope of this assessment. These activities are performed by a different group of workers arise from a different practice which is designed to a much lower dose-rate criteria that is much closer to background in these locations. This results in the estimated individual doses being less than the 2mSv/y ALARA objective.

For the calculations it is assumed that the reactor will operate in cycles of 33 days with ten of these cycles in a normal year. Maintenance of equipment could be performed during the cycle except in cases where a significant reduction in the hazards is achieved if the reactor is in Shutdown State.

It is anticipated that every ten to twelve years a major overhaul will be performed where complex maintenance activities involving relevant doses will be performed. These activities will be planned and performed under a close radiological survey ensuring that doses will be ALARA. Nevertheless preliminary assessments have been performed and will be used in the present evaluation.

Seven groups of activities are foreseen to produce doses on the Reactor Facility operation staff. These are:

1. Daily plant walkthrough with the reactor at power.
2. Operation of pieces of equipment in radiation areas with the reactor operating at

full power.

3. Fuel Assemblies management for core refuelling.
4. Pool Top Production Operations
5. Pneumatic and Loading Hot Cell Operations
6. Maintenance activities performed with the reactor operating at full power.
7. Maintenance activities performed during refuelling or major overhaul.

It is possible that additional activities will be identified as operational experience is gained. Any actual exposures associated with them are will be quantified and assessed at that time. These activities may result in an increase in the dose estimates performed in this version of the SAR. Assessments of any additional activities may need to be performed to ensure that ALARA is maintained.

For the activities included into group one, the proposed pathway of checks inside the containment areas, the average dose rate and the time spent are assessed. One walkthrough per day is envisaged totalling 330 walkthroughs at Power in a normal year. Just the walkthrough involving rooms inside the containment is considered since no significant doses produced by the reactor are expected outside the containment areas.

For the group two, the dose rate in the area where the components should be operated, the time spent in every operation and the number of operations per year is assessed.

In the group three, the movement of three FAs at the end of each cycle is foreseen as well as other manoeuvres associated with major maintenance tasks.

The group four encompasses the activities performed from the Operation Bridge in relation with the movement of irradiation targets (BIF and LVIF).

Group five complements the previous one considering the doses produced by the utilisation of Hot Cells in the management of radioactive materials.

As has been mentioned, some maintenance activities can be performed with the reactor at power. The dose rate in the area where these activities are performed, the averaged time spent and the approximated occurrences per year are assessed. These activities are accounted into the group six

Major maintenance involves planned activities and exposures that have been determined for the following activities and are considered in group seven:

1. Neutron guide in pile replacement
2. Cold Neutron Source in pile replacement
3. Control Rods replacement

The following sections detail and assess the collective doses associated with these seven groups of activities.

12.5.1 Doses from Reactor Operations

12.5.1.1 Daily plant walk-through with the reactor at power

The following paragraphs summarise the activities that should be routinely performed in radiation areas during the daily walkthrough, while at Power. The capabilities embedded into the RCMS and the adequate instrumentation of key processes has been also considered in the operational plans for the facility reducing greatly the field surveillance requirements.

Walk-throughs performed with the reactor in shutdown state produce marginal doses if compared with the ones made in operation.

Table 12.5/1 details the calculation of the dose associated with every step of the daily walkthrough in the Basement

The time and dose involved are detailed in the table 12.5/2. A total value lower than 1 µSv was obtained.

The total estimated dose involved in the daily walkthrough is less than 15 µSv. For 330 operational days, the resulting estimated annual collective dose is less than 5 man-mSv. It should be noted that two thirds of this dose arise from the in service inspection of the PCS pumps. The instrumentation included in the PCS main pumps is so complex that it is still under consideration if this in service visual inspection is required. The requirement for this visual inspection will be determined once operational experience and radiological survey data has been performed. Hence a reduction in doses received during the walkthough will be achieved if this task is not required.

12.5.1.2 Operations to be performed during the operation cycle in Radiation Areas

Some systems require actions to be performed in the field with the reactor operating at power. These operations are:

1. Discharge of Liquid Waste Storage Tanks: This operation includes the valves line-up, tank homogenisation, sampling and the LEM operation. The activities take approximately one hour per tank. The number of discharges per year from each tank has been conservatively assumed as 50.
2. Set up of Heavy Water Ventilation: some modifications in the valve line-up can be performed with the reactor in operation inside the Heavy Water Ventilation Room. The time taken to perform this manoeuvre is approximately 4 minutes and is estimated to be performed only once per year.

The change of RWPS train (duty to standby) is not included in this group since the behaviour (efficiency) of the train is a parameter that evolves slowly in time. Therefore, the change to the stand by unit could be performed during the 2 days shutdown between operation cycles. The same applies for some other systems.

The other systems can be operated remotely from the RCMS reducing in such a way the need for operations in radiation classified areas.

The dose rate in the area of the Liquid Waste Storage Tanks can be calculated assuming a concentration of 2500 Bq/l of ^{24}Na and 2500 Bq/l of ^{137}Cs in the B tank and neglecting the C tank contribution. In this condition the dose rate calculated is 0.19 µSv/h, therefore in the 1 hour per discharge per tank a dose of 0.38 µSv is delivered for both tanks. The transit through classified areas up to the tanks is calculated from the assessments made for a walkthrough. Considering 50 discharges of both tanks per year, the annual collective dose for this activity is less than 0.04 man-mSv.

The set up of Heavy Water ventilation involves the operator receiving similar doses. Therefore, for this group of activities, a total annual collective dose of less than 40 man-µSv can be assigned.

12.5.1.3 Fuel Assembly Management

The estimated doses attributable to the movement of fresh FAs is negligible if compared with the ones produced during the handling of irradiated fuel. Moreover, the most significant doses appear during the transportation of irradiated fuels from the RPO storage rack to the SPO assigned position through the transfer channel.

The FAs being moved from the RPO to the SPO are the ones removed in the previous cycle, therefore a 33 day decay is considered. From the data available in Table 12.3/6 and assuming 1 hour for each operation, these manoeuvres produce a total dose lower than 0.03 mSv (0.01 mSv per FA). This operation is performed 10 times in a year during the planned shutdown for refuelling. Therefore the annual collective dose from fuel assembly management is 0.30 man-mSv.

The estimated doses attributable to the manoeuvres performed in the CRD Room during refuelling are negligible when the reactor is shutdown.

Manoeuvres to store the spent FAs in the transport cask for further conditioning is performed underwater delivering marginal doses with a frequency of 10 years. Refer to section 12.5.5 for doses from fuel management during major maintenance overhauls.

12.5.2 Doses from Production Activities

Two locations have been selected to consider the doses associated with production activities, the pool top area, where production activities are carried out from the Operation Bridge and the pneumatic and loading hot cells.

12.5.2.1 Pool Top Production Operations

The loading and unloading of products into the Bulk Irradiation Facilities (BIF) and the Large Volume Irradiation Facilities (LVIF) are performed from the Operation Bridge. It is not anticipated that there will be any change in the ambient dose rates above the RPO during these operations hence the doses received by staff are directly related to:

a) the dose rate on top of the operation bridge,

b) the time it takes to perform the activities, and

c) the frequency that these operations are performed.

For the purpose of these estimations the following assumptions have been made:

a) a dose rate of 8 μSv/h

b) each change takes 20 minutes

c) 1 staff member on the operations bridge at a time

d) the rigs in all 17 locations are changed once every 7 days

e) the current ^{99}Mo ‘rocket cans’ are used and not the proposed LEU foils.

Using the above assumptions, the total number of rig load/unloads is estimated to be ~80 per program. The dose to a staff member on top of the operations bridge is calculated be ~2.6 μSv for each operation and the collective dose for a program will be 0.213 man-mSv and 2.1 man-mSv for a year.

The flux in the LVIFs is approximately 3 times that in the vertical graphite facilities in HIFAR, which means that the Reactor Facility has the capacity to triple the production of silicon. HIFAR currently averages approximately 120 silicon facility irradiations per program, hence the Reactor Facility capability is assumed to be approximately 360 irradiations in the LVIFs per program.

For the purpose of these estimations the following assumptions have been made:

a) A dose rate of 8 μSv/h

b) Each change takes 30 minutes

c) 1 staff member on the operations bridge at a time

d) The Reactor Facility LVIF irradiations for NTD silicon is running at a capacity of 360 irradiations per program

Using the above assumptions, staff members spend approximately 180 hours on the Operation Bridge per program loading/unloading the LVIFs. At 8 μSv/h this results in a collective dose of ~1.44 man-mSv per program and 14.4 man-mSv per year.

The annual collective doses for pool top production operations will approximately 16.5 man-mSv. This estimated dose is extremely dependant on the utilisation of the production facilities of the Reactor Facility and the number of staff present on the operations bridge during the operations.

12.5.2.2 Hot Cell Production Operations

The pneumatic and transfer hot cells located in the Reactor Hall and the Loading hot cell also form part of the production facilities that are utilised in the Reactor Facility.

Dose rate modelling has been performed to provide estimates of the maximum dose rates that may be detected on the outside of the pneumatic, transfer and loading hot cells. The maximum estimated dose rates for the front wall of the hot cells are:

Pneumatic hot cell:	3.0 μSv/h
Transfer hot cell:	7.0 μSv/h
Loading hot cell:	7.4 μSv/h

For the purpose of these calculations the following assumptions have been used:

a) 330 days of at power operation of the reactor facility

b) An occupancy factor for the use of the cells of 0.3 over a 24 hour period

c) The maximum dose rate of 7.4 μSv/h has been assumed for all cells

Using these assumptions the total time spent working in front of the hot cells is 2376 hours per year. From the above assumptions it has been estimated that the collective annual dose from hot cell operations is 17.6 man-mSv.

The actual value of this estimation can vary significantly as the pneumatic and transfer hot cells are used for the vast majority of operations. The actual collective dose depends on the actual dose rates on the outside of the cells, the activity of radioactive materials in the cells and the utilisation of the cells. This will be determined from operational experience.

12.5.3 Doses from Maintenance Activities

12.5.3.1 Routine Maintenance Activities at Power

During the normal operation of the facility certain failures can be produced, provisions were taken to limit the need of intervention with the reactor operating at significant power levels. These provisions include:

1. Stand by units in relevant systems

2. Automation of processes
3. Operative features that simplify equipment operation (eg check valves that allow startup of redundant trains with discharge valves open in the duty unit).

On the other hand, the layout facilitates the execution of maintenance activities and reduces the contribution from other sources through the use of suitable shields.

The location of instruments and the remote sensing through the RCMS reduce dramatically the doses associated with surveillance activities, therefore the assessment for this group of activities is limited to the following corrective maintenance that can be accomplish with the reactor at power in radiological areas:

Isolation of faulty components

Isolation and replacement of I&C sensors and modules

Tighten fittings

Lamp replacements

Dismantling of components

a) Isolation of faulty components

The data of all the components located in the plant are loaded into a database. From this database, less than 90 components that can fail requesting a service isolation with the reactor at Power have been identified in the basement. Not all of these components are permanently working during the cycle (some of them are stand by units) or require a special isolation from the field but some conservative assumptions can be developed using this number.

The hypothesis that these 90 components can fail once per year and that every isolation procedure takes approximately 20 minutes as an average produce an occupancy of 1800 minutes (30 hours) per year.

The average dose can be derived from the values used in the assessments for the walkthrough considering that the process corridors are the commonly accessed areas. Therefore 20 µSv/h times 30 h produce an annual dose of 600 µSv.

b) Isolation and replacement of I&C sensors and modules

From the database mentioned above, the number of sensors, transducers, indicators, etc located in the basementrounds 600 units. These components are passive components (electronic) with a very low failure rate. Additionally, the safety related variables are measured using triplicate sensors with “fail to safe” features reducing in this way the need of immediate replacement. Therefore, it is assumed that this kind of tasks is performed less than 100 times per year averaging 15 minutes each, ie the total time spent is 1500 minutes (25 hours).

The average dose rate assumed is similar than in the previous case (20 µSv/h) and the annual dose resulting from this activity is 500 µSv.

c) Fittings tighten

Supporting services are connected to the systems through fittings that can present leaks during operation. To keep the reactor available, some fittings shall be tightened in operation. It has been assumed that once per cycle eight hours are used for this activity resulting in 80 hours per year.

The average dose rate assumed is the same used in the previous two cases (20 µSv/h) and the annual dose resulting from this activity is 1.6 mSv.

d) Lamps replacement

There are approximately 270 lamps that can be replaced with the reactor in service. Not all of them require a replacement with the reactor in operation (eg those located into access restricted areas).

It has been conservatively assumed that one third of them shall be replaced per year (considering the typical life of bulbs) with the reactor in operation consuming 5 minutes per replacement and averaging a dose rate of 20 µSv/h. Therefore, the annual dose associated with this activity is 150 µSv.

e) Dismantling of components

The PCS Main pumps are the components that can be disassembled with the reactor in service producing the highest dose.

The dose rate in the room where the pump not running is located is around 40 µSv/h. The time for disassemble the pump is around 4 hours. It is not expected that these pumps shall be disassembled more than one time per year. Therefore, the annual dose is 160 µSv.

12.5.3.2 Routine Maintenance Activities during refuelling

Maintenance activities during refuelling will produce marginal doses if compared with the ones produced during major overhauls or with the reactor in operation. The appropriated material selection ensures that the radiation fields will drop to background values shortly after the reactor reaches the shutdown status.

Just two operations justify a further analysis, the change over of resins and the handling of tritiated heavy water.

In the first case, the transport method selected to drive the resins from the storage tank to the transport flask and the possibility to allow one year of decay before discharge, ensure a negligible dose over the involved staff. Other precautions taken in the system like the appropriated routing of pipelines or the use of remotely actuated valves also support this statement.

Nevertheless, a preliminary assessment can be done where a peak of 20 µSv/h is calculated for the process area corridor close to the active resins train. Considering conservatively that an operator is there during four hours during the transport of the resins to the transport flask, and that this operation is performed once per year, the annual dose associated is 80 µSv.

For the case of the heavy water, the whole system is located inside a dedicated room whose access is only allowed after a radiological survey is performed. Portable tritium monitoring units are used during manoeuvres that can produce a release, warning the operators. The access to this room is performed through a dedicated Airlock where precautions have been taken to reduce the possible hazards (eg breathable air supply, PPE, emergency shower, contamination check point, etc).

A preliminary assessment considers that eight hours employed per year in an environment with 0.1 DAC of tritium (higher concentrations in air require the utilisation of PPE). With this value, the dose committed is 8µSv.

12.5.4 Summary of doses for Normal Operation

The estimated annual collective and average doses for operation of the Reactor Facility have been summarised in Table 12.5/3. The estimations are all based on the assumptions and calculations performed within this section. An additional assumption on staffing has been made to calculate the estimated annual individual dose for reactor operation and production activities. The actual individual and collective doses of staff will not be accurately known until operational data is available through radiological surveys, individual and area monitoring, and experience gained on the utilisation and operation of tasks.

The Reactor Operators and Utilisation staffs share the operational and production tasks between the two groups. For this reason they have been grouped together as one worker category for the purpose of these estimations with a total of 20 workers sharing these tasks between them. The estimated collective dose from the identified activities has been estimated at 39.4 man-mSv with an estimated average individual dose to these staff of 1.97 mSv, which is less than the ALARA objective. The actual breakdown of doses between these two groups will be very dependent on the frequency that members of the groups do certain tasks, especially production related tasks.

The estimated annual collective dose from identified Maintenance tasks that are performed during routine operation in the Power and Shutdown states is 3.1 man-mSv. The manning of maintenance staff is approximately 10, resulting in an estimated average individual dose of 0.31 mSv per year. This figure can change dependant on any additional maintenance tasks that need to be performed, the radiological exposures of the tasks and the frequency required.

12.5.5 Activities during major overhauls

Major overhauls of the reactor facility are required approximately every ten years, when the Control Rods (CR), Cold Neutron Source (CNS) and the in-pile neutron guide shutters (NG) all will need to be replaced. The actual timing of such overhauls depends on utilisation and depletion/degradation of the respective components.

At this time, 16 FAs will be moved from the RPO to the SPO but just 13 will returned later accounting 29 transports through the transfer canal in total. This manoeuvre will produce less than 290 µSv since some FAs have lower burn-up and longer decay.

On the other hand, the maintenance activities performed during major overhauls and involving radiological hazards identified up to now are:

a) CNS in pile assembly replacement.

b) NG in pile replacement (including shutters removal/reinstallation)

c) Control Rod replacement.

The dose associated with the replacement of the CNS and NG in-piles are calculated in the documents describing the manoeuvres:

CNS: estimated from 57 to 111 man-mSv. (RRRP-0057-3BEIN-037)

NG: approximately 49 man-mSv. (RRRP-0057-3BEIN-066)

For the case of the CR plates replacement, the activity and isotopes considered at the time of the manoeuvre are:

a) 1.31E+05 TBq de Hf^{181}

b) 2.44E+04 TBq de Ta^{182}

The CR plates remain underwater during the whole manoeuvre since the core and the storage rack are located in the Reactor Pool. The operation is performed from the Operation Bridge keeping always more than 250 cm of water between the operator position and the plate. In this condition, the dose rate calculated is less than 5 µSv/h.

Considering that one hour is enough to remove the 5 CR plates, the total dose involved is 5 µSv for this operation performed every ten years.

Using the conservative assumptions and models used in these calculations the collective dose for the ten-yearly major overhaul is approximately 308 man-mSv. All planning and preparation for major overhauls will be subject to detailed review and consultation with the Radiation Protection Adviser and optimisation will be employed wherever possible. Where further optimisation is not possible then methods such as dose sharing shall be employed to ensure that no staff member exceeds the annual dose constraints for the organisation.

12.5.6 Dose in Experimental Related Areas

The Reactor Beam Hall is permanently monitored by the Area Radiation Monitoring System gamma and neutron detectors. Nevertheless, portable instruments to monitor for gamma and neutron doses are used every time that an experimental device is assembled/disassembled in the area. In such a way, dose rates applicable to a radiation white area are guaranteed over routine operation of the facility. If a beam is not in use, the respective shutter is closed and its position locked and verified during the start-up procedure in the applicable form (checklist).

Reactor Facility staff may be provided with personal dosimeters following assessment of potential exposure. Dosimetry used takes into consideration that the expected neutron spectra may vary in the experimental areas.

The doses in the Neutron Guide Hall are expected to be quite similar to the background values as the experimental devices are assembled and used appropriately. Radiation surveys and review of staff dose records are performed on a routine basis as determined by the Radiation Protection Advisor in consultation with the Reactor Facility staff.

Doses due to external radiation in the Neutron Activation Analysis (NAA) and Chemistry Blue labs are, in normal operative conditions (i.e. following respective procedures and requirements), in agreement with their classification as radiation blue. Detectors of the Area Radiation Monitoring System are located in both labs. Regarding contamination hazards, appropriate glove boxes and fume cupboards have been located in those labs.

12.5.7 Dose Estimates for the Public

Calculations of dose in populations have been carried out using PC-CREAM (Consequences of Releases to the Environment Assessment Methodology. NRPB, June 1997).

The source term (i.e. the activities releases per year) assumed are the values tabulated in Table 12.2/5, which were obtained by maximising values calculated and extrapolated from measurements in other similar facilities.

Data collected between 05-04-91 and 22-11-00 at the Lucas Heights station from 10m above ground level considered representative of a typical Reactor Facility operation year has been used in the calculations performed with PC_CREAM.

The categorisation of atmospheric conditions assumed by PC-CREAM is given in Table 12.5/4.

The Pasquill atmospheric dispersion stability categories used is determined on the basis of USEPA (1987 - Report N° EPA-450/4-87-013) with data from 10 m above ground level. This type of data suits the format to be used in PC-CREAM assessments. The Table 12.5/5 shows the frequency of Pasquill categories.

At LHSTC, the locally experienced mean wind speeds are usually lower than the PC-CREAM defaults, so this code would tend to overestimate the mixing and therefore underestimate the dose received. This issue was addressed by applying a correction factor to the dose of 1.5 (Macnab, 1998; Fayers, NRPB 1998). Results presented here are obtained after multiplying the code output by this factor.

In order to allow an easy comparison with the present LHSTC doses reports, the location of relevant receptor points are the same as employed in “Estimates of Doses from Routine Airborne Emissions of Radionuclides from LHSTC during 1997 and 1998” (ANSTO ref. SD/SR/TN 99-11 rev 0). In such a way the polar co-ordinates of receptor points as well as the Reactor Facility stack position showed are referred to HIFAR stack.

The release height of Reactor Facility stack is relative to HIFAR is 175 metres at a bearing of 285° .

Default PC-Cream data libraries are used except where indicated. Some relevant parameters are shown in further paragraphs.

The pathways used to calculate the dose are shown in the Table 12.5/6. Consumption pathways are ignored for receptors closer than 1.6 km since this is the extension of LHSTC buffer zone.

The ingestion data used is shown in the Table 12.5/7 for adult consumers. The scaled ingestion rate is computed automatically by PC-Cream as the product of the ingestion rate and the fraction of ingested food grown locally for each food type. These data came from Agriculture Australia, 1994-95, Australia Bureau of Statistic Catalogue No. 7113.0.

The occupancy factors have been maximised to obtain conservative results. Occupancies lower than a full year are justified as follows:

a) 2607 hours represent the working time spent in the corresponding areas

b) 3910 hours are the number of hours while the Waste Service Facility is open

c) 1042 h represent 10 hours per weekend during every weekend of the year

Table 12.5/8 shows the assumed values.

The assumed integration time is 50 years and the breathing rate for adults is 8,500 m^3y^{-1}.

Table 12.5/9 shows the assumed deposition rate and washout coefficient.

Results of PC-CREAM Calculations (Table 12.5/10) show that the closest points with the higher dose are located in the North direction (in agreement with ANSTO reports SD/SR/TN 99-11 rev 0 and SD/SR/TN 98-7 rev 1). The nearest resident (1704 meters 82°) receives lower doses than this point located in the boundary of the present buffer zone.

The obtained value is well below the applicable limits (see Section 12.1.1.2) and below the design objective (see Section 2.2.1.2). The result has been obtained under conservative assumptions as can be seen, for example, in the occupancy factors, which assume a person would be outdoors all the time neglecting the shielding provided by buildings.

The values presented for the Reactor Facility are a few percent of the values reported as being produced by HIFAR and the LHSTC facilities in ANSTO reports. Reduction in the discharges are the main reason, since the relative position between receptor points and stacks (HIFAR and the Reactor Facility) are quite the same. A difference in the assumed distribution of meteorological stability categories exists (meteorological data used for

HIFAR and other LHSTC facilities are the actual meteorological conditions for the period of release, whereas this assessment uses a long-term average) but does not significantly affect the results.

The isotope and pathway contributions for the maximum calculated dose are shown in the Table12.5/11.

Pathways Contributions to Dose in Location N at 1.6 km and Nuclide Contributions to Dose in Location N at 1.6 km are shown in Figures 12.5/7 and 12.5/8.

Argon 41 is the most significant nuclide, with almost 95% of the total dose (delivered through the cloud gamma pathway). A variation in the discharge data of this nuclide will represent a significant modification in the total dose. The assigned value was the biggest value obtained from extrapolated measurements in other facilities and from calculations performed via a suitable model as has been mentioned. That means that it is almost certain that the release will be lower than that used in calculations if the operational conditions are kept within design range. The foreseeable releases of Ar^{41} are more than 30 times lower than the present HIFAR reported values.

The ventilation systems handling tritium in the Reactor Facility are designed to constrain the tritium levels to below 155 $GBqy^{-1}$. The estimated annual emission of tritium adopted for the calculation (37 $GBqy^{-1}$) is based on projected operation and maintenance schedules of the heavy water systems. The actual values of tritium released will be determined on the basis of operational experience. It should be noted, however, that releases up to 155 $GBqy^{-1}$ would increase doses to members of the public by less than 1% (see Table 12.5/11).

Finally it should be noted that the calculated doses are less than 0.1 μSvy^{-1}, well within the Reactor Facility design objective of <5 μSvy^{-1}.

End of Section

Table 12.5/1 Doses associated with daily walkthrough in Basement A

Position	Dose rate [µSv/h]	Time	Dose [µSv]
1	≈ 0	0	0.00
2	8	2 min + 30 sec	0.33
3	8	10 sec	0.02
4	8	2 min + 15 sec	0.30
5	8	2 min	0.27
6	8	10 sec	0.02
7	≈ 0	0	0.00
TOTAL		< 8 min	< 1 µSv

Table 12.5/2 Doses associated with daily walkthrough Reactor Hall B

Position	Dose rate [µSv/h]	Time	Dose [µSv]
1	≈ 0	0	0.00
2	8	2 min + 30 sec	0.33
3	8	10 sec	0.02
4	8	2 min + 15 sec	0.30
5	8	2 min	0.27
6	8	10 sec	0.02
7	≈ 0	0	0.00
TOTAL		< 8 min	< 1 µSv

Table 12.5/3 Estimated Annual collective and average doses for staff during normal operations and major overhauls.

Reactor Operation and Production Activities	
Group of activities	Estimated Annual Dose
Daily plant walkthrough	5.0 man-mSv
At Power Operations	0.04 man-mSv
Fuel Assemblie Movements	0.3 man-mSv
Open Pool Production	16.5 man-mSv
Hot Cell Production	17.6 man-mSv
Collective Dose	**39.4 man-mSv**
Average Individual Annual Dose	**1.97 mSv**
Maintenance Activities	
Group of activities	Estimated Annual Dose
Maintenance Activities at Power	3.0 man-mSv
Maintenance Activities during Refuelling	0.1 man-mSv
Collective Dose	**3.1 man-mSv**
Average Individual Annual Dose	**0.31 mSv**
Major Overhaul Activities	
Activities	Collective Dose for a Major Overhaul
Refuelling for Major Overhaul	0.3 man-mSv
Control Rod Replacement	0.005 man-mSv
Cold Neutron Source Replacement (RRRP-0057-3BEIN-037)	~57 to 111 man-mSv
In-Pile neutron Shutter Replacements (RRRP-0057-3BEIN-066)	197 man-mSv
Collective Dose	**~308 man-mSv**

Table 12.5/4 Wind Speed and Depth Mixing Layer Values Used by PC-CREAM

Stability category	Typical Wind Speed at 10 meters [ms^{-1}]	Typical mixing layer depth [m]	Rain
A	1.0	1300	No
B	2.0	900	No
C	5.0	850	No
D	5.0	800	No
E	3.0	400	No
F	2.0	100	No
C rain	5.0	850	Yes
D rain	5.0	800	Yes

Table 12.5/5 Stability Category Distribution

	A	B	C	D	E	F	C (rain)	D (rain)
NNE	1.29×10^{-3}	9.71×10^{-4}	3.87×10^{-3}	4.42×10^{-2}	2.32×10^{-2}	5.61×10^{-3}	1.88×10^{-4}	2.58×10^{-3}
NE	1.44×10^{-3}	9.76×10^{-4}	2.57×10^{-3}	2.55×10^{-2}	3.44×10^{-2}	6.10×10^{-3}	1.32×10^{-4}	1.25×10^{-3}
ENE	1.61×10^{-3}	1.38×10^{-3}	4.60×10^{-3}	3.18×10^{-2}	3.30×10^{-2}	3.50×10^{-3}	1.09×10^{-4}	1.05×10^{-3}
East	2.41×10^{-3}	2.34×10^{-3}	6.49×10^{-3}	3.44×10^{-2}	1.61×10^{-2}	3.10×10^{-3}	1.36×10^{-4}	6.63×10^{-4}
ESE	4.08×10^{-3}	3.39×10^{-3}	6.10×10^{-3}	1.94×10^{-2}	7.41×10^{-3}	3.66×10^{-3}	1.52×10^{-4}	3.97×10^{-4}
SE	1.22×10^{-2}	7.60×10^{-3}	9.28×10^{-3}	2.32×10^{-2}	1.05×10^{-2}	7.86×10^{-3}	2.59×10^{-4}	6.78×10^{-4}
SSE	1.20×10^{-2}	3.48×10^{-3}	3.51×10^{-3}	9.14×10^{-3}	5.50×10^{-3}	9.03×10^{-3}	2.80×10^{-4}	6.59×10^{-4}
South	7.67×10^{-3}	2.74×10^{-3}	1.32×10^{-3}	2.56×10^{-3}	2.05×10^{-3}	4.75×10^{-3}	2.39×10^{-4}	3.95×10^{-4}
SSW	8.61×10^{-3}	2.90×10^{-3}	1.90×10^{-3}	5.16×10^{-3}	5.89×10^{-3}	7.13×10^{-3}	3.41×10^{-4}	5.67×10^{-4}
SW	7.73×10^{-3}	3.46×10^{-3}	3.64×10^{-3}	2.27×10^{-2}	2.00×10^{-2}	6.00×10^{-3}	3.21×10^{-4}	1.34×10^{-3}
WSW	2.60×10^{-3}	1.49×10^{-3}	4.18×10^{-3}	4.31×10^{-2}	9.49×10^{-3}	2.23×10^{-3}	1.76×10^{-4}	1.12×10^{-3}
West	1.58×10^{-3}	1.33×10^{-3}	4.90×10^{-3}	1.53×10^{-2}	3.24×10^{-3}	2.20×10^{-3}	1.97×10^{-4}	6.68×10^{-4}
WNW	1.43×10^{-3}	1.72×10^{-3}	5.61×10^{-3}	1.31×10^{-2}	2.52×10^{-3}	2.27×10^{-3}	2.82×10^{-4}	1.10×10^{-3}
NW	1.60×10^{-3}	1.96×10^{-3}	8.26×10^{-3}	3.29×10^{-2}	5.50×10^{-3}	2.33×10^{-3}	2.93×10^{-4}	1.96×10^{-3}
NNW	1.69×10^{-3}	2.37×10^{-3}	1.32×10^{-2}	5.27×10^{-2}	1.23×10^{-2}	4.27×10^{-3}	3.96×10^{-4}	3.26×10^{-3}
N	1.35×10^{-3}	1.68×10^{-3}	9.96×10^{-3}	8.69×10^{-2}	3.87×10^{-2}	1.27×10^{-2}	2.91×10^{-4}	5.67×10^{-3}

Table 12.5/6 Exposure Pathways

Pathway consider for all receptors	Receptor points located at 1600 and 4800m, Woronora Valley and Nearest Resident	Settings for Stevens Hall, Waste Services, BMX track, Library, Building 9 and the Main Gate
Consumption of green vegetables	ON	Not Applicable
Consumption of root vegetables	ON	Not Applicable
Consumption of fruit	ON	Not Applicable
Inhalation of radionuclides in the plume	ON	ON
External gamma from airborne radionuclides	ON	ON
External beta from airborne radionuclides	ON	ON
External gamma from deposited radionuclides	ON	ON
External beta from deposited radionuclides	ON	ON
Inhalation of re-suspended radionuclides	ON	ON

Table 12.5/7 Ingestion Data for Adult Consumers

Ingestion (*i.e.* food type)	Ingestion rate ($kg.y^{-1}$)	Fraction of ingested food grown locally
Green vegetables	46.6	0.25
Root vegetables	84.3	0.25
Fruit	147.7	0.25

Table 12.5/8 Occupancy Data

Receptor	Fraction Spent Indoor	Location Factor Cloud Gamma	Location Factor Deposited Gamma	Occupancy (hours/year)
Library	0.0	1.0	1.0	2607
Building 9	0.0	1.0	1.0	2607
Main Gate	0.0	1.0	1.0	2607
Stevens Hall	0.0	1.0	1.0	8760
Waste Services	0.0	1.0	1.0	3910
BMX Track	0.0	1.0	1.0	1042
Woronora Valley	0.0	1.0	1.0	8760
Nearest Resident	0.0	1.0	1.0	8760
All receptors at 1.6 and 4.8 km	0.0	1.0	1.0	8760

Table 12.5/9 Assumed Deposition Rate and Washout Coefficient

Nuclide	Deposition Rate (ms^{-1})	Washout Coefficient (s^{-1})
H-3	0	0
Ar-41	0	0
Kr-85m	0	0
Kr-85	0	0
Kr-87	0	0
Kr-88	0	0
Sr-90	0.001	0.0001
Y-90	0.001	0.0001
I-131	0.01	0.0001
I-133	0.01	0.0001
Xe-133	0	0
Xe-135	0	0
Xe-135m	0	0
Xe-135	0	0

Table 12.5/10 Results of PC-CREAM Calculations – Annual Dose

Receptor Points and Reactor Facility stack locations	Bearing	Estimated dose [µSv]
N at 1600 m	0º	8.9×10^{-2}
NNE at 1600 m	23º	4.5×10^{-2}
NE at 1600 m	45º	4.2×10^{-2}
ENE at 1600 m	68º	4.2×10^{-2}
Woronora Valley (5657 m)	45º	1.0×10^{-2}
E at 1600 m	90º	3.2×10^{-2}
ESE at 1600 m	113º	2.1×10^{-2}
SE at 1600 m	135º	3.3×10^{-2}
SSE at 1600 m	158º	2.3×10^{-2}
Nearest Resident (1704 m)	82º	3.0×10^{-2}
S at 1600 m	180º	1.1×10^{-2}
SSW at 1600 m	203º	2.0×10^{-2}
SW at 1600 m	225º	3.9×10^{-2}
WSW at 1600 m	248º	3.6×10^{-2}
W at 1600 m	270º	1.7×10^{-2}
WNW at 1600 m	293º	1.7×10^{-2}
NW at 1600 m	315º	3.0×10^{-2}
NNW at 1600 m	338º	4.8×10^{-2}
N at 4800 m	0º	2.3×10^{-2}
NNE at 4800 m	23º	1.2×10^{-2}
NE at 4800 m	45º	1.2×10^{-2}
ENE at 4800 m	68º	1.2×10^{-2}
E at 4800 m	90º	8.6×10^{-3}
ESE at 4800 m	113º	5.6×10^{-3}
SE at 4800 m	135º	8.9×10^{-3}
SSE at 4800 m	158º	6.5×10^{-3}
S at 4800 m	180º	3.2×10^{-3}
SSW at 4800 m	203º	5.3×10^{-3}
SW at 4800 m	225º	1.0×10^{-2}
WSW at 4800 m	248º	8.3×10^{-3}
W at 4800 m	270º	3.8×10^{-2}
WNW at 4800 m	293º	3.6×10^{-3}

Receptor Points and Reactor Facility stack locations	Bearing	Estimated dose [µSv]
NW at 4800 m	315º	6.6×10^{-3}
NNW at 4800 m	338º	1.1×10^{-2}
Stevens Hall (971 m)	61º	6.8×10^{-2}
Waste Services (1007 m)	312º	2.4×10^{-2}
BMX Track (887 m)	345º	2.1×10^{-2}
Building 9 (368 m)	49º	4.4×10^{-2}
Main Gate (841 m)	61º	2.3×10^{-2}
Library (587 m)	68º	3.0×10^{-2}

Table 12.5/11 Segregation per Pathway and Isotope for the Highest Calculated Dose (N 1600 m) in µSv

Nuclide	Parent	Inhalation.	Cloud gamma	Deposited gamma	Resuspended	Cloud beta	Deposited beta	Green vegetable	Root vegetable	Fruit	Total
H-3		1.65 x10^{-4}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	1.23 x10^{-5}	1.95 x10^{-5}	3.45 x10^{-5}	2.31 x10^{-4}
AR-41		0.00 x10^{+0}	8.40 x10^{-2}	0.00 x10^{+0}	0.00 x10^{+0}	4.50 x10^{-4}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	8.45 x10^{-2}
CR-51		1.65 x10^{-8}	1.95 x10^{-9}	1.65 x10^{-7}	7.05x10^{-12}	4.95x10^{-14}	0.00 x10^{+0}	2.10 x10^{-9}	1.19x10^{-13}	6.30x10^{-10}	1.86 x10^{-7}
KR-85M		0.00 x10^{+0}	6.15 x10^{-5}	0.00 x10^{+0}	0.00 x10^{+0}	1.31 x10^{-6}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	6.28 x10^{-5}
KR-87		0.00 x10^{+0}	2.55 x10^{-4}	0.00 x10^{+0}	0.00 x10^{+0}	6.00 x10^{-6}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	2.61 x10^{-4}
KR-88		0.00 x10^{+0}	1.65 x10^{-3}	0.00 x10^{+0}	0.00 x10^{+0}	4.50 x10^{-6}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	1.65 x10^{-3}
SR-90		6.60 x10^{-6}	0.00 x10^{+0}	3.45x10^{-11}	8.25 x10^{-9}	6.45x10^{-11}	1.46 x10^{-7}	6.00 x10^{-6}	1.50 x10^{-6}	4.05 x10^{-6}	1.83 x10^{-5}
I-131		9.30 x10^{-6}	6.90 x10^{-8}	1.32 x10^{-5}	1.95 x10^{-8}	5.10x10^{-10}	6.75 x10^{-6}	1.65 x10^{-5}	6.15 x10^{-6}	3.90 x10^{-5}	9.10 x10^{-5}
I-133		6.15 x10^{-5}	3.45 x10^{-6}	7.80 x10^{-5}	2.10 x10^{-8}	3.45 x10^{-8}	1.50 x10^{-3}	1.50 x10^{-5}	2.10 x10^{-7}	4.20 x10^{-5}	1.70 x10^{-3}
XE-133		0.00 x10^{+0}	3.30 x10^{-4}	0.00 x10^{+0}	0.00 x10^{+0}	1.13 x10^{-5}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	3.41 x10^{-4}
XE-135		0.00 x10^{+0}	2.40 x10^{-4}	0.00 x10^{+0}	0.00 x10^{+0}	4.65 x10^{-6}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	2.45 x10^{-4}
CS-137		1.01 x10^{-8}	1.65x10^{-10}	1.65 x10^{-6}	1.26x10^{-11}	1.07x10^{-12}	3.15 x10^{-9}	7.65 x10^{-9}	1.29 x10^{-8}	1.25 x10^{-8}	1.70 x10^{-6}
BA-140		8.25 x10^{-8}	4.20x10^{-10}	2.25 x10^{-7}	2.85x10^{-11}	9.60x10^{-12}	4.05 x10^{-8}	4.05 x10^{-9}	2.85x10^{-12}	2.40 x10^{-9}	3.55 x10^{-7}
LA-140		2.70 x10^{-8}	7.80 x10^{-9}	4.20 x10^{-8}	2.70x10^{-12}	2.70x10^{-11}	1.80 x10^{-7}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	2.57 x10^{-7}
KR-85D	KR-85M	0.00 x10^{+0}	8.70x10^{-14}	0.00 x10^{+0}	0.00 x10^{+0}	1.11x10^{-13}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	1.98x10^{-13}
Y-90D	SR-90	1.80x10^{-10}	8.25x10^{-19}	0.00 x10^{+0}	0.00 x10^{+0}	2.10x10^{-13}	1.65 x10^{-9}	0.00 x10^{+0}	0.00 x10^{+0}	0.00 x10^{+0}	1.83 x10^{-9}
LA-140D	BA-140	1.95x10^{-11}	6.15x10^{-12}	2.70x10^{-11}	1.65x10^{-15}	1.95x10^{-14}	1.13x10^{-10}	6.90x10^{-13}	1.02x10^{-16}	4.65x10^{-13}	1.67x10^{-10}
Total		9.53 x10^{-3}	8.66 x10^{-2}	1.33 x10^{-2}	1.95 x10^{-5}	4.78 x10^{-4}	8.25 x10^{-3}	1.65 x10^{-2}	6.17 x10^{-3}	3.91 x10^{-2}	8.91 x10^{-2}

End of Tables

Figure 12.5/7 Pathways Contributions to Dose in Location N at 1.6 km

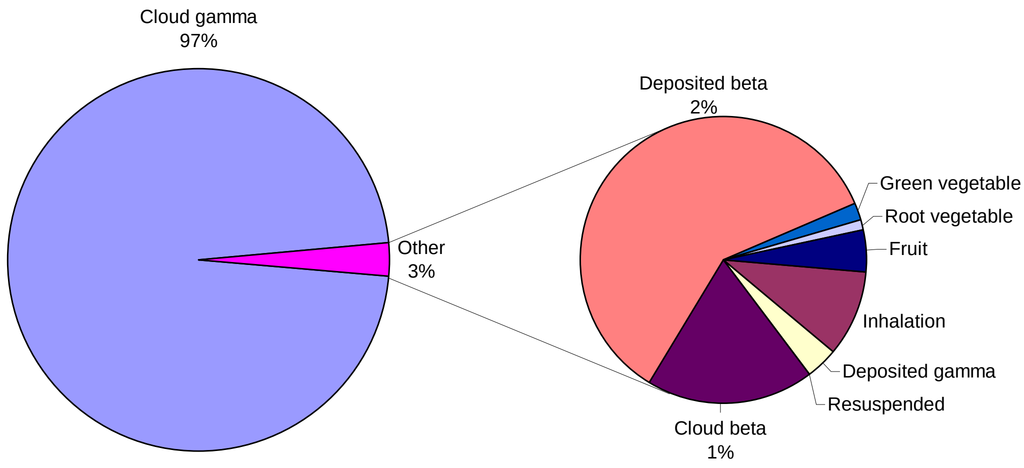
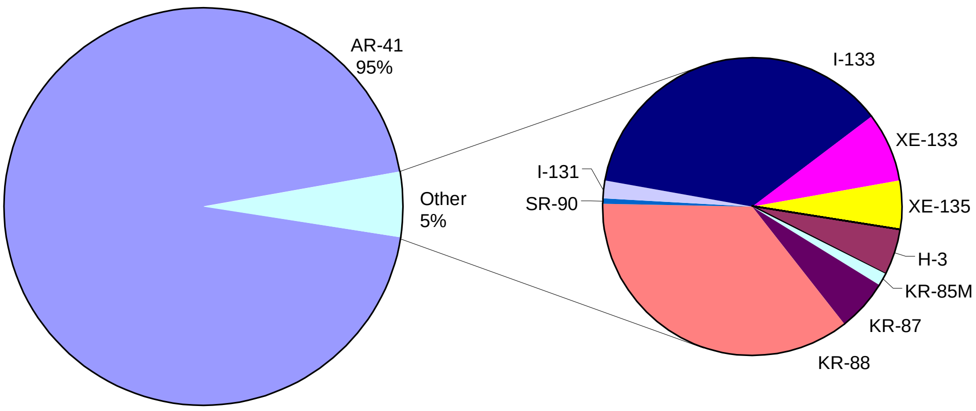


Figure 12.5/8 Nuclide Contributions to Dose in Location N at 1.6 km



End of Figures

12.6 CONCLUSIONS

12.6.1 Calculated and Expected Doses

The information contained in this Chapter shows that the expected radiation doses produced by the normal operation of Reactor Facility to staff members and the public are below the applicable limits and constraints. Moreover, in most cases the expected doses are below the ALARA objective.

The dose received by the most exposed worker is in the order of 2 mSv per year, while for a member of the public located at the buffer zone boundary the dose is less than 0.1 µSv per year.

The reduction of the releases from the Reactor Facility in comparison with those for HIFAR, shows that the Reactor Facility represents a significant decrease in the potential radiation exposures. The releases through the stack were estimated using conservative values of measurements in similar plants and the calculations performed using appropriate models and extrapolations. This conservatism in the source term indicates that actual values will less than those reported in this chapter.

12.6.2 Reactor Facility Features Related with Radiological Protection Issues

This section describes several engineering features included in the Reactor Facility design that makes it possible to develop activities with a maximum benefit to Australia, while reducing the hazards to the minimum achievable. Some examples are summarised in the following paragraphs.

The instrumentation and equipment related with radiological protection are appropriate for a plant with the performance of the Reactor Facility. They allow a close and fast control of radiological parameters such as dose rate, liquid waste streams concentrations, liquid and airborne releases (rates, trends and accumulated values), surface and airborne contamination and cooling water circuits activity (PCS, RSPCS and SCS). Systems are provided with operating ranges and/or redundancies that cover normal and abnormal radiation fields with the capability to trigger safety features.

The instruments used for radiological protection measurements have been specified and/or designed in such a way that they correspond to the state of the art in the area. They have been equipped, to the maximum possible extent, with self-test features and/or low-level alarms in order to prevent unnecessary exposures due to malfunctioning.

The plant layout fulfils requirements related with area classification and access/egress arrangements while at the same time allows future modifications once operational experience confirms that a downgrade in the classification of certain areas is adequate and possible.

Moreover, the plant layout allows appropriate contamination control to prevent unnoticed spread and facilitates the required corrective actions. In all areas with a potential contamination risk, the surface finishing, floor drainage slopes and collection systems have been located to allow effective corrective action.

Personal decontamination facilities have been distributed adequately in the building and a dedicated area for object decontamination is provided, where appropriate elements and instrumentation will be available for first decontamination practices.

The shielding and ventilation of potential hazardous areas provide protection to the staff members. Proven calculation techniques and codes and conservative source terms have been used to guarantee that results fulfil the specified dose rate requirements.

The hot water layer at the surface of the Reactor Pool minimises the transport of nuclides to the water surface and provides a reactor pool water purification stage since the system has its own dedicated resin bed.

Safety requirements regarding alternative escape routes have been matched with radiological protection characteristics.

The waste treatment system deals with the entire potential waste source foreseeable in the Reactor Facility. The best practice solutions have been applied to ensure a minimisation of volumes and activities generated, as well as segregation as close as possible from the source. All effluents released from the plant are controlled and accounted for adequately.

In addition to the comprehensive engineering features described for the Reactor Facility, there is a complementary organisational radiation safety structure that implements an international best practice radiation protection plan. These radiation safety systems ensure compliance with standards and regulatory requirements on radiation protection and the application of optimisation of protection, which contribute to the development of a safety culture in the Reactor Facility at ANSTO.

End of Section