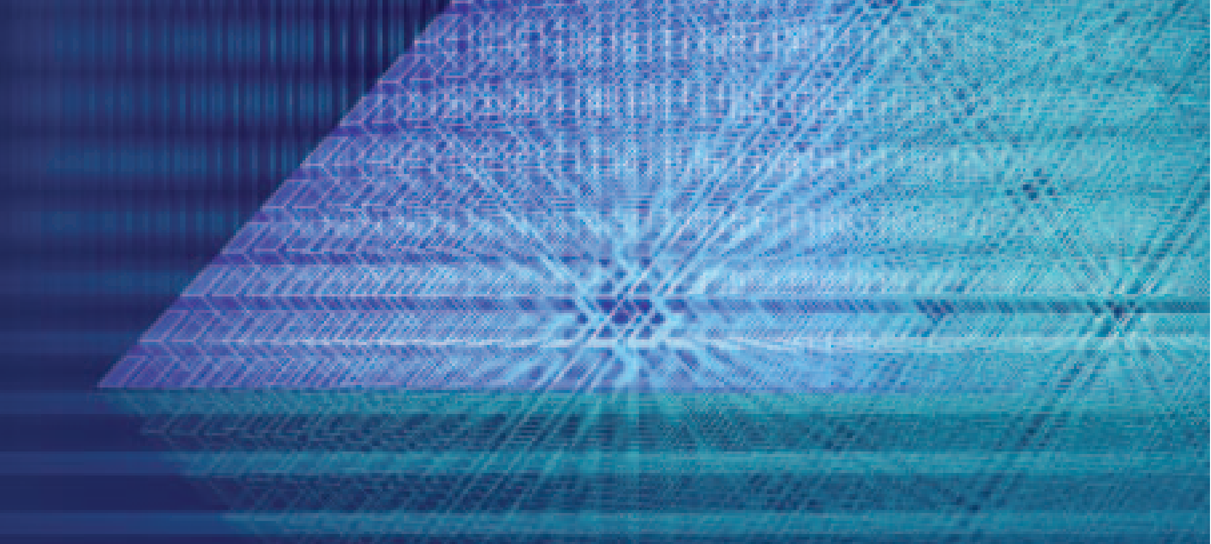

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Decommissioning of Small Medical, Industrial and Research Facilities

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2003

DECOMMISSIONING OF SMALL MEDICAL, INDUSTRIAL AND RESEARCH FACILITIES

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FOREWORD

Most of the existing literature on decommissioning addresses the technological and other aspects of decontaminating and dismantling large nuclear facilities such as nuclear power plants, reprocessing plants and relatively large prototype, research and test reactors. However, the majority of nuclear facilities are smaller in size and complexity and may present a lower radiological risk in their decommissioning. Such facilities as critical assemblies, radiodiagnostic and radiotherapy hospital departments or laboratories, factories using radioactive material, etc., are often associated with the erroneous perception that their decommissioning is a trivial, low priority activity. Under these circumstances the possibility exists that even minimum requirements and strategies may be disregarded in decommissioning, resulting in unnecessary costs, delays and, possibly, safety issues such as the loss of radiation sources. The objective of this report is to highlight important points in the decommissioning of small nuclear facilities for policy makers, operators, waste managers and other parties, drawing on the experience of some Member States.

Two IAEA Technical Reports were published in the 1990s to address decommissioning planning and management, Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities (Technical Reports Series No. 351, 1993) and Decommissioning Techniques for Research Reactors (Technical Reports Series No. 373, 1994), but no IAEA documents have been specifically aimed at the decommissioning of smaller facilities. A preliminary draft was prepared by an IAEA consultant, A. Brown of the United Kingdom, who also chaired subsequent review meetings, and the IAEA Scientific Secretary, M. Laraia of the Division of Nuclear Fuel Cycle and Waste Technology. Following the preliminary drafting a series of consultants meetings was held to review and amend this report, which included the participation of a number of international experts. Further contribution to the drafting and review of the report was also provided by other experts.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Most of the technical literature on decommissioning addresses the regulatory, organizational, technical and other aspects for large facilities such as nuclear power plants, reprocessing plants and relatively large prototype, research and test reactors. There are, however, a much larger number of licensed users of radioactive material in the fields of medicine, research and industry. Most of these nuclear facilities are smaller in size and complexity and may present a lower radiological risk during their decommissioning. Such facilities are located at research establishments, biological and medical laboratories, universities, medical centres, and industrial and manufacturing premises. They are often operated by users who have not been trained or are unfamiliar with the decommissioning, waste management and associated safety aspects of these types of facility at the end of their operating lives. Also, for many small users of radioactive material such as radiation sources, nuclear applications are a small part of the overall business or process and, although the operating safety requirements may be adhered to, concern or responsibility may not go much beyond this. There is concern that even the minimum requirements of decommissioning may be disregarded, resulting in avoidable delays, risks and safety implications (e.g. a loss of radioactive material and a loss of all records). Incidents have occurred in which persons have been injured or put at risk.

It is recognized that the strategies and specific requirements for small facilities may be much less onerous than for large ones such as nuclear power plants or fuel processing facilities, but many of the same principles apply. There has been considerable attention given to nuclear facilities and many IAEA publications are complementary to this report [1–6]. This report, however, attempts to give specific guidance for small facilities. ‘Small’ in this report does not necessarily mean small in size but generally modest in terms of complexity, safety risk and radiological inventory.

1.2. OBJECTIVES

The key objective of this report is to provide information, experience and assistance on what is appropriate and sufficient for policy makers, regulators and operators of small facilities. It is intended to promote timely and cost effective decommissioning and waste management at the end of the life of a facility so as to render such a facility harmless. No statements in the report are intended to be prescriptive.

As mentioned above, there is significant documentation on the decommissioning of large nuclear facilities but the more modest requirements of small facilities have received little attention. If users of small facilities only have available to them published information for large, complex facilities, then there may be a tendency to overreact and engage in elaborate or unnecessary studies and activities. They also may shy away from important issues and do too little, either because they are not trained or advised properly or they do not have a decommissioning plan or adequate human and financial resources. They also are often unaware of the requirements, both legal and technical, of decommissioning and waste management. Some decades ago, when the first large power and demonstration reactors started to be shut down, there were many unknowns and uncertainties on how to proceed. There is now a significant experience database on common problems, which has been shared by the international nuclear power industry. This is not necessarily the case for small facilities, however, and this report is intended to encourage the interchange of information and experience.

1.3. SCOPE

It is intended in this report to cover all aspects of decommissioning small facilities in which radioactive material and radiation sources are produced, received, used and/or stored. Power reactors, prototype and demonstration reactors, larger research reactors, fuel processing and reprocessing plants and their associated large nuclear chemical facilities, and all forms of waste disposal are outside the scope of this report and have been covered adequately elsewhere [1, 3, 4]. While a clear cut definition of small facilities that distinguishes them from large facilities is not possible, this report provides information that warrants consideration for all nuclear facilities that do not fall into the specific categories given above. Facilities covered by this report include the large number of licensed or registered nuclear facilities in which nuclear science and technology is often used as an aid or tool (e.g. radioactive sources), particularly in industry, medicine, universities and research establishments. These types of facility are elaborated upon further in the next section.

Three important aspects are dealt with in this report: strategy, planning and technical issues. Each of these aspects are elaborated upon in more detail in subject areas such as organization and responsibility, waste management, quality assurance and costs. For safety related matters the reader is encouraged to consult Ref. [5] for further guidance. Excerpts from Ref. [5] are given in Section 7.

This report and its annexes also report on various small decommissioning projects that have been completed and on lessons learned, where this information has been made available.

2. TYPES OF FACILITY

The main focus of this report is on all aspects of the decommissioning of medical, industrial and research facilities in which radioactive material and sources are produced, handled or used. The facilities covered in this report include:

- Medical facilities with radiotherapy units, as well as those using radioisotopes for diagnosis and treatment.
- Industrial facilities such as those that use irradiation (irradiators) and radiography. Uses of radionuclides may include the production and labelling of compounds, sterilization, water treatment, food irradiation and using radioactive material (mainly sealed sources) for measurements and/or calibrations, oil exploration, process and plan control, non-destructive testing and quality control and the manufacturing of specialized products (such as smoke detectors and anti-static bars).
- Research facilities such as those associated with the nuclear industry (e.g. low and zero power reactors, critical and subcritical assemblies), pharmaceuticals and medicines, the development and labelling of compounds, the study of metabolic, toxicological and environmental pathways, the development of clinical processes and the applications of prepared compounds.
- Teaching, research and analytical laboratories and facilities at universities and schools doing basic research in the fields of physics, chemistry, engineering, medicine and biology.
- Facilities in which small cyclotrons and particle accelerators are used for the production of radionuclides[1].

It is difficult to establish how many small facilities exist worldwide since no comprehensive register exists. In almost all Member States there is a system of licensing or registering operators for the possession and use of radioactive material. In the United States of America licences are issued either by the Nuclear Regulatory Commission (NRC) or by individual states. The total number of licensees in the USA alone has been estimated to be about 24 000 [7]. The NRC publishes a regular report on the status of sites that are contaminated with radioactive material and that require special attention to ensure their timely decommissioning and release for unrestricted use. There were in 1995 51 sites listed, which stem mainly from previous commercial

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[1] For the purposes of this document very large research particle accelerators, for example CERN (Switzerland), JET (United Kingdom), Triumph (Canada), CEA Saclay (France) and Dubna (Russian Federation), are not included.

activities involving various applications of radioactive material [7]. However, this is a small subset of the sites covered by radioactive material licences.

Data of a few years ago indicate that a total of nearly 700 000 sealed sources are in use worldwide, of which nearly 200 000 are in the USA [8]. In 1991 about 500 000 were reported to be in use in industrial gauges and it was estimated that there could be up to 30 000 spent radiation sources in developing countries [9]. In addition, it has been estimated that not any of the global radium sources now in use will be needed within the next decade and hence will require safe long term storage and disposal. The use of radium has been replaced by other materials, such as ^{60}Co, ^{137}Cs and ^{192}Ir. Many of the radium removal operations will require decommissioning activities.

It has been reported that 206 cyclotrons for radionuclide production, particularly for medical imaging technology, are operating in 34 Member States [10]. The bulk of these are in the USA, Germany and Japan. There are 48 in the European Union (EU).

There has been a large increase in the use of charged particle accelerators in recent decades and it is estimated that there are about 250 operating within the EU [11]. There are also a significant number of irradiators containing powerful sources (e.g. ^{60}Co) that have been used in research institutes but that now need decommissioning [9]. There are estimated to be about 54 commercial large irradiators in the USA, of which nine are already shut down in preparation for decommissioning.

It was reported in 2000 that about 107 small research reactors (<250 kW)[2] were in operation [12], but this number is decreasing owing to ageing and other factors. A higher number of small research reactors are already decommissioned or permanently shutdown and awaiting decommissioning. Many research reactors are dedicated to radionuclide production; typical applications and statistical data are given in Ref. [13].

Sealed sources are used in medicine, industry and research and their characteristics are reported in Tables I–III [14]. Similar information for unsealed sources is given in Table IV (see also Refs [15, 16]).

2.1. MEDICAL FACILITIES

2.1.1. Nuclear medicine departments

In nuclear medicine departments unsealed sources are used both for diagnostics and therapy (in vivo techniques). These types of source are also used in laboratory

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[2] This is not intended to be an upper limit of the research reactors addressed in this report.

TABLE I. SEALED SOURCES USED IN MEDICINE [14]

Application	Radionuclide	Half-life	Source activity	Comments
Bone densitometry	^{241}Am	433.0 a	1–10 GBq	Mobile units
	^{153}Gd	244.0 d	1–40 GBq	
	^{125}I	60.1 d	1–10 GBq	
Manual brachytherapy	^{137}Cs	30.0 a	50–500 MBq	Small portable sources
	^{226}Ra	1600 a	30–300 MBq	
	^{60}Co	5.3 a	50–500 MBq	
	^{90}Sr	29.1 a	50–1500 MBq	
	^{103}Pd	17.0 d	50–1500 MBq	
	^{125}I	60.1 d	50–1500 MBq	
	^{192}Ir	74.0 d	200–1500 MBq	
	^{252}Cf	2.6 a	50–1500 kBq	
Remote after loading brachytherapy	^{60}Co	5.3 a	~10 GBq	Mobile units
	^{137}Cs	30.0 a	0.03–10 MBq	
	^{192}Ir	74.0 d	~400 GBq	
Teletherapy	^{60}Co	5.3 a	50–1000 TBq	Fixed installations
	^{137}Cs	30.0 a	500 TBq	
Whole blood irradiation	^{137}Cs	30.0 a	2–100 TBq	Fixed installations

tests (in vitro techniques). Among medical facilities, nuclear medicine departments generate the largest quantity and widest variety of radioactive waste. Most medical radionuclides are, however, short lived and will decay quickly to acceptable levels (see Table IV). Exceptions are ^{14}C (routinely used for in vivo studies of disorders of the stomach and intestine) and ^{3}H (routinely used for liquid scintillation counting in medical assays of drugs, hormones, etc.).

All nuclear medicine departments consist of restricted access areas, such as areas for the reception, storage and handling of radioactive material, areas for administering radiopharmaceuticals to patients, waiting areas for injected patients and rooms for imaging installations. Typical laboratories in which radioactive material is handled are shown in Figs 1 and 2. A comprehensive description of nuclear medicine services, including instrumentation, radiopharmaceuticals, training and setting up nuclear medicine centres, is given in Refs [13, 17, 18].

TABLE II. SEALED SOURCES USED IN INDUSTRY [14]

Application	Radionuclide	Half-life	Source activity	Comments
Industrial radiography	^{192}Ir	74.0 d	0.1–5 TBq	Usually portable units
	^{60}Co	5.3 a	0.1–5 TBq	
	(^{137}Cs, ^{170}Tm)			
Well logging	^{241}Am/Be	433.0 a	1–800 GBq	Portable units
	^{137}Cs	30.0 a	1–100 GBq	
	(^{252}Cf)			
Moisture detectors	^{241}Am/Be	433.0 a	0.1–2 GBq	Portable units to measure moisture content and density normally contain both a neutron and a gamma emitter
	^{137}Cs	30.0 a	400 MBq	
	(^{252}Cf, ^{226}Ra/Be)		3 GBq	
	^{239}Pu			
Conveyor gauges	^{137}Cs	30.0 a	0.1–40 GBq	Fixed installations to measure the density of coal, silt or ores
Density gauges	^{137}Cs	30.0 a	0.1–20 GBq	Fixed installations to measure the density of material in a constant volume
	^{241}Am	433.0 a	0.1–10 GBq	
Level gauges	^{137}Cs	30.0 a	0.1–2 GBq	Fixed installations to measure the level of material in tanks, silos or packages
	^{60}Co	5.3 a	0.1–10 GBq	
	(^{241}Am)			
Thickness gauges	^{85}Kr	10.8 a	0.1–50 GBq	Fixed installations to measure the thickness of paper, plastic or similar material
	^{60}Sr	29.1 a	0.1–4 GBq	
	(^{14}C, ^{147}Pm, ^{241}Am)			
Static electricity eliminators	^{241}Am	433.0 a	1–4 GBq	Fixed installations and portable units
	^{210}Po	128.0 d	1–4 GBq	
	(^{226}Ra)			
	^{60}Co			
Lightning preventers	^{241}Am	433.0 a	50–500 MBq	Fixed installations
	(^{226}Ra)	1600 a		
	^{60}Co	5.3 a	3–7 GBq	

TABLE II. (cont.)

Application	Radionuclide	Half-life	Source activity	Comments
Electron capture detectors	^{63}Ni ^{3}H	96.0 a 12.3 a	200–500 MBq 1–7.4 GBq	Fixed or portable equipment
X ray fluorescence analysers	^{55}Fe ^{109}Cd (^{238}Pu, ^{241}Am, ^{57}Co)	2.7 a 463.0 d	0.1–5 GBq 1–8 GBq	Often portable units to analyse alloys by stimulating fluorescence X rays
Sterilization and food preservation	^{60}Co ^{137}Cs	5.3 a 30.0 a	0.1–400 PBq 0.1–400 PBq	Fixed installations (individual source activity up to 600 TBq)
Calibration facilities	^{60}Co ^{137}Cs	5.3 a 30.0 a	1–100 TBq	Fixed installations
Smoke detectors	^{241}Am ^{226}Ra (^{239}Pu)	433.0 a	0.02–3 MBq	Fixed (easily removed)
Dredgers	^{60}Co ^{137}Cs	5.3 a 30.0 a	1–100 GBq 1–100 GBq	Fixed installations for silt density measurements
Blast furnace control	^{60}Co	5.3 a	2 GBq	Fixed

2.1.2. Radiotherapy departments

Radiotherapy departments use both sealed and unsealed sources. Hospitals are still among the largest users of sealed radiation sources [9], which are mostly used for teletherapy and brachytherapy. Unsealed sources are generally short lived and thus do not constitute a decommissioning problem.

2.1.2.1. Teletherapy

Teletherapy machines are used mainly for treatments for cancer patients and are found in the radiotherapy departments of hospitals. The sources (typically ^{60}Co and ^{137}Cs) for these units are of a high activity (100–500 TBq) and are held in a shielded head, which often contains depleted uranium.

TABLE III. SEALED SOURCES USED IN RESEARCH [14]

Application	Radionuclide	Half-life	Source activity	Comments
Calibration sources	Various	Variable	<0.1 GBq	Small portable sources
Electron capture detectors	^{3}H ^{63}Ni	12.3 a 96.0 a	1–50 GBq 200–500 MBq	Can be used in portable units and in gas chromatography detectors
Irradiators	^{60}Co ^{137}Cs	5.3 a	1–1000 TBq	Fixed installations
Calibration facilities	^{137}Cs ^{60}Co ^{252}Cf (^{241}Am/Be, ^{238}Pu/Be) ^{226}Ra/Be	30.0 a 5.3 a 2.6 a	<100 TBq <100 TBq <10 GBq	Fixed installations
Tritium targets	^{3}H	12.3 a	1–10 TBq	Fixed installations for the production of neutrons

These machines present challenges when being considered for decommissioning, largely owing to the age of the sources and mass of the inherent shielding. An old teletherapy machine is depicted in Fig. 3.

2.1.2.2. Brachytherapy

Brachytherapy sources are small needles or seeds that contain radioactive material. A hospital can have an inventory of more than a hundred or even thousands of sources in use or storage. Until the late 1940s the most significant radionuclide used was radium. When particle accelerators and research reactors became available, during the 1950s, sources containing various types of radionuclide could be produced. These were safer and easier to handle than those containing radium, and could be used for many new purposes. Old radium sources have presented one of the largest problems, owing to their widespread distribution. A particular problem is the tendency of these old

TABLE IV. PRINCIPAL RADIONUCLIDES USED IN BIOLOGICAL RESEARCH AND MEDICINE (UNSEALED SOURCES)
(adapted from Ref. [14])

Principal application	Radionuclide	Half-life	Typical quantity per application	Waste characteristics
Biological research	^{3}H	12.3 a	Up to 5 MBq	Solid, liquid
Labelling			Up to 50 GBq	Organic solvents
Biological research	^{14}C	5730 a	Up to 10 MBq	Solid, liquid
Labelling			Less than 1 GBq	Exhaled CO_2
Biological research	^{32}P	14.3 d	Up to 200 MBq	Solid, liquid
	^{33}P	25.4 d	Up to 50 MBq	
	^{35}S	87.4 d	Up to 5 GBq	Solid, liquid
	^{51}Cr	27.7 d	Up to 5 MBq, up to 100 kBq	Solid, liquid
	^{125}I	60.1 d	Up to 500 MBq	Solid, liquid, occasionally vapour
Nuclear medicine diagnostics	^{18}F	1.8 h	Up to 500 MBq	Solid, liquid
	^{67}Ga	3.3 d	Up to 200 MBq	Solid, liquid
	^{81}Kr^{m}	13.3 s	Up to 2 GBq	Gaseous
	^{99}Tc^{m}	6.0 h	Up to 1 GBq	Solid, liquid
	^{111}In	2.8 d	Up to 500 MBq	Solid, liquid
	^{123}I	13.2 h	Up to 500 MBq	Solid, liquid, occasionally vapour
	^{131}I	8.0 d	Up to 500 MBq	
	^{133}Xe	5.3 d	Up to 400 MBq	Gaseous
	^{201}Tl	3.0 d	Up to 200 MBq	Solid, liquid
Nuclear medicine therapy	^{32}P	14.3 d	Up to 200 MBq	Solid, liquid
	^{89}Sr	50.5 d	Up to 300 MBq	Solid, liquid
	^{90}Y	2.7 d	Up to 300 MBq	Solid, liquid
	^{131}I	8.0 d	Up to 500 MBq, up to 10 GBq	Solid, liquid, occasionally vapour
	^{153}Sm	1.9 d	Up to 8 GBq	Solid, liquid

sources to leak due to internal radon overpressure. The casings are made of precious metal and hence susceptible to theft. The use of ^{226}Ra in radiotherapy has been discontinued for safety reasons and in accordance with international recommendations;

FIG. 1. View of a medical biological laboratory (courtesy of the Department of Pathophysiology, University of Vienna Medical School). Typical elements include the fume hood, sink and waste containers.

these sources are being replaced by other radionuclides (mainly ^{137}Cs, ^{60}Co and ^{192}Ir). Most ^{226}Ra sources have been collected from hospitals and managed in accordance with the recommendations of the IAEA [9, 19–21][3].

The control of brachytherapy sources must include tracking patients who have had sources implanted. This tracking ensures that the sources are retrieved at the conclusion of the treatment period. Retrieval includes the removal of sources from patients who die with the sources still in place. Failure to track and retrieve brachytherapy sources can lead to unnecessary exposure to the patient and his or her family. Loss of control of sources in a patient's body can cause unnecessary exposure when the body is released to the family or to a mortuary. In addition, there is the potential for significant contamination problems, depending on the disposal of the body (e.g. if the body is cremated).

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[3] The IAEA supports Member States in resolving the problem of spent radium sources with proven methods and technology within the IAEA Interregional Programme (INT/4/131). The target objective of this programme is to render radium sources safe worldwide by the year 2005.

FIG. 2. Radioactive waste containers at a medical biological laboratory (courtesy of the Department of Pathophysiology, University of Vienna Medical School).

Brachytherapy installations normally have at least four well defined and clearly different areas: a storage area, a preparation room, an application room and a special ward. Radioactive material is supplied in lead containers, which may be retained for use as storage containers. In the past few years brachytherapy departments have been decommissioned in hospitals in some Member States. Some examples are Cuba (see Annex I.B), the Dominican Republic (see Annex I.D), Colombia and Costa Rica. However, other countries are currently constructing additional brachytherapy units, especially in cardiology and vascular departments. For example, in the USA and the United Kingdom there is a vast expansion occurring in the field of vascular

FIG. 3. A teletherapy unit being decommissioned.

brachytherapy. The principal radionuclide usually used is ^{90}Sr, as a train of 16 sources, each with an activity of 1.85 MBq [22].

2.2. INDUSTRIAL FACILITIES AND APPLICATIONS

There are two aspects of the industrial uses of radioisotopes to be considered: the facilities in which sources and equipment containing radioactive material are manufactured and a separate topic of the diverse applications of the radioisotopes.

The production of radionuclides is generally undertaken using high powered reactors or multipurpose research reactors. The decommissioning of these is not in the scope of this report. Facilities in which sources are incorporated into industrial equipment are described below.

The industrial applications of radiation sources are both varied and numerous, and are based on the interaction of radiation with material and the resulting effects. Table II gives a list of typical sealed sources used in industry.

2.2.1. Manufacture of sources

In facilities in which sources are manufactured the radioactive material is initially received in a suitable shipping cask. The material is first removed from its shipping container within some form of containment system. The type of containment used to handle the material depends on the radiation(s) emitted from the radionuclides being handled. For gamma emitters and some beta emitters all work with the radioactive material is conducted inside hot cells. Work with low energy beta emitters and alpha emitters may be conducted inside either a glovebox or a hood.

The ultimate form of the source for gamma sources and volumetric beta sources is a capsule. After assaying the radioactive material to determine its activity, appropriate aliquots are placed inside an open capsule. The capsule is then welded shut (depending on the source capsule design, there may be one or two layers of encapsulation). Finally, the sealed capsule is leak tested to ensure its integrity (including an external test for contamination) and assayed to confirm its radiation emission rate.

The most common form of low energy beta and alpha emitting source is for the source to be placed on a plate or disc. The initial step for plated sources may involve dissolving the source material in an appropriate solution (if the source material is not already in the necessary chemical form). An aliquot is then electroplated on to an appropriate backing material (typically stainless steel). The source is then wipe tested to ensure that there is no loose contamination present, and assayed for activity. Once the manufactured source passes all quality control tests it is packaged for shipment to the user.

Radioactive contamination at facilities in which sources are manufactured is largely a result of handling the radioactive material in its unsealed form. This contamination normally resides within the containment system(s) in which the unsealed material is handled and the sources are manufactured. There is a risk, owing to the movement of material in and out of the hot cell or glovebox, that contamination may be spread outside the containment and into the enclosures of the facility; this should be taken into consideration when planning a decommissioning.

2.2.2. Applications based on detection and measurement

The interaction of radiation with material results in phenomena such as absorption and scattering; the results of measurements of these phenomena provide information on the material in which this interaction occurs. The sources used in these applications are usually sealed sources. Examples of these applications are given below.

- Industrial radiography (gamma and neutron radiography). The most widely used isotopes for gamma radiography are ^{60}Co, ^{137}Cs and ^{192}Ir. A research reactor or neutron generator is used for neutron radiography. These are the industrial sources of major interest, owing to their high specific activity.
- Measurements of thickness and density. Depending on the thickness to be measured, beta or gamma radiation sources may be used. The most widely used isotopes are ^{14}C and ^{63}Ni for thin plastics, ^{147}Pm for photographic films and fabrics, ^{85}Kr for paper and plastics, ^{90}Sr/^{90}Y for thick paper and tapes, ^{90}Sr and ^{133}Ba for aluminium and copper films and ^{60}Co and ^{137}Cs for dense materials.
- Measurements of surface levels. The measurement procedures using gauges are varied. The most widely used isotopes are ^{60}Co and ^{137}Cs.
- Humidity measurements. The process is based on the moderation of neutrons colliding with atoms of hydrogen in water. This process is widely used in the construction of roads and in soil analysis. The most widely used sources are ^{241}Am/Be and ^{239}Pu/Be.
- Well logging. This method is used for hydrogeological exploration and monitoring.

Sources for radiography are usually contained in portable and/or mobile equipment and have been used extensively in many countries, particularly for pipeline welding inspections. It is suspected that significant numbers of sources have been left behind after their use or reported lost. Many portable radiography units contain valuable heavy metal, which can be sold as scrap, which can produce a risk of the exposure of people to radiation. There are also large neutron and gamma sources used in the oil and mining industry for well logging that constitute a risk of exposure if not properly managed.

Instrumentation containing radioactive sources constitutes a significant risk if the sources are not properly documented or are abandoned or disposed of as non-radioactive waste.

2.2.3. Applications based on the effects of radiation

The absorption of radiation in material has many applications. There are two categories to be considered:

- High activity gamma emitting sources;
- Low activity alpha and beta emitting sources.

A typical gamma irradiator consists of several tubes or rods containing powerful radioactive sources, mainly ^{60}Co for industrial purposes or ^{137}Cs in laboratory irradiators. The radioactive source is located inside a radiation shield, which could be, for example, a 2 m thick concrete chamber. The product to be irradiated is moved into the irradiator cell by means of a conveyor, or alternatively the source is placed into the cell. Typical applications are sterilizing medical, pharmaceutical and laboratory devices and supplies, sterilizing food ingredients and cosmetic base materials, protecting archives and antiques and modifying the properties of materials (e.g. their colour, mechanical behaviour or gas production). These applications require doses varying from 1 to 25 kGy. A seed irradiator is depicted in Fig. 4.

FIG. 4. A seed irradiator being decommissioned.

It is reported in the USA that sterilizing applications amount to about 85% of the capacity of irradiators [8]. There are about 150 irradiators located around the world [9].

So long as the sources have not leaked, the decommissioning of an irradiation facility is straightforward. If a leak has occurred there is the potential for a contamination of the facility and the environment, with important consequences for decommissioning. Access control systems and procedures should be in place to prevent personnel from entering the chamber when the sources are exposed.

Applications using low activity alpha and beta emitting sources are based on the ionizing and luminescing effects of radiation. The following uses may be singled out.

- The elimination of static electricity in processes used in, for example, the textile, paper or glass industries and in processes using large volumes of inflammable material. The most widely used isotopes are ^{3}H, ^{85}Kr, ^{90}Sr and ^{241}Am.
- Lightning conductors. The most widely used isotopes are ^{14}C, ^{90}Sr, ^{214}Am and ^{226}Ra.
- The production of luminescent material for signalling for, for example, aircraft or ships. The most widely used isotopes are ^{3}H, ^{85}Kr, ^{90}Sr and ^{147}Pm.
- Smoke detectors. The most widely used isotopes are ^{63}Ni, ^{85}Kr, ^{226}Ra and ^{241}Am.

2.2.4. Applications based on the use of radioactive tracers (unsealed sources)

For applications based on the use of radioactive tracers the radioactive material, generally an unsealed source, is incorporated into or attached to a material in order to track and study events by detecting the radiation emitted. The application may or may not involve a chemical reaction.

The possibilities for these applications are extremely varied, the following being examples.

- The transport of fluids: for example for flow measurement or residence times.
- Studies of wear and friction: for example the wear of metallic parts or the behaviour of lubricants.
- Research into chemical processes: for example kinetics studies and the mechanisms of chemical reactions.
- Environmental pollution and dispersion.
- The detection and location of leaks in pipes and tanks.
- Monitoring the homogeneity of mixtures.

Examples of the radionuclides used in radiotracer technologies are given in Table V.

2.3. RESEARCH FACILITIES AND APPLICATIONS

The field of research using radioactive material directly or indirectly is varied. It is possible to divide nuclear research into three major groups:

- Small research reactors and critical assemblies;
- Nuclear research laboratories and hot cells;
- General research laboratories that use radionuclides.

2.3.1. Small research reactors and critical assemblies

Critical assemblies usually consist of an array or pile in which the neutron multiplication associated with nuclear fuels can be investigated. As the fission product yields of such facilities are low, their radiological inventories are also low. The IAEA database on such facilities indicates that 52 were operational in September 2000 [12].

Small research reactors (e.g. having thermal powers less than 250 kW) will have relatively low radiological inventories after the spent fuel is removed prior to the start of their decommissioning and are therefore included within the scope of this report. Typical of these types of reactor are Argonaut, MNSR, SLOWPOKE and TRIGA. About 107 small research reactors (<250 kW) were operational in September 2000 [12].

2.3.2. Nuclear research laboratories and hot cells

Research laboratories are usually associated with research reactors, universities and industrial facilities. Research facilities are typically equipped with fume hoods, gloveboxes and/or hot cells. A wide range of radionuclides may be handled.

Fume hoods are mainly used for the handling of tracers and radioactive material that presents a low risk for workers and the environment. Fume hoods are associated with active ventilation systems, which can become contaminated.

Material presenting a risk of inhalation for workers or a significant contamination problem for the environment is usually handled inside a glovebox. Typical applications occurring inside gloveboxes are the mechanical and chemical preparation of fissile material for the fabrication of fuel or the handling of very low activity radioactive material for radiological characterization.

TABLE V. PRINCIPAL RADIONUCLIDES USED IN RADIOTRACER TECHNOLOGIES

Application	Radionuclide	Half-life	Chemical form
Hydrology	^{3}H	12.3 a	Tritiated water
	^{24}Na	15 h	Sodium carbonate
	^{51}Cr	27.7 d	EDTA[a] complex
	^{58}Co	71 d	EDTA complex
	^{60}Co	5.3 a	EDTA complex
	^{82}Br	36 h	Ammonium bromide
	^{131}I	8 d	Potassium iodide
Agriculture	^{14}C	5730 a	Glyphosate
	^{99}Tc^{m}	6 h	Sodium technetate
Industrial processes	^{46}Sc	84 d	Scandium oxide
	^{64}Cu	12.7 h	Metal
	^{69}Zn^{m}	13.8 h	Metal
	^{140}La	40 h	Lanthanum chloride
	^{197}Hg	2.7 d	Metal
	^{198}Au	2.7 d	Chloroauric acid
	^{203}Hg	46.6 d	Metal
Gas flow	^{41}Ar	110 minutes	Gas
	^{79}Kr	35 h	Gas
	^{85}Kr	10.8 a	Gas
Fluid flow	^{137}Cs/^{137}Ba^{m} generator	2.5 minutes	Ion exchange resin

Note: The data in this table have been elaborated from tables in Refs [86–88].

[a] EDTA: ethylenediamine tetra-acetic acid.

Material that presents a higher risk of both external and internal (inhalation, ingestion and skin contamination) radiation exposure is handled in hot cells. A hot cell consists of a chamber (usually made of stainless steel) surrounded by a radiation shield (of lead and/or concrete). It is equipped with lead glass windows, tongs and manipulators and is usually operated at a sub-atmospheric pressure. Typical activities carried out inside hot cells are, for example, post-irradiation analyses on fuel rods, the small scale reprocessing of spent fuel, analyses of reactor material, material and waste characterization and the packaging and sealing of radioactive sources recently produced in a high energy neutron flux reactor. Hot cells can be located inside a research reactor facility.

Fume hoods, gloveboxes and hot cells have connections to an active ventilation system and may also have a connection to an active drainage system. The drains may become contaminated with any or all of the radionuclides that were used in these enclosures. Active drains are therefore an important component of the decommissioning process. The spread of airborne contamination in the ventilation ducts associated with hot cells and gloveboxes is also a potential issue for decommissioning.

For decommissioning purposes it is important to record not only the type of contamination (beta, gamma and/or alpha) but also whether the facility was used for mechanical (e.g. cutting) or chemical activities. If chemical activities are carried out inside a hot cell or a glovebox the residual material and equipment may be more difficult to decontaminate, especially if a buildup of radioactive substances occurs over many years of operation.

2.3.3. General research laboratories

The term general research laboratories encompasses a wide range of laboratories that use small quantities of radioactive tracers or sources, including for the following applications.

- Medical and pharmaceutical research applications. These may be used in the study of metabolic and toxicological pathways.
- Veterinary research applications.
- Environmental pathway studies that involve the dispersion of pesticides, fertilizers and other chemicals.
- Basic and applied research in the fields of physics, chemistry, engineering and biology at universities and research institutions.
- Agricultural research in which sealed sources using isotopes such as ^{241}Am/Be and ^{137}Cs are used.

In all the above applications the monitoring and control of the radioactive material and radiation sources is difficult, owing to the changing nature of research projects and their applications. This will impact upon radiological characterization in decommissioning and waste characterization.

2.4. PARTICLE ACCELERATORS

Accelerator types include:

- Van de Graaff accelerators;

- Linear accelerators;
- Cyclotrons;
- Synchrotrons.

All these accelerators have similar features and physical characteristics important for decommissioning.

In a Van de Graaff accelerator a moving belt between electrodes a few metres apart continuously transports electric charges. The discharge tube is encased inside a large pressure vessel charged to several atmospheres of gas to inhibit breakdown. Positive ions are accelerated up to 10 MeV.

In a linear accelerator low energy electrons generated by an electron gun are transmitted to an accelerator tube structure. This tube structure is surrounded by quadrupole magnets to ensure sufficient focusing of the electron beam. Proton linear accelerators use the same principles. A linear accelerator complex consists of a shielded enclosure and a target room. Both are surrounded by thick concrete walls.

In a cyclotron charged particles are injected into a vacuum chamber subjected to a magnetic field of 1 to 2 tesla. The particles are accelerated by an alternating electric field driven by a radiofrequency wave. The energy of the accelerated particles ranges from a few MeV to 300 MeV, depending on the type of the cyclotron (i.e. whether it is a medical or research cyclotron). When the desired energy is reached, the charged particles interact with a target. A cyclotron complex consists of a vault of typically 2 to 4.5 m wall thickness in which the cyclotron is located, several irradiation rooms with very thick walls and a control room. Cyclotrons used in nuclear medicine applications are often standalone, self-shielded units. Typical applications of cyclotrons are for nuclear cross-section measurements and the production of radionuclides for radiopharmaceuticals, which are used in nuclear medicine procedures. A comprehensive directory of the characteristics of cyclotrons used for the production of radionuclides is given in Refs [10, 13].

Synchrotrons consist of a toroidal vacuum tube producing proton or electron beams of several GeV. Synchrotrons are generally used for fundamental particle physics purposes.

For decommissioning purposes it is important to stress that only charged particle accelerators delivering beams with energies higher than a few MeV per nucleon and a beam power of at least 100 W are able to induce significant activation in building structure materials.

In contrast to the generally well characterized waste generated in nuclear reactors, the characterization of contaminated or activated material in accelerator facilities may suffer from poor records of the experiments performed as well as of beam times, beam currents, the materials used and suppliers. The large volume of the materials used and a complicated nuclide inventory increases the problem of inadequate information.

For radiation shielding, accelerators are sometimes housed in large thick walled concrete structures, and the activation of trace elements in the construction materials gives rise to large quantities (possibly thousands of cubic metres) of low active waste. Decay periods to achieve conditions that will allow for removal from regulatory control can be several decades. Large accelerator facilities can become activated by neutrons to levels many times higher than the permitted release criteria (levels of up to 300 Bq/g have been reported).

3. DECOMMISSIONING STRATEGIES

3.1. INTRODUCTION

The possession, handling and use of radioactive material is controlled in most Member States by a competent authority that issues a licence or similar authorization for the possession and use of material and that ensures regulatory control. A national register of radiation sources is often kept. Before the 1940s radioactive material, particularly radium, was thought to be beneficial to health and not subject to controls (see appendix I of Ref. [9]). Since the 1950s radionuclides have been used for a wide range of applications and it is by no means certain that all these activities have been under proper regulatory control. This also applies to decommissioning and waste management activities. The situation is exacerbated by the large number of small items, the varying magnitude of hazards and the fact that users with no further use for a material may not have made adequate financial, organizational or technical provisions for its decommissioning and disposal. Some sources have also been passed on to new owners who lack training and experience in managing radioactive material. A discussion on facilities operational before the establishment of radiation protection regimes is given in Section 3.5.

The following steps for establishing a strategy for decommissioning are common to all strategies:

- Appointing or identifying a responsible person;
- Communicating with the regulatory authorities;
- Establishing timing and schedules;
- Gathering radiological data on the facilities to be decommissioned;
- Identifying the alternatives for the dismantling;

- Identifying the alternatives for waste management;
- Establishing the staffing and financial resources necessary;
- Collating the records and archives.

3.2. OBJECTIVES AND TIMESCALES OF DECOMMISSIONING

One of the objectives of decommissioning is to reduce the risk of harm caused by radiation to workers and the public to acceptably low levels. The importance of planning is to achieve this objective, while minimizing the risk to the personnel engaged in these activities. The formulation of an appropriate strategy is an important part of planning. Actions are carried out to achieve a progressive and systematic reduction in radiological hazards. These actions are taken on the basis of preplanning and assessments to ensure safety during decommissioning actions. Decommissioning also includes waste management involving conditioning followed by interim storage, if necessary, and eventual disposal.

The timescale to complete a decommissioning will depend on the type of the facility, the radioactive material inventory, the half-life of the radionuclides, the chosen decommissioning strategy and the techniques employed. Decommissioning can sometimes be accomplished in a few days if it only requires the removal of radioactive material (e.g. sealed sources) followed by radiological surveys. There will then only be a need to decontaminate as necessary and apply for the removal of the facility from regulatory control. There could, however, be major dismantling and decontamination activities required over a number of weeks or even months for some small research reactors and laboratories.

Immediate dismantling is often the optimal strategy, as it is generally the most appropriate for small facilities because the radioactive inventory is small and prompt action makes the best use of key operational staff familiar with the facility. However, decommissioning strategies for multiunit sites that include both small and large facilities may be dictated by the dominant facility and may include significant deferral periods. A strategy of delay or deferral by default is not condoned or encouraged as it can lead to many problems and undesirable situations (see Section 3.3).

The wide range of small facilities described in Section 2 will require different approaches to decommissioning commensurate with the type and status of the facility concerned and the nature of the radioactive waste inventory. Strategies that may be appropriate for the various types of facility are outlined below.

3.3. THE NO ACTION STRATEGY

Following the permanent shutdown of a nuclear facility a hazardous situation can eventually arise if no action is taken. The no action strategy is unfortunately common practice for many shut down small facilities, and often occurs because they can, by their nature, be easily shutdown for periods of non-use or maintenance but thereafter never restarted for commercial, obsolescence or other reasons. No action being taken often results from the erroneous perception that the risks associated with the shutdown facility are trivial and therefore can be disregarded. In other cases no action being taken may be due to a lack of funds for the decommissioning of the facility. Eventually, no action being taken may end up with the abandonment of the plant. An extreme example is given in Ref. [23], in which a radium facility was abandoned. The loss of records can be particularly serious.

There are various risks associated with this no action strategy. For example, the knowledge of the construction of the plant and its operational features tends to fade away quite quickly, both in terms of the dispersal of staff familiar with the plant and the loss of documentation. The historical memory of key staff plays a major role in decommissioning, particularly for facilities erected and operated in the 1960s and 1970s, when records were rarely archived properly. However, re-assembling a team of competent people a few years after the shutdown of a plant may be almost impossible, because the former staff rapidly disband and find new employment or retire. Planning for and implementing decommissioning without the necessary information is a complicated matter, particularly in satisfying regulators and other interested bodies. Long periods of no action being taken inevitably leads to higher costs when the decommissioning is finally undertaken. Timely planning and implementation may prevent these difficulties.

Other risks arising from a policy of no action are, for example, a lack of adequate maintenance, which allows systems and components to deteriorate, contaminated fluids to leak and drain pumps and sumps to become inoperative. This will eventually result in a spreading of contamination, with consequent risks to workers and the general public. It may happen that rainwater or groundwater finds a path into and out of the facility. Another even more serious risk is that, owing to a lack of adequate surveillance, contaminated material or even radioactive sources are stolen because of their perceived salvage value. Failure to protect radioactive sources has already resulted in fatal accidents worldwide [16, 24–26]. A policy of no action is generally not approved by regulators and is not recommended by the IAEA. Decommissioning implies positive management action together with adequate resources and initiatives.

In successful decommissioning projects the duration of any period of no action following the permanent shutdown of a facility has to be kept to a minimum. The no action strategy does not preclude deferral for the decay of short lived isotopes under properly controlled conditions (see Section 9.5 for more details of deferral for radioactive decay). During this transition period, planning for decommissioning should be initiated as soon as possible, while the operational staff are available. Priority should be given to collecting, in a systematic way, all information relevant to the decommissioning.

3.4. ESTABLISHING A DECOMMISSIONING STRATEGY FOR INDIVIDUAL FACILITIES

The selection of a suitable strategy for small facilities is typically simpler than for power reactors or fuel cycle facilities. There may be variations in detail between the strategies for medical, research and industrial facilities and variations within these types of facility, but these are less important than the general approach given in this report.

Facilities have been categorized for the purposes of this report according to characteristics related to decommissioning:

- Facilities with small portable or mobile sealed sources;
- Facilities with high activity sources, including irradiators;
- Facilities with particle accelerators;
- Research facilities, hot cells, radiochemical laboratories and medical facilities;
- Small research reactors and critical assemblies;
- Large sites incorporating many diverse facilities;
- Manufacturing facilities.

It is of particular note for sealed sources that alternative exposure scenarios could be considered according to the potential harm the source could cause to workers, the public or the environment. There are three main general exposure scenarios that result from a loss of control of sealed sources:

- Non-uniform external exposure to a source in close proximity to one or more individuals;
- Whole body external exposure to an unshielded source (from a few to many individuals);
- Exposures (external and internal) following the rupture of a source casing or containment.

These three generic scenarios were derived from experience documented in case histories [16]; for example, an external exposure to a single individual from an unshielded source occurred in Georgia in 1997 and in Peru in 1999. The exposure of multiple individuals to an unshielded source has occurred in various locations, such as Estonia in 1994, Algeria in 1996, China in 1996, Turkey in 1998 and Thailand and Egypt in 2000. Several of these incidents resulted in deaths of members of the public. Significant exposure following the rupture of a source occurred in Juarez, Mexico, in 1983 and in Goiânia, Brazil, in 1987. All the above are cited in Ref. [16].

Tables I–III give information concerning the applications and some characteristics of the types of radioactive material used in sealed sources.

3.4.1. Strategy for facilities using small portable or mobile radiation sources

These facilities include sealed radiation sources for medical, research and industrial uses, for example:

- Radiotherapy facilities, specifically for brachytherapy;
- For gamma radiography;
- For measurements of, for example, thickness, density, fluid level and humidity;
- Universities with small portable sources (calibration sources).

These sources can give rise to immediate hazards because of their small size, apparently benign appearance and sometimes high scrap value. Establishing the total inventory of these sources is usually the most difficult problem and in many cases it is impossible to quantify accurately (e.g. for old radium sources).

Spent sealed sources pose significant hazards that must be addressed. When a radiation source is no longer to be used for its original purpose, the following management options may be considered [21]:

- Transfer to another user for application elsewhere;
- Return to the manufacturer or supplier;
- Storage for decay of sources containing radionuclides with a short half-life, followed by disposal as non-radioactive material;
- Transport to a centralized interim storage facility until a conditioning facility is available;
- Transport to a central conditioning facility, followed by interim storage;
- On-site conditioning of the source followed by interim storage until a centralized storage or disposal facility is available;
- Transport of the conditioned source to a disposal facility, if available;
- Final disposal in a licensed repository.

The preferable strategy for spent sealed sources is to send them back to the manufacturer; this, however, is often not possible. An appropriate strategy would therefore be their interim storage at a suitable third party location. This third party organization would specialize in the handling and treatment of spent sources and may be a national enterprise. Examples of current initiatives for the management of spent and disused sources are given in Refs [9, 14, 19, 26]. An interim storage facility would need to take account of shielding requirements, especially if there is a large accumulation of sources, and of the control of any possible leaks or emissions. It is important that the facility operator has an up to date register of the sealed sources to use for the planning of decommissioning.

Adopting a shutdown and decommissioning strategy for a facility using sealed sources would prompt the following activities:

- Notifying the appropriate authorities of intentions.
- Appointing a person(s) responsible for the decommissioning.
- Counselling staff.
- Initiating public relations, as necessary.
- Determining, as far as possible, the historical context of the use of sealed sources at the facility.
- Making available a register or an update of an existing register of sources, including the characteristics of the sources.
- Arranging a thorough radiological survey of the facility buildings wherever it is suspected that spent sources may have been stored, used or leaked. This may require special equipment and specialist skills may be needed.
- Establishing a disposal or storage route for disused or orphan sources. These options are discussed further in Section 6.2.
- Notifying the appropriate authorities of the proposed actions, including the transport, design and construction of any special shielding devices needed and obtaining approval to proceed.
- Removing all sources to interim storage or off the site.
- Initiating and completing decontamination and dismantling activities.
- Conducting a final radiological survey as necessary, especially if any decontamination was undertaken, and obtaining agreement for removal from regulatory control.
- Updating records and the register of sources to reflect the current situation.

It is appreciated that certain difficulties are likely to arise in undertaking what would appear to be a simple plan of action. Some of these would be:

- Obtaining the necessary financial and human resources;
- Locating sources that have been stored or used in the past and for which no records were kept;
- Keeping track of sources used in remote locations.

Sealed sources in redundant equipment are often overlooked (e.g. built-in radioactive standards in gamma counters). Anatomical markers attached as a wand for gamma cameras, which may have sources of up to 400 MBq of ^{241}Am, are a particular problem. The latest gamma cameras now have attenuation correction gantries with an array of about 24 different activity sources of ^{153}Gd, with activities in the range of 74–800 MBq [22].

3.4.2. Strategy for facilities using high activity sources, including irradiators

High activity sources are beta/gamma emitting radionuclides with inventories of the order of 100 PBq and are enclosed in thick metal or concrete shielding. Some are kept in pools to provide protective shielding when not in use. Their handling, transport and disposal are particularly difficult without proper equipment and training. A typical large irradiator would be housed in a building of possibly 30 000 m^2, although the irradiated cell may only be 10 to 15% of this [8].

The most appropriate and usual strategy is to remove the whole source in a special package that is suitable and approved for transport to a major nuclear research centre or to a centralized facility that has the appropriate equipment [9]. The possibility of the reuse of the source at another facility may be considered. One specific case history is described in Ref. [27]. The loss or theft of these types of source is less likely, and the source may be able to be retained in storage until a suitable disposal route and disposal contract has been negotiated. Some facilities may become contaminated owing to an internal leakage of the source, and consequently may require an alternative decommissioning strategy.

If the source itself is to be replaced with a new one for the continued operation of the irradiator, a strategy of its return to the vendor or manufacturer is preferable. If this option turns out to be impracticable and no approved storage site is available then interim storage is the appropriate strategy. The absence of a readily viable strategy can become serious, especially if there is lack of preplanning, information, expertise and suitable equipment at the facility concerned.

Interim storage can be on the site or at a separate centralized facility. If the source is to be retained on the site for an extended period then a safety case will have to be prepared for this that addresses the problems of deterioration, leakage and safe storage (see Section 7). For example, some facilities incorporate a pool, which may become contaminated if any of the sources develop leaks. One example of water contamination as a result of failed source encapsulation and its impact on decommissioning is given in Ref. [28].

If a large source is to be disposed of the suggested list for decommissioning activities for small sealed sources given above is still applicable, although handling the item and securing an approved transport package would also have to be addressed.

An example of the successful decommissioning of a ^{60}Co irradiator in New Jersey and its removal from the NRC contaminated site list is given in Ref. [29].

A further problem is that of decommissioning facilities that have large sealed sources. The original transport container in which the source was supplied may no longer be suitable to package the source for removal from the site, owing to changes in transport legislation, which can result in protracted delays in the removal of the source from the site if there is a need for the competent authority to approve a Type B container for transport. In the UK the decommissioning of a vacated hospital site was delayed for more than one year while awaiting the disposal of a teletherapy source. The hospital incurred significant additional costs because the land had been sold to a builder that was unable to take possession. The hospital vacated the entire site but had to employ 24 hour security to guard the radiotherapy department until the source could be removed [22].

3.4.3. Strategy for particle accelerators

Van de Graaff accelerators, linear accelerators, cyclotrons and synchrotrons are grouped together because they have some similar features and physical characteristics. For radiation shielding, accelerators are sometimes housed in large thick walled concrete structures, and the activation of trace elements in the construction materials gives rise to large quantities (possibly thousands of cubic metres) of low active waste. Decay periods to achieve the conditions that will allow removal from regulatory control can be several decades. Large accelerator facilities can become activated by neutrons to levels many times higher than the permitted release criteria (levels of up to 300 Bq/g have been reported). Decommissioning can thus give rise to large waste volumes [30]. In addition, there could be internal system contamination problems (e.g. tritium release from targets). A comprehensive analysis of particle accelerator decommissioning is given in Ref. [11]. The decommissioning of a medical cyclotron was successfully completed in 1993 and the site licence terminated [31]. A number of additional cyclotrons are under construction in Member States to meet the increasing demand for PET (positron emission tomography) radionuclides. In many instances these will need a phased decommissioning plan [22].

The above mentioned factors prompt a consideration of alternative decommissioning strategies similar to those for large nuclear facilities such as power or research reactors, which can include immediate or deferred dismantling to allow for the decay of activity. Numerous IAEA publications discuss optimum or preferred decommissioning strategies [1–4] and in addition there is an NRC publication on decommissioning non-fuel-cycle nuclear facilities that discusses decommissioning alternatives [32]. In general, if resources are available (i.e. financial and trained human resources) and there is an approved waste disposal route, the facility may be decommissioned promptly. In exceptional cases deferred decommissioning is viable provided that adequate safe enclosure conditions are established that meet regulatory and public approval. The cost of care and maintenance over the storage period should be considered [4].

Regardless of whether immediate or deferred decommissioning is chosen, it is desirable that a full decommissioning plan be prepared as early as possible to ensure that all relevant data, studies and proposals are recorded and to allow the interested bodies (e.g. regulators and environmental protection agencies) to give their approval.

For smaller, more compact devices, such as medical cyclotrons, the problem is less onerous, as many components can be removed intact to facilitate decommissioning. Immediate decommissioning is probably the best strategy. Decommissioning plans for smaller facilities are generally much simpler than for larger facilities.

3.4.4. Strategy for research facilities, hot cells, radiochemical laboratories and medical facilities

To this group of facilities belong, for example:

- Nuclear medicine, veterinary and pharmaceutical laboratories;
- University and research institute laboratories;
- Industrial research and development laboratories;
- Hot cells, gloveboxes and fume hoods.

All the above types of facility have a number of features in common that allow similar decommissioning strategies to be considered. The common features are that they are all likely to be contaminated with a wide range of radioactive material, which may have a variety of chemical forms that can influence their solubility and their ability to become airborne. Some material can also have bacterial or infectious contamination. Activation of enclosure walls or containment is not likely to be a problem. The facilities are likely to be extensive, covering large areas and numerous buildings.

Soil contamination of the site is also possible, especially at old facilities. Many older facilities are in various degrees of neglect and are legacies of decades of nuclear and medical research that began in the 1950s or 1960s without proper provision or consideration for decommissioning. There are also a number of much older facilities (pre-1939), particularly those built for the use and manufacture of radium-containing devices such as brachytherapy needles and luminous coatings on instruments [23, 33].

In general, most of these facilities exist in industrialized countries, although there are well used research facilities in some developing Member States for which decommissioning will also have to be planned for. As one example, in the USA the decommissioning of the above types of facility has received considerable attention. Up to about 1994 there was no clear guidance on when facilities should be decommissioned, and many stood idle for long periods of time. The NRC has more recently amended the licence rules to include the timeliness of submitting decommissioning plans and the completion of decommissioning. A maximum delay period of two years after shutdown has been specified by the NRC before it is obligatory to begin the decommissioning process [34].

In developed Member States there appears now to be a tendency to address the decommissioning of contaminated facilities on a more urgent basis. This generally tends to suggest a strategy of immediate decommissioning after shutdown or as soon as possible thereafter if facilities and sites have remained idle for many years. Examples of these projects are described later in this report. Increases in regulatory pressure and adverse public opinion have forced decommissioning strategies to be addressed.

In general, the increased degree of care and maintenance of shutdown facilities has resulted in a strategy of more prompt decontamination and dismantlement, unless there is a technical reason for further delay, such as the appreciable decay of radioactive material in the short term. The absence of a suitable waste disposal route is not usually considered a reason for delay. In many instances decommissioning has proceeded with waste being conditioned and placed into interim storage if, for instance, suitable disposal sites are not available [8].

It is concluded that the currently preferred decommissioning strategy for contaminated medical, research and industrial facilities is to perform dismantling activities promptly and to store waste in an approved manner until disposal routes become available.

3.4.5. Critical assemblies and small research reactors

The decommissioning strategies for critical assemblies and small research reactors may draw upon the experiences and lessons learned from the decommissioning of many similar reactors in recent years [1–3, 5, 6]. A useful checklist for decommissioning small facilities is given in Appendix III.

Critical assemblies are generally classified as having power levels of less than 5 kW, and therefore their radiological inventories are low. Consequently, the decommissioning strategy is straightforward. Following the removal of the fuel, dismantling and waste management can be carried out under the radiological control procedures that were in place for normal operations. The removal of the fuel will of course be planned, approved and implemented in accordance with the regulations set in individual Member States.

The radiological inventories of research reactors are usually not insignificant and the selection of the most appropriate strategy is a challenging process. When the fuel is evacuated most activity is associated with highly irradiated parts and components (e.g. reactor internals and vessels) or with components contaminated by strong gamma and/or alpha emitters, which requires additional shielding, which must be considered when setting up a decommissioning strategy. Indeed, the necessary tools and equipment for remote handling and for shielding must be available when dealing with this type of material. Access for material, waste and personnel must be taken into account. It is also important that the radioactive waste management policy and waste acceptance criteria be established before dismantling the reactor. The dismantling and size reduction processes often require significant investment costs, and sometimes infrastructure changes or modifications.

The decommissioning and dismantling of research reactors will produce a large amount of waste of various types. However, most material released from dismantling falls into a category of non-radioactive waste or potentially contaminated material that must be measured for clearance. Clearance levels are usually very low (of the order of a few Bq/g or tenths of Bq/g), which thus often means the use of decontamination processes. The measurement and characterization process for clearance is a specific and challenging topic. This subject is dealt with in Ref. [26]. A good example of decommissioning a small research reactor is that given for the Jason Argonaut reactor (10 kW) in Annex II.

For critical assemblies and small research reactors the decommissioning strategy can be significantly simplified if the final removal of the nuclear fuel from the facility is planned and carried out under the provisions in force during operation. The decommissioning safety case then focuses on the decontamination and dismantling of the facility, in which most of the waste can be classified as either short lived low and intermediate level waste (LILW–SL), cleared waste or material for reuse or recycling with no radiological restrictions. Experience indicates that very little long lived low and intermediate level waste (LILW–LL) is generated by small research reactors and critical assemblies, unless there have been problems with damaged fuel assemblies.

3.4.6. Strategies for large sites incorporating many diverse facilities

There are a number of large sites with diverse nuclear facilities in most of the industrialized countries that helped to pioneer nuclear technologies. Examples of these are Argonne, Cadarache, Harwell, the Japan Atomic Energy Research Institute, Karlsruhe, Mol and Oak Ridge. Facilities in this group include, inter alia, research reactors and hot cells.

Most large sites have generally been under State control, either for defence purposes, the power industry or general nuclear research. There are many published documents on decommissioning proposals, the progress of the decommissioning and partial works that have been carried out [7, 32, 35–40]. Some of these facilities are old and have been awaiting decommissioning for many years. It is observed that many of the sites have numerous relatively small, uncomplicated facilities, each of which could be classed as a small decommissioning project. The total amount of work and resources required for some sites is, however, extremely large and is generally being met by government funding.

The overall strategy being adopted for large sites is one of defining the problem and priorities, followed by extensive planning and the work being progressed slowly as funds and resources allow. The strategy adopted for each small decommissioning project can be taken from the various strategies outlined above for specific types of facility. The co-ordination of individual projects to account for common services and facilities such as waste treatment and interim storage is advantageous. Many sites will take 20 to 30 years, or longer, to remove from regulatory control. However, many useful lessons and experience is being gained from recent and current decommissioning projects.

3.4.7. Strategy for manufacturing facilities

There are diverse manufacturing facilities that use radionuclides as a component of their products. These products include smoke detectors, gauges, calibration and measuring devices of various types, as well as devices that use radiation sources for illumination, fluorescence, static elimination and well logging. There are two forms of material used in the manufacturing process. One form is completely sealed sources, for which the likelihood of leakage is low. The second is open sources, for which the risk of contamination may be significant. The strategies adopted for dealing with these forms are likely to be different. A licence or registration document may be used to regulate the manufacture and distribution of these types of material.

An example of the successful decontamination of a facility using radium for instrument dials is reported in Ref. [41].

The strategy for decommissioning typically includes those common features listed in Section 3.1.

Appendix III gives a more comprehensive list of the aspects to be considered.

3.5. STRATEGY FOR FACILITIES OPERATED BEFORE RADIOACTIVE MATERIAL CONTROLS WERE ESTABLISHED

When addressing a facility in which either no controls have existed or control has been lost, the essential first step is to characterize the nature of the hazards (both radiological and non-radiological) that currently exist there. Characterization activities may take many forms, including:

- A review of the historical records of the layout of the facility and activities at the facility, which materials were brought to the site, which materials left the site (e.g. products and waste) and the current conditions of structures, containers and piping systems.
- Interviewing former staff, if they are available.
- Scoping surveys to obtain basic data important to both characterization and to health and safety planning (e.g. direct exposure rates, oxygen levels and the presence of noxious or flammable gases).
- The collection of samples for analysis.

Planning for initial characterization activities often requires the primary emphasis to be on the safety of the workforce. High levels of personal protection (e.g. respiratory protection and protective clothing) may be needed for initial entries into an inactive or abandoned facility. In extreme cases utilities (lighting, electrical power, ventilation and communications) may also have to be supplied. As information is gathered on the facility and its current conditions, the scope of characterization activities and the health and safety requirements for them may change based on the hazards actually present.

Once sufficient information is gathered on the facility, the traditional approaches described for decommissioning planning in earlier sections may be utilized. However, the planning may need to include an assessment of the adequacy of existing facilities to support the decontamination (e.g. clothes changing facilities, decontamination areas and administrative support offices). If existing facilities are not adequate, planning for decommissioning will have to address how and where the needed facilities will be provided.

The planning effort may also be complicated by the availability of funding to support the decommissioning activities. Close co-ordination with the current (and possible past) facility owners, the competent regulatory authority and local government agencies is important to ensure the success of the project. In many cases the public may also have a strong interest in the status of the facility and plans for its decontamination. In such instances the use of a professional community relations professional is often necessary.

4. REGULATORY ASPECTS

4.1. INTRODUCTION

The possession of radioactive material for whatever use usually involves a licence or registration document issued by a regulatory body or national authority, unless the facility has a radiological inventory below the exemption levels. A licence or registration document for the possession of radioactive material establishes the identity of the licence holder (licensee) or registrant who possesses and uses it. Some details concerning the application for a licence or registration are given in Appendix I. The regulatory requirements for decommissioning may, however, vary widely within national legislation, but there are numerous IAEA documents that can be consulted [5, 6, 42–45].

An important step in the decommissioning process is to identify all the regulations, standards and laws that will be applicable to the decommissioning, including those for:

- The health and safety of workers and the public;
- Safe radioactive waste management practices;
- The criteria for removing controls on or for authorizing the reuse of material;
- The removal from regulatory control or the authorized reuse of the site or facility, as required;
- Industrial hygiene;
- The protection of the environment.

Reference [5] requires that a decommissioning plan “be developed for each nuclear facility [, unless otherwise required by the regulatory body,] to show that decommissioning can be accomplished safely”. This plan may vary in detail and complexity, but no matter how small or trivial the facility may appear a plan focuses all proposed activities in a single document for the information of all those concerned and allows the regulator to give the necessary approvals. It allows all safety and radiological risk issues to be resolved in advance of the decommissioning. It also ensures that waste management issues are adequately addressed. Planning also addresses the financial and resource needs. An example of the contents of a decommissioning plan is given in Appendix II.

During the decommissioning programme the regulatory body may audit to ensure compliance with existing national regulations. When decommissioning activities are near completion or finally completed the regulatory body reviews an application for site clearance or an authorized reuse of the site or facility.

The types of facility licensed to use radioactive material discussed in this report vary greatly. The extent of the regulations issued by authorities may also therefore be different. These differences will impact upon the planning and arrangements for the decommissioning.

4.2. LEGISLATION AND DOCUMENTATION

In many countries legislation is comprised of primary legislation (the law) and secondary legislation (regulations), the provisions of which are legally binding. Primary and secondary legislation are usually supplemented by guidance documents and codes of practice. The manner in which laws, regulations and guidance documents are combined to provide an effective regulatory framework for decommissioning and waste management is a matter for the State concerned.

Typical examples of legislative frameworks in two Member States are given in Annexes I.C and I.F.

There are numerous documents produced for international use by organizations such as the International Commission on Radiological Protection [46] and the IAEA [42, 44, 47]. As one national example, in the USA the NRC produces numerous regulatory guideline documents, for example for licensees on the standard format for decommissioning plans [48].

Another type of document that has been developed to comply with regulations is a national register of sealed radiation sources. Such a register can be used by the authorities as a check for the regular control of sources that may lead to serious risks in their handling, storage or disposal. It is often the responsibility of the regulatory body or other competent authority to compile and maintain the register of sources. The IAEA has developed a simple computerized registry, which is available to Member States. The registry system, the Sealed Radiation Sources (SRSs) Registry, has been specifically designed to track and store relevant data about SRSs [49].

The transport of radioactive waste is also subject to close regulatory control. This may involve different authorities than those regulating fixed installations. The IAEA has produced detailed recommendations for ensuring the safe transport of radioactive material [50–52].

4.3. PENDING ISSUES

Although regulatory requirements for the design, construction, operation and maintenance of nuclear facilities have been established in many Member States, the specific requirements for decommissioning have yet to be developed or finalized in some of them. In the absence of such requirements the regulation of decommissioning could be subject to the regulations for operational facilities. In general, experience shows that the analogous application to decommissioning of the regulations intended for the construction or operation of nuclear facilities may result in a convoluted approach, be overly bureaucratic and may, eventually, result in delays and a loss of effectiveness [53].

It is necessary for policy makers to look at the decommissioning and management of spent sealed sources in the small user sector. Priority should be given to considering the decommissioning of small facilities at the beginning of their operation, in particular the provision of funding to meet the costs of the decommissioning when the time arrives. Legislation is important to encourage users to manage such liabilities. Large sealed sources are supplied for irradiators in the full knowledge that their activity levels may still be of the order of TBq at the end of their working life. A possible approach for policy makers and source manufacturers is to build in a system in which charges for the eventual return of the redundant source to the manufacturer are set as part of the initial purchase, with further payments at appropriate intervals to keep pace with the eventual projected disposal cost. This is especially important as many sources are in use for periods substantially longer than the initial recommended working life. This issue is, however, controversial and no firm positions of generic applicability have yet been established.

5. PLANNING AND MANAGEMENT

5.1. GENERAL ASPECTS OF MANAGEMENT

The licensee or responsible person often appoints an agent or project manager for the decommissioning of a small facility, regardless of whether the project is to be undertaken in-house or by contractors. For very small facilities the project manager may be the facility manager or radiation protection officer. If the project manager is appointed or nominated before or at the latest as soon as the facility is shut down, this will ensure the continuous and committed management of, motivation for and understanding of the requirements of the programme. Depending on the magnitude of the project this may or may not be a full time activity (e.g. the disposal of disused sources may not be a full time activity, while the decontamination and dismantling of a large research laboratory will be). The project manager, acting on behalf of the licensee or registrant, would be in charge of all aspects of the planning, management and implementation of the decommissioning. Assistance should be given to the licensee or registrant in negotiations with regulators and inspectors on safety and regulatory matters. The licensee, however, always remains responsible for achieving the legal aspects of safe decommissioning and waste disposal.

A problem that often arises when appointing a project manager is that the person with the most knowledge of the facility may not be committed to or in favour of its decommissioning, owing to the disruption to staff that usually results. This person may also not necessarily have the right expertise in decommissioning. It is desirable that these factors be recognized by higher management. Aspects of decommissioning management are outlined in Refs [2, 3]. Particular guidance for the safe decommissioning of medical, industrial and research facilities is given in Ref. [5] (see also Appendix III).

As one national example, to assist and give guidance to those engaged in the planning for the decommissioning of non-reactor facilities the NRC has published a number of guides, which are discussed in Ref. [35].

5.2. TECHNICAL PLANNING AND MANAGEMENT

Early planning for decommissioning is endorsed by the IAEA [5]; this planning should lead to a formal decommissioning plan that describes which activities are intended. The technical aspects within the document or plan are important for ensuring that all those involved are aware of the goals and the methodology to achieve those goals. Approval of the document or plan will generally allow financial and human resources to be allocated to the project. The regulatory aspects of a plan are dealt with in Section 4.1.

The typical contents of a comprehensive decommissioning plan are given in various IAEA publications [1, 2, 5]. The contents of a typical plan are reproduced in Appendix II. One example of planning for the decommissioning of an industrial facility and the related regulatory interactions is given in Ref. [54]. The level of detail will vary according to the type of facility and national legislation and regulations.

The technical topics and responsibilities to be considered in implementing and completing a decommissioning project successfully and within budget include those listed below (some of these topics, however, may not apply to very small, simple facilities):

- Definitions of job specifications for key staff;
- The identification and appointment of the decommissioning team members;
- Assigning roles and responsibilities to all parties, including contractors;
- Setting qualifications and undertaking training;
- Routine inspections and maintenance;
- The specification of work packages (for in-house work or for outside contractors);
- Work progress and reviews;
- Data collection, records and reports;

- Licensing and regulatory aspects;
- The selection and acquisition of special equipment;
- Safety management, including radiological and non-radiological hazards;
- Emergency planning;
- Establishing the necessary and appropriate quality assurance programme;
- Project completion records and archiving.

In addition to the technical planning and management of the project there will be a need for administrative support. The administrative management should address the following:

- The provision and approval of the project budget;
- Equipment procurement;
- The recording and monitoring of expenditure;
- Cost and schedule control;
- The placing and control of contracts (if any);
- Personnel services;
- Publicity and external communications, if necessary;
- The management of a records database.

For all decommissioning work, planning is needed, resources appropriate for the project need to be allocated and there needs to be a clear allocation of responsibilities. Even the relatively simple disposal of spent sources needs to be planned, costed and implemented in a responsible manner.

There are sometimes benefits to be gained from a pilot study being carried out before proceeding to the full decommissioning of a facility. One example is given in Ref. [55], in which some uranium glazed tiles were removed from a wall as a pilot study in order to assess the hazards and potential spread of personal and work area contamination.

5.3. INDUSTRIAL SAFETY

Decommissioning activities can and often will give rise to varying conditions in the facility concerned and dangers may not be immediately apparent. However trivial the decommissioning activities may seem, identifying potentially hazardous conditions and practices permits planning and preparedness for abnormal events. Physical hazards can result from, for example, uneven surfaces, operations on elevated platforms, noise and extremes of temperature. Dangerous chemicals may be present as a result of the operations of a facility or brought on-site for decontamination operations. Biological hazards include disease-causing organisms, toxic plants, biting insects and animals. Other hazards include fire, flooding, dropping loads and the theft of material.

Failing to identify the hazards present and to implement programmes to address them has resulted in serious consequences. Control programmes can in many cases be implemented and, in other cases, personal protective equipment may be used. It may even be necessary to change the way decontamination operations are conducted (e.g. taking frequent breaks to avoid heat stress). These programmes have been very effective in reducing the adverse impact that disease and injury can have on decommissioning operations.

5.4. TRAINING

Training is an essential part of an effective decommissioning programme. It is often assumed that, because of their intimate knowledge of the facility and equipment, experienced facility operators will suffice for a decommissioning. This is often not the case, however. A particular problem that often arises is the predictable resistance to a final shutdown of a facility and the resulting loss of jobs. This needs to be recognized by management and appropriate operators should be selected as candidates for the decommissioning and the necessary training and motivation should be provided.

It is important that contractors and subcontractors employed on a decommissioning project be adequately trained and demonstrate the necessary experience. The entire workforce for a decommissioning project should be derived from suitably qualified and experienced personnel.

The complexity of a decommissioning project will depend on the radiological inventory and on the physical complexity of the facility to be dismantled. While radiation protection training is a mature discipline and well prescribed [56], industrial training requirements are less well defined. It is therefore important when developing a training programme for decommissioning to carry out an initial job analysis to provide an appropriate and systematic approach to training.

Training can be general, management level or specific, the latter relating to the technical aspects of the decommissioning project. The first two aspects (general and management) are well covered within the training activities of the IAEA, especially in regional training courses. In addition, the safety guides and technical documents of the IAEA can be used in developing general and more specific training requirements. It is important that the specific training requirements of a training programme be appropriate for the tasks to be undertaken, which can be achieved by reference to the job analysis process discussed above. Some of the following aspects relevant to the training to be carried out during the decommissioning of a small nuclear facility [57] may be considered:

- General training required for unescorted access to the facility; for example, facility orientation, contractor safety orientation, the conduct of operations,

basic general and radiological safety, hazardous waste and environmental safety.
- Hazard recognition for construction.
- Radiological protection training for health physics and related functions, including site characterization and decontamination [58].
- Radioactive waste labelling, inspection and packaging.
- Industrial safety, including fork-lift truck and crane operations, rigging and hoisting.
- Industrial safety, including the hazards of materials (e.g. lead and asbestos) and working in confined spaces [59].
- Specialist training for various tools and equipment.
- Fire protection and security.

The list above is not exhaustive, or in a specific order of importance, and other factors, such as nuclear criticality safety and/or emergency planning training, may be included if required. Finally, it is important that supervisors and managers of work programmes within the project (i.e. for health and safety) be duly authorized and approved by the licensee or registrant.

5.5. EMERGENCY PLANNING

As in other human activities, incidents or accidents can occur during the decommissioning of a nuclear facility. At the beginning of a decommissioning project the risks of incidents or accidents are similar to those that existed when the facility was operated. Subsequently, with the progress of the decommissioning project, some risks decrease or disappear. This is the case, for example, after defuelling or the removal of sources. Other kinds of risks increase by, for example, the spread of contaminants during cutting activities or in the transfer of waste.

Appropriate and comprehensive procedures addressing the actions to be undertaken in the event that incidents or accidents occur are the best way to proceed in order to minimize the occurrence of such events, to limit their consequences and to manage the situation. Setting up an emergency plan for a decommissioning project generally involves the national and local emergency services as well as regulatory bodies and other competent authorities. Such services may include the police, fire departments, ambulance services and hospitals. However, it is likely that in a decommissioning project for a small facility most emergencies can be dealt with on the site.

It is important that the media and the public be kept informed of the facts during an emergency, and a person may be nominated in the emergency plan to deal with communication.

5.6. SITE SECURITY

In establishing the preferred decommissioning option, alternatives are often evaluated to determine their impact on site security and physical protection. If a facility contains material that could be considered valuable and consequently at risk of theft, then increased security for this material and early safe storage or disposal are in order. Many of the cases cited in Ref. [9] have been due to the unauthorized removal of material from a site. Security is therefore always a concern on a decommissioning site.

5.7. RECORD KEEPING

An important part of decommissioning is records and record keeping. Experience has shown that in many cases the records for nuclear facilities have been badly managed. There is a historical problem if records from earlier periods (e.g. the design, construction and operation of a facility) have been lost or not adequately documented. There may also a problem with an overwhelming volume of records for facilities if everything has been kept but not properly indexed and recorded. The specific records for decommissioning need to be clearly identified.

The type and quantity of the records associated with decommissioning will vary considerably between the various types of facility described in this report. The selection of the appropriate records and the system to manage them are both important.

The IAEA has produced a technical report on record keeping guidelines and experience for decommissioning purposes [60] to give guidance on this subject. Failure to give adequate attention to records management is likely to lead to delays and additional costs for a decommissioning and unnecessary risks from unexpected hazards. It may also be difficult to satisfy regulatory requirements with inadequate records.

One specific example is given in Ref. [22] to illustrate the problems caused by poor record keeping. It was reported that, in disposing of old laboratory equipment, mistakes were made that resulted in heavily contaminated laboratory refrigerators ending up in scrapyards or on landfill sites. In this particular case laboratory workers using ^{125}I and ^{57}Co were not aware that, ten years earlier, GBq quantities of high specific activity tritium radiochemicals were in use. It is now common practice that refrigerators are gamma surveyed with suitable instruments; wipe tests and beta counting of water from defrosting are necessary to detect tritium prior to disposal.

6. TECHNICAL ASPECTS

6.1. GENERAL ASPECTS

Small facilities are diverse in terms of their design, construction and operation; they also use a wide range of radionuclides. The physical condition of a facility awaiting decommissioning may vary from one of neglect to well managed. All these factors will influence the technical problems that will be encountered.

6.2. DECOMMISSIONING OF FACILITIES CONTAINING SPENT SEALED SOURCES

Sealed sources are used in a large variety of equipment. The hazard presented by a source varies according to the isotope it contains and the total activity present. When in service, the source is usually protected by the installation and surrounding equipment. When removal is required, a consideration is whether the device is to be removed intact with its shielding or whether a separate transport container for the source is required. A particular problem faced is that when sources are removed after their use they may be stored in unsatisfactory conditions. If sources are stored in uncontrolled environments for long periods then the degradation of the devices and their support systems is likely to become a problem. Long periods of storage may also result in a loss of their identification and of the appropriate records. There may no longer be a possibility of their return to the original supplier. In general, it is never recommended to dismantle these sources, as this operation requires special facilities. Various strategies present themselves, as outlined in Section 3.4. If interim storage has to be resorted to, then the design of the storage facility will usually undergo a regulatory review and approval. Particular aspects of the design of an interim storage site include its ventilation, security, inspection, maintenance and radiological surveillance. The types of container to be used will also need to be considered, along with the plans for the eventual emptying of the storage site. The use of an interim storage site at a facility does not allow decommissioning to be fully completed.

If all sources are identified and characterized and are in a good physical condition (i.e. not leaking) then decommissioning involves only their safe handling and transport. If, however, the sources are leaking or suspected to be leaking then more complex decommissioning techniques will have to be applied. This generally involves radiological surveys and the design and provision of secondary containers and decontamination techniques, including for the handling of contaminated items and material. All these techniques are well established and advice can be obtained from numerous organizations and publications [19, 21].

Although final disposal is the preferred option for sources containing long lived isotopes, it may be necessary to use interim storage if approved disposal sites are not available, in which case suitable containers and a location will have to be established. Sources containing short lived isotopes may also have to be stored to allow for decay.

Special conditioning techniques are required for spent radium sources and their facilities, which are described in detail in Ref. [20]. Spent radium sources should be stored in an appropriate interim storage area that has strict control for access and radiation monitoring.

6.3. DECOMMISSIONING OF MORE COMPLEX FACILITIES

This section describes the decommissioning of more complex facilities, such as small research reactors, particle accelerators, cyclotrons and irradiators. These types of large and slightly more complex facility are grouped in this section because it is believed that similar decommissioning techniques can be applied to all or some of them. The particular distinctive features of each type of facility are outlined below.

The technical aspects associated with the decommissioning of small reactors have been well documented [1]. The most significant technical aspect at the first stage of decommissioning small reactors is the removal of the spent fuel to an interim storage facility and the final transport of the fuel to storage or to a repository. In some cases, if new provisions for the storage and transport of the spent fuel are required, a new criticality assessment may have to be carried out. Other key technical aspects associated with the various stages of decommissioning are:

- Initial radiological characterization;
- The dismantling of reactor internals;
- The disposal of operational fluids, including primary and secondary coolants;
- Personnel access and clothes changing facilities;
- The decommissioning of secondary containments;
- Waste management;
- Temporary storage areas;
- Waste transport;
- Decontamination facilities;
- The decommissioning of ventilation systems;
- Radiological monitoring.

The design of the reactor will lead to the choice of the specific technique to be used for the dismantling of small research reactors such as pool reactors (e.g. TRIGA, SLOWPOKE and MNSR reactors), tank reactors or graphite reactors (e.g. Argonaut type reactors). Details of the techniques that may be used are given in Ref. [1]. The dismantling of activated concrete may have to be considered.

As stated previously, the decommissioning of critical assemblies is straightforward, as the radiological inventory is much lower than for research reactors. The technical aspects of decommissioning may incorporate some or all the aspects given above.

The decommissioning of particle accelerators and cyclotrons is addressed in Ref. [11]. One common characteristic associated with their decommissioning is that material with (very) low levels of activation has to be managed (i.e. generally less than 300 Bq/g for metal pieces of particle accelerators and less than 100 Bq/g for infrastructure materials such as concrete and reinforcement rods). The use of recycle and reuse techniques greatly minimizes the volume of radioactive waste that must go to a regulated disposal facility. An overview of these techniques is given in Ref. [61]. Some of the decommissioning techniques listed in Ref. [1] can also be used in the decommissioning of particle accelerators and cyclotrons.

The decommissioning of irradiators is quite straightforward once the sources have been sent back to the supplier or disposed of properly by other means. Indeed, the main specific decommissioning aspects are then limited to the management of the water in the pool (as radioactive waste, if it became contaminated during operation) and the characterization of and, if necessary, the decontamination of the infrastructure.

The decommissioning practices for facilities such as small research reactors, particle accelerators, cyclotrons and irradiators are similar to those that have been developed for the decommissioning of larger research reactors and associated research facilities. There is an abundance of published data and reports on successful small decommissioning projects. These descriptions should be treated as generic guides because techniques for small facilities are likely to be simpler. Nevertheless, the use of an optimal decommissioning technique meets both the ALARA (as low as reasonably achievable) principle and ensures the minimization of both primary and secondary waste. Reference [32] gives practical information and guidance. Examples of successful decommissioning projects are given in the annexes of this report.

6.4. DECOMMISSIONING OF RESEARCH FACILITIES

The main types of equipment used in medical, pharmaceutical, university and industrial research facilities are fume hoods, gloveboxes and hot cells. Contamination rather than radiation is often the main concern. The decommissioning of such equipment has to be prepared carefully. Research projects using different types of radioactive material (not always in a similar chemical form) may have been carried out by several people using the same equipment. Without well defined and controlled procedures for care and maintenance, this can lead to the presence of a wide range of contaminants.

An example of unsatisfactory care and maintenance after a research project is to fix the contamination by means of a coating, instead of removing it. This is exacerbated by inadequate records or history of use. At the end of the lifetime of the equipment those who do the decommissioning are faced with several contaminated layers of different coatings, which generally leads to difficulties in the decontamination of the equipment. Specific decommissioning actions are:

- To remove any potential sources of radiation exposure remaining on the equipment;
- To carry out a preliminary decontamination process to decrease the risks and exposure of the workers during the subsequent decommissioning phases (i.e. at least eliminating removable contamination);
- To perform a dismantling of the hot cells and gloveboxes in a ventilation controlled chamber or tent or, for fume hoods, in a controlled area;
- To minimize waste production by addressing appropriate waste management techniques (i.e. recycle and/or reuse, decontamination to meet release criteria or removal as radioactive waste).

The recommendations stated in Section 3 for the selection of decommissioning strategies and techniques are also appropriate for research facilities.

Successful decommissioning projects have been reported. For example, the decommissioning of the facility for the production of radioisotopes for medical, scientific and industrial applications at Oak Ridge National Laboratory has been carried out without significant technical problems [62]. Boston University Medical Center has also published a report on its successful decommissioning experience [63].

Other successful decommissioning projects have been reported at the Argonne National Laboratory [64, 65]. These were for a waste ion exchange facility building and a contaminated pneumatic transfer tube. No special techniques other than the use of some pipe cutting equipment were used. Two minor electrical problems were reported, as well as a problem with water contamination in the pipe trench, which forward planning could have avoided.

Another successful project for dismantling a heavily contaminated hot cell, as well as research and development laboratories, carried out by the Belgian Nuclear Research Centre (SCK•CEN), is reported in Ref. [39] (see Annex I.A). It can be noted that conventional cutting and decontamination techniques were used. A detailed account of decommissioning an industrial research, development and production facility in the Quehanna Wild Area in Pennsylvania is reported in Ref. [40].

6.5. CONCLUSIONS

From the discussion in this section and the documents cited it can be seen that the dismantling and decontamination of small facilities does not in general present significant technical problems. This is particularly true if experienced operators are employed together with the appropriate equipment and facilities in accordance with an established plan. In some cases decommissioning is best done by those having existing experience with the facility, but motivation and training are always important. Where it is necessary to conduct on-site decommissioning operations there is a large amount of successful experience to draw upon.

7. SAFETY IN DECOMMISSIONING

The IAEA has published a safety guide on the decommissioning of medical, industrial and research facilities [5]. The radiological protection of workers involved in decommissioning activities is addressed in Refs [44, 66]; the complementary safety guidance on radiation sources is given in Ref. [67]. Pertinent extracts from Ref. [5] are given below.

Section 2.8 of Ref. [5] states that “During all phases of decommissioning, workers, the public and the environment should be properly protected from hazards resulting from the decommissioning activities. A thorough safety assessment of the hazards involved during decommissioning (including accident analysis, where necessary) should be conducted to define protective measures, part of a defence in depth system that takes into account the specifics of decommissioning. In some cases, such measures may be different from those in place during the operation of the facility.”

Section 2.9 of Ref. [5] states that “Decommissioning of nuclear facilities often involves the removal, at an early stage, of significant quantities of radioactive material, including sources and operational waste. Even after this step, the total contamination and activation of the facility has to be taken into account in the safety assessment.”

Section 2.10 of Ref. [5] states that “Activities such as decontamination and the progressive dismantling or removal of some existing safety systems are also of importance. These activities have the potential for creating new hazards. An important objective during decommissioning is, therefore, that the safety aspects of such activities are adequately assessed and managed so as to minimize the impact on safety.”

Section 2.11 of Ref. [5] states that “In the course of decommissioning, consideration should be given to the radiation protection of both workers engaged in the decommissioning operations and the public who may be exposed to radiation from discharges to the environment, from the release of solid materials, and as a result of any subsequent occupancy of the decommissioned site.”

8. SPENT FUEL MANAGEMENT

The quantity of spent fuel from small research reactors is modest but can present problems. If no off-site disposal route is available, then interim storage must be considered either on or off the site. Storage in the existing reactor pool is not encouraged, as this will interfere with and delay decommissioning. Both wet storage (for example for the Jason reactor) and dry storage (for example for the École Polytechnique, Montreal, SLOWPOKE first core) have been successfully demonstrated at small research reactor facilities.

As stated previously in this report, the removal of the spent fuel, under the provisions in force during operation, prior to decommissioning simplifies the decommissioning task. Decommissioning cannot reasonably be started until the spent fuel and sources have been dealt with.

Developments worldwide have resulted in a situation in which the removal and disposal of spent fuel has become a serious problem. Many research reactors, for example, were provided with their fuel by suppliers in other countries, to which the operators of the reactors had planned to return it when spent. In many cases, however, this has now become impracticable. As this situation was unforeseen, few operating organizations have their own off-site spent fuel storage facilities. There is also often a delay period for cooling and decay before the fuel can be moved, which will affect decommissioning plans.

A return programme for fuel of US origin is presently underway for a fixed period, requiring a final discharge from reactors during 2006 and a final shipment to the USA during 2009. A return programme for fuel of Russian origin is currently being considered.

For long term fuel storage, factors such as its physical form, chemical composition, enrichment and burnup, and potential criticality, as well as any damage incurred during operation, are important. Two additional conditions for fuel storage are the ability to remove decay heat from the fuel storage area, if necessary, and maintenance of the water quality if the fuel is stored in water. Poor water chemistry could lead to fuel cladding failure, which could complicate fuel shipment and storage area decommissioning [1].

Spent fuel may not previously have been removed from the site of some research reactors and hence a spent fuel transport cask may not be available locally. If none is available in a given State, then a cask will have to be designed and manufactured. Alternatively, it can be hired or purchased from another State. Also, appropriate licences and/or authorizations for the use of the cask should be obtained from the regulatory body, and a specific safety assessment may be needed to cover the transport and reception of the fuel into storage [1].

9. WASTE MANAGEMENT

9.1. GENERAL

The three major objectives of waste management within the context of this report are:

- To minimize quantities of radioactive waste at all stages of decommissioning;
- To prevent the mixing of waste of different categories;
- To comply with all applicable regulations in the handling, storage and disposal of the waste.

There are numerous publications available that give guidance on the conditioning and packaging of waste for storage and disposal. In particular, a number of reports on waste management for small users have been published by the IAEA [20, 45, 68–72]. Relevant safety aspects are also dealt with in Ref. [5]. The principles of waste management are given in Ref. [73].

There are special risks associated with the management of decommissioning waste. Decommissioning waste may be different from operational waste and involve different handling, processing and disposal techniques.

The handling, treatment, conditioning, storage, transport and disposal of waste are generally well regulated within the nuclear industry. The methods applied to decommissioning waste covered by this report will, in general, be similar to those used in other parts of the nuclear industry during the operation, management and refurbishment of facilities. However, the logistics related to the management of waste from decommissioning are somewhat different from that from operational waste programmes. Decontamination may result in the generation of secondary waste. Decommissioning itself results in the generation of radioactive solid and liquid waste (for example equipment, components, sealed sources, scrap and cutting fluids). The rate and volume of waste generation can be much higher than during operation, but this, of course, is dependent on the type of the facility concerned.

Decommissioning activities can be only initiated when well defined waste management arrangements have been made. A summary of the waste management plan is usually part of the decommissioning plan presented to the regulatory body. The waste management plan is based on national radiological and waste management regulations, if available, or on international recommendations.

9.2. DECONTAMINATION AND WASTE MINIMIZATION

An important requirement for decommissioning is to minimize the generation of radioactive waste. This can be achieved by a careful study of the operational history of the facility and planning waste management before the decommissioning commences. Judgement, experience and characterization will determine the best way to minimize the waste volume, which can be by, for example, decontamination or segregation for recycling, reuse or removal from regulatory control by clearance.

The choice of a decontamination technique is determined by the following considerations:

- The nature of the radionuclide to be removed (its type and physical and chemical forms);
- The nature of the material to be decontaminated;
- The release criteria that must be met;
- The fate of the decontaminated material (i.e. release, recycle, reuse or disposal as waste);
- The location of the decontamination operation;
- The nature, quality and treatment of the secondary waste;
- Operator exposure and safety;
- Cost.

Decontamination sometimes involves elaborate equipment and operator action, while segregation can require expensive measurement equipment; although the objective is to reduce waste quantities, the cost, exposure and risk to the operator must be taken into account. It is desirable that the selection of the methods and material (solvents) for decontamination accounts for secondary waste, in order not to have waste different from the normal or operational radioactive waste.

The principle of ALARA is used to compare risks to operators and the public as compared with savings in waste volumes and costs. The application of ALARA requires a monetary value to be put on dose, which involves input from the regulator. Judgement should be exercised on the extent to which the cost–benefit analysis is applied [46].

Dismantling techniques can affect the quantity of waste generated. It is important to plan all cutting operations so that cutting and dismantling is optimized to avoid cross-contamination or the spread of contamination and thus achieve a minimization of the waste produced.

9.3. RADIOACTIVE WASTE CATEGORIES

It is necessary and common practice to classify waste into different categories. The categories are related to the hazards and risks to humans and the environment. Different waste materials present different degrees of hazard. Some types of material contain short lived radionuclides, while other types contain very long lived radionuclides. Some types of material are very mobile and easily dispersed, while other types are fixed. The categories and activity levels in each category tend to differ in different States [1]. To try and harmonize such national categorizations the IAEA promulgated a standard categorization in Ref. [74], which is shown in Table VI. The

TABLE VI. TYPICAL CHARACTERISTICS OF WASTE CLASSES [74]

Waste class	Typical characteristics	Disposal options
1. Exempt waste (EW)	Activity levels at or below clearance levels that are based on an annual dose to members of the public of less than 0.01 mSv	No radiological restrictions
2. Low and intermediate level waste (LILW)	Activity levels above clearance levels and thermal power below about 2 kW/m^3	
2.1. Short lived waste (LILW–SL)	Restricted long lived radionuclide concentrations (limitation of long lived alpha emitting radionuclides to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g per waste package)	Near surface or geological disposal facilities
2.2. Long lived waste (LILW–LL)	Long lived radionuclide concentrations exceeding the limitations for short lived waste	Geological disposal facilities
3. High level waste (HLW)	Thermal power above about 2 kW/m^3 and long lived radionuclide concentrations exceeding the limitations for short lived waste	Geological disposal facilities

types of facility under consideration in this report are likely to yield waste in all categories, including high level waste (HLW) (this includes minor amounts of HLW and spent fuel fragments that may exist even in small facilities, for example in laboratories in which analyses of waste matrices and spent fuel are carried out). Other terminologies and classifications are used in other countries [2, 75]. It should be noted that criteria for exempt waste are being developed by the IAEA [76] and other international organizations [61]. Reference [45] is particularly applicable to the release and/or disposal of airborne, liquid and solid waste from small nuclear facilities and hence is of relevance to this report.

The waste arising from the decommissioning of small facilities can be characterized in various ways:

- By the quantity and type of its content (e.g. whether it is low or intermediate level waste);
- By its physical and/or chemical properties (e.g. whether it is a solid or liquid waste and/or whether it is an aqueous and/or organic liquid);
- By its suitability for particular waste treatment techniques (e.g. whether it is compactible or non-compactible, combustible or non-combustible).

Waste classification is important to:

- Reduce hazards from different classes of waste;
- Minimize volumes of waste;
- Separate waste into separate categories for its handling and packaging;
- Distinguish low hazard and high hazard activities;
- Determine shielding and containment requirements;
- Meet the regulations for the transport of waste;
- Meet the regulations for clearance, waste treatment, storage and disposal (e.g. shallow burial or disposal in deep geological repositories);
- Reduce the costs of waste handling, storage, transport and disposal.

The quantity of waste should therefore be estimated during the planning of the decommissioning, then categorized and segregated by measurement and sampling as soon as it is generated during the decommissioning activities. Some waste may be suitable for disposal in normal landfill sites and some material, such as steel and concrete, may be suitable for recycling or reuse outside the facility.

9.4. GENERIC WASTE MANAGEMENT STEPS

An overall waste management scheme generally includes several or all of the following steps:

- Minimization and segregation;
- Characterization and monitoring;
- Treatment;
- Conditioning;
- Transport and disposal.

It may be necessary to store waste between the above steps, and record keeping is required for every step. If the steps in waste management are complementary and compatible with each other, then a more satisfactory overall strategy arises.

Minimization and segregation take place during the decommissioning of a facility (during the dismantling and decontamination activities). The minimization and segregation of radioactive waste is discussed in Ref. [15].

The remaining steps take place during the subsequent waste management activities. Figure 5 [14] shows an overall scheme for waste management from waste generation (dismantling) through to disposal. Some waste (e.g. LILW–LL) can generally only be taken into an interim storage facility if disposal sites for long lived waste do not exist. The following sections provide generic information on the management of typical waste streams in the decommissioning of small nuclear facilities.

9.5. MANAGEMENT OF WASTE CONTAMINATED BY RADIONUCLIDES WITH A HALF-LIFE OF LESS THAN 100 DAYS

An effective and economical waste management procedure is to allow for the waste to decay. This is a procedure in which waste containing small amounts of short lived radionuclides is stored until the activity naturally decays to such a level that it can be considered inactive in compliance with the national regulations.

As one example of national regulations the NRC has concluded that material with a half-life of less than or equal to 120 days is appropriate for decay in storage. The minimum holding time is ten half-lives of the longest lived radioisotope in the waste. Such waste may be disposed of as ordinary refuse if radiation surveys (performed in a low background area without interposed shielding) of the waste at the end of the holding period indicate that radiation levels are indistinguishable from the background level [77].

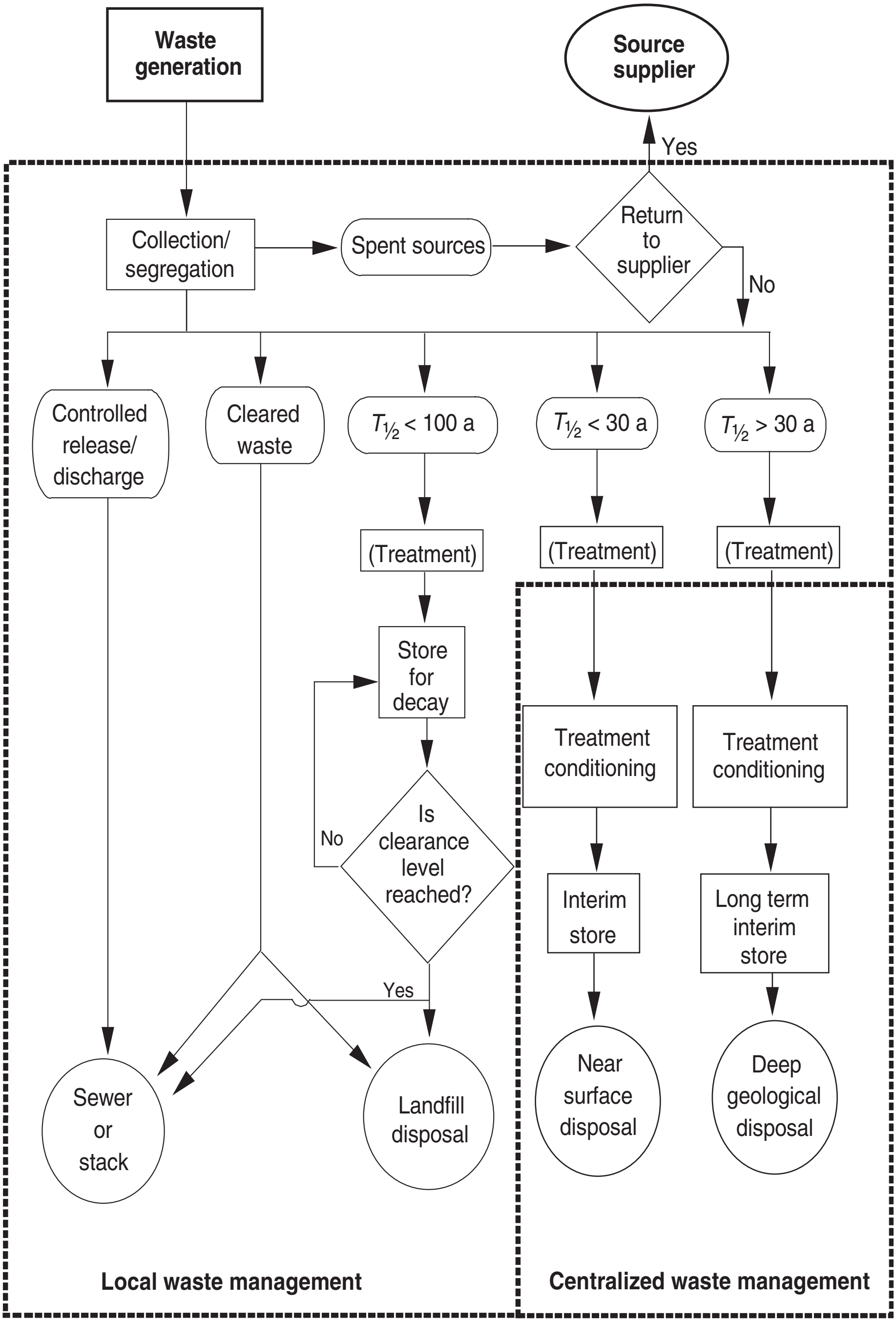

FIG. 5. An example of a waste management strategy [14].

9.6. MANAGEMENT OF SPENT SEALED SOURCES

In a State with an inadequate and incomplete radioactive waste management infrastructure there may be a need to apply simple but effective methods of enhancing the safety of spent sealed sources.

A technical manual for the handling, conditioning and storage of spent sealed sources has been published by the IAEA [21] that contains technical procedures for the conditioning of spent radioactive sources. It also describes the means to assure the quality of the resulting package and discusses the measures to prepare waste packages with the flexibility to accommodate possible future disposal requirements.

A potentially suitable method of securing small spent sources from loss or theft is to encapsulate the spent sources or source holders in a suitably sized metal drum. A convenient way to embed the source would be to place it in the centre of a drum and to fill the drum with a suitable matrix (e.g. cement). Appropriate technology and quantity control will ensure the long term stability of the package. When immobilizing spent sealed sources the need for security in storage and the long term retrievability of the drum is important.

9.7. MANAGEMENT OF OTHER RADIOACTIVE SOLID WASTE

The types of solid active waste dealt with in this section are restricted to those arising from the dismantling of small facilities. They are generally classified as LILW. The major types of solid material arising from dismantling operations are:

- Combustible waste, such as protective clothing, wood from ventilation hoods and laboratory furniture;
- Low to high active metallic waste, such as reactor internals, reactor pressure vessels, primary pumps, tanks, valves, structural materials, electric cables and contaminated instrumentation;
- Mass concrete waste from slightly activated or contaminated slabs, floors, shielding walls and room walls;
- Concrete and bricks as super-compressible rubble from the demolition activities of activated or contaminated material;
- Various light material, such as thermal insulation;
- Special waste, such as lead bricks, graphite, beryllium, shielding or waste that contains asbestos.

Incineration may be the most efficient volume reduction procedure for combustible waste if incineration facilities are available and the regulations allow it. The decommissioning of incinerators at the end of their useful life is a special problem that requires attention. One example of the decommissioning of incinerators is given in Ref. [78].

Compaction is a well demonstrated and proven volume reduction method that ideally fits the needs of solid, compactible waste. A well proven method for conditioning non-compactible waste is encapsulation in a cement matrix. The technical aspects related to the treatment and conditioning of radioactive solid waste are described in more detail in Ref. [68].

9.8. MANAGEMENT OF RADIOACTIVE LIQUID WASTE

9.8.1. Aqueous liquid waste

Decontamination activities during the decommissioning of small facilities can generate a range of aqueous waste streams that need treatment and conditioning. The need for decontamination is likely to occur if operations with unsealed sources have been carried out (e.g. in hot cells). Radiological characterization activities may also generate aqueous waste streams when equipment is cleaned or surplus sample material is handled. The low activity concentrations associated with these types of waste will generally permit contact handling and avoid the need for shielding. The selection of the treatment for liquid waste involves a set of decisions related to factors described in Ref. [69], which is a technical manual that provides reference material and direct step by step assistance.

9.8.2. Organic liquid waste

A number of options are available for the treatment of organic liquid waste such as scintillation cocktails, organic extractants, contaminated oils and miscellaneous solvents. Incineration is potentially the most attractive option for the complete destruction of organic material, and provides a large volume reduction [70]. However, because of the significant investment costs it may not be justified to install an incinerator for small volumes of organic liquid waste. Absorption on an inert absorbent material followed, if necessary, by cementation is an option in many States, even though this leads to a volume increase.

9.9. MANAGEMENT OF RADIOACTIVE WASTE CONTAINING OTHER HAZARDOUS MATERIAL

A final consideration is of waste for which no direct solution exists or for which there are other concerns to be addressed, such as the presence of toxic waste or material that is otherwise dangerous to handle. Research facilities and non-power applications often produce such types of waste.

Within such a category of waste the most frequent types encountered in non-power applications are:

- Alkali metals (e.g. sodium and potassium);
- Graphite;
- Beryllium;
- Lead;
- Asbestos;
- Various types of organic material (e.g. methanol and polymers);
- Various chemicals used in laboratories;
- Chlorine-containing plastics.

9.10. CONCLUSIONS

Treatment processes are already available for some of the types of material discussed, while research and development is needed for others. Special arrangements need to be made in the event that radionuclides become mixed with the types of material or waste discussed in Section 9.9. The segregation of radioactive waste should be optimized both to avoid its mixing with non-radioactive waste and to adopt different and more appropriate ways for its management (i.e. for its treatment, conditioning and disposal). It is also necessary to refer to the laws and regulations for conventional (non-radioactive) waste, as applicable. If there are no disposal facilities that will accept radioactive waste, then interim storage may be the only option. Comprehensive examples of all the issues covered in this section are given in Ref. [26]. Experience on simple decontamination methodologies and techniques for small facilities is given in Refs [79, 80].

10. COSTS

The various types of facility covered in this report are different in size, complexity and nature, and hence costs must be dealt with on a case by case basis. Cost estimating is an important part of planning for decommissioning. Cost estimates were often not made in the past and funds for decommissioning were therefore seldom provided. This has been of particular concern to regulators and often the public as it becomes apparent that safety could be neglected when a facility shuts down and no longer generates benefit or income and hence becomes a liability to the owner. For this reason most regulators now insist on seeing arrangements and funding for decommissioning before licences or registration documents are issued for new facilities. In some States, including the USA, the regulator has taken particular interest in funding [81].

Many techniques (often computer aided) have been developed to assist in the costing of decommissioning. In the USA, prompted by the NRC, costing has been studied extensively for power reactors and for non-fuel-cycle facilities [32]. The European Commission has also given attention to costing and the economics of decommissioning, for example, particle accelerators [11]. There is also an analysis of the variability of decommissioning costs, which concentrates more on reactor decommissioning, but may nevertheless provide useful guidance [82]. These cost comparisons and concern about variability have resulted in the OECD Nuclear Energy Agency, the IAEA and the European Commission producing a standardized list of items for costing purposes [83].

Cost centres for dismantling and decommissioning can be grouped as follows:

- Labour;
- Material and equipment;
- Contracted services;
- Waste management;
- Overheads.

There will in addition be indirect costs, such as the costs of licensing and regulation, financing costs, insurance, maintenance, site security, surveillance and radiological monitoring.

A contingency is normally added to all costs as an identifiable item against each cost group. In general, the contingency allowance will diminish as cost estimates become more refined and are based on actual costs and the feedback of actual experience. Many of the unit costs of activities are time dependent and it will usually be more cost effective to complete decommissioning in as short a time as possible. In this respect careful and thorough planning will help to reduce costs. A contingency fund may also be needed to pay for additional technical expertise and the waste management costs associated with non-radiological hazards (e.g. asbestos in the linings of ducts from fume hoods, asbestos in ceiling tiles or fibreglass in pipe lagging and insulation) [22].

The cost of waste disposal and interim storage can be significant and, as mentioned above, the reduction of the quantity of radioactive waste by segregation, decontamination and size reduction can be cost effective.

11. QUALITY ASSURANCE

Some small users of radioactive material may not have identifiable quality assurance procedures in place. All nuclear activities typically involve some form of quality assurance programme. An identifiable procedure should be used for registering the radioactive material held by the licensee. This can consist of a record of radioactive material, such as sealed sources, that is periodically verified. For decommissioning, when a larger volume of radioactive material may be handled, it is important to enhance the documentation procedures and work practices to minimize any loss of material, accidents or abnormal occurrences. Quality assurance also encompasses lines of responsibility and the training of operators.

A fully comprehensive quality assurance programme should include the following basic elements:

- Design control;
- Document control;
- Procurement control;
- Component identification;
- Inspection;
- Calibration procedures;
- Non-conformance procedures;
- Control procedures for changes, records and audits.

Many small decommissioning projects will, however, not require such a comprehensive system. Guidelines have been compiled on the application of quality assurance to decommissioning [84, 85].

12. SUMMARY AND CONCLUSIONS

This report covers a variety of small decommissioning projects that range from those that use radionuclides in measuring devices and for inspection or medical needs, to research facilities containing a number of hot cells and radiochemical and similar facilities. It was believed necessary to compile a report such as this to concentrate on smaller facilities to describe how the well established and comprehensive decommissioning practices for large facilities can be adapted for smaller, more modest ones.

Although the facilities covered by this report are varied, many of the decommissioning principles that are now well established for power reactors, research reactors and fuel cycle facilities can be drawn upon. There is a wealth of published information on good and bad practice to draw upon and many documents are available for guidance in which, in many cases, lessons learned are reported.

This report and the references cited demonstrate that the dismantling and decontamination of small facilities does not present significant technical problems provided that experienced operators are used together with the appropriate equipment and facilities and in accordance with an established plan. In some cases decommissioning is best done by those having experience with the facility in question, but motivation and training are important. The return of radioactive sources to the original suppliers or other authorized bodies is good practice. Where it is necessary to conduct decommissioning operations on the site, there is a large amount of successful experience to draw upon.

It is recognized that owners of smaller facilities may not have large resources to draw upon for decommissioning and it is reasonable to reduce the demands of decommissioning to a level that is appropriate to smaller facilities that have lower safety risks.

This report has identified a wealth of reported information and practical experience on the successful implementation of decommissioning projects. Important features and aspects that have special relevance to small facilities have been highlighted. It is concluded that careful planning and attention to detail, together with the implementation of the safety guidance given in Ref. [5], will result in the successful and safe implementation of decommissioning.

Appendix I

APPLICATION FOR A LICENCE OR REGISTRATION DOCUMENT TO POSSESS NUCLEAR MATERIAL

Applications should contain at least the following information:

- The name and address of the responsible person;
- The name, position and qualifications of the person nominated for the purchase, storage and disposal of the radioactive material;
- A description of the work and the use of the material;
- A quantification and characterization of the radionuclides involved;
- A description of the apparatus containing and housing the sealed sources and copies of the test certificates;
- A description of the facility or facilities in which the radionuclides will be used;
- The number of persons involved;
- Details of the instruments available for measuring dose rates and contamination levels and contamination control measures;
- Details of the personnel dosimetry services available;
- A description of the documentation and record keeping to be used;
- A description of the storage facilities and security arrangements;
- A description of the proposed waste management system, including for the disposal of the waste;
- Considerations for future decommissioning.

Appendix II

EXAMPLE OF THE CONTENTS OF A DECOMMISSIONING PLAN

TABLE VII. EXAMPLE OF THE CONTENTS OF A DECOMMISSIONING PLAN

Section	Contents
Introduction	Objectives, scope and goals to be achieved
Description of the facility	Physical description of the site and the facility and its operational history Inventory of the radioactive and toxic material
Decommissioning strategy	Objectives and decommissioning alternatives Selection and justification of the preferred option
Project management and planning	Resources Organization and responsibilities Review and monitoring arrangements Detailed estimates of waste quantities Training and qualification Reporting and records Risk and hazard management Scheduling
Decommissioning activities	Decontamination and dismantling activities Waste management Maintenance programmes
Safety assessment	Dose prediction for tasks and demonstration of ALARA for tasks Non-radiological hazards Risk, hazard and uncertainty analyses Operating rules and instructions
Environmental impact assessment	Demonstration of compliance with environmental standards and criteria
Quality assurance programme	Setting up a quality assurance and/or quality control programme Verification of compliance with established quality assurance requirements

TABLE VII. (cont.)

Section	Contents
Radiation protection and safety programme	Radiation monitoring and protection systems Physical security and material control Emergency arrangements Management of safety Justification of safety for workers, the general population and the environment (use of ALARA)
Continued surveillance and maintenance	Development of surveillance and maintenance programmes
Final radiation survey	Demonstration of compliance with the clearance criteria
Costs	Cost estimate Provision of funds

Appendix III

CHECKLIST OF THE DECOMMISSIONING REQUIREMENTS FOR SMALL FACILITIES

As highlighted in this report, certain requirements are important for the decommissioning of small nuclear facilities. The lack of adequate provisions for decommissioning may lead to undesirable effects in terms of timeliness, cost effectiveness and safety. The following checklist is intended to be a useful reference for those involved in the decommissioning of small facilities.

- Identify all regulatory requirements, not only from the licensing agency but also from other authorities (i.e. local, regional and national) that have jurisdiction over the facility to be decommissioned.
- Convert decommissioning criteria into units that can be measured in the field, and obtain regulatory concurrence with that conversion.
- Identify and retain key staff at the facility whose participation is important to the success of the decommissioning effort.
- Maintain good communications with regulatory authorities on plans for the decommissioning, the implementation of those plans and any unforeseen events that may impact upon the scope or schedule described in the plan.
- Assess the need to establish lines of communication with the local community and elected officials about the decommissioning project. Identify any necessary public relations campaigns.
- Ensure that there is adequate knowledge about the facility and the history of its operations, surveys and decontamination projects.
- Carry out a detailed physical and radiological characterization of the plant, including an inventory of all radioactive material and/or radiation sources.
- Identify all radiological, physical, chemical and biological hazards that exist at the facility and establish the best means to address these hazards (including monitoring, as needed) as part of the planning.
- Evaluate candidate decontamination methods for their applicability to the facility and their effectiveness in removing the contamination present.
- Document all requirements and how those requirements will be satisfied in a detailed decommissioning plan that receives regulatory approval in accordance with the national legislation, including milestones and the scheduling of activities.
- Assign roles and responsibilities covering the implementation of the decommissioning plan to all parties, including contractors.
- Provide or extend all utilities and services needed to conduct decontamination activities safely and effectively.

- Develop procedures to specify how the work is to be accomplished, which steps are necessary to document the success of the work and which documentation must be maintained.
- Define disposal routes for all waste generated (both radioactive and non-radioactive) as part of the characterization and decommissioning of the facility, including any necessary treatment required prior to disposal.
- Prepare detailed estimates of the work to be performed and generate cost estimates.
- Ensure that sufficient funds are available to complete the project.
- Establish the quality assurance programme needed to ensure that the decommissioning will be successful and integrate these requirements into all planning and implementation documents.
- Implement and complete the decommissioning as planned.
- Identify any facility conditions that can change the scope of the decommissioning effort, modify the plan to account for those conditions and obtain regulatory approval for the modifications.
- Confirm the successful completion of the decommissioning through termination surveys.
- Ensure records are made and kept for all activities.
- Request regulator approval of the licence termination effort, and support any verification activities that they conduct.
- Archive essential records for long term storage and establish a system for the retrieval of the records, if necessary.

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Annex I

EXAMPLES OF NATIONAL EXPERIENCE

Examples provided in this annex range from national policies and legislation to detailed technical and organizational aspects. It is felt that both approaches are useful to provide practical guidance and information on how such decommissioning projects are planned and managed in various Member States. The examples given are not necessarily best practices, nor has their consistency with the IAEA's guidance been tested in detail, rather they reflect a wide variety of national policies, social and economic conditions, nuclear programmes and traditions. Although the information presented is not considered to be exhaustive, the reader is encouraged to evaluate the applicability of these cases to a specific decommissioning project.[1]

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[1] Data and statements were mostly provided by national contributors and are not necessarily endorsed by the IAEA.

Annex I.A

MANAGEMENT AND DECOMMISSIONING OF SMALL NUCLEAR FACILITIES AT SCK•CEN, BELGIUM

I.A–1. INTRODUCTION

The Belgian Nuclear Research Centre (SCK•CEN) in Mol is comprised of:

- BR 1, a graphite moderated, gas cooled reactor with power of up to 3.5 MW, which has been in operation for research purposes since 1956;
- BR 2, a materials testing reactor (MTR) with power of up to 125 MW, which has been in operation for research purposes and the production of isotopes since 1961;
- BR 3, a 40 MW(th) pressurized water reactor, which operated as both a pilot and as an experimental reactor between 1962 and 1987 and which has been in the process of being dismantled since 1989;
- VENUS, a zero power reactor located inside the BR 1 facility and used for tests on fuel and benchmarks for neutronic computer codes;
- Research laboratory buildings with hot cells, gloveboxes and fume hoods, used for post-irradiation research on fuel and reactor material, tests on fuel reprocessing, the characterization of waste and studies and analyses of the effects of radiation and contamination on animals and plants;
- An underground laboratory for research on geological waste disposal.

This annex presents the organization of the decommissioning activities at the SCK•CEN, with specific attention given to the management and the decommissioning of small nuclear facilities. The management and the decommissioning of large nuclear facilities are reported elsewhere [I.A–1]. Results and lessons learned from past experience in the decommissioning of small nuclear facilities are also presented.

I.A–2. ORGANIZATIONAL ASPECTS

The Site Restoration Department of the SCK•CEN is in charge of the management of the decommissioning of the SCK•CEN’s nuclear installations and plays the role of a decommissioning consultant and/or manager for decommissioning activities in Belgium and abroad.

Among the responsibilities of the department are:

- The development and maintenance of the physical and radiological inventory of nuclear installations;
- The development of decommissioning plans, including cost evaluations;
- The management of the decommissioning projects performed by its personnel or by subcontractors;
- Communication with the Health Physics Department, which reports to the authorized control body and the regulatory body;
- Reporting to the Belgian Agency for Radioactive Waste and Enriched Fissile Materials (ONDRAF/NIRAS);
- The evaluation of decommissioning costs for new installations and the yearly evaluation of the two decommissioning funds for the SCK•CEN's facilities.

The financing of the decommissioning of the SCK•CEN's nuclear installations is ensured by two funds. One fund is managed by ONDRAF/NIRAS and covers the decommissioning of the SCK•CEN's nuclear installations created for research performed before 31 December 1988. The second fund is managed by the SCK•CEN and covers the decommissioning of the other nuclear installations. Both funds are evaluated at least once a year.

The management of both decommissioning activities is centralized in the Site Restoration Department.

I.A–3. PRACTICAL EXPERIENCE

I.A–3.1. Decommissioning of nuclear laboratories

In the past the SCK•CEN has been involved in both nuclear and non-nuclear research programmes, but in the early 1990s the Belgian government decided to restrict the operational scope of the SCK•CEN to strictly nuclear programmes. A new research centre, VITO (the Flemish Institute for Technological Research), was founded and took over all non-nuclear activities. VITO is housed in former SCK•CEN buildings. In addition to well equipped non-nuclear laboratories and offices, these buildings contain laboratories and installations with a history of radiological material use [I.A–2, I.A–3].

I.A–3.1.1. Safety reports

Before starting with the actual decommissioning activities, a safety report has to be prepared and approved by the Health Physics Department. This safety report describes the installation to be decommissioned, identifies the problems in terms of radiation doses and contamination and describes the organization of the decommissioning activity. The safety report also serves as the technical scope for the call for proposals if an external company is to carry out the decommissioning activity.

I.A–3.1.1.1. Definition of the unrestricted release and/or reuse limits

In the absence of a well defined Belgian regulation for the unrestricted release of potentially contaminated and/or decontaminated material, the SCK•CEN Health Physics Department, in association with the Association Vinçotte Nuclear (the authorized control body (AVN)), specifies the following release limits:

- 0.4 Bq/cm^2 surface contamination for beta and gamma emitters;
- 0.04 Bq/cm^2 surface contamination for alpha emitters;
- The residual radioactivity of the radionuclides present in representative samples of the construction material of the building must be similar to that of corresponding construction material originating from a non-nuclear zone.

The SCK•CEN and AVN based the development of these limits on existing Belgian rules and regulations in preparation by the IAEA [I.A–4] and other international organizations. They also defined the methods that must be used to analyse and monitor the radioactivity from the start of the decommissioning until the final release, namely:

- Monitoring the entire wall and floor surfaces;
- Radiological analyses of the washwater from walls and floors;
- The measurement of selected core samples.

If the results of these measurements and analyses are below the release limits, the Health Physics Department prepares a request for an unrestricted reuse and submits it for approval to the independent control body.

I.A–3.1.1.2. Gathering physical and radiological characteristics of the buildings

The buildings that needed to be decommissioned were the physics building, the metallurgy building, block 3 of the chemistry building and two radiobiology buildings.

The physics building consists mainly of laboratories, offices and a waste storage room. Only a few laboratories were used for experiments and measurements with radioactive material such as ^{14}C, ^{137}Cs, ^{60}Co, ^{133}Ba and ^{90}Sr. The total wall and floor surface of these laboratories covers approximately 700 m^2. Average contamination levels were below 2.5 Bq/cm^2, with hot spots of up to 30 Bq/cm^2 for beta and gamma emitters and 0.1 Bq/cm^2 for alpha emitters.

The decommissioning of the metallurgy building was, compared with the physics building, somewhat more complicated. In addition to a number of conventional laboratories, a large hall for material testing, some cellars and a storage room were contaminated. In these laboratories characterization programmes were carried out on different kinds of fuel, fissile material and waste, which resulted in contamination with isotopes of thorium and uranium. The total wall and floor surface of potentially contaminated areas is approximately 4800 m^2. Average contamination levels were below 2 Bq/cm^2, with hot spots of up to 60 Bq/cm^2 for beta and gamma emitters and 0.5 Bq/cm^2 for alpha emitters.

Block 3 of the chemistry building consists mainly of laboratories and offices. The laboratories were used for experiments and measurements with radioactive material, fuel and fissile material. The total wall and floor surface of these laboratories covers approximately 4300 m^2. Average contamination levels were below 500 Bq/cm^2 for beta and gamma emitters and 0.12 Bq/cm^2 for alpha emitters.

Two radiobiology buildings needed to be decommissioned, namely the bio-lab and the bio-animals 1 buildings. Both buildings contain laboratories, offices and storage rooms. In addition, the bio-animals 1 building contained several animal cages. Extensive experiments and measurements of the impact of radiation and contamination on plants and animals were carried out in these laboratories. A whole range of isotopes, including ^{3}H, ^{14}C, ^{32}P, ^{238}U, ^{239}Pu and ^{241}Am, were used. The total wall and floor surface of these laboratories and cages covers approximately 3500 m^2. Average contamination levels for the surfaces were below the limits for unrestricted release and/or reuse. Hot spots of 1.5 Bq/cm^2 for beta and gamma emitters and 0.15 Bq/cm^2 for alpha emitters were found.

I.A–3.1.1.3. Decommissioning strategy

In general, the goal of the common decommissioning strategy is the unrestricted release of the site by dismantling the equipment, removing contaminated parts of walls and floors using appropriate techniques and finally demolishing the remaining structures using conventional techniques.

Since important parts of some VITO buildings (the former SCK•CEN buildings) contain valuable non-nuclear infrastructure and equipment, the SCK•CEN selected a different strategy of limiting the decommissioning activities to the radioactive parts of the nuclear installation. The final goal of the decommissioning project was to obtain a release for the unrestricted reuse of the building after removing the radioactivity. After refurbishment, the building can then be used for new industrial purposes outside the nuclear field; that is, without radiological protection measures and without regulatory control.

The specific strategy used for decommissioning the above described facilities followed multiple steps. The zone to be decommissioned was first isolated from the rest of the building and equipped with hand and foot monitors and an air monitoring device. In this zone an area was set up for decontamination, material reduction (see Fig. I.A–1) and sorting and packaging the waste produced during the dismantling and demolition activities.

The first step in the decommissioning process consisted of scanning for alpha, beta and gamma contamination on loose material and equipment inside the controlled area. Non-contaminated objects were released from the controlled zone. Potentially contaminated or contaminated items were brought to the decontamination area and treated.

Next, the devices anchored to the walls and floors, such as ventilation pipes (see Fig. I.A–2), fume hoods and waste-piping, were demolished and removed to the decontamination area for treatment.

Based on the results obtained by scanning the dismantled and/or decontaminated items, the agent of the Health Physics Department decided whether the objects could be freely released or released with restrictions (i.e. whether they could be disposed of as industrial waste on a public dumping ground). Material that could not be decontaminated was treated as radioactive waste following the specifications of ONDRAF/NIRAS.

FIG. I.A–1. Reduction of contaminated material.

FIG. I.A–2. Dismantling ventilation pipes.

The next step included vacuum cleaning and washing the floors and walls, followed by mapping all surfaces to identify which parts were contaminated. The contamination was removed by scabbling, shaving and/or drilling. This sequence of washing, mapping and decontamination was repeated until the radioactivity was removed.

All the surfaces were then washed again. The washwater was collected and analysed by alpha and gamma spectroscopy. Core samples were taken at random from floors and walls for further measurement by alpha and gamma spectroscopy. Once the spectroscopy results proved that the release criteria were met, the demarcation of the zone was removed.

I.A–3.1.1.4. Decommissioning tools and equipment

The choice of decommissioning tools was based on their simplicity in manipulation and their ability to minimize cross-contamination (see Table I.A–I).

Standard saws were used for the dismantling and size reduction of contaminated items. Ventilation pipes were cut by means of electrical nibblers and shears. Contaminated wall plaster, concrete, stone and tiles were removed using electric and pneumatic scabblers and drills. Water and common decontamination solutions were used for washing walls, scrubbing floors, cleaning apparatus and

TABLE I.A–I. OVERVIEW OF DECOMMISSIONING TOOLS AND DECONTAMINATION PRODUCTS

Tool or decontamination product	Treatment
Standard saws	The dismantling and size reduction of contaminated pieces
Nibblers and shears (electric)	Cutting ventilation pipes
Scabblers and drills (electric or pneumatic)	Removing contaminated wall plaster, concrete, stone and tiles
Water and commercially available products	Washing walls, scrubbling floors, cleaning apparatus, etc.

equipment, etc. When using these tools, the decontamination crew wore special overalls, safety shoes, gloves and full-face respirators, when needed.

I.A–3.1.1.5. Material stream management

All solid radioactive waste was packed and managed in accordance with the specifications of ONDRAF/NIRAS. The washwater collected during decontamination was sampled, measured by alpha and gamma spectroscopy and managed as liquid waste.

I.A–3.1.1.6. Radiation protection

During all the dismantling or decontamination operations the crew wore safety clothing and full-face respirators. The crew removed their safety clothing and were checked for contamination by means of a hand and foot monitor when they left the demarcated zone. At the end of the decommissioning of the building every crew member had a check up for possible internal contamination using a whole body counter.

I.A–3.1.1.7. Results obtained and lessons learned

The decommissioning of the physics building, executed by the decontamination crew of the SCK•CEN, was used as a test case. During the decommissioning the daily activities in the non-contaminated parts of the building went on as usual. The decommissioning activity caused some stress for personnel not familiar with radioactivity. It was therefore important to hold an information meeting for the employees before each decommissioning phase.

No major problems were encountered during the decommissioning of the building itself. The contamination levels of some parts of the infrastructure, such as the window ledges and doors, were, however, higher than expected.

The metallurgy building, block 3 of the chemistry building and the two radiobiology buildings were decommissioned by a company selected on the basis of a competitive bid. The SCK•CEN Technical Liabilities team and the Health Physics Department supervised all the activities. Before each decommissioning phase a meeting was held to inform the employees of VITO about the decommissioning activities and the safety conditions.

After the complete decommissioning of the buildings the transferable alpha, beta and gamma contamination was below 0.001 Bq/cm^2. The core samples had a similar radionuclide spectrum as corresponding material coming from a non-nuclear zone. The total alpha, beta and gamma activity measured on those samples was below 1 Bq/g.

The decommissioning certifications were obtained in 1993 for the physics building, in 1994 for the metallurgy building and in 1995 for block 3 of the chemistry building. The certification for the radiobiology buildings was obtained at the end of 1996. At the time of each certification VITO formalized its agreement for the transfer of the decommissioned buildings.

In the parts of the laboratory buildings in which radionuclides were used in small amounts the surfaces and equipment were generally more extensively contaminated than would be expected from the history of the buildings. The opposite was observed in the laboratories in which high amounts of activity were handled; that is, contamination was, contrary to all expectations, more limited.

Table I.A–II gives a short summary of the main data for the four buildings decommissioned up to the unrestricted reuse level. An indication is also given of the cost breakdown between the dismantling operations and the waste costs (including conditioning, storage and disposal costs).

I.A–3.2. Decommissioning of contaminated hot cells and gloveboxes

In the past the common strategy for decommissioning hot cells and gloveboxes was to transport them intact to a waste treatment facility and then dismantle them. This decommissioning strategy was not totally satisfactory, as:

- Only small hot cells and gloveboxes could be managed this way;
- There was no attempt to reduce the waste volume by using decontamination processes and recycling techniques;
- The waste costs became increasingly high.

The first time that this strategy was not applied was for the decommissioning of the oldest and largest hot cell, C10 [I.A–5, I.A–6].

TABLE I.A–II. SUMMARY TABLE

	Building			
	Physics	Metallurgy	Chemistry block 3	Biology
Total wall and floor surface (m^2)	700	4800	4300	3500
Contamination				
Average beta and gamma contamination level (Bq/cm^2)	2.5	2	500	—
Beta and gamma hot spot contamination (Bq/cm^2)	30	60	—	1.5
Average alpha contact level (Bq/cm^2)	0.10	0.50	0.12	0.15
Material stream management				
Free released material (%)	98	97.5	95.4	98.5
Radioactive material (%)	2	2.5	4.6	1.5
Cost breakdown				
Dismantling (%)	39	37	39	55
Waste (%)	61	63	61	45

I.A–3.2.1. Decommissioning of hot cell C10

The decommissioning of the oldest hot cell within the laboratory, C10, for low, high and medium activity was completed a few years ago. Over a 20 year period the cell was used for post-irradiation research on irradiated material and fuel pins. The cell consisted of an L-shaped box of 18 m^3 that was airtight for alpha contamination and surrounded by a lead biological shield. Twelve tongs and two MA11 machine slave manipulators operated the cell equipment, which included a small lathe and several cutting devices.

I.A–3.2.1.1. Decommissioning

The decommissioning of hot cell C10 was accomplished in four main phases:

- Phase 1: Dismantling of the equipment. The initial quantity of equipment was reduced by 65%, and the resulting waste was placed in tin cans and transferred to a waste processing facility by means of shielded containers.

- Phase 2: Decontamination of the cell. Decontamination of the cell was required before any structural dismantling activity began, owing to the high level of contamination and dose produced by isotopes such as ^{60}Co, ^{134}Cs, ^{137}Cs and ^{241}Am. Mechanical polishing combined with intensive vacuum cleaning was used to reduce the contact radiation levels (down to 1.4 mSv/h). It must be noted that limitations in the reach of the tongs and manipulators of cell C10 did not permit all internal surfaces to be adequately decontaminated. During the decontamination of the remaining cell surfaces the crew wore the normal safety clothes for working in controlled areas (i.e. safety shoes and protective garments).
- Phase 3: Dismantling of the cell. The cell was surrounded by a contamination controlled enclosure with an access airlock. Seventy tonnes of lead was first removed. The cell walls and its work table were then cut using a plasma torch. This technique was selected to reduce the exposure time of the workers during the cutting phase of the cell dismantlement.
- Phase 4: Waste management. After phase 1 a chart was created of the radiological data and physical information on the items remaining after decontamination. A major consideration in defining the decommissioning strategy was keeping the decommissioning costs as low as possible. Since the decommissioning costs are proportional to the amount of radioactive waste produced, it was necessary to define the different waste types in advance, to estimate their quantities and to select the most efficient technique in order to minimize the production of waste. There is a correlation between minimizing the total cost (i.e. the dismantling and waste treatment) and the dismantling techniques used in producing the waste, which couples waste management to the decommissioning strategy used.

Taking into account the specific radiological and physical restrictions, the following waste streams were relevant:

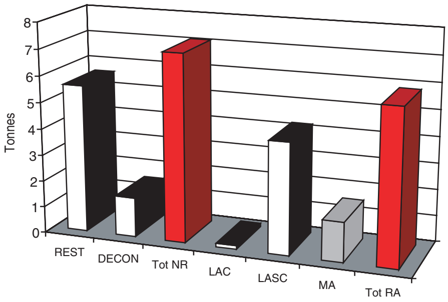
- Free release. The SCK•CEN’s Safety Service, in association with the authorized control organization for radiation protection, specifies the limits for the free release of material at 0.4 Bq/cm^2 for beta and gamma emitters and 0.04 Bq/cm^2 for alpha emitters. A complete file for release was prepared for each item or group of similar items. Material for free release was stored for approximately three months and monitored again before a final release could occur. For this project all 70 t of lead could be free released.
- Restricted reuse. If the radiological restrictions established for dose, activity and isotope content are met, steel and stainless steel can be recycled for use in the nuclear industry after melting. For this project 5.6 t of steel were within the specified restrictions for restricted reuse.

TABLE I.A–III. MAIN RADIOLOGICAL RESTRICTIONS

Type	Maximum beta and gamma	Maximum alpha	Maximum dose
Low active combustible solid (LAC)	40 GBq/m^3	40 MBq/m^3	2 mSv/h
Low active super-compressible solid (LASC)	40 GBq/m^3	5 MBq/m^3	2 mSv/h
Medium active solid (MA)	100 TBq/m^3	200 GBq/m^3	200 mSv/h

- Decontamination. Grit blasting can be used on free release steel and stainless steel. This blasting technique is able to clean only the accessible parts; its efficiency is in inverse ratio to the surface activity. A test on two barrels showed that as the blasting grit became saturated with removed radioactivity the secondary disposal costs (for disposing of the grit) rose to such an extent that the total disposal cost (for decontaminated material and secondary waste) became equal to the cost for standard radioactive waste. Nevertheless, 3.6 t could be considered for restricted reuse.
- Radioactive waste. Depending on the radiological and physical criteria, ONDRAF/NIRAS distinguishes 25 standard types of radioactive waste. The types relevant to this case are shown in Fig. I.A–3. A total of 6 t of radioactive waste (MA, LASC and LAC) was produced. See Table I.A–III for the main radiological restrictions.

Figure I.A–3 gives an overview of the quantities of each decommissioning material stream.



- 70 t could be free released. This material contains the periphery of the cell box, which has normally never been in contact with irradiated materials.
- 50% radioactive.
- The 50% includes radioactive waste from the cell box and secondary waste that was produced during decommissioning.

FIG. I.A–3. Overview of the different quantities of waste produced (except free release). REST = restricted reuse materials, DECON = decontaminated materials, Tot NR = total non-radioactive material, LAC = low active combustible solid waste, LASC = low active super-compressible solid waste, MA = medium active solid waste, Tot RA = total radioactive waste.

I.A–3.2.1.2. First conclusions drawn

By linking the waste management and decommissioning strategies together the overall decommissioning cost can be reduced by finding the optimal (lowest) overall cost for different amounts of decontamination and for the resulting waste streams, noting that decommissioning costs are usually inversely related to waste costs.

In addition:

- Preparation costs can be reduced by an organized planning of the decommissioning. In such a way batches of hot cells can be treated together, which means that the preparation costs are apportioned to several cells.
- Dismantling in situ allows the waste management and decommissioning strategy to be coupled. This coupling should result in finding the best solution for each cell.

I.A–3.2.2. Decommissioning of hot cells and gloveboxes

I.A–3.2.2.1. Safety report

The planned decommissioning activities are first described in a safety report. The safety report describes:

- The hot cell or glovebox, including the physical and radiological inventory;
- The organization of the decontamination works.

The safety measures include an estimation of the collective dose (i.e. the ALARA approach). This safety report has to be approved by the Health Physics Department.

I.A–3.2.2.2. Preparation of the hot cells and gloveboxes

If necessary, the decommissioning of the gloveboxes and/or hot cells is preceded by a post-operational phase that allows the transfer of the waste generated during the operational period. The equipment inside is then dismantled using tongs, manipulators or gloves. The box or the cell is decontaminated after the equipment is dismantled and the waste generated by these activities is removed. The goal of this decontamination process is to minimize the dose and the spread of contaminants when the box or cell itself is dismantled.

I.A–3.2.2.3. Decommissioning of hot cells and gloveboxes

The decommissioning of small hot cells and gloveboxes can be carried out in the SCK•CEN Central Buffer Zone (CBZ), a building licensed in 1999. The alpha zone of the CBZ has a permanent airtight construction with an airlock for access, which facilitates the decommissioning of small gloveboxes and hot cells. After the removal of the biological shield (if any), the box or the cell is packaged and transported into the alpha zone of the CBZ for further dismantling activities.

If the decommissioning work cannot be performed in the CBZ, an airtight enclosure is constructed around the hot cell or the glovebox. Before the construction of the enclosure the biological shield (if any) is removed. The dismantling activities can then be carried out.

I.A–3.2.2.4. Decommissioning tools and equipment

Mechanical polishing combined with intensive vacuum cleaning is the usual technique employed for the preliminary decontamination of cells and gloveboxes. Pastes and high pressure cleaning and the cerium process can be used on metallic pieces to reach the free release limits.

Disk grinders are common tools for cutting metallic pieces. Plasma-arc cutting can also be used if no attempt is to be made to decontaminate the metallic pieces to free release limits.

I.A–3.2.2.5. Radiation protection

During the preparation of a hot cell and glovebox for decommissioning the crew wears the normal safety clothes for working in controlled areas (i.e. safety shoes and protective clothes). During the decommissioning activities performed inside the airtight chamber the crew wears safety clothes in combination with an airtight suit (Fig. I.A–4).

I.A–3.2.2.6. Reporting

Reports documenting the progress of decommissioning and the individual dose are frequently prepared and communicated to the Health Physics Department. At the end of a decommissioning the collective exposure and the individual doses are compared with the provisional values contained in the safety file.

The performance and the results achieved during the decommissioning activities are analysed and recorded in the Decommissioning Management Tool (DEMATO [I.A–7]) of the Site Restoration Department for use in improving the management of future decommissioning activities.

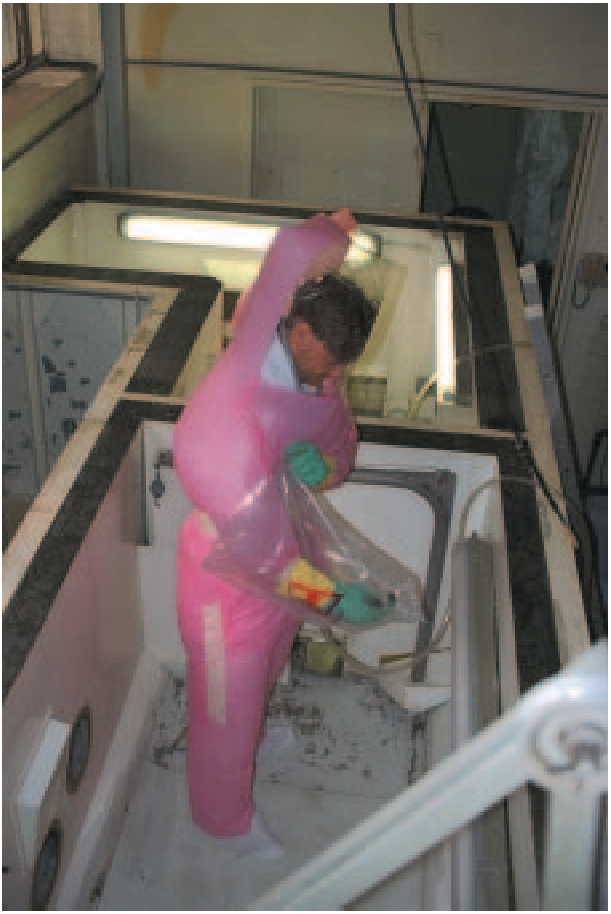


FIG. I.A–4. Protective clothes worn during the dismantling of a hot cell.

I.A–3.3. Decommissioning of an underground 10 km pipe

A 10 km underground polyethylene pipe was installed between the collection tank of one laboratory and the waste processing facility. The pipe, for which only some summary location maps existed, was mainly used for conveying liquid samples and was only considered to be slightly contaminated.

The main objective of decommissioning was to remove this piping and to produce as little radioactive waste as possible. The whole pipe was thus excavated, with all the necessary precautions taken (see Fig. I.A–5), and then directly cut into

FIG. I.A–5. Excavation of the pipe.

50 cm segments and placed into plastic bags (see Fig. I.A–6). The total volume produced amounted to about 50 m^3 (approximately 10 t).

As leak sites were found along the pipe route samples of water and soil surrounding the sites of these leaks were taken. These samples were fortunately not contaminated, although if they were the soil would have been excavated and managed as radioactive waste.

In order to minimize the generated radioactive waste volume, the subsequent handling of the piping was first tested on a small batch (about 5 m^3). The pipes were first cut open longitudinally to expose the internal surface for direct contact measurements.

For the test batch:

FIG. I.A–6. The segments are cut on the site and placed in plastic bags.

- A direct contact measurement was used to sort the pipe sections into contaminated (i.e. above the free release limit levels) and non-contaminated batches;
- The contaminated parts were decontaminated (by simple means, such as by the use of detergents);
- A portion (20%) of the non-contaminated parts was granulated for sampling and measurement to allow a second bulk activity measurement;
- A granulation (or shredding) operation was performed on this test batch to validate the sampling and detailed measurement protocols;
- A final direct contact measurement was carried out on the decontaminated parts to assess the effectiveness of the treatment;
- Finally, the complete test batch was free released and a small fraction of the total was once again granulated for a final assessment of the process.

Drawing upon the lessons learned from this test batch, a complete procedure for the remaining 45 m^3 of piping was set up and approved by the Health Physics Department and the control body.

For the whole pipeline (see Fig. I.A–7):

- The pipe will be cut into four parts using hydraulic shears (to avoid aerosol production and thus all the secondary waste associated with it);

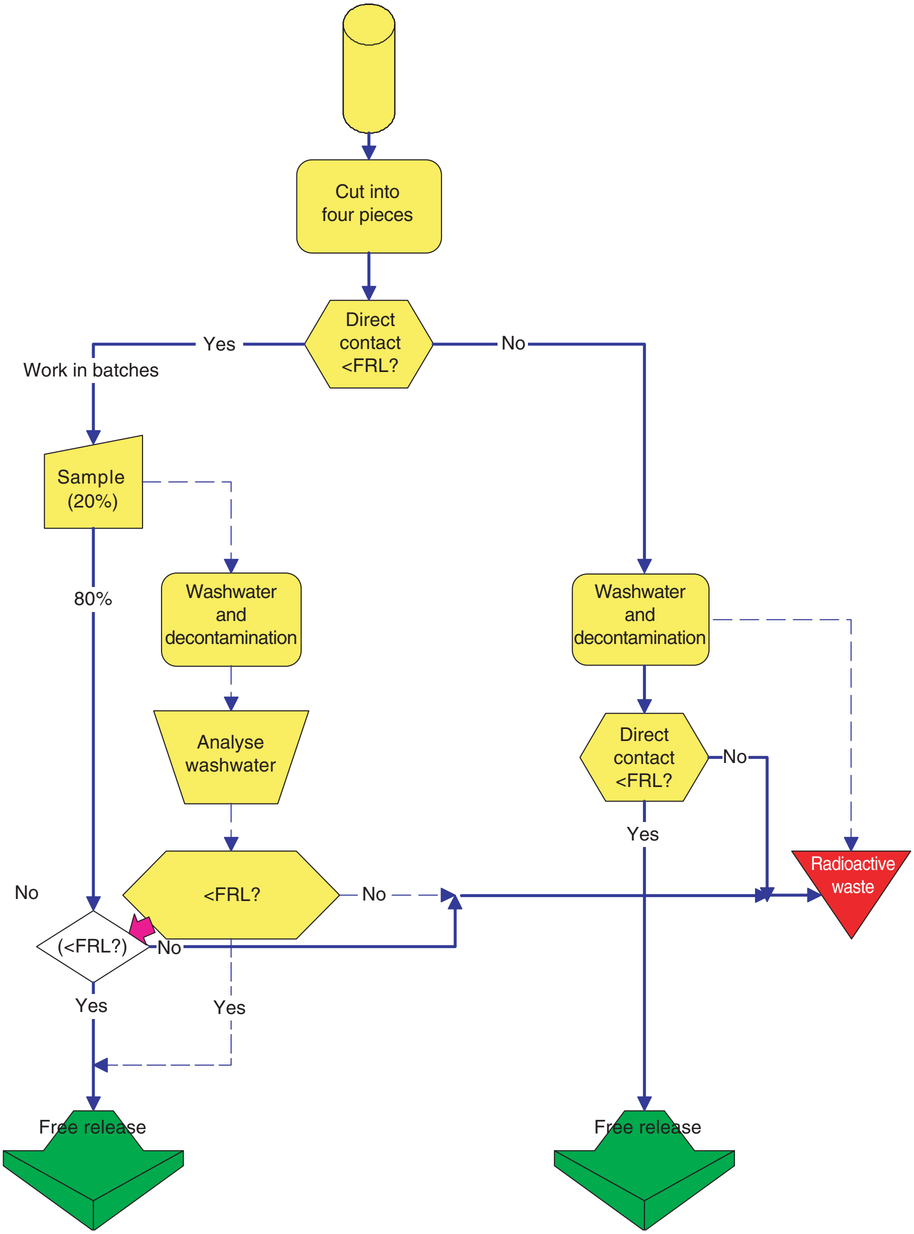


FIG. I.A–7. Measurement principle for the pipeline. FRL = free release level.

- A direct contact measurement will be taken to allow the separation of contaminated and free releasable material (note that the general SCK•CEN procedure for free releasing material foresees two separate measurements on the material to be taken);
- The contaminated pieces will be decontaminated using water and a decontamination solution (the washwater will be considered as secondary waste);
- A new direct contact measurement will be carried out after the decontamination of the cleaned pieces;
- A portion (20%) of the free releasable pieces will be selected and washed in the same way as the contaminated ones and the washwater will be analysed and the measurement used (this is a proposal) as a second measurement for the free release.

I.A–4. CONCLUSIONS

As a general conclusion it can be stated that the decommissioning of contaminated laboratories and hot cells can be carried out simply, with existing tools, provided that the work is efficiently organized.

The management and minimization of the generated waste is important, as this part of the operation has a large impact on the total cost of the decommissioning (this, however, can depend on the waste disposal policy and the costs in a particular country).

For a successful and economical decommissioning two topics need consideration:

- The overall organization of the operation, including for radiological safety and waste management;
- The characterization before, during and after the operation, to establish the scope of the decommissioning, to sort material by its radioactivity content and to characterize the flow of the material generated during the decontamination.

Finally, for research centres it is important to know that the technologies for the complete decommissioning of nuclear laboratories and facilities are available and have been field tested and that the costs can be estimated based on existing experience.

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Annex I.B

DECOMMISSIONING OF A BRACHYTHERAPY FACILITY AT THE ONCOLOGY HOSPITAL IN HAVANA, CUBA

I.B–1. DECISION TO DECOMMISSION

The National Institute of Oncology and Radiobiology (INOR) used to use ^{226}Ra sealed sources for brachytherapy, but, for technical obsolescence and safety reasons, the brachytherapy facility has been shut down. Most of the ^{226}Ra sources were collected from the hospital in 1996, but for various reasons not all the radium sources could be removed at that time. Some of these sources leaked, which caused a spread of contamination. In May 1997 the directorate of the hospital requested the Centre for Radiation Protection and Hygiene (CPHR) to evaluate the radiological situation in the contaminated areas and to carry out the decontamination of the rooms and the decommissioning of the brachytherapy facility for unrestricted use. Contamination surveys conducted during that year confirmed the extent of the contamination of the facility.

The decontamination of the rooms and the decommissioning of the facility took place in June 1999, once all the necessary conditions were met. In order to perform this work the hospital received authorization from the National Centre for Nuclear Safety (CNSN) (Cuba's regulatory body) in the form of a licence for decommissioning.

I.B–2. INITIAL PREPARATIONS

The decommissioning process greatly benefited from early planning. This planning included an assessment of the available documentation and operational history of the brachytherapy facility, a definition of the responsibilities for each activity and technical seminars with personnel from the hospital and the specialists from the CPHR in charge of the decommissioning activities. A safety assessment of the radiological and non-radiological hazards was carried out together with an evaluation of the available waste management provisions and of the availability of financial resources, as well as a review of the lessons learned from previous projects.

I.B–3. RESPONSIBILITIES

The licensee must be responsible for the safety management system and be in control of the day to day activities of the facility until the facility licence is terminated. Therefore, although CPHR workers carried out all the decommissioning activities, the Radiological Protection Department of the hospital was actually the body responsible for the decommissioning.

I.B–4. DESCRIPTION OF THE FACILITY

The brachytherapy facility is a small part of a larger non-nuclear facility (the hospital). It was located in Section D of the Oncology Institute. There were four rooms on the third floor (Fig. I.B–1) used for the treatment of patients. The well used to store and handle the radiation sources was in room 3, behind a wall 1 m high and 20 cm thick.

The well was a 1.5 m deep underground hole that had concrete walls. The bottom was filled with sand. There were 60 PVC pipes inside the well, each 10 m long and 10 cm in diameter. Radium sources within lead containers were raised and lowered through the PVC pipes by means of wire cables. The well had an exit door on the first floor (Fig. I.B–2).

There were various contaminated objects in the rooms: lead pieces, clothing, overshoes, gloves, a table and seats.

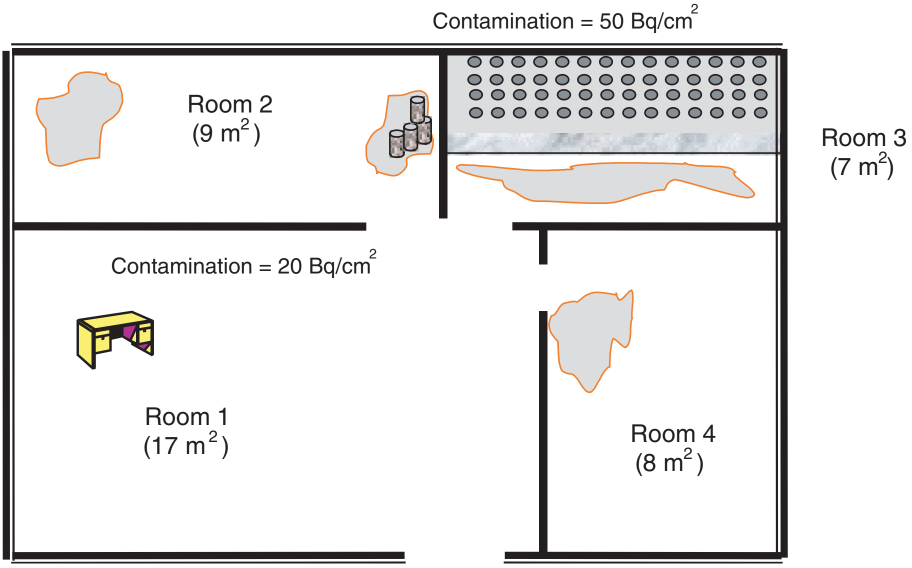


FIG. I.B–1. Contaminated rooms on the third floor.

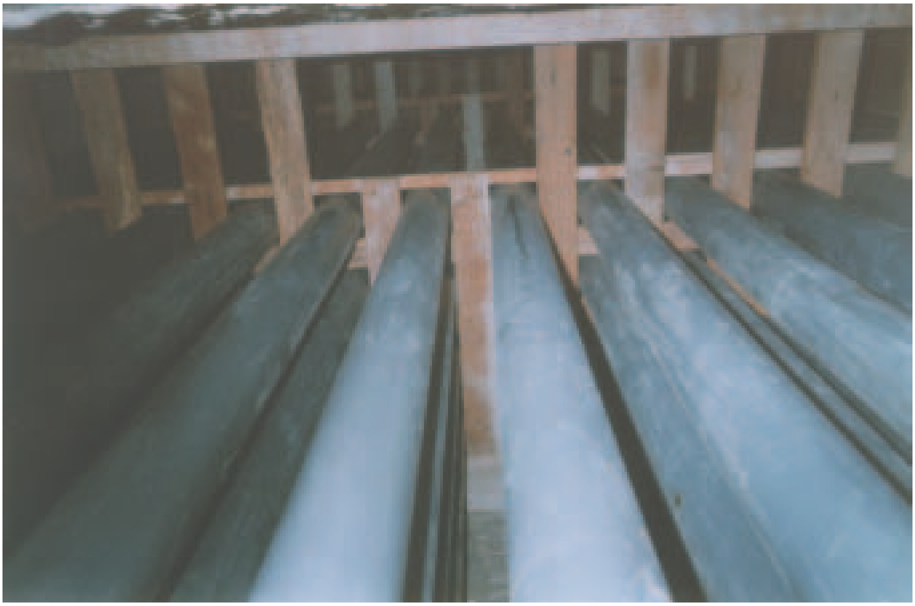


FIG. I.B–2. View of the well showing the pipe distribution. The length of the pipes was 10 m.

The characterization of the facility was an important task in the selection of the decommissioning option. The characteristics of the rooms, equipment and devices, such as their material composition, volumes, geometric shapes and their accessibility for disassembly, were carefully considered in the work planning.

I.B–5. SAFETY AND RADIATION PROTECTION CONSIDERATIONS

The workers were monitored for radiation while the decommissioning operations were carried out. The four crew members who participated in the decommissioning were monitored using a whole body counter. All the measures required for individual protection were considered: protective clothing and disposable overalls, gloves, caps, overshoes and respiratory protection. Personal dosimeters to record the radiation dose received by workers were used. Each operator used two extremity dosimeters, a whole body thermoluminescent dosimeter and a personal dosimeter.

A dose rate monitor and a surface contamination monitor were used in the work area and for checking objects during their decontamination. These monitors were previously calibrated and verified in the CPHR secondary laboratory of dosimetric calibration.

Some arrangements were made on the first and third floors for the zoning of areas, according to radiation and contamination levels. Physical protection systems against unauthorized human intrusion and for the physical security of contaminated areas were in place during the decommissioning.

The ^{226}Ra sources were carefully managed to minimize the exposure of workers to radiation. Special tongs, remote manipulators, a mirror and protective lead shielding were used for handling sources. The length of time of exposure to radiation was limited to the minimum necessary.

Tasks involving radiation exposures were carefully planned in advance and the doses were estimated. An estimation of accumulated person-doses was carried out prior to each activity. These rough estimates were used for planning and budget purposes and also indicated where extra radiation protection measures should be taken.

I.B–6. RADIOLOGICAL CHARACTERIZATION

In order to evaluate the radiological situation in the contaminated rooms, measurements of the dose rate and surface contamination were carried out. Surface contamination was measured directly using a contamination monitor. The wipe test method was used on highly contaminated surfaces and to evaluate the removable contamination. The activity on the swabs was measured using a gamma spectrometer. Significant levels of surface contamination were reported in some lead bricks in room 2 (20 Bq/cm^2) as well as in the walls and on some objects in room 3 (on clothes, a desk, lead containers and a lead apron). Contamination was detected in the floor of room 4. Average contamination levels were around 5 Bq/cm^2, with hot spots of up to 50 Bq/cm^2.

I.B–7. CRITERIA FOR RELEASE FROM REGULATORY CONTROL

The criteria for release were established by the CNSN, as the national regulatory authority, for the unrestricted release of the brachytherapy facility after decommissioning. These criteria were that the residual ingrained surface contamination should be fixed and that the level of surface contamination should be $\leq$0.04 Bq/cm^2.

I.B–8. SELECTION OF DECOMMISSIONING OPTIONS

The selection of the most convenient strategies for the decommissioning required a justification, a proposed timetable and a demonstration of adequate financial provision. Various factors were considered, such as the future use of the facility, the availability of a national waste treatment and storage facility, the technical feasibility and a cost–benefit analysis.

I.B–9. PREPARATION OF A DECOMMISSIONING PLAN

To ensure its success, sufficient time and effort for planning were a requirement before the decommissioning started. The information collected during the physical and radiological characterization was used for the detailed planning of the decommissioning activities. A detailed timetable was prepared for each dismantling and decontamination activity. It was estimated that all the activities would be performed within a month. The procedures used for the radiological evaluation and decontamination of the rooms were drafted prior to the decommissioning, reviewed technically and approved.

I.B–10. DECOMMISSIONING IMPLEMENTATION

The dismantling and decontamination activities began on the first floor. The first operation was to recover the radium sources from the well, which was necessary because the removal of these sources would give a considerable reduction in the radiation exposure of the workers and because the well sources could have influenced the radiation measurements to be carried out on the third floor. Dismantling was necessary to gain access to the radium sources and for the size reduction of the contaminated material to facilitate its handling. The dismantling strategy was straightforward and used simple equipment (Fig. I.B–3). The sand in the well was removed and monitored. The first decontamination activity on the third floor was vacuum cleaning to avoid the spread of contaminated powder. The main contaminated spots on the floor were then cleaned with 5% detergent solution; more specific decontamination reagents were used as required. The floor plastic covering was removed to avoid fixed contamination remaining on the floor. It was necessary to dismantle part of the floor in room 2 because of its contamination. The levels of surface contamination before and after each decontamination activity were recorded in a register. After the decontamination of rooms 1, 2 and 4 a general cleanup was carried out.

Items inside room 3 were carefully monitored. No contaminated objects were directly removed from the controlled zone. Contaminated items were placed into different containers according to their physical and radiological characteristics: one container for compactible solid radioactive waste and another for non-compactible solid radioactive waste. The removable contamination on the floor and walls was successfully dealt with.

FIG. I.B–3. The pipes being cut at the lower level after the external surface contamination is measured. The contamination on the tools used was also checked.

The next step was to demolish the parts of the wall that had fixed ingrained contamination. Firstly, the covers of the well were properly covered with nylon in order to avoid the spread of contamination. The demolition debris was piled up into plastic trays and placed above the wooden covers of the well. The wall was demolished progressively until the contamination was totally eliminated (Fig. I.B–4). After each step the surface contamination was measured and recorded in a register.

The next step was to remove the wood from the top of the well. In order to avoid possible contamination of the well and pipes inside, any powder or dust was removed with a vacuum cleaner. All pieces of wood removed from the controlled zone were carefully monitored. The contaminated parts were placed into the container for non-compactible solid waste. The plastic pipes were also removed by raising them and cutting them into pieces of convenient size. Contaminated pieces were placed in the container for compactible solid radioactive waste. Size reduction was important for minimizing the volume of waste.

I.B–11. EMERGENCY PLANNING

The decommissioning plan specified provisions to mitigate the consequences of possible incidents during the decommissioning process. Since the workplace was

FIG. I.B–4. The contaminated tiles being removed.

a hospital the emergency plan covered the whole medical establishment. Medical personnel were also trained for the treatment of radioactively contaminated persons.

I.B–12. WASTE MANAGEMENT

It was appreciated that decommissioning is inextricably linked with the management and ultimate disposal of radioactive waste. The generation of radioactive waste from the decommissioning process was kept to the minimum practicable by using appropriate decontamination and dismantling techniques. Before decommissioning, consideration was given to the different categories of waste to be generated and their safe management. Radioactive waste was adequately segregated and characterized to facilitate the overall safe management of conditioning and long term storage.

I.B–13. FINAL RADIOLOGICAL EVALUATION

The purpose of the final radiological measurements (Fig. I.B–5) was to demonstrate that the requirements established by the regulatory authority were met, and therefore that the facility could be released from regulatory control. The final radiological status report was submitted to the CNSN, which reviewed the report and inspected the facility. The CNSN did not identify a need for a further decontamination or further surveys.

I.B–14. TRACEABILITY

The progress of the decommissioning was documented in detail. All radiological measurements were recorded in a register. Containers with radium

FIG. I.B–5. Final survey of the well after removing all radiation sources, pipes and sand. The measurements show the natural background levels.

sources and radioactive waste were properly identified. Their radiological and physical characteristics were also reported in a register.

I.B–15. CONCLUDING REMARKS

The decommissioning project was successfully completed. Adequate project management was applied for safety assurance, radiation protection and waste management.

The following was achieved:

- One hundred and thirty six spent ^{226}Ra sources were recovered from the facility and properly managed.
- The volume of waste generated during the decommissioning was:
  - Non-radioactive waste: around 5 m^3 (for example sand, tubes and wood).
  - Compactible radioactive solid waste: 0.4 m^3 (for example PVC tubes, gloves and material used for decontamination).
  - Non-compactible radioactive solid waste: 1 m^3 (for example wood, lead and sand).
- The requirements established by the regulatory body to release the facility from regulatory control were met.
- Simple and effective decontamination and dismantling technology was applied for the decommissioning of the facility, resulting in minimal radioactive waste being generated and the immediate site release from regulatory control.

The Oncology Institute received the authorization from the regulatory body for the unrestricted use of the facility upon the successful completion of the decommissioning.

Annex I.C

DECOMMISSIONING OF SMALL MEDICAL, INDUSTRIAL AND RESEARCH FACILITIES IN THE CZECH REPUBLIC

I.C–1. GENERAL INFORMATION

The management of nuclear installations and facilities that use radiation sources in the Czech Republic is regulated by what is know as the Atomic Law of 1997 (Law No. 18/1997, Sec. on the Peaceful Uses of Nuclear Energy and Ionising Radiation and on Amendments and Additions to Related Acts), which sets out the principles for decommissioning. Responsibilities in this field are divided among three bodies: the State Office for Nuclear Safety (the Czech independent regulatory body), the operator of the facility concerned and the Radioactive Waste Repository Authority (the Czech waste management agency).

- The State Office for Nuclear Safety (the Czech independent regulatory body):
  - Approves specific documents and issues licences for the decommissioning process.
  - Supervises radiation and nuclear safety during the performance of the process and collects and archives relevant information.
  - Issues regulations (Regulation No. 196/1999, Sec. on Decommissioning of Nuclear Facilities or Facilities with Significant or Very Significant Radiation Sources) to ensure radiation protection during the decommissioning process, to set out the decommissioning methods to be used and to specify the documentation required for the licensing procedure.
- The operator of the facility:
  - Plans and is responsible for executing the decommissioning process.
  - Based on the plan, suggests the total required for the relevant financial reserve fund for the preparation and performance of the decommissioning.
  - Creates a reserve fund for the decommissioning of its facilities.
- The Radioactive Waste Repository Authority (the Czech waste management agency):
  - Verifies that the reserve fund for decommissioning is properly estimated.
  - Controls the operator’s payments to the reserve fund.

Legislation demands that an operator should plan the decommissioning process in the early stages of the development of its facilities. The first plan must be delivered as part of the siting procedure. An updated decommissioning plan is required in all later licensing steps, namely permission for construction, the commissioning of the facility and the termination of its operation. The level of a reserve fund must be updated at five-year intervals and also after changes that may influence significant aspects of the plan.

The level of the involvement of the regulator in the preparation, performance and evaluation of decommissioning activities depends on the type of the facility. A facility is placed into one of the following categories based on its potential for risk to human health and the environment due to ionizing radiation from the sources it handles:

- Insignificant sources: sources that in their handling are not associated with the possibility of a radiation accident or with the generation of radioactive waste.
- Minor sources: sources that in their management are not associated with the possibility of a radiation accident but that may generate radioactive waste.
- Simple sources: sources that in their management are associated with the possibility of a radiation accident, although these radiation accidents will have no acute health effects.
- Significant sources: sources that in their management are associated with the possibility of radiation accidents that have acute health effects, although there is no possibility of a radiation emergency.
- Very significant sources: sources for which consideration must be given to the possibility of a radiation emergency.

Facilities in the latter two categories require a licence prior to the start of their decommissioning. For the other cases the only requirement is to notify the regulator once decommissioning is completed.

This brief description of the institutional structure is based on relatively recent acts. Some practical examples that deal with decommissioning cases performed both prior to and after the introduction of these acts are presented below to demonstrate the development of the formal approaches to the decommissioning process.

I.C–2. CASE HISTORIES

I.C–2.1. Case I: A repair shop for measuring devices

In a repair facility was a laboratory in which dials painted with phosphorescent paints spiked with ^{226}Ra were renovated and maintained; the laboratory also dealt with radioimmunoassay sources containing ^{90}Sr isotopes. As these types of measuring device were banned from further use it was decided to close the laboratory in 2000, after some 40 years of operation. On the basis of the amounts and types of radionuclide used, the laboratory was considered to be in the ‘simple radiation sources’ category, which means that the role of the regulator is limited only to the final confirmation that the facility is ready to be released for unrestricted use. However, documentation was prepared in accordance with that required by the Czech regulations for a higher category of facility.

I.C–2.1.1. The planning stage

After surveillance of the laboratory the implementer (a company with licences for, for example, the management of radiation sources and radioactive waste and for transportation) developed, in co-operation with the operator, a plan for the decommissioning process. The plan contained the information shown in Sections I.C–2.1.1.1– I.C–2.1.1.9.

I.C–2.1.1.1. Identification of the implementer and the operator

Both the companies and the responsible persons were identified.

I.C–2.1.1.2. Assignment of the project

The project goals were identified as:

- The decontamination of all laboratory surfaces, equipment, floor material and walls;
- The segregation of the waste to be treated as radioactive from waste that could be released for unrestricted use or dumped as municipal waste;
- The treatment and conditioning of all radioactive waste generated during the decommissioning.

I.C–2.1.1.3. Description of the laboratory

The location, layout and dimensions of the laboratory were provided. The equipment and furniture to be treated were listed.

I.C–2.1.1.4. Radiocontaminant specification

All radionuclides in the laboratory were listed with qualitative (relative) estimates of their activity, nature, forms and state.

I.C–2.1.1.5. Monitoring of radiation parameters

Devices and relevant methods were selected for the determination of key radiological parameters (e.g. the personnel dosimetry required, exposure rate and surface contamination measurements and the sensitivity and effectiveness of the measurement techniques to be used).

Criteria for the decision on the release of material into the environment, based on legislative levels (Regulation No. 184/1977, Sec. on Requirements for Assuring Radiological Protection), were listed for the radiocontaminants in the laboratory. They were expressed both as surface contamination (3 kBq/m^2) and as mass activity (0.3 kBq/kg).

I.C–2.1.1.6. Work procedures

The work procedures and equipment to be used for the contamination identified on laboratory surfaces and on items in the laboratory were proposed. These were for:

- The detection of contaminated material and the evaluation of the level of their contamination (a storage area for those items that could be released was found and adopted prior to the start of the decommissioning);
- The disassembly and segregation of contaminated and non-contaminated material at the laboratory and at the implementer’s facility;
- The decontamination techniques to be applied at the operator’s and the implementer’s facilities (i.e. the material, agents, techniques and devices to be used);
- The transport of contaminated material for treatment by the implementer and for processing as radioactive waste.

I.C–2.1.1.7. Radiological safety

Radiation protection, the principles of which were based on the monitoring programme approved by the regulator prior to the start of the technical activities (both a system of personal dosimetry and a system of quality control during the implementation of the decommissioning process), was assured by the consistent supervision of all activities by an independent skilled dosimetrist, who certified any measurement performed by the implementer’s staff.

I.C–2.1.1.8. Metrology

All devices used during the decommissioning were calibrated and linked to standardized devices in accordance with the relevant law (Law No. 505/1990, Sec. on Metrology).

I.C–2.1.1.9. Records and documentation

Methods and procedures were proposed for the storage of documentation on major activities, primarily for the management of radioactive material and for the release of material out of the jurisdiction of the Atomic Law.

Protocols were proposed to document all the decisions, namely those dealing with:

- The separation of dismantled material to be treated as radioactive waste or to be released into the environment;
- The management of radioactive waste to be disposed of (a waste package passport);
- The final radiological survey in the laboratory after the decommissioning, which is an obligatory input for any decision on the release of a facility for unrestricted use.

The decommissioning plan was submitted to the regulator for information purposes (its approval was not required for this type of facility).

I.C–2.1.2. Implementation of the decommissioning plan

When performing the planned activities the step by step procedure outlined below was implemented. All significant information regarding radiological safety was recorded.

The procedure involved fulfilling the following tasks:

- Releasing all the equipment and furniture that did not need to be dismantled (i.e. the devices and furniture that were not fixed to the structure of the building).
- Dismantling the equipment and fixed furniture (laboratory tables, sinks, drainpipes and laboratory equipment supports) with the use of hand tools, a saw equipped with an exhaust system and/or hydraulic scissors.
- Continuous vacuum cleaning to collect any particulate material.
- Removing the wall protective coatings.
- Decontaminating small items on the site and transporting larger ones to the implementer’s specialized facilities. The decontamination solutions typically used on the site included a 5% Syntron solution (a mixture of complexing agents) and a 2% surfactant solution. Similar solutions, degreasing agents, high pressure water and active decontamination foam were utilized at the implementer’s facilities.
- Decontaminating the floor and walls, mostly by the mechanical removal of contaminated layers.

The proposed procedures were modified when needed; for example, a floor covering made of a rubber-like material was to be ripped up. It was learned that the concrete below the covering was contaminated (probably by contaminated water penetrating through it while the floor was being washed), so the removal of the concrete surface was added to the scope of the project.

The activities of the staff were monitored throughout the process, focusing primarily on:

- The radiation safety of the personnel (e.g. the use of protective clothes and masks, dosimetric monitoring during the work and whole body measurements after the completion of the procedure);
- The protection of the surrounding environment (e.g. limiting the ventilation of the laboratory during the destructive processes and the transport of contaminated items in overpacks and plastic bags);
- Controlling dose rates from the contaminated equipment, furniture and construction material;
- The removal of surface contamination from processed items.

All monitoring data were collected and recorded. An assessment protocol was developed for the radiation measurements. The protocol typically involved collecting the following data:

- The place and the time of the measurement;
- The characteristics of the measured subjects;
- A short description of the method used;
- A specification of the devices used, including their measuring capabilities;
- A description of the work procedure;
- The results of the measurement;
- The names and signatures of the persons involved in performing and supervising the measurement.

All the waste (both radioactive and non-radioactive) generated during the decommissioning was described and characterized and the records were kept for eventual inspection.

Copies of these data were distributed to the operator, the implementer and the supervisor (the State Office for Nuclear Safety).

I.C–2.1.3. Post-decommissioning activities

After the removal of all radioactive material from the laboratory and after the decontamination of all surfaces to below the limit values, the final report was issued.

The final report documented that the set goals were reached and that the defined release criteria were met. This report contained information summarizing the following topics:

- The identification data of the decommissioned facility;
- The main characteristics of the activities performed;
- A list of the radioactive and non-radioactive waste produced (see Table I.C–I);
- The final status survey of the laboratory (see Table I.C–II);
- The principles of the methods applied during the process (as described above);
- The period in which the procedure was performed (the laboratory was decontaminated between 8 June and 21 July 2000);
- The final statement on the release of radioactive material from the laboratory, which allows the laboratory to be used without restriction.

This document was submitted to the regulator, which then approved the unrestricted use of the facility.

TABLE I.C–I. TYPICAL EXAMPLES OF THE WASTE GENERATED

Identification No.	Type	Volume (L)	Generation date	Description	Nature
901.01	Solid	50	8 June 2000	Rubbish, dials, devices and glass	Radioactive
901.07	Solid	200	12 June 2000	Iron scrap	Radioactive
901.23	Liquid	30	13 June 2000	Organic liquid	Radioactive
901.31	Solid	1234	19 June 2000	Wooden drawers	Radioactive
901.71	Solid	60	28 June 2000	Iron radiator	Non-radioactive
901.76	Solid	20	11 July 2000	Washbasin	Non-radioactive
901.91	Solid	20	13 July 2000	Wall paint and wall debris	Radioactive
901.92	Solid	120	13 July 2000	Rubber floor covering	Radioactive
901.99	Liquid	40	18 July 2000	Rinse water	Radioactive
Total					
		1720		Non-radioactive waste	Non-radioactive
		7510		Radioactive waste	Radioactive

TABLE I.C–II. SUMMARY OF THE FINAL STATUS SURVEY

Date of measurement	18–19 July 2000
Method employed	Direct measurement by aerial detectors
Type of device	Radiometer RP 114A (scale: 0.3–300 Bq/cm^2) Berthold LB 1210D (scale: 0.2–1999 Bq/cm^2)
Release limit	Surface contamination: 0.3 Bq/cm^2
Investigated area	Floor, walls, windows and doors (total area of 240 m^2)
Measured contamination	<0.3 Bq/cm^2

I.C–2.2. Case II: An irradiation facility

A ^{60}Co source (with a diameter of 15.5 mm × 140.4 mm) was installed in a chemical factory that produced tetrabromoxylene: its activity on the date of dismantling was 33.3 TBq. The irradiation facility was located in a separate small building–bunker with concrete walls of a thickness of 1.8 m. The cobalt source was inserted in an S-shaped stainless steel pipe. The source capsule was raised into the irradiation chamber mechanically by means of a cable and pulley.

The underground part of the facility consisted of two blocks, an inner and an outer block. Each block was composed of two stainless steel pipes of different diameters, which allowed both the safe handling of the source and the physical protection of the personnel. The inner block, which could originally be removed as a whole, included an operational S pipe (diameter approximately 50 mm and approximately 3 m deep), which was surrounded by concrete inside another pipe (diameter approximately 30 cm). This pipe was in turn placed inside another pipe belonging to the outer block; the void space between them originally allowed the vertical movement of the inner system. Finally, all these pipes were placed in a stainless steel vessel (diameter approximately 1 m), in which concrete was used as a filler (see Fig. I.C–1).

The source capsule became bonded to the bottom of the S pipe, probably as a consequence of corrosion. The cable used to raise the source into the irradiation chamber broke when attempts were made to dislodge the source. The system of emergency source retrieval was also rendered ineffective by the corrosion. As a consequence, the source was fixed in the S pipe at a depth of 2.8 m. In conjunction with a change in the production programme of the factory, it was decided to decommission the facility. The decommissioning work was completed in the mid-1980s.

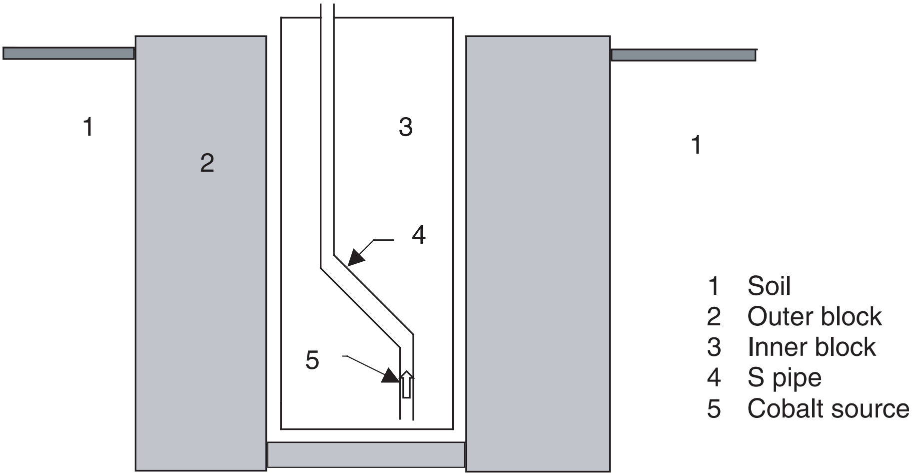


FIG. I.C–1. A simplified scheme of the irradiation facility.

This radiation source was in the ‘significant radiation sources’ category, but at the time of the decommissioning this classification was not in use. The Ministry of Health and regional hygiene offices were responsible for radiation protection, and were entrusted to provide the State’s independent supervision. The valid regulation of the State Office for Nuclear Safety (Regulation No. 59/1972, Sec. on Health Protection Against Ionising Radiation, among others) listed the institutions permitted to manage radiation sources. As the selected implementer was on this list, the administrative approval procedures were simplified. This company provided regular services in waste management and possessed all the necessary licences (e.g. for the management of radiation sources, waste management and the transport of radioactive material). The company’s staff was experienced in the practical implementation of the principles of radiation safety.

I.C–2.2.1. The planning stage

The decommissioning steps used for an irradiation facility consist of the removal of all radiation sources, checking whether the work surfaces are contaminated (and their decontamination, if necessary) and the demolition of any special equipment. The radiation sources are normally kept in leak-tight capsules, which prevents the contamination of surfaces.

The main problem in this case consisted of the retrieval of the source from the facility and its placement into a transport container. During the planning stage several options were considered, all of which were based on the removal of the inner block, its insertion into a heavy concrete transport container and its disposal in a repository. As the dose rate of the source was high, a thorough evaluation of potential accident situations had to be performed.

A plan for the decommissioning was developed based on evaluations of the situation. This included:

- Safely fixing the capsule in the S pipe;
- A procedure to release and retrieve the inner block containing the radiation source;
- A procedure to insert the inner block into the transport container;
- A calculation of the shielding capacity of the container;
- Safety and emergency considerations for an unplanned event;
- The transport to and disposal of the container in a repository.

The safety documentation for the plan was verified by the regional hygiene office; technical activities could start once it was approved.

I.C–2.2.2. Implementation of the decommissioning plan

The first phase of the approved procedure consisted of fixing the source in the S pipe. Concrete mortar with a plasticizer was poured into the pipe and the whole void volume of the pipe was filled using a vibration machine.

The bunker was destroyed using explosives to allow heavy machinery to have access to the pipe system. The inner block was to have been released by drilling out the surrounding pipe, but drilling tools failed to penetrate its metallic surface. The planned procedure was therefore changed so that the upper part of the outer block was destroyed, followed by manipulation hooks being welded to the inner block to enable it to be moved with the use of a special machine and lubricants. The whole inner block was then lifted into a transport container made of heavy concrete and its upper part was cut out so that only the active zone was left in the container.

The dose rate was measured while these procedures (i.e. destroying the bunker and the outer block and inserting the inner block into the container) were carried out. Dose commitments were recorded and provided to the hygiene authorities. The shielding thickness required for the shipping container was calculated and additional shielding was found to be necessary; a layer of lead shielding was therefore inserted into the critical part (the level at which the cobalt source was fixed) of the shipping container.

I.C–2.2.3. Post-decommissioning activities

The level of radiological contamination on the site was checked after the inner block was released. The result was negative and thus the site could be released for unrestricted use. Documentation containing the following information was prepared:

- The place and the date of the measurement;
- The characteristics of the measured subjects;
- A short specification of the device and the method used;
- The results of the measurement;
- The names and signatures of the persons involved in performing and supervising the measurement.

This documentation was used as part of the final report, which also contained an evaluation of the costs and a summary of the technical procedures applied.

The container was transported to a repository, where it is still stored in an isolated location and where physical inspection and radiation monitoring continues.

I.C–2.3. Case III: Nuclear medicine department

A hospital department in which radiopharmaceuticals were administered to patients was moved to a new building after 40 years of operation. It had been decided that the original area would be reused for non-radiological purposes and therefore had to be decommissioned. The work was completed in 1998.

The department was located in two identical one-storey buildings with outer dimensions of 15 m × 45 m. The first building contained an endocrinology laboratory, which used only very short lived radionuclides (typically $^{99}Tc^{m}$). In the other building (see Fig. I.C–2) thyroid gland endocrinic ophtalmopathy and thyreotoxicosis were diagnosed and treated with the use of ^{131}I. Some other isotopes were permitted to be used in the department (see Table I.C–III), but the only contamination of any significance was caused by radioiodine. On the basis of the type and amount of isotopes used the department was classified as a workplace with ‘significant radiation sources’.

Solutions of radioiodine (in the form of NaI) were diluted in an air hood (see Fig. I.C–3) to a concentration of 7–72 MBq/dose for diagnostic purposes and 3–10 GBq/dose for therapeutic purposes. About 37 GBq of ^{131}I solution was processed in a typical week. Patients provided urine samples in containers, which were then measured. After measurement the containers were emptied and washed in a special automatic device (see Fig. I.C–4).

Urine and other active solutions were collected in three tanks built in the basement of the building, each with a volume of 10 m^3: while one was in use (the usual

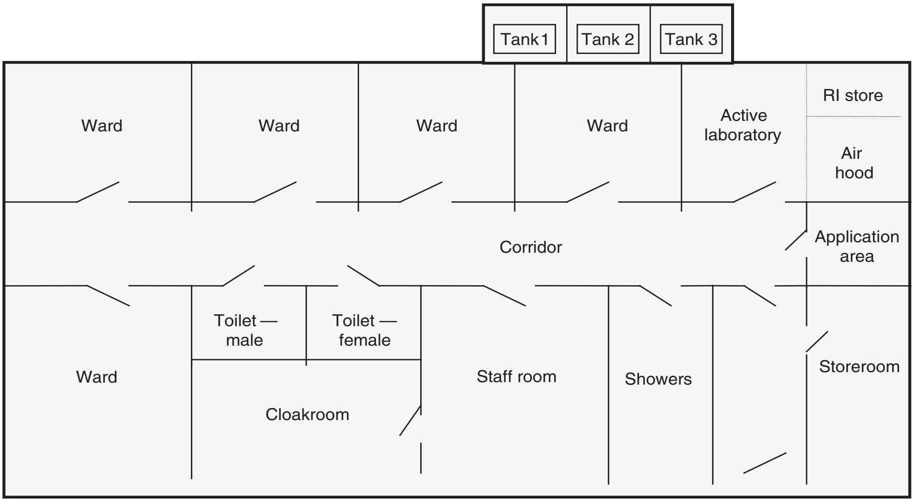


FIG. I.C–2. A simplified schematic layout of the nuclear medicine building (not to scale).

filling period was approximately three months) the other tanks were left until the radioactivity in the liquids decayed. Solid waste contaminated by radioisotopes was collected in polyethylene bags and stored in a shelter adjacent to the building. After several months of decay the waste was checked for contamination and, when releasable, was treated as non-radioactive waste and incinerated or disposed of at the municipal dump.

I.C–2.3.1. The planning stage

Owing to the short half-lives of the radionuclides used it was decided to decommission the building using the delay–decay method. This method involved three major steps:

- The transfer of the sources (both opened and sealed) to the new facility;
- The identification of contaminated surfaces and equipment;
- Regular checking of the contamination levels of the contaminated surfaces and equipment until they reached the prescribed limit.

TABLE I.C–III. RADIOISOTOPES APPROVED FOR USE IN THE DEPARTMENT

Isotope	^{3}H	^{14}C	^{32}P	^{67}Ga	^{90}Y	^{99}Tc^{m}
Allowed activity	40 MBq	10 MBq	10 MBq	200 MBq	4 GBq	40 GBq
Isotope	^{113}In^{m}	^{123}I	^{125}I	^{131}I	^{198}Au	^{201}Tl
Allowed activity	4 GBq	2 GBq	70 MBq	40 GBq	400 MBq	100 MBq

FIG. I.C–3. A laboratory processing radiopharmaceuticals.

The decommissioning plan was submitted to the regulator (the State Office for Nuclear Safety) for approval, together with a radiological map of the building. The following limits were set for the release of the facility:

- Surface contamination limit: 0.5 Bq/cm^2;
- Specific activity limit for liquids: 50 Bq/m^3.

I.C–2.3.2. Implementation of the decommissioning plan

The procedures applied consisted of checking the surface contamination and the specific radioactivity of the items measured. All moveable furniture and equipment, after their radioactivity content decayed to below-limit levels, were released for reuse or for treatment as non-radioactive waste. Each measurement was documented on forms containing information concerning the item, the measuring device and method used, the measured values and a final statement on the release of the item.

These results are summarized in Table I.C–IV.

FIG. I.C–4. A device for emptying and washing urine pots.

I.C–2.3.3. Post-decommissioning activities

Based on the radiological survey completed in the building, a proposal for its release for unrestricted use was submitted to the regulator. It contained a summary of the activities performed, a summary of the measurements of the contamination of surfaces and waste liquids and information concerning the doses received by the personnel during the decommissioning.

I.C–2.4. Case IV: Decommissioning of a radioisotope workplace

The institute for importing, producing, labelling and distributing radioisotopes and managing institutional radioactive waste for the whole of Czechoslovakia was

TABLE I.C–IV. SUMMARY OF CONTAMINATION CHECKING IN THE DEPARTMENT OF NUCLEAR MEDICINE

Checked space and/or item	Surface contamination level at the date of measurement (Bq/cm^2)		
	9 March 1998	5 June 1998	25 August 1998
Floor — female cloakroom	<0.5	<0.5	<0.5
Floor — wards	1	<0.5	<0.5
Floor — corridor	<0.5	<0.5	<0.5
Floor — isotope application area	1	<0.5	<0.5
Device for emptying urine pots	150	30	<0.5
Air hood	200	40	<0.5
Shower — patients	2	<0.5	<0.5
Floor — male patients’ toilet	2	<0.5	<0.5
Toilet bowl — male patients’ toilet	3	<0.5	<0.5
Floor — female patients’ toilet	1.5	<0.5	<0.5
Toilet bowl — female patients’ toilet	3	<0.5	<0.5
	Specific activity of liquid waste in tanks (Bq/m^3)		
Tank 1	<50	<50	<50
Tank 2	100	<50	<50
Tank 3	400	50	<50

located from 1959 in a building that had formerly been used by the State Radiological Institute (SRI). Scientific and research activities were carried out using radiation sources for non-energy applications. About 1 to 10 PBq of radioisotopes were processed annually in its laboratories, mostly in a non-sealed form.

A new facility was built in 1979, which meant that the original building had to be decontaminated, decommissioned and renovated for non-radioactive uses.

The main building stands in a municipal housing estate, surrounded by multistorey blocks of apartments and offices. The main part of the building consists of five floors, including a basement. The skeleton of the building is made of reinforced concrete, while the walls are built of bricks. The area of each floor is approximately 400 m^2. Behind the main building are two auxiliary two-storey buildings that were used as garages and non-radioactive storage areas.

I.C–2.4.1. The planning stage

To prepare the building for renovation, all radioactive material and non-radioactive equipment and furniture had to be removed. The remaining contamination of the building was estimated at 100 GBq, with the radionuclides ^{3}H, ^{14}C, ^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra and ^{241}Am identified as the principal sources.

The procedure adopted for the decommissioning of the building consisted of the following steps:

- The transfer of all activities and personnel to new facilities and the release of the existing building for decontamination;
- The detailed mapping of contaminated areas, surfaces and movable items;
- The creation of a new support area for conducting the decontamination activities and a new laboratory for personal dosimetry;
- The decontamination of the radioisotope laboratories;
- Clearing out the excess material and equipment from the building;
- The dismantling of equipment;
- The systematic decontamination of all surfaces and equipment to be left on the site;
- An evaluation of the effectiveness of the decontamination;
- The submission of an application for the release of the building for non-radioactive uses (i.e. closing down the radioisotope workplace).

I.C–2.4.2. Implementation of the decommissioning plan

Transferring the operation to a new facility was relatively simple, since all the furniture and most of the equipment and laboratory items, with the exception of specialized equipment and radioisotopes, was to remain in the old building. Since the new owner also planned to install new equipment, most of the material removed from the old buildings was shipped for disposal.

The determination of contaminated areas was carried out using tested and routinely employed portable instrumentation for detecting surface alpha, beta and gamma contamination, gamma dose rates and concentrations of radionuclides in the air. Spectral analyses and the total activity of contamination samples were measured in a laboratory installed in the new facility. The regulatory limits were set at 0.3 Bq/cm^{2} for radium and americium and 3 Bq/cm^{2} for the other nuclides. The maximum effective dose rate was set at 0.3 μSv/h. The most contaminated places were clearly marked to reduce the risk of undesirable doses to the staff.

Most of the equipment had to be dismantled prior to the decontamination of the building. A support area for the personnel involved in the decommissioning therefore had to be built. This facility, which consisted of cloakrooms, showers, and dosimetry and administrative spaces, was installed in the non-contaminated part of the building.

Initial decontamination was carried out by laboratory personnel employing the usual tools and equipment available at all radioisotope workplaces. The dismantling of equipment such as air hoods, alpha boxes and special devices was completed at this stage. Any radioactive waste found was collected and treated. Staff doses were monitored by personal dosimeters and by whole body measurements.

All equipment released from the building was monitored for contamination before being disposed of at a municipal dump or in the repositories of Bratrství and Richard (when found to be contaminated with radium and with other isotopes, respectively). Some larger items contaminated with radium were placed in the disposal pond of a uranium ore milling plant.

The dismantling of special systems that had been in contact with radioactive material began at the same time. These systems included the air ventilation and filtration system, active sewage treatment equipment, drains, retention tanks and the cooling system. Once these systems were removed, the contaminated floor coverings were removed and disposed of.

After the equipment and fixtures were removed from the laboratories all the empty spaces were systematically decontaminated, generally with the use of wet and non-invasive mechanical processes. Great emphasis was placed during this time on the detection of, and protection of personnel against, airborne contamination.

The final evaluation of surface decontamination was performed in the presence of independent hygiene institution representatives. It was found that, in certain areas, radiological contamination exceeded the regulatory values. The ^{226}Ra isotope was identified as the principal source of the contamination. It was found that in some places contamination of structural material had occurred, which was probably due to leaks from radioactive drain systems. This had probably occurred during the operation of the SRI, which had been the monopoly producer of this radioisotope.

The only way to solve this radiological problem was to demolish large volumes of contaminated structural material. Plaster, flooring and fill layers were removed, contaminated soil in the basement was excavated, and the material disposed of as radioactive waste. Subsequent confirmatory measurements showed that, in all the affected places, contamination by ^{226}Ra and its progeny was below regulatory limits.

I.C–2.4.3. Post-decommissioning activities

Based on measurements performed after the completion of the decontamination, an application was submitted to the responsible hygiene authorities for closing down the radioisotope building. A set of confirmatory radiological measurements was collected. After proving that the remaining concentrations of radioactive substances were below limits and would not result in doses to the public exceeding the legal dose limits, the authorities granted the application.

The problem remained of persuading the new user of the building that the remaining contamination posed a negligible risk to the personnel working in it. Additional measurements were therefore performed in the presence of the persons who would be located on the premises. To demonstrate how negligible the radiation levels were, parallel measurements were taken at another building in which radioactive substances had never been used and in the open air. Further confirmatory measurements taken by an independent radiological laboratory showed that the radiological situation was within the prescribed limits.

Annex I.D

DECOMMISSIONING OF A BRACHYTHERAPY FACILITY AT THE DR. HERIBERTO PIETER ONCOLOGY HOSPITAL IN SANTO DOMINGO, DOMINICAN REPUBLIC

I.D–1. INTRODUCTION

In the past the Dr. Heriberto Pieter Oncology Hospital in Santo Domingo used ^{226}Ra and ^{137}Cs sealed sources for brachytherapy. One of the ^{137}Cs sources leaked and caused contamination on some areas, including some of the equipment and devices used (for example patient beds, the bathroom, containers, medical material and instruments). Owing to safety considerations the facility was shut down.

In 1996 the National Commission of Nuclear Affairs (CNAN) of the Dominican Republic requested the IAEA to evaluate the radiological situation in the contaminated areas in the hospital and to carry out a decontamination of the rooms and the decommissioning of the brachytherapy facility for unrestricted use.

The decommissioning activities were carried out in November 1996.

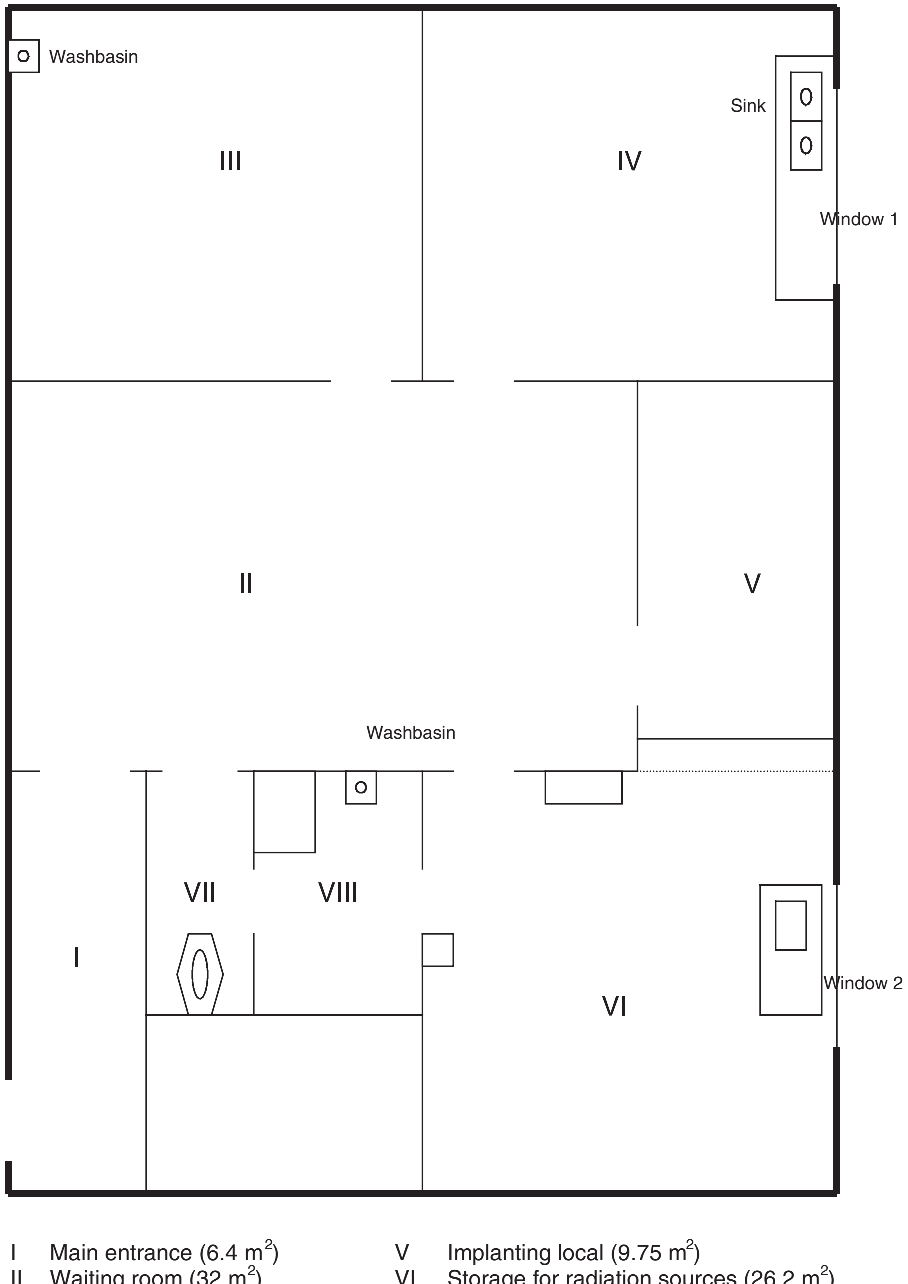
I.D–2. DESCRIPTION OF THE FACILITY

The brachytherapy facility was a small part of a larger non-nuclear facility (the hospital).

The contaminated areas are shown in Fig. I.D–1. There were eight rooms, with a total area of 130 m^2, and which were identified as rooms I–VIII. These rooms were locked for a long period of time, so a considerable amount of dust had accumulated.

I.D–3. SAFETY AND RADIATION PROTECTION CONSIDERATIONS

During decommissioning operations the monitoring of workers for radiation was carried out. The crew members who participated in the decommissioning were monitored using a whole body counter. All the measures required for individual protection were considered: protective clothing and disposable overalls, gloves, caps, overshoes and respiratory protection. Personal dosimeters to record the radiation doses received by workers were used. Each operator used two extremity dosimeters, a whole body thermoluminescent dosimeter and an ionization dosimeter (model FJ-351) to control the daily received dose.



I Main entrance (6.4 m^2)
II Waiting room (32 m^2)
III Patient room (23.4 m^2)
IV Patient room (23.4 m^2)

V Implanting local (9.75 m^2)
VI Storage for radiation sources (26.2 m^2)
VII Toilet
VIII Toilet

FIG. I.D–1. Contaminated rooms at the Dr. Heriberto Pieter Oncology Hospital.

A dose rate monitor and a surface contamination monitor were used in the work area and for checking objects during the decontamination. These monitors had been previously calibrated and verified in the CPHR Certified Secondary Laboratory of Dosimetric Calibration in Havana (see Annex I.B). The surface contamination monitor was periodically checked using a plane calibration radiation source of ^{204}Tl with an effective area of 210 mm × 130 mm and a surface activity of 8.87 Bq/cm^2.

Radium-226 and ^{137}Cs sources were carefully managed to minimize radiation exposure to workers. Special tongs, remote manipulators, a mirror and protective lead shielding were used for handling sources. The radiation exposure time was limited to the minimum necessary.

I.D–4. RADIOLOGICAL CHARACTERIZATION

In order to evaluate the radiological situation in the contaminated rooms, measurements of the dose rate and surface contamination were carried out. Surface contamination was measured directly using the contamination monitor. The wipe test method was used on highly contaminated surfaces and to evaluate the removable contamination. The activity on the swabs was measured using a gamma spectrometer.

I.D–5. RADIOLOGICAL RELEASE CRITERIA

The criteria defined by the Dominican Republic regulatory authority for the unrestricted release of this brachytherapy facility after decommissioning were that the residual ingrained surface contamination should be fixed and that the level of total surface contamination should be $\leq$0.04 Bq/cm^2.

I.D–6. DECOMMISSIONING IMPLEMENTATION

Before entering each room the dose rate was measured from the door and a PVC pathway was prepared on the floor. The next step was to make a preliminary evaluation and to check all details inside the room (for example the radiation and contamination levels, other possible accesses and/or exits, construction details, devices and the need for additional illumination and/or ventilation). Surface contamination was measured directly wherever possible. The wipe test procedure was used in highly contaminated areas, particularly to determine whether the contamination was fixed. Thirty six wipes were taken from all contaminated areas, which were measured at the Physical Institute of the Science Faculty at the

Autonomous University of Santo Domingo. Some wipes were also measured at the CPHR in Havana. The wipe tests confirmed that the contaminant was ^{137}Cs.

Removing the dust and non-fixed contamination was the first operation undertaken in each room. These operations were carried out manually and in accordance with the established procedures. Conventional methods of wet cleaning were then used, with the wastewater volumes being carefully minimized. Detailed monitoring was carried out in the rooms that contained a lot of furniture. No contaminated items were removed from the control zone. In the event of contamination a quick and simple cost–benefit analysis of their decontamination was made.

Either paper towels or abrasive steel wool, depending on the surface, were used for the manual decontamination. Pickaxes, hammers, hand saws and other tools were used for dismantling contaminated furniture and floors.

After the furniture was removed from each room and the floors were cleaned and covered with PVC sheets; the floor and walls up to a height of 2.20 m were divided into grids. The area of each grid ranged from 1 m^2 to 3 m^2.

Each grid was carefully monitored in accordance with the established procedure. The most contaminated areas in the grid were additionally marked with the value of the surface contamination.

The removable contamination on the floor and walls was dealt with successfully. Parts of the floor surface with fixed ingrained contamination were removed by scabbling.

The floors of rooms I and II (which were used as a control zone) and room III (which was used as a temporary store for radioactive waste and spent sources) were covered with PVC sheets after their decontamination, as thick PVC allows people and objects to transit while minimizing the risk of the spread of contamination. The floors were also covered with PVC sheets before furniture, walls, windows or other objects were destroyed or demolished.

Furniture, medicines, medical devices, tools and other objects from the contaminated areas were carefully monitored, classified and segregated. Contaminated items were removed and collected as radioactive waste and handled under controlled conditions.

After the dismantling and decontamination work each room was again surveyed to verify the remaining levels of radiation and contamination. The decontamination work continued until the level of contamination was at the background level.

I.D–7. BRIEF DESCRIPTION OF DECOMMISSIONING ACTIVITIES IN EACH ROOM

I.D–7.1. Room II: Waiting room

The floor and walls of room II, which had areas of 32 m^2 and 53 m^2, respectively, were monitored. Some furniture, a mobile shielding device and other objects were in this room. All the furniture and objects were monitored, but only the wheels of the mobile shielding device were found to be contaminated. The contamination was fixed, so the items were removed as radioactive waste. Some points on the floor had fixed ingrained contamination (50 Bq/cm^2). These points were marked and later removed.

The room was used as a control zone. The floor was cleaned and covered with PVC sheeting.

I.D–7.2. Room III: Room for patients with implants

The floor and walls of room III, which had areas of 23.4 m^2 and 43 m^2, respectively, were monitored. Three beds, three small tables, a refrigerator, a seat and a washbasin were in this room. A mattress with a contaminated cover was also found in the room, which had a contamination level of around 10 Bq/cm^2; the contaminated parts were cut out and removed as radioactive waste. Two small tables had a contamination level of around 10 Bq/cm^2 and around 2 m^2 of floor surface near the washbasin had to be removed. It appears that the contamination in this area was caused by a break in the washbasin drainpipe.

After its decontamination, this room was used as a temporary store for radioactive waste and spent sealed sources.

I.D–7.3. Room IV: Room for patients

The floor and walls of room IV, which had areas of 23 m^2 and 43 m^2, respectively, were monitored. Three beds, three small tables, a table, a sink and other objects were in this room. One of the metallic beds had to be decontaminated. The three small tables were contaminated: one was decontaminated and the other two were demolished and removed as radioactive waste. Two contaminated pillows were also removed as radioactive waste.

There was a small contaminated cardboard box under the workbench, which had a contamination level of around 400 Bq/cm^2, and a metallic drawer with some contaminated medical devices, which had a contamination level of around 500 Bq/cm^2.

The most critical situation in this room was found in the sink. The dose rate above the sink was 55 μSv/h and in the drainpipe was 150 μSv/h. The sink was dismantled, with all the necessary protection measures being taken (Fig. I.D–2). A ^{226}Ra spent sealed source was found in the drainpipe. The sink was decontaminated and removed. Some parts of the workbench were contaminated: these parts were cut out and removed as radioactive waste.

The wall under the windows and the floor showed high contamination levels (of 500 Bq/cm^2 and 100 Bq/cm^2, respectively). The wall surface and 11 m^2 of the floor surface were therefore removed by scabbling. The external eaves were also contaminated (around 20 Bq/cm^2), and the contamination was successfully removed.

I.D–7.4. Room V: Implanting room

The floor and walls of room V, which had areas of 10 m^2 and 27 m^2, respectively, were monitored. Special beds, a lead sheet and two tables with medical devices were in this room. Some contaminated medical devices (such as clips) were removed as radioactive waste. The bed was decontaminated with a detergent solution. Some pieces of the lead sheet were contaminated at around 30 Bq/cm^2; these were removed as non-compactible radioactive waste. The floor and the walls of this room had varying contamination levels, from 10 Bq/cm^2 to 500 Bq/cm^2. The contamination was fixed and 6.5 m^2 of the floor surface was removed.

FIG. I.D–2. Patient room IV. Decontamination activities: dismantling the sink and looking for the sealed source.

I.D–7.5. Room VI: Radiation sources storage room

The floor and walls of room VI, which had areas of 26.2 m^2 and 43 m^2, respectively, were monitored. A seat, a table, drawers, lead aprons, two small tables, a workplace for handling radiation sources, an empty container and a box containing contaminated devices were in this room. There was also a closet with other contaminated objects. A safe box to store radioactive sources was built into the wall of this room.

A PVC cover sheet was placed on the floor before it was entered. The radiation sources were removed from the room and a radiological evaluation was performed that measured the dose rate 1 m from the floor surface before and after removing the sources.

Surface contamination was also measured. The workplace on which the radiation sources were handled was found to be highly contaminated, with surface contamination higher than 540 Bq/cm^2 and total activity around 900 kBq. Five contaminated (300 Bq/cm^2) lead trays were found in this room and removed as radioactive waste.

Five of the eight drawers were contaminated, which were dismantled and removed as radioactive waste.

Some small tables, trays and other objects were found in the closet and carefully monitored. Three trays were found to be contaminated at a level of 100 Bq/cm^2.

A cardboard box covered with black polyethylene was also in this room. The dose rate on the surface of the box was 323 μSv/h. A highly contaminated spring was found inside the box. The dose rate 2 cm from the spring was 1300 μSv/h. Some other contaminated medical devices were also found and removed as radioactive waste.

The table placed in this room was contaminated. It was dismantled and the upper part and wheels were removed as radioactive waste.

The window and the wall under it were also contaminated (500 Bq/cm^2). Part of the window was dismantled and removed as radioactive waste.

The surface contamination on the floor of this room was around 400 Bq/cm^2. The walls were less contaminated than the floor (around 150 Bq/cm^2). The floor surface was removed, including the floor inside the closet. Figure I.D–3 is a map of the dose rates in this room before and after removing the radiation sources.

I.D–7.6. Rooms VII and VIII: Toilets

The floor and walls of rooms VII and VIII, which had areas of 10 m^2 and 40 m^2, respectively, were monitored. Most of the objects found in this area were not contaminated; however, a towel and washbasin were contaminated (60 Bq/cm^2), as

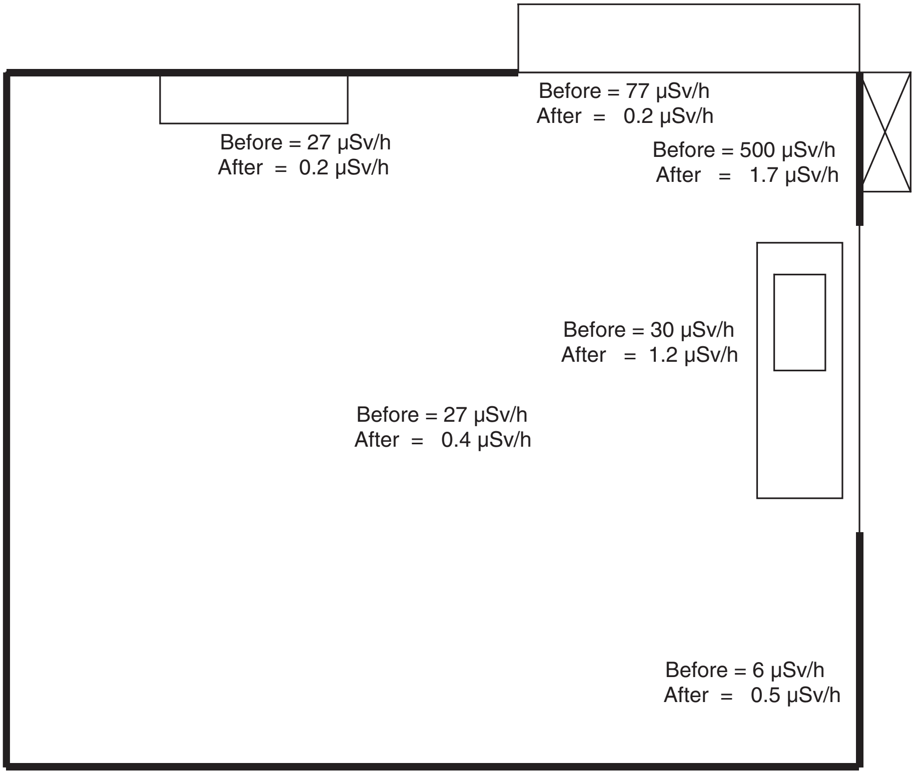


FIG. I.D–3. Dose rate levels in room VI before and after the removal of the radiation sources.

well as a latchkey, which was removed from the door and removed as radioactive waste. The contamination on the floor was fixed and part of the floor was removed.

I.D–8. WASTE MANAGEMENT

The decontamination works were performed in a manner so as to avoid the generation of liquid radioactive waste. The generated solid radioactive waste was segregated according to its physical characteristics and possible future treatment into compactibles (paper and plastics) and non-compactibles (wood, metal and debris). Radioactive waste was placed in plastic bags and then put into 200 L drums. Seventeen drums of radioactive waste were generated during the decontamination activities, which were identified and marked with the trefoil symbol. They were temporarily stored in room III, and since 2001 have been stored in a national centralized storage facility.

I.D–9. HANDLING OF SPENT ^{137}CS AND ^{226}RA SOURCES

More than 50 spent sealed sources were stored in a safe, and other sources were found during the decontamination work. Seven stainless steel capsules and a lead container were designed and constructed for the temporary storage of these spent sources. The sources were placed into capsules (Fig. I.D–4) and conditioned in accordance with IAEA recommendations. Some spent sources were stuck inside four applicators (i.e. small metallic pipes), the dimensions of which were larger than the diameter of the stainless steel capsules; the applicators had therefore to be cut before the sources could be conditioned. The sources were glued together but had to be

FIG. I.D–4. Conditioning the disused ^{226}Ra radiation sources into stainless steel capsules.

removed one by one. Auto-radiography was carried out to locate the sources exactly; photographic films mounted on a mammography chassis were used for this purpose. The optimal exposure time for the films was found to be 2 to 3 seconds. Once the exact position of the sources was defined the applicator was carefully cut.

I.D–10. CONCLUSIONS

The decommissioning project was successfully completed. Adequate project management was applied for safety assurance, radiation protection and waste management.

The following results were achieved: 97 spent radiation sources were recovered and properly managed, the volume of waste generated during decommissioning was 3.4 m^3 (i.e. 17 200-L drums). The decontamination and decommissioning activities were carried out in a month.

Upon the successful completion of the decommissioning, the Oncology Institute received the authorization for the unrestricted use of the facility from the regulatory body.

Annex I.E

DECOMMISSIONING OF ORIS CELLS 22, 23 AND 24 IN SACLAY, FRANCE

Cells 22, 23 and 24 at the Oris plant at Saclay in the Paris suburbs were the prototype for the industrial production of radioactive sources at ELAN IIB at La Hague, France. These facilities, which covered an area of some 150 m^2 of an active production site, have now been dismantled.

The three cells were built between 1966 and 1968 and were in use until 1972. They produced sealed sources of ^{137}Cs, which is a gamma emitter, and ^{90}Sr, which is a beta emitter. The plant was not contaminated with alpha emitters.

The irradiated material used in the production process was brought in through the rear of the cells. The packaging used in the transport of the irradiated material was attached to the rear doors of the steel-clad enclosures in which the operations were carried out. The packaging and waste created in the production process were taken out of the enclosures in the same way that they were brought in. Liquid effluents were drained to storage tanks located at the basement level before being sent to the Saclay treatment facility.

The three enclosures were separated from the forward cell area by 80 cm thick barite concrete walls. Barite concrete, which has a very high density, provided the operators with sufficient biological protection. Handling was by existing remote controlled manipulators and viewing was possible through 80 cm thick lead glass windows.

The cells remained unused from 1972 until 1987, at which time a cleaning campaign was launched to dismantle the effluent storage tanks located under the cells.

Dismantling operations on the majority of the site, including the production area, started in 1989. The plant was at this time quite old and its drawings were not necessarily up to date. The descriptions of the operating history were poor. Not everything was documented, so sometimes contamination was found in places in which there should have been none. Moreover, objects were discovered that were not assembled as indicated in the drawings.

Decommissioning the plant posed two problems:

- Firstly, a high level of radioactivity inside the enclosures and in the rear cell area restricted access for the dismantling operations;
- Secondly, the decommissioning work, which was within an existing production zone, was not to interfere with the work of staff nearby.

The area had to be isolated from the rest of the plant by setting up cladding panels and clearly marking directions for personnel and waste. Waste evacuation was often performed out of normal working hours, including weekends.

The first task was to clear the cells of all contaminated material. For that purpose the remote controlled manipulators and other operating equipment had to be restored to good working order. Access paths to the site, located in the forward cell area and leading to the corridors of the plant, were unfortunately closed off. Paths that could be used by workers and for transporting material were established. Changing rooms were also installed for the operators to change into and out of their protective clothing.

In order to remove the waste during the operation without interfering with the production staff working nearby, a conveyor belt was installed at the far end of the rear area.

The next stage consisted of decontaminating the internal parts of the enclosures by making use of remote controlled manipulators operated from the front of the cells. After radiological monitoring to ensure safety, the biological shielding elements that isolated the front cell area were dismantled to enable operators to work inside the enclosures. The enclosure concrete walls were cut into blocks by means of various methods, such as the use of diamond saws and expanding concrete grout. The blocks were then covered with a coating of plastic paint, which fixed any residual contamination on the exposed surfaces. As the enclosure walls were dismantled, walls of steel cladding were erected to separate the front cell area from the rear area.

The decommissioning work of the cells was completed in 1994. A few minor incidents occurred in the project, but there was no contamination of personnel.

A monitoring campaign was conducted to ensure that there was no residual contamination, after which Oris was able to use the plant as a production site again.

In conclusion, the special conditions encountered during the project were that the decontamination and dismantling work areas were enclosed within an operating production area. The dismantling team sometimes worked less that two metres away from laboratories in which technicians were working. They, nevertheless, completed the dismantling project under these conditions over a period of four and a half years.

Annex I.F

DECONTAMINATION AND DECOMMISSIONING OF SMALL MEDICAL, INDUSTRIAL AND RESEARCH FACILITIES IN HUNGARY

I.F–1. INTRODUCTION

The use of radioisotopes in Hungary started on a wide scale in the 1950s, although ^{226}Ra has been used there in medicine since the 1930s. The establishment of new laboratories and institutions to a large extent drove the increase of the use of radioisotopes. This growth took place throughout the areas of research, industry and medicine and continued until the end of the 1980s, at which time about 1300 users were registered. In the 1990s, after the change of the economic and political system in Hungary, many institutions closed down, merged or altered their activities, and today the number of users is only 400 to 500.

I.F–2. LEGAL BACKGROUND

The use and handling of radioisotopes in Hungary has been always regulated. Today, the Atomic Energy Law, Law No. CXVI of 1996, together with its implementation orders, provides the legal framework for the use of atomic energy. Among these orders, Decree No. 16/2000, issued by the Ministry of Health, regulates all activities related to radiation protection, and Decree No. 23/1997, issued by the Ministry of Welfare (today the Ministry of Health), covers radioisotope exemption levels. These orders are based on international regulations and recommendations, such as Refs [I.F–1–3]. The Hungarian orders also describe the requirements for activities involving radioactivity for particular cases (e.g. the classification of laboratories based on the amount and usage of radioisotopes, storage conditions and radiation and contamination levels for different workplaces).

The closure of facilities that used radioactive material before the 1990s was not a control or regulatory issue since the institutions were State owned and centralized, and hence closure was solely a technical matter. Since the 1990s, however, private ownership has become more common and has caused new concerns for the new owners. Decree No. 72/2000 was issued in order to avoid serious consequences arising in the event of a change of ownership; such consequences could include a loss of radioactive sources or unacceptable uses of facilities and laboratories. The authorities have had to increase control of the process of the closure and alteration of facilities.

Specific descriptions or orders for the decontamination and decommissioning of small facilities have not been issued in Hungary, so the competent authorities prescribe their requirements and stipulations on a case by case basis. The competent authority in the field of the decontamination and decommissioning of small facilities is the State Public Health and Medical Officer Service, acting on behalf of the Ministry of Health. There is no funding system for the decontamination and decommissioning of small facilities and thus the operator or owner has to finance the work.

I.F–3. MOTIVE AND INITIATION OF DECONTAMINATION AND DECOMMISSIONING

As mentioned above, closures or changes in the use of facilities increased in the 1990s among the users of radioactive material, principally because of economic conditions and obsolescence. Owing to the change in economic conditions, many new processes and developments have been introduced.

Decontamination and decommissioning procedures have been initiated most frequently in Hungary by the owners terminating their activities themselves, mostly to meet the demands of new owners or new technologies. The authorities can also take initiatives for closure if the use of a facility is redundant or does not comply with the requirements.

I.F–4. DECONTAMINATION AND DECOMMISSIONING

Facilities that do not handle radioactive material have been established after most cases of decontamination and decommissioning being partially or completely accomplished. In the majority of cases, if there is no serious contamination and the facility mainly used sealed, registered sources, decontamination and decommissioning can be carried out simply. If low levels of contamination are present, and the sources are adequately registered, the situation can also be managed easily without the aid of special techniques and expertise. Problems are caused, however, if contamination is from particular radioisotopes (e.g. ^{3}H, ^{14}C and ^{232}Th), if it covers a large area, if a large amount of radioactive waste is expected or if there are significant issues to overcome. If this is the case decommissioning becomes complex and licence termination can only be achieved by using experts and special equipment.

The decontamination and decommissioning of a facility can be achieved by using various methods:

- The unit is completely demolished (i.e. full decommissioning);
- The use of radioactive material at the facility is stopped and the facility is transformed into one that does not use radioactive material;

- The activity at the facility is replaced by a new, up to date technology that uses radioactive material;
- A part of the facility is closed and moved to another location.

The authorities have been involved to varying degrees, according to the complexity, in a variety of cases of decontamination and decommissioning. Thus the role and the licensing requirements of the authorities can vary according to the specific case; that is:

(a) A licence for the decontamination and decommissioning is given with no stipulations;
(b) A licence for the decontamination and decommissioning is given with stipulations;
(c) In addition to the administration of the licensing, the authorities may have an active participation in and control of the various steps of the decontamination and decommissioning;
(d) The authority initiates the decontamination and decommissioning.

All the decontamination and decommissioning methods listed above have been employed in Hungary, but cases (b) and (c) have been used most often, while cases (a) and (d) have been used only infrequently.

I.F–5. TECHNICAL REQUIREMENTS OF DECONTAMINATION AND DECOMMISSIONING

The authorities assess many steps during the licensing process for the decontamination and decommissioning of a facility. The requirements cover:

- The authoritativeness and reliability of the companies that do the work;
- The reasonableness of the scheduling, timing and planning;
- The determination of the quality and number of experts to be used;
- The training plan for the workforce;
- Special procedures for the measurement of radiation levels (in the preparatory, intermediate and final phases);
- The procedures for taking samples and measurements in laboratories;
- Making a complete inventory of the radioisotopes at the site;
- The clearance procedures;
- An assessment of the quantity of waste;
- Working out a suitable waste classification system;
- The provision of special dismantling and demolishing equipment;

- The selection of the decontamination processes to be used;
- Ensuring appropriate protection measures;
- Planning for waste handling, packaging, storage, treatment, transport and disposal;
- Keeping records (i.e. documentation of the different measurements, logs and operational procedures);
- Preparation for special access tools (plus shielding material);
- Maintaining a relationship with the authorities during the work;
- Preparedness for abnormal situations.

The circumstances may change from situation to situation and thus all the above steps should be considered for a given decontamination and decommissioning project. Since in Hungary there are no specific regulations for decontamination and decommissioning, the authority requires these steps to be evaluated by the owner and/or licensee. The more complex a case, the more steps are necessary, with each step being applied to the appropriate degree.

I.F–6. CASE STUDIES

I.F–6.1. Decommissioning of a contaminated laboratory building

I.F–6.1.1. Introduction

The multistorey laboratory building of the Central Mining Development Institute was in operation between 1962 and 1996. The laboratory building was initially licensed at the highest category (level A) of the laboratory classification of the Hungarian standards, but in 1982 the laboratory was reclassified to level B. A wide range of unsealed radioactive material was used, but only three radioisotopes were dominant. For many years unsealed ^{131}I had been applied for tracer investigations in the oil industry in the form of an oil solution. Approximately 55 GBq of ^{131}I was used per month. In other processes, cold beacons for illumination and other long lived marker lights were produced for deep level mines. Promethium-147 was used in a powder form, with a total activity of about 37 GBq. In the production of stronger illumination marker lights, ^{90}Sr was utilized in the form of a nitrate solution. Generally, each ampoule filled with a solution of ^{90}Sr contained 37 GBq of activity. The nitrate solution was in an enamel layer sintered on to an iron surface by heating.

There were several types of contamination: ^{147}Pm surface dust had to be occasionally removed, but ^{131}I decays rapidly and therefore caused no problem; the only serious contamination was therefore caused by ^{90}Sr. Over many years, routine operations and a few unexpected events resulted in the laboratory building becoming heavily contaminated, especially the air conduits. There were three origins of the ^{90}Sr contamination:

- In the 1960s the exteriors of the ampoules were contaminated during their production in the former USSR.
- This contamination increased, as the ampoules were susceptible to breaking in strong sunlight or heat. The most serious contamination incident occurred in a hot cell when an ampoule containing 3.7 TBq of activity broke. Owing to a malfunction of the ventilation system the laboratory became heavily contaminated with ^{90}Sr.
- During the production of the enamel layer for marker lights a large amount of a byproduct (cinder) that contained a residue of ^{90}Sr was created. On one occasion the roof of the storage area for these byproducts leaked and ^{90}Sr migrated into the concrete floor and the brick walls. This contamination was later covered with a new layer of concrete and plaster.

I.F–6.1.2. Rooms and equipment of the laboratory building

There were several rooms in the laboratory building next to the main rooms:

- A level B laboratory with three hot cells, a chamber for a radioisotope service trolley, shielded cabinets, laboratory furniture and tools;
- A level C laboratory with a hot cell, shielded cabinets, laboratory furniture and tools;
- A store for radioactive waste;
- A compartment for treating and storing liquid radioactive waste;
- Several further rooms.

The infrastructure of the building and sanitary engineering (health physics) areas included:

- Wall surfaces, flooring covered with plastic material and painted areas that were easy cleaned by washing and wet rubbing;
- Hand washing basins, emergency showers and water and sewer pipe systems;
- Tanks for the treatment of liquid waste;
- A ventilation system with a self-contained plant compartment.

I.F–6.1.3. Preparation for the decommissioning

The new owner (the Land Agency) of the laboratory building ordered and financed the decommissioning work. The owner searched for and selected a reliable firm that had a licence for radiological activities and decommissioning expertise.

The firm first entered into a contract with the Püspökszilágy Radioactive Waste Treatment and Disposal Facility (RWTDF) for the transport and disposal of accumulated and packaged radioactive waste. In addition, the firm engaged a few skilled experts for measuring radiation levels and contamination. Throughout the decommissioning the firm maintained close co-operation with the competent authority. The decommissioning started with a radiological survey, followed by the following steps, in accordance with the decommissioning plan:

- The collection, sorting, packaging and transport of movable radioactive material;
- Gradual demolition activities, together with the collection of contaminated material;
- The taking of detailed radiological measurements after the dismantling;
- The clearance process.

I.F–6.1.4. Results of the radiological surveying

Checking and monitoring measurements were carried out in each phase of the decommissioning in conjunction with the dismantling work. The contamination was measured using surface monitoring instruments. In accordance with the Hungarian standards for beta and gamma emitter radioisotopes, the intervention level for contamination is above 0.5 Bq/cm^2, thus the surface was not considered contaminated if the measurements were below this level (the clearance level).

The main results of the radiological survey in the preparatory stage close to the hot spots were as shown in Table I.F–I.

It was realized that larger quantities of contamination had been covered with various radiation attenuation layers (e.g. plaster); these contaminated surfaces only came to light during the demolition. The measured values, for example, after the removal of a 20 cm thick concrete layer in the store for radioactive waste were above 1000 Bq/cm^2.

I.F–6.1.5. Radiation protection

Dust respirators, protective clothing, gloves and helmets were worn during all phases of the decommissioning. No one received radiation doses above the detection level.

TABLE I.F–I. MAIN RESULTS OF THE RADIOLOGICAL SURVEY IN THE PREPARATORY STAGE CLOSE TO THE HOT SPOTS

Location	Contamination level (Bq/cm^2)
Air funnel in the ventilation fan compartment	150–300
Chamber for the service trolley	40–70
Manipulator and preparation room	5–25
Level B laboratory	5–70
Hot cells	2–8

I.F–6.1.6. Transfer of radioactive material and waste

The legal successor (owner) of the laboratory transferred for disposal those sealed sources that had a high activity and the majority of the standard sources. The successor continued with a number of the earlier activities in a new laboratory in another location. The rest of the radioactive sources were transported to the Püspökszilágy RWTDF. In the whole decommissioning process 23.5 m^3 of debris and 7 m^3 of metal waste were produced that had contamination above the clearance level. The amount of free release waste cleared was about 10 m^3, which was disposed of in the bottom of a mine pit and covered with non-radioactive debris.

I.F–6.1.7. Control measurements on the area

After the facilities were dismantled in situ gamma ray spectrometry was carried out and soil samples were taken to determine the radioactive concentrations. The results of these measurements showed that the area was not contaminated, and thus the authority declared the area as clear and ready for free release.

I.F–6.2. Clearance and decontamination of medical research radioactive waste storage areas

I.F–6.2.1. Introduction

Examinations and research activities have been performed using various radioisotopes (^{3}H, ^{14}C, ^{60}Co, ^{137}Cs, ^{75}Se, ^{65}Zn, ^{54}Mn, ^{109}Cd) at the Pécs Medical University (PMU) since the middle of the 1960s. A significant quantity of hazardous liquid waste (organic and toxic solvents) was produced. The waste from these examination and research activities was collected and stored in two storage rooms in the basement of the central building. The majority of the waste was hazardous, biological and contaminated with ^{3}H and ^{14}C radioisotopes, together with some very short lived radioisotopes. Other universities had incinerated this type of waste, but this had not been done at the PMU for more than 30 years. The authority responsible set a deadline of 30 June 1998 for dealing with the unsatisfactory storage situation.

I.F–6.2.2. Storage conditions

The storage rooms were situated in the basement and had smooth (plaster finish) concrete floors and tiling on the walls. The hazardous waste was collected initially in plastic cans or in metal drums of various sizes. The biological waste was mostly put into drums filled with formol, although some was also placed into large preserving jars or milk cans made of aluminium. These rooms were nearly full after more than 30 years of waste accumulating in them, and in some places the packages of waste were piled on top of each other. Owing to these unsatisfactory storage conditions (and the long duration of time), many packages were damaged and in a few cases their identification was missing or illegible.

I.F–6.2.3. Preparation for clearance

The Püspökszilágy RWTDF was only obliged to accept the radioactive waste that conformed with the RWTDF acceptance criteria, which was accepted by the PMU as a first step. Thus 150 pieces of solid waste in plastic bags and 32 radium applicators with an activity of 11 GBq were packaged and transported to the Püspökszilágy RWTDF. The Püspökszilágy RWTDF was then engaged to remove the rest of the waste and decontaminate the storage rooms. The use of a registration book for radioisotopes to clarify which types of radioisotope were present and their quantities was introduced. A radiological survey was performed to determine the radiation and contamination levels on and around the packages. It was assumed that the activity concentration of the waste packages was below the exemption or clearance levels. The waste volume was about 8000 L, which included 2000 L of biological material, and it was considered reasonable to apply for approval for incineration. The authority specified the main requirements for the approval process, which were that:

- The activity concentrations were to be determined by taking samples of each waste stream;
- The waste was to be treated and transported to the Püspökszilágy RWTDF if the activity concentration was above the exemption level;

- An assessment of the radiological consequences and doses to the population from the incinerator was to be made to ensure that doses were less than 10 mSv/a;
- The activity of the ash and slag was to be measured during and after the waste being burnt.

I.F–6.2.4. Waste removal

After getting the licence for decommissioning and waste removal the waste was sorted according to the readable labels and signs. From each waste group a sample was then taken (a total of 55) to determine the activity concentration of the radioisotopes at a suitable laboratory. In accordance with the results of the measurements all the groups could be ranked as clearance waste, as the concentrations were mostly below the release levels by three to four orders of magnitude. Only in a few cases were the concentrations close to the release limits.

It was found during the sampling that, owing to the corrosive nature of the compounds, the tools used (automatic pipettes) became damaged rapidly; it was therefore decided to use less sophisticated tools (rubber bulbs with glass tubes) instead.

The operator of the incinerator requested a radiological assessment to be made. The release calculations from incineration predicted that the dose to the population would be below 1 μSv/a.

At the beginning of the removal of the waste a plan was prepared that took account of the type of waste and its condition. In practice, however, waste package movements were influenced by their condition, and thus sometimes the plan was altered to suit an alternative approach. Additional drums had therefore to be used for repackaging, and extra plastic bags were used for secondary waste.

All those involved in the removal of the waste were required to wear protective garments (boots, overalls, gloves), and the flow rate of the ventilation system was set at the maximum level. The storage rooms were emptied one after the other. Each package was measured in order to determine its radiation dose rate and surface contamination. The radiation levels around the packages were mostly around the background level, although in a few cases they were three to four times the background level.

Any damaged package was repacked into a drum and a sorbent containing perlite was continuously applied on to the floor to absorb spilled liquid waste. Once a storage room was completely emptied the wet perlite and protective equipment were collected and placed into a drum, which was incinerated.

Transport to the incinerator was carried out using normal hazardous waste procedures. Measurements of the radioactive content of the ash and slag samples indicated that their values were below the clearance level.

I.F–6.2.5. Decontamination of the storage areas

Surface monitoring measurements were performed after the waste packages were removed. Contaminated surfaces were cleaned several times by being washed with detergent. After this decontamination process smear samples were taken for laboratory measurements, which showed that the contamination was below the clearance level by many orders of magnitude.

I.F–6.3. Decontamination of industrial premises

I.F–6.3.1. Introduction

Instruments for sintering (i.e. thorium oxide implanted into a tungsten filament) had been operated at the works of GE Lighting Tungsram Ltd for a long time. The equipment for the sintering process was situated in a hall, together with other equipment, which resulted in the whole hall becoming gradually contaminated with thorium. The contamination increased when a hydrogen fire caused an intense spread of contaminated dust. It was required that the plant be decontaminated, as the plant was to be transferred to a new owner. The hall was reconstructed at the same time in order to separate the sintering operation area from the other workplaces by creating an enclosed airspace dedicated to sintering.

I.F–6.3.2. Surveying the contamination

The main results of the contamination measurements are shown in Table I.F–II.

The radioactivity of dust samples taken ranged from 2 to 3 MBq/kg. Thorium-232 and its progeny (^{228}Ac, ^{212}Pb, ^{212}Bi and ^{208}Tl) were present. The age of the dust samples was found to be between 8 and 13 years.

TABLE I.F–II. THE MAIN RESULTS OF THE CONTAMINATION MEASUREMENTS

Location	Contamination level (Bq/cm^2)
Waste around the vice and bench	15–25
Tools around the vice and bench	30–40
Vacuum apparatus for hydrogen	5–12
Storage cupboard	300–400
Metal waste with thorium rods	250–300
Tool box	80–100
Ventilation system	2–4
Surfaces of the work area	1–3

I.F–6.3.3. Decontamination

Loose contamination on tools and equipment was removed using a vacuum cleaner. The surfaces of the hall were also vacuum cleaned and then washed either once or twice. The contaminated surfaces were cleaned by washing with detergent or by wet rubbing. The collected dust, washing water and highly contaminated tools were packaged into drums and disposed of at the Püspökszilágy RWTDF in six consignments. A controlled surface monitoring showed that the radiation levels were mostly around the background level, although in some places the values were twice the background level.

I.F–6.4. Decommissioning of a level B laboratory

I.F–6.4.1. Introduction

A level B laboratory was operated at the Applied Chemistry Branch of the Budapest Technical University (BTU) from the middle of the 1950s. Radioactive and nuclear substances were used in both research and education. Since the BTU was in the neighbourhood of a training nuclear reactor, joint research activities were conducted, and a close co-operation between the two institutions developed. Owing to a decrease in the amount of work carried out in the level B laboratory, the BTU decided that it should be decommissioned.

I.F–6.4.2. Decommissioning

The decommissioning consisted of the collection of the radioactive material and waste and the decontamination of the laboratory. The identification of the radioactive substances was a challenge, partly because the head of the laboratory had left the institute. This situation was fortunately resolved to a great extent by the existence of a national register of the radioisotopes in circulation.

I.F–6.4.3. Taking an inventory

Twenty four sealed sources had certificates, but 62 items did not. Of these 62, only 25 were identified in the national register. Tracing the identity of the rest of the sources was unsuccessful, although all these sources were standard or reference sources and their activities were below the exemption levels. The unsealed radioactive substances consisted of ^{60}Co, ^{210}Pb, ^{204}Tl, ^{137}Cs, ^{58}Co, ^{88}Y and ^{90}Sr, with a total activity was 65 MBq. Liquid radioactive waste had been stored in the laboratory.

The nuclear material included uranium and thorium (in the form of oxides, nitrates and acetates), of which there were 500 g and 400 g, respectively.

I.F–6.4.4. Treatment of the waste and the decontamination

The low activity sources were all cemented into drums. The high activity sources were transported in shielded containers to the Püspökszilágy RWTDF. Liquid waste was absorbed with perlite and then also cemented into drums.

Radiological surface monitoring was carried out after the radioisotopes were transported. In a large number of cases small contaminated stains were discovered on the surfaces of the floors, walls, shielding material and furniture. These contaminated areas were removed by chiselling and wet abrasion. Any part of equipment having fixed contamination was cut out for disposal. After the final control measurements, the authority considered the laboratory as inactive.

REFERENCES TO ANNEX I.F

[I.F–1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 1990 Recommendations of the International Commission on Radiological Protection, Publication 60, Pergamon Press, Oxford and New York (1991).

[I.F–2] EUROPEAN UNION, Council Directive 96/29/Euratom of 13 May 1996, Basic Safety Standards for the Protection of the Health of Workers and the General Public against the Dangers arising from Ionizing Radiation, Official Journal of the European Communities No. L 159/1–114, Luxembourg (1996).

[I.F–3] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

Annex I.G

DECOMMISSIONING OF A ^{60}Co RADIOACTIVE PANORAMIC IRRADIATOR FACILITY IN ITALY

I.G–1. GENERAL DESCRIPTION

In 1960 a panoramic irradiation unit for use for research was installed in a gamma cell at the Institute of Chemistry, University of Genoa, Italy. The sealed ^{60}Co source had an activity of 37 TBq (1000 Ci) and was, at that time, the largest available in Italy for chemical research.

The source was contained in a shielded underground well and could be raised from its rest position into an 8 m^3 shielded room in which various types of sample holders, heated pressure vessels or pilot plants could be placed. The position of the source could be controlled with great accuracy and adjusted in accordance with the position of the samples. Except for a small amount of condensation water, the well was operated dry. The source, formed by three disks of activated cobalt contained in sealed aluminium capsules, was obtained from Amersham in the UK. The producer inserted the radioactive capsules into a source holder prepared by the university.

The source was shipped from the UK to Italy in a lead container (thickness about 220 mm). The dose rate at the surface of the shield was 2–3 Gy/h. No special rules, authorization or control were required by the existing regulations at the time of shipping and disembarkation (1960).

The container was placed in the underground well below the handling apparatus. The source holder was connected to the supporting rod and extracted from the container, which was then moved aside in order to allow the introduction of the source into the shielded well. The procedure for installing the source (the introduction of the container into the irradiation room and the connection of the source holder to the supporting rod) took about 30 minutes. The maximum dose rate was 10 mGy/h and the dose received by the most exposed operator during the installation was 2.2 mSv.

When the source was at the bottom of the well, the shielding (lead within the supporting rod and 1200 mm of magnetite filled concrete under the floor) permitted the operators to enter the irradiation room and receive only a negligible dose. Safety interlocks avoided the extraction of the source when the sliding door was open and the opening of the door when the source was not in its rest position.

Research on the radiolysis of hydrocarbons and polymers, radio-induced alkylation, the polymerization of gaseous and liquid monomers, graft co-polymerization and dosimetry was carried out using this source over many years.

Owing to natural decay (residual activity 270 GBq (7.6 Ci)) the dose rate for research became insufficient and the source was therefore removed and the building used for other purposes.

The following describes the activities taken to remove the source safely and to eliminate all restrictions on the reuse of the room and building.

I.G–2. DECOMMISSIONING

The first part of the work was dedicated to the acquisition of all the basic information necessary to characterize the irradiation source as precisely as possible. It was found that this was not easy, owing to the long period of utilization of the equipment and its subsequent removal.

In practice, it was difficult to find sufficient documentation describing the equipment itself or to establish the technical history of the equipment, as the maintenance procedures were not all recorded or traceable. In addition, there were some difficulties in getting information from previous users or workers, who were often retired or dead.

An on-site survey was carried out to obtain all the technical information necessary to plan and design the removal procedures, taking into account the radiological boundary conditions.

The physical condition of the source holder had been monitored by a closed circuit television unit and some traces of corrosion were observed, probably due to the combined effects of the gamma irradiation and the humid underground environment the stainless steel was in. The possibility of contamination of the source holder and of the well liner was therefore taken into account in the project.

After verifying the good condition of the lifting mechanism and establishing the absence of any contamination (due to the source capsule still being sealed) it was decided to remove the source by lifting it and taking away only the active part (i.e. the source holder).

In order to minimize the dose to the workers involved in the decommissioning and to prepare a package suitable for safe transport, it was decided to remove the source by surrounding the source supporting rod with a lead shield, which was also used as the inner shield for the package. This was accomplished by designing and manufacturing an inner shield in four parts, which were assembled around the supporting rod in order to allow the extraction of the source from the well without exposing the personnel to a high radiation dose. The shield had a lower sliding door that could be closed after the source was in place.

After the construction of the shield, a general test was conducted in the workshop in order to verify the removal procedures. This test was essential to train the operators and to prove all the operating procedures in order to avoid problems during the actual decommissioning activity.

The thickness of the lead shielded container was 127 mm and the maximum value of the dose rate at the surface was 6 mGy/h.

The sequence of the operations is shown in Figs I.G–1–I.G–10.

The inner shield was assembled around the supporting rod of the source holder while the source was still at the bottom of the underground well (Figs I.G–1–I.G–4). The source was then raised from the well up into the central cavity of the shield, which was in the irradiation chamber (Fig. I.G–5). The sliding bottom door of the shielding assembly was inserted and the supporting rod was then cut through using hydraulic shears (Fig. I.G–6). The shielding cylinder containing the source was finally closed and sealed (Fig. I.G–7) and removed from the irradiation chamber (Figs I.G–8, I.G–9).

The shielding assembly was designed to fit exactly within the cavity of a Type A transport container (a CF66 container), which complies with the IAEA regulations for the transport of radioactive material (Fig. I.G–10).

The internal part of the underground well was found to be slightly contaminated by corrosion products in condensation water (about 100 MBq (3 mCi) of ^{60}Co), and therefore the water was pumped out and the steel liner was removed and disposed of; the proper safety procedures were observed.

The total time required for the decommissioning was two days, but the work with the partially shielded source was limited to less than five minutes. The whole process, including the planning, authorization, design, manufacture and testing, took a little less than two months

The activities performed were:

- Cleaning the irradiation cell mounting of the shield;
- Extracting the source from the well;
- Cutting the source holder;
- Sealing and removing the shielding container;
- Preparing the waste package for transport;
- Decontaminating the bottom of the well;
- A final check and the unrestricted release of the site.

By following all the safety procedures the dose received by the most exposed operator during the entire decommissioning work was 54 μSv.

Since there is no disposal site in Italy the source, the contaminated parts removed during the decommissioning and all secondary waste were transported to an authorized interim storage site. By law, the work plan was subject to approval by the health physicist responsible for the operation of the facility. Archive records included the description of the project, workers’ dosimetric records and details of the source and its packaging.

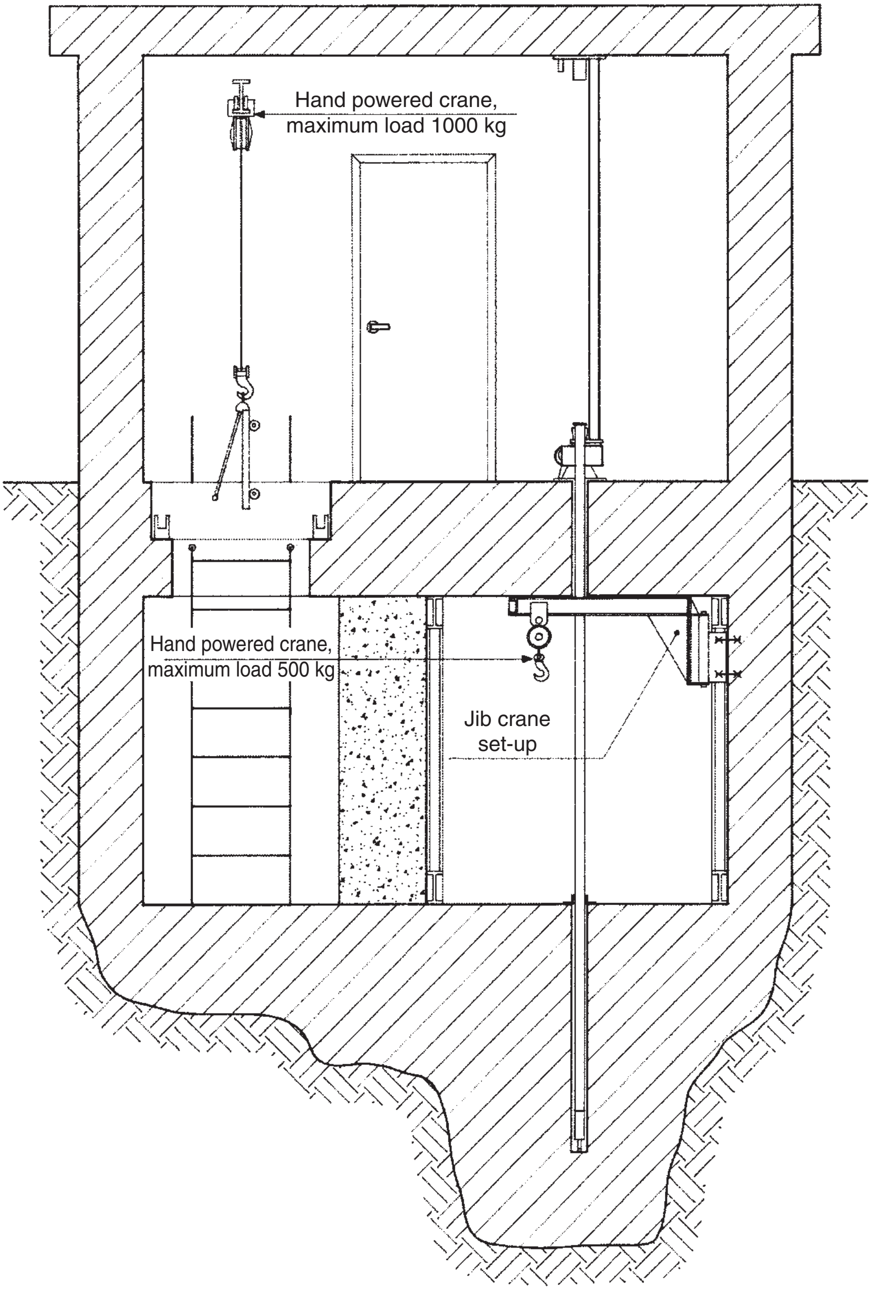


FIG. I.G–1. Initial situation. Ground floor and underground bunker, cross-section.

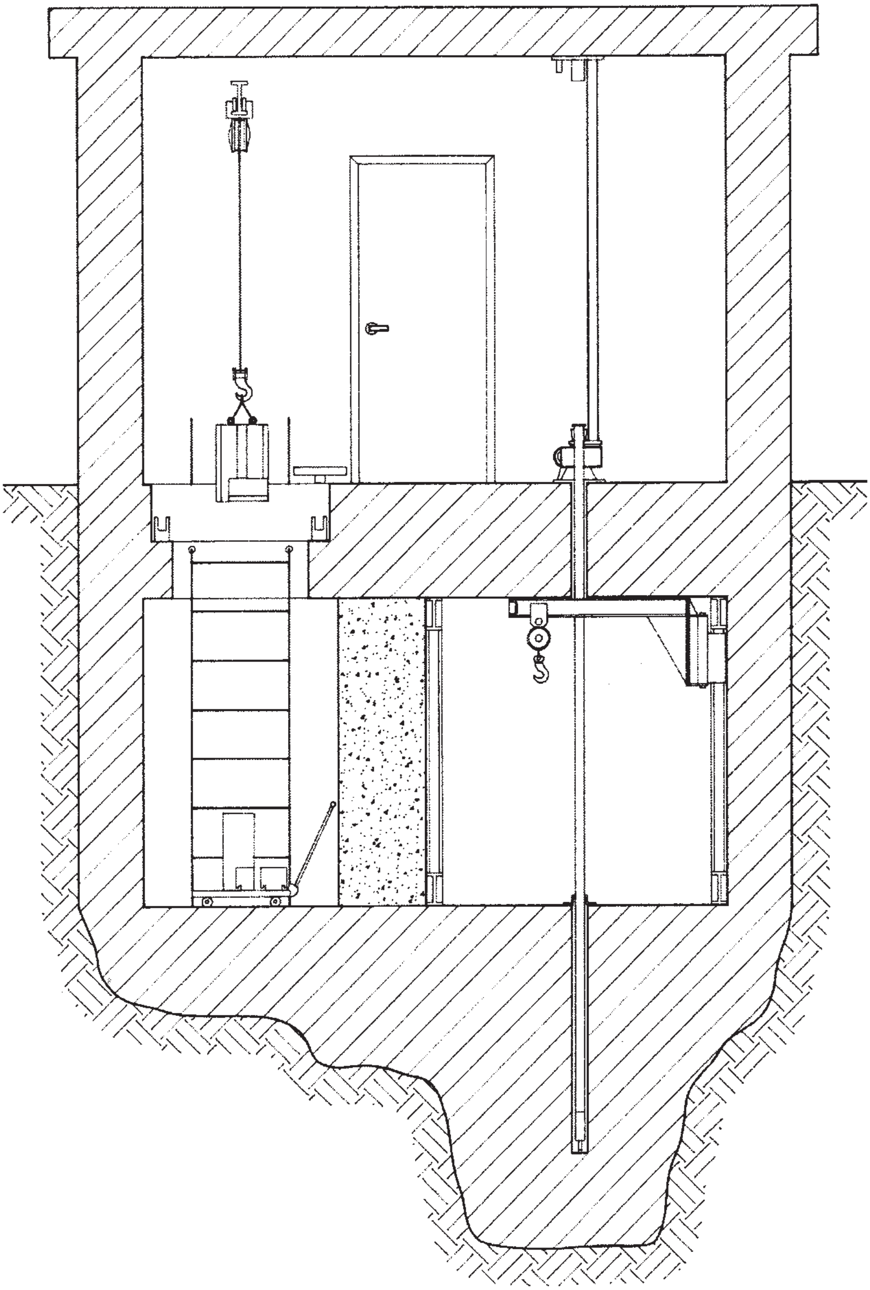


FIG. I.G–2. Shield parts being moved to the bunker.

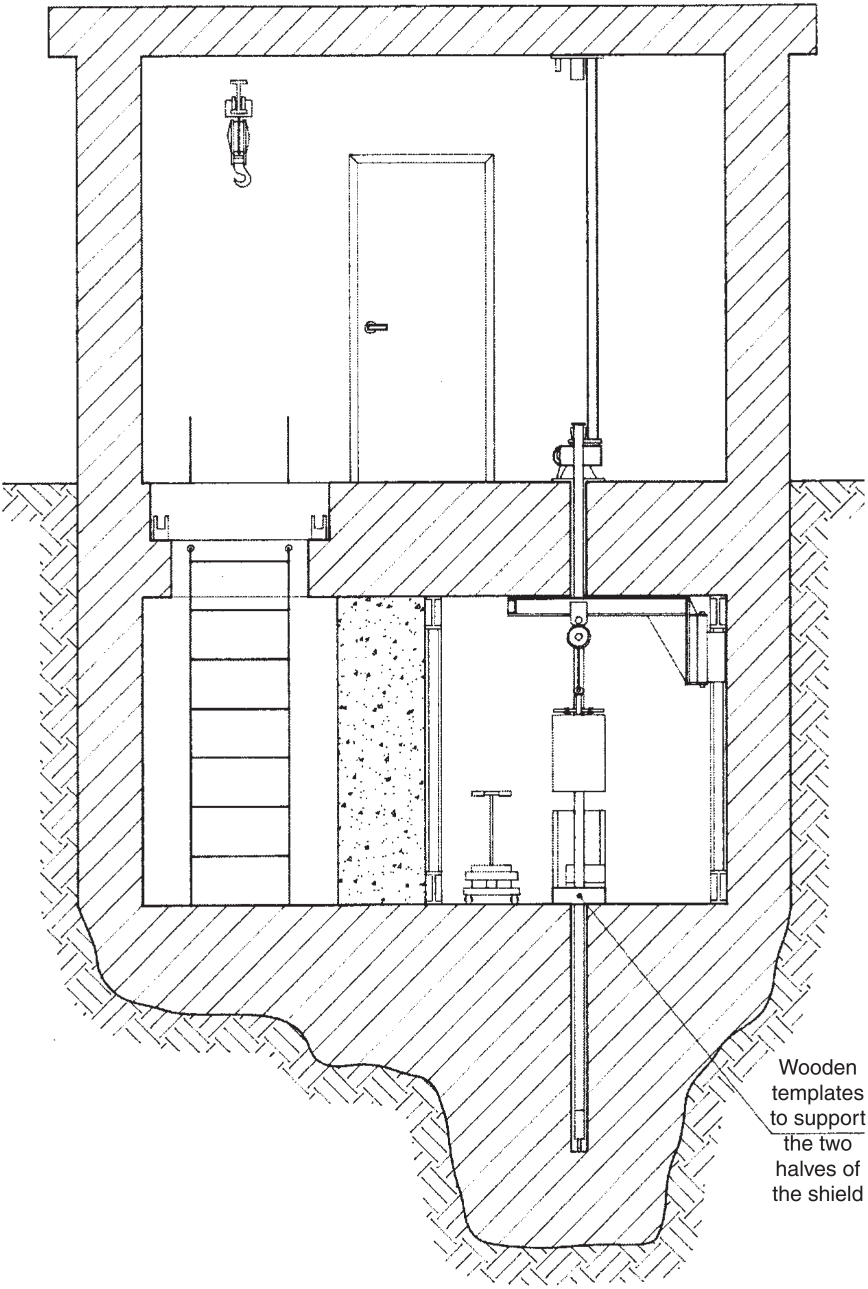


FIG. I.G–3. Shield positioning and assembling around the guide tube.

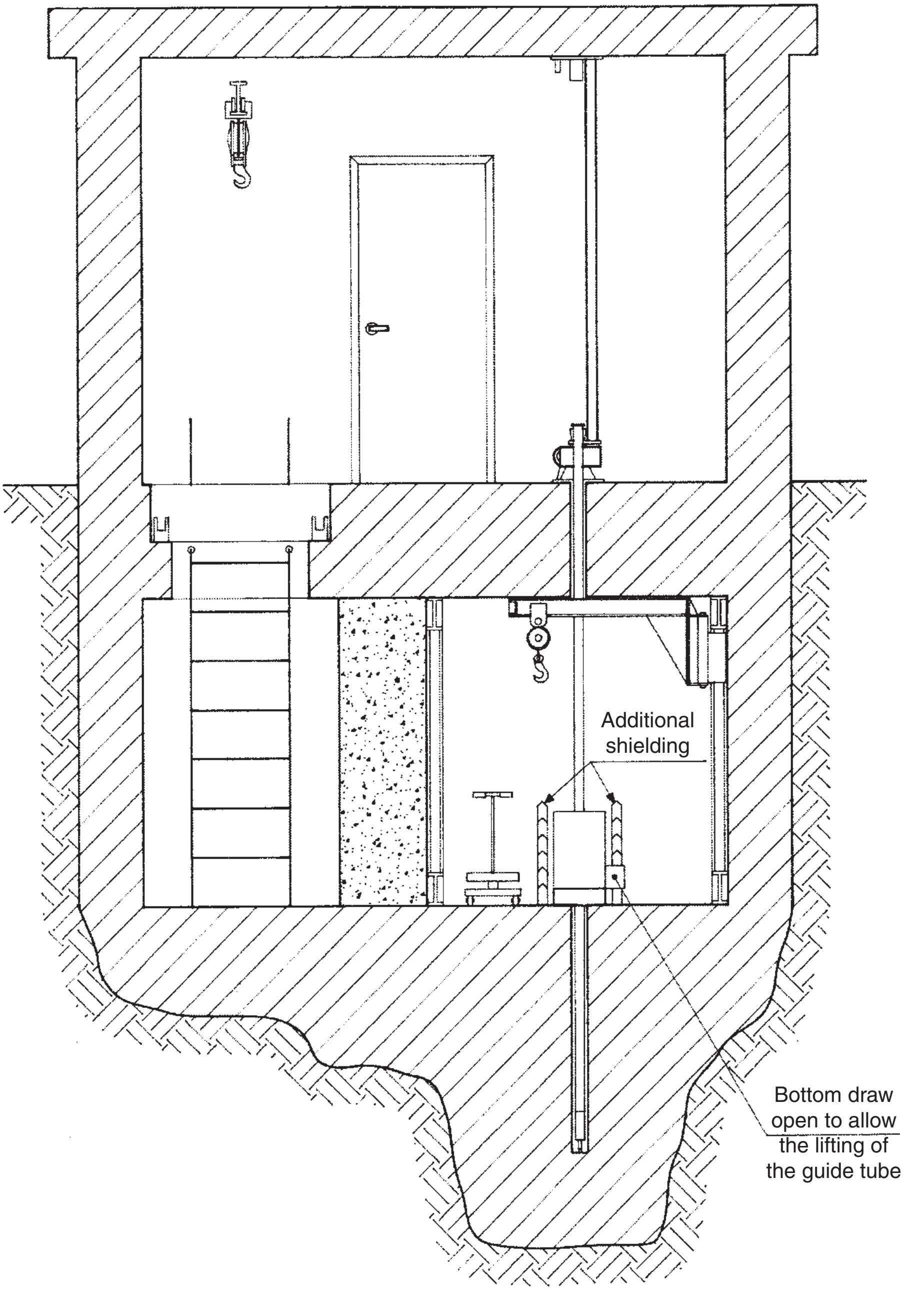


FIG. I.G–4. Mounting additional shielding around the inner shield.

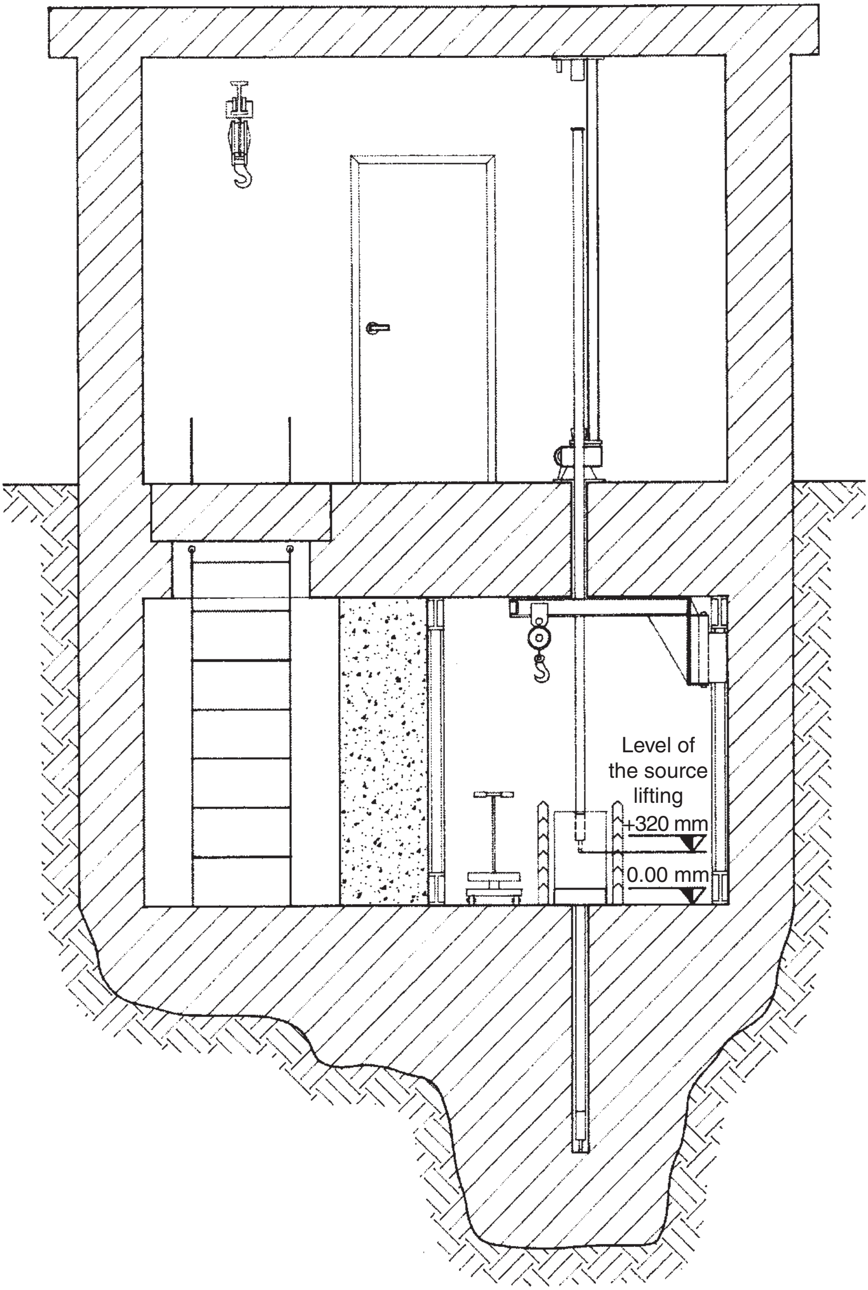


FIG. I.G–5. Lifting the guide tube with the source at the bottom up to the middle of the cavity of the assembled shield and closure of the bottom lid of the inner shield.

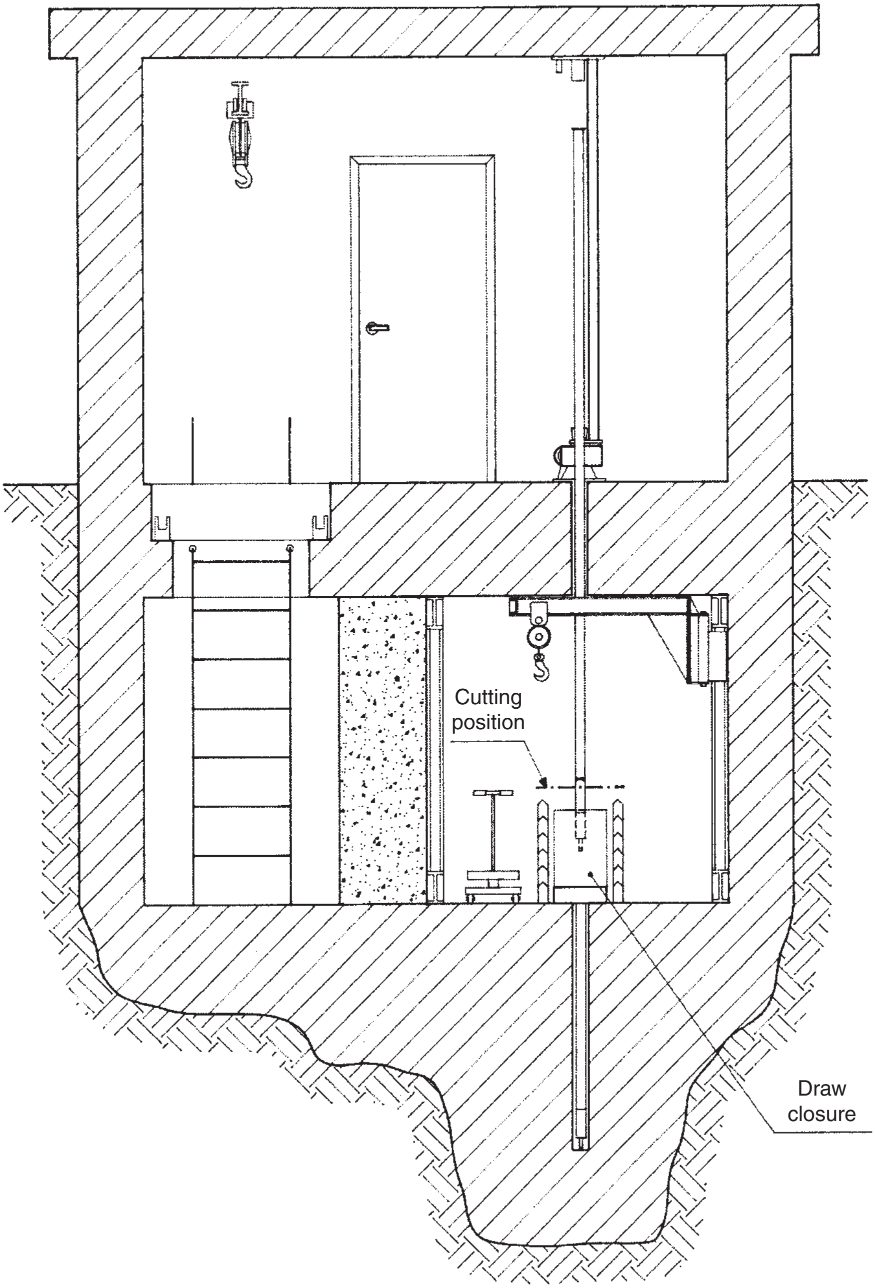


FIG. I.G–6. Cutting the guide tube. The end with the source remains in the shield cavity.

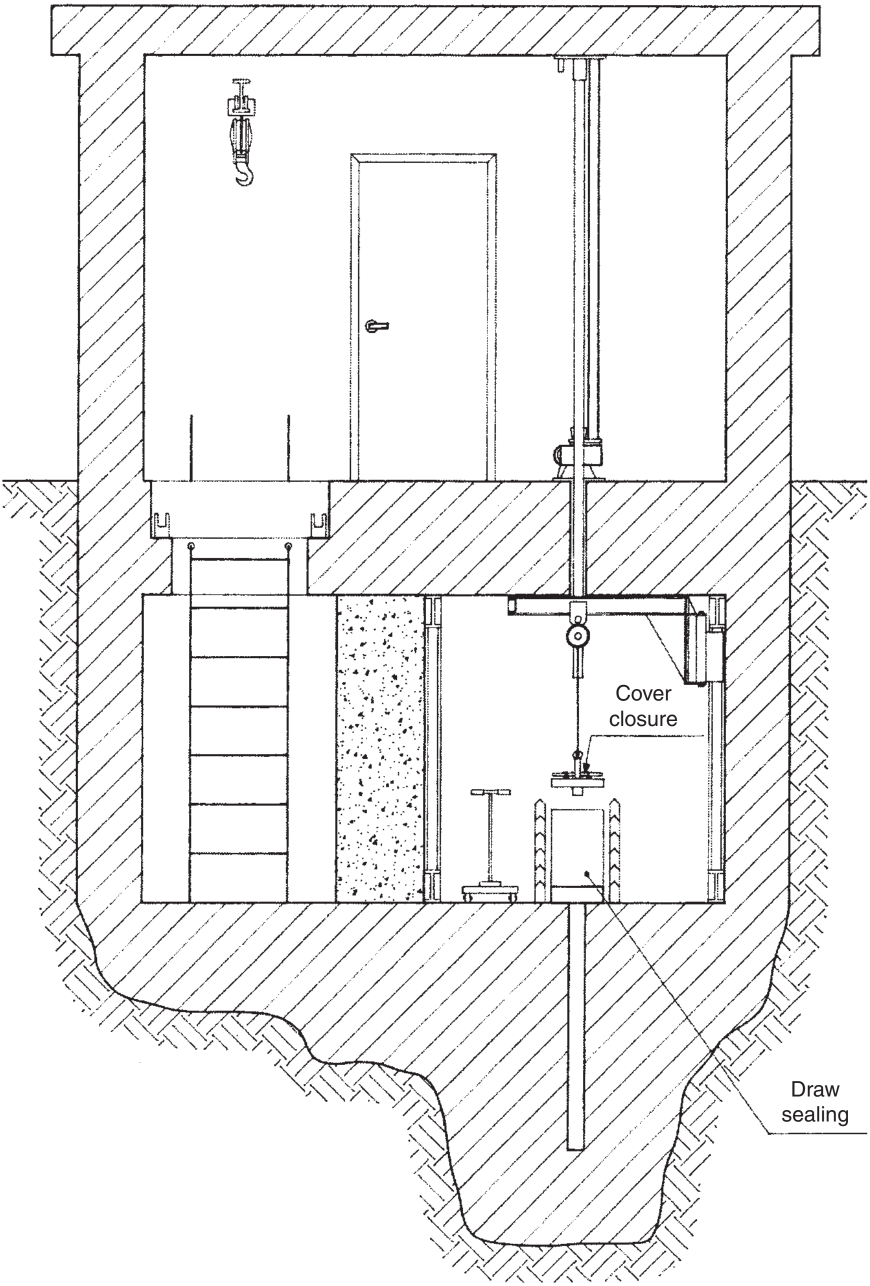


FIG. I.G–7. Closing the shield with the upper cover.

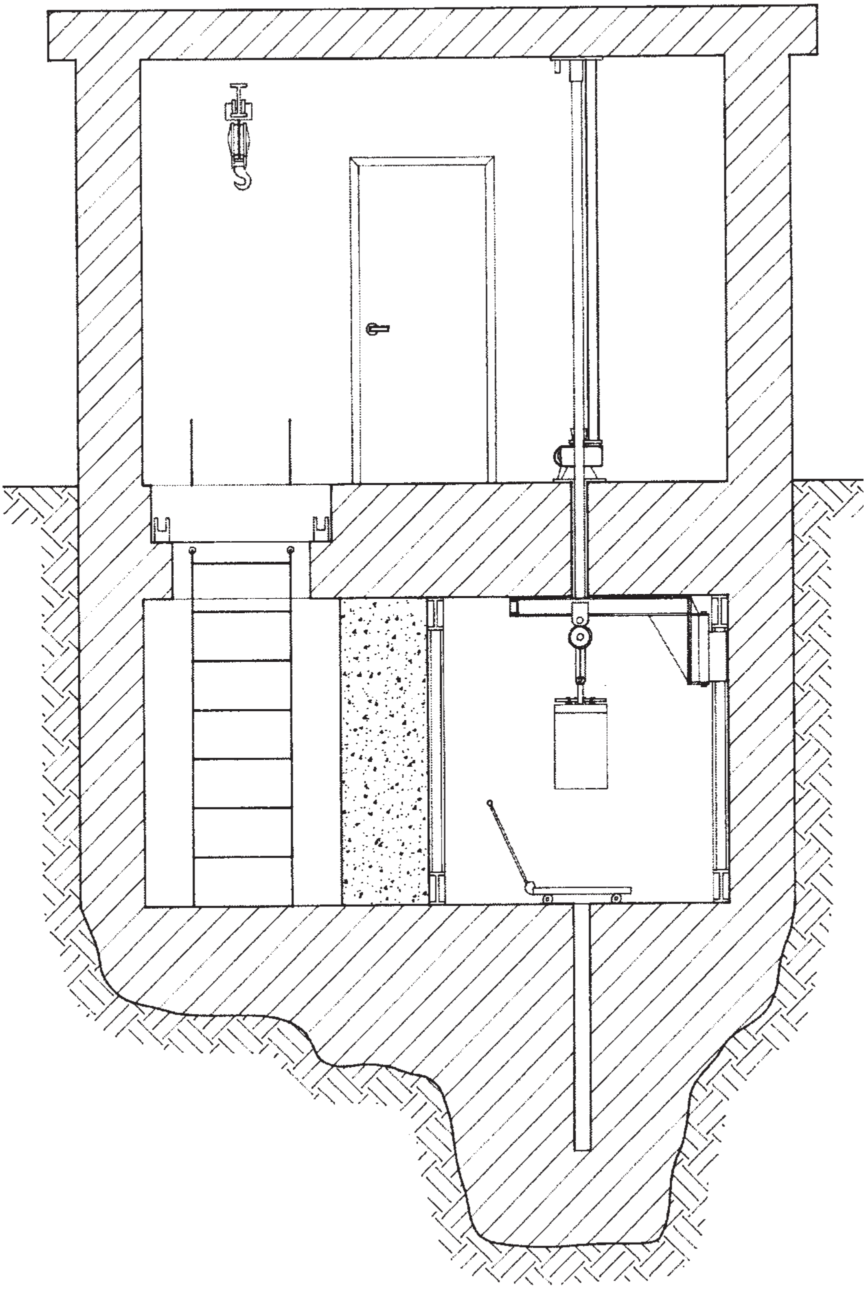


FIG. I.G–8. Moving the assembled shielding container.

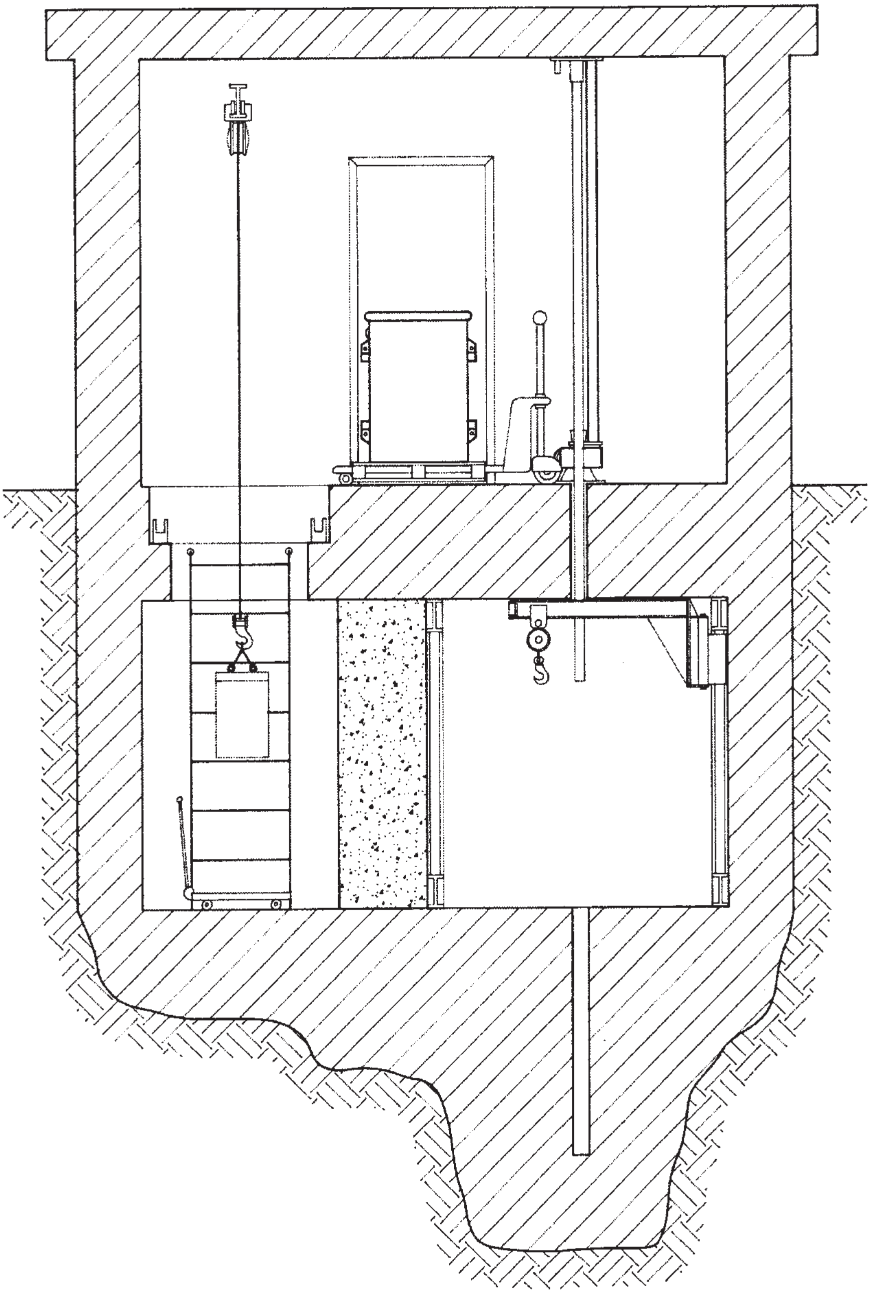


FIG. I.G–9. *Lifting the assembled shielding container to the ground floor.*

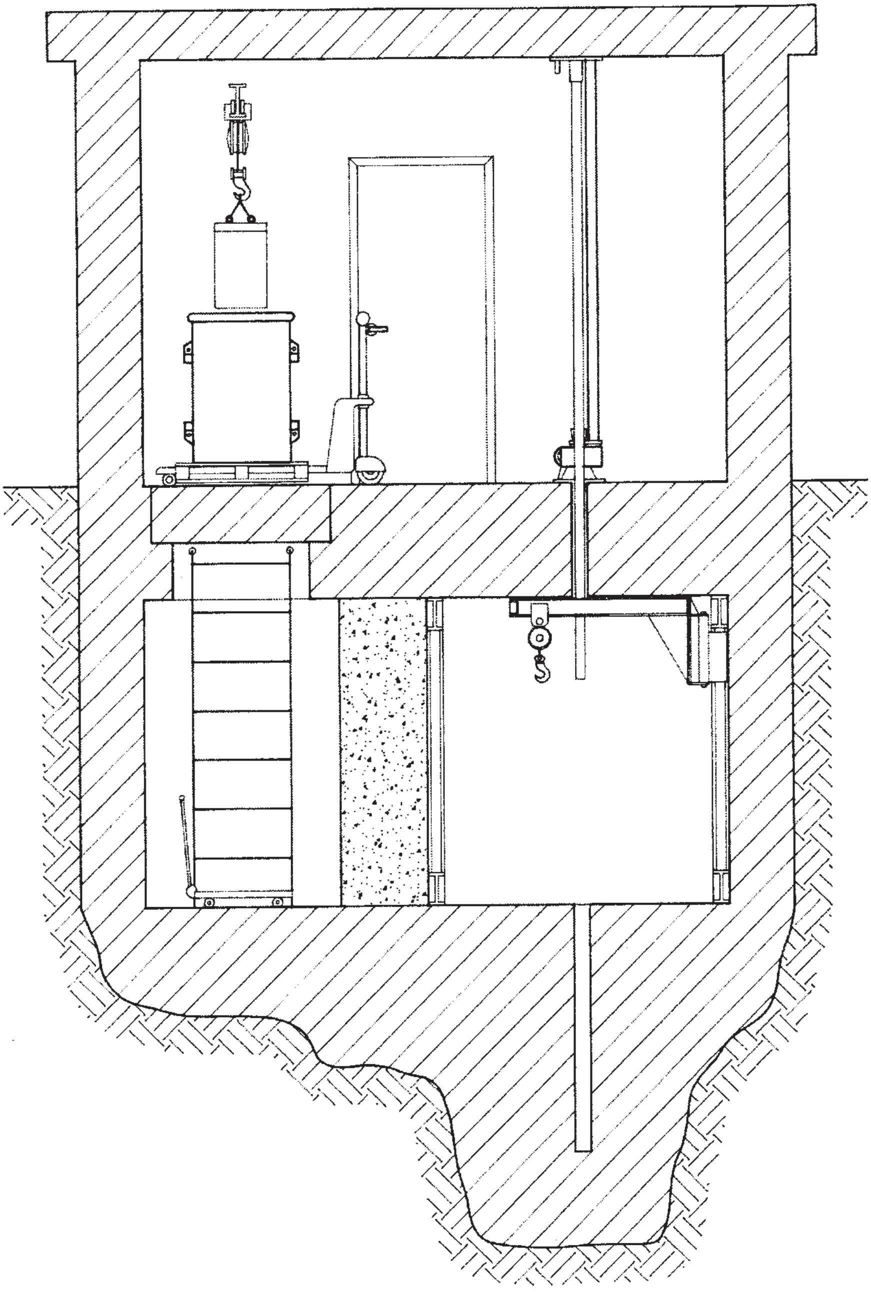


FIG. I.G–10. Packaging of the shielding container into a transport (Type A) container.

Annex I.H

EXAMPLES OF DECOMMISSIONING SMALL RESEARCH FACILITIES IN THE RUSSIAN FEDERATION

I.H–1. DECONTAMINATION AND REFURBISHMENT OF RADIOCHEMICAL PREMISES

The Radium Institute Laboratory (RIL) operated from 1922 to 1962. In 1973 a decommissioning plan was adopted to clean, decontaminate and convert the RIL into a laboratory in which work with neutrons and gamma spectrometry could be carried out. The RIL, which was situated near to an urban area, contained more than 20 g of radium and various other radioisotopes. It performed many radiochemical tasks, including both experimental work and the industrial production of alpha and neutron sources for the purposes of science and defence. Decommissioning work was carried out between 1973 and 1974 by RIL personnel, including some of the former staff of the laboratory.

Contamination levels (of mostly radium, radon and their progeny) varied from several to many becquerel per square metre. Floors and abandoned equipment contaminated by radium spills and dust contained roughly 1 g of radium in various forms. The air was continuously contaminated by radon from these radium remains, with concentrations of up to 40 kBq/L.

The decommissioning activities undertaken included the following:

- The continuous monitoring of the radioactivity levels, both inside and outside the facility. The preliminary testing of various decontamination techniques and the associated liquid and airborne effluent control. These tests addressed effectiveness and safety requirements and the specific nature of the various structures and materials present.
- Providing for the safe transfer of personnel and material into and out of the restricted area.
- The use of a tented containment with airlocks and fitted with showers for the decontamination of protective clothing and personnel.
- The use of powerful air ventilation with particulate filters and recirculation (of the cleaned air). Personnel also used respirators, as required.
- Decontamination work, involving dismantling first the external and then the internal ventilation units; the removal of in-fill gravel from attics; the removal of the plaster covering of the walls and demolishing parts of the inner walls, where necessary; stripping linoleum and other floor coverings; the disassembly and removal of the water supply and drainage systems.

- The removal of the parquet floors, basement floors and in-fill gravel on the floors; the scabbling of contaminated concrete; the removal of the doors and their frames; the replacement of the windows and window frames.
- Rebuilding and recovering the floors and walls with new material and the painting or plastering in of residual fixed alpha contamination.

The doses accumulated by the decommissioning personnel (32 in total) ranged from 2 to 50 mSv during the work time of approximately half a year. The total collective dose was about 800 man mSv.

As a result of this decommissioning operation the use of the premises as a designated class III laboratory (i.e. for six hours per day, five days per week, for qualified laboratory personnel) was permitted.

In the Arkhangelsk region a similar situation existed with the abandoned auxiliary compartments of an unloaded spent fuel storage facility, a radiochemical laboratory, cloak and shower rooms and a dosimetric monitoring station — all were contaminated with radionuclides and required refurbishment. Decontamination and repair works were carried out in 1992 and 1993 with the aim of reusing these facilities as a storage complex.

Unlike the case described above, there was no preliminary testing or evaluation of decontamination technologies. At the planning stage preference was given to the mature technology of decontamination using strippable coatings with additives of plasticizers and anti-adhesive substances.

Another important peculiarity of the project was the careful sorting of the waste generated during decommissioning operations to minimize the production of radioactive waste. All the non-radioactive waste or waste containing radionuclides below permissible concentrations (which was mostly building refuse) was disposed of at the normal disposal facilities. Radioactive waste (some equipment, strippable coatings, laboratory glasses, etc.) were containerized and transported to the regional radioactive waste storage facility.

Decontaminated and refurbished premises are now ready for the planned re-utilization (see Ref. [I.H–1]).

I.H–2. DECOMMISSIONING TECHNOLOGY FOR CONTAMINATED LABORATORY AND PILOT INSTALLATIONS AT THE A.A. BOCHVAR INSTITUTE OF INORGANIC MATERIALS (VNIINM)

The chemical decontamination of plutonium- and americium-contaminated structures using solutions of alkaline or organic or mineral acids prior to decommissioning resulted in the generation of large amounts of liquid radioactive waste. There was a risk of contaminating the containment air and surfaces by transuranic aerosols, for example by the high solution vapour pressure.

A low waste dry decontamination technique based on the application of readily strippable polymeric (protecting, decontaminating, immobilizing) coats was developed and tested over extended periods at the VNIINM.

Water soluble polymers have been found to be the most suitable for forming protective coatings. Organic solvents, although advantageous in some ways, are unsuitable for the purpose because of their toxicity and, if they are contained, their inherent potential for explosion and fire.

A few transuranic contaminated facilities have been decommissioned so far. In 1999 a large scale decommissioning programme was initiated for a complex of work areas used between 1945 and 1964 for performing experiments on the recovery of plutonium. The rehabilitation programme covered process areas, gloveboxes, two hot cells containing manipulators, support areas and two wells accommodating experimental equipment. Plutonium-239, ^{90}Sr, ^{137}Cs and ^{241}Am were found to be major contributors to the contamination.

Techniques tested and optimized in this project included the use of decontaminating coats. Polymeric films also served effectively to immobilize contamination to prevent its spread during dismantling activities. Because of this approach, airborne contamination remained lower than the maximum permissible concentrations. More details are given in Ref. [I.H–2].

REFERENCES TO ANNEX I.H

[I.H–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Facilities Other Than Reactors, Technical Reports Series No. 386, IAEA, Vienna (1998).

[I.H–2] MAMAEV, L.A., et al., D&D Technology for Contaminated Laboratory and Pilot Installations, Waste Management ’00 (Proc. Symp. Tucson, AZ, 2000), Arizona Board of Regents, Phoenix, AZ (2000).

Annex I.I

DECOMMISSIONING OF SMALL MEDICAL, INDUSTRIAL AND RESEARCH FACILITIES IN THE UNITED KINGDOM

I.I–1. INTRODUCTION

The number of small medical, industrial and research facilities in the UK is quite large. Some of these facilities have already undergone complete decommissioning, some are undergoing decommissioning now, but the large majority are still in operation and therefore awaiting decommissioning. The word ‘small’ here does not necessarily mean small in size, but generally modest in terms of complexity, radiological risk and the quantity of waste arising. These facilities range from small radiochemical laboratories to isotope production facilities and small research reactors.

The regulatory control of the decommissioning operations for nuclear facilities, small or large, is quite exhaustive but non-prescriptive. A brief description of the powerful but flexible regulatory regime to control decommissioning processes is given in Section I.I–2.

The preparation of a decommissioning strategy and decommissioning plan that takes full account of all relevant factors, such as the regulatory requirements, hazard assessments, resource availability and proposed timescale, is carried out prior to the start of a decommissioning operation. This aspect is addressed in Section I.I–3. Once the optimum decommissioning option is identified, the actual implementation of the decommissioning is carried out.

It is essential that there is planning and proper management in compliance with the approved plan for a decommissioning project for small medical, industrial and research facilities. This aspect is addressed in Section I.I–4.

The management of radioactive waste is an important element in the decommissioning process. The segregation, categorization, treatment and conditioning of radioactive waste falls within the waste management strategy, and this aspect is considered in Section I.I–5.

Section I.I–6 is a summary of this annex. A list of decommissioned small facilities is given in Table I.I–I.

I.I–2. REGULATORY ASPECTS OF DECOMMISSIONING

The decommissioning of a nuclear facility, whether large or small, is controlled in the UK by a powerful but non-prescriptive regulatory policy. The decom-

TABLE I.I–I. SELECTION OF DECOMMISSIONED SMALL FACILITIES IN THE UK

Name	Description	Operations start	Delicensing of the site
Marston Excelsior	Nuclear fuel processing plant, Wolverhampton	1960	February 1973
IRD nuclear research laboratories	International Research and Development Co., Fosseway, Newcastle-upon-Tyne	1963	December 1975
Vickers test rig	Test rig installations, Vickers, South Marston, Wiltshire	1962	September 1976
Universities Research Reactor	Argonaut type, 300 kW thermal power, Risley, Cheshire	1963	July 1996
Critical assembly	Queen Mary College, London	1963	March 1967
Fuel element plant	Nuclear fuel processing installation, Melton Mowbray, Leicestershire	1966	April 1973
Research reactor	Queen Mary College, London	1966	November 1983
Fuel storage installation	Nuclear fuel storage, Cammell Laird and Co., Birkenhead	1967	February 1972
Test rig	Vickers test rig (fissile material) installation, Vickers, South Marston, Wiltshire	1977	July 1981
Reactor access house	Reactor access house, Devonport, Plymouth	1987	March 1997
Jason research reactor	Argonaut type, 10 kW thermal power, Langley, Slough	1960	January 1962
Jason	Argonaut type research reactor, 10 kW thermal power, Greenwich, London	1962	November 1999

missioning process is in general considered an extension of the operating regime if a facility goes smoothly from its operating phase to its decommissioning; decommissioning can also, however, be a standalone operation.

The overarching enabling legislation in the UK governing the health and safety of workers and of the general public is the Health and Safety at Work etc. Act 1974 (HSWA74). Under the statutory provisions of the Nuclear Installations Act 1965 (as amended) (NIA65) and the Nuclear Installations Regulations 1971 (NIR71) no civil nuclear installation may be set up or operated in the UK unless a nuclear site licence has been granted by the regulatory body, which is the Health and Safety Executive (HSE). However, some of the Ministry of Defence (MoD) nuclear installations operate under the regulatory control of the MoD regulator, the Chairman of the Naval Nuclear Regulatory Panel, within the Conditions of Operation, which are similar to a site licence. The HSWA74 provisions, nonetheless, are applicable to both civil and military establishments.

Nuclear facilities (as distinct from nuclear installations) do not require site licences to operate. These nuclear facilities are required to be registered with the environment agencies (the Environment Agency in England and Wales and the Scottish Environment Protection Agency in Scotland) as far as the storage, handling, accumulation and disposal of radioactive material is concerned under the provisions of the Radioactive Substances Act 1993 (RSA93) and its proposed modification (the Proposals for the Radioactive Substances (Basic Safety Standards) (England and Wales) Regulations 2000 and the Radioactive Substances (Basic Safety Standards) (England and Wales) Direction 2000). The environment agencies are, however, responsible for the control of the dispersal and disposal of radioactive waste from all types of nuclear installation using certificates of authorization under the RSA93.

The control of exposures of workers and members of the public to radiation is exercised under the provisions of the Ionising Radiation Regulations 1999, Statutory Instrument No. 3232 (IRRs99), which were promulgated under the HSWA74. Thus the IRRs99 provisions are applicable to both civil and military nuclear facilities, irrespective of whether they require site licences. The IRRs99 are consistent with the recommendations of the International Commission on Radiological Protection (ICRP) given in ICRP Publication 60 [I.I–1] as reflected in the EU Council Directive 96/29/Euratom [I.I–2]. In particular, the IRRs99 include a requirement to keep doses as low as reasonably practicable (ALARP), which is similar to the ALARA approach of the ICRP.

Once the site licence for a civil nuclear installation is granted, the licensee is under regulatory control until the termination of the licensee's period of responsibility following decommissioning or the revocation of its site licence. The termination of the site licensee's period of responsibility is only made when, in the opinion of the regulator (i.e. the HSE), there has ceased to be any danger from ionizing radiation from anything on the site.

The decommissioning operation is considered complete for small nuclear facilities without site licences when the environment agency is satisfied that there is no danger from ionizing radiation on the site. The 'no danger' criterion implies that the dose level is no more than the background radiation level; when this is ascertained by an independent radiological survey, the environment agency issues a 'Notice of Revocation'.

The HSE’s policy on decommissioning and radioactive waste management at licensed nuclear sites as well as the policies of the environment agencies for non-licensed sites reflect the policy of the UK Government (as set out in the Review of Radioactive Waste Management Policy: Final Conclusions, Cm 2919, 1999, Statutory Instrument No. 3232). The key elements are that:

- The process of decommissioning should be undertaken as soon as reasonably practicable, taking account of all relevant factors;
- Operators should draw up strategies for decommissioning their redundant facilities, which should include a justification of the proposed timetable and a demonstration of adequate financial provisions;
- The strategies should ensure that the hazards are reduced in a systematic and progressive way;
- Radioactive waste should be segregated and characterized to facilitate the overall safe management of its conditioning, storage, retrieval and subsequent disposal.

I.I–3. DECOMMISSIONING STRATEGY AND DECOMMISSIONING PLAN

Once the decision to terminate the operation of a facility is taken, an overall decommissioning strategy is drawn up by the operating organization. This strategy should consider all issues of decommissioning, such as regulatory requirements, financial provisions, human resources requirements and technical issues. This strategy should serve as a guide to the detailed decommissioning plan that will be drawn up later.

A detailed plan covering all aspects of decommissioning is drawn up nearer the time of the decommissioning. The plan considers various options, with full consideration being given to the health and safety of workers and of the public, technical issues associated with decontamination and dismantling, environmental protection, proposed waste management provisions and, of course, human resources requirements. A significant part of the plan should go into assessments of both the radiological and the non-radiological consequences to the workforce and to the public during the implementation of the decommissioning plan using a probabilistic and deterministic risk assessment methodology. A key element of this plan would be the demonstration of the ALARP principle. The ALARP principle is very much embedded in the UK regulatory regime. It is insufficient for regulatory approval just to manage to demonstrate that all the basic tenets of the regulations have been met; the regulators would like to see that all possible steps have been taken to reduce exposures and risks to ALARP.

This plan, sometimes called the decommissioning safety report (DSR), when fully written and quality assured, is submitted to the nuclear safety committee of the originating organization for approval. After this approval it is submitted to the regulatory body for approval. Until this approval of the DSR is received, decommissioning work cannot take place.

It should, however, be pointed out that the DSR of a small facility can be prepared within a short period of time, as the complexity of the facility and the associated hazards are far less extensive than those of large nuclear installations. Nonetheless, the same overall procedure is followed for both small and large facilities in order to maintain high standards of safety.

I.I–4. PROJECT MANAGEMENT

A project manager is appointed and a project management team is formed to carry out a decommissioning project. The project management team members are selected from the existing employees on the basis of their skills, qualifications, familiarity with the plant and the management expertise needed to undertake the responsibility of the decommissioning operation. The team size is likely to be smaller than that of the operational workforce. However, key expertise, such as project management, radiological safety, waste management, quality assurance, purchasing and contractor control, must be retained for the decommissioning programme.

This project management team is obviously formed prior to the preparation and submission of the decommissioning plan to the regulatory body for approval. Depending on the complexity of the operation of the facility, the project management team may come up with a plan quickly or it may take a significant length of time. Once the plan is drawn up and approval is received from the regulatory body the plan effectively becomes frozen.

The next task of the project management team is to implement the approved plan. Any variation of the plan, in terms of work scheduling, timescale, etc., needs to be communicated to the regulator, and some variations may even require further approval. To progress with the decommissioning work smoothly and without regulatory holdup, considerations are therefore given at the initial stage to all aspects of decommissioning. Major problems can arise with regard to decontamination and dismantling operations, which are normally carried out by specialist contractors, and so early arrangements with these contractors could be beneficial.

I.I–5. RADIOACTIVE WASTE MANAGEMENT

Radioactive waste management is an integral part of the decommissioning process. Planning is made at an early stage to dispose of the various categories of waste arising. The accumulation, handling and storage of radioactive material and waste at non-licensed sites is controlled by the environment agency. The transport of radioactive material is controlled by the Department for Transport, Local Government and the Regions (DTLR). Approval must be received from the DTLR before radioactive waste is transferred from a site.

There is currently no disposal route for HLW or heat generating waste in the UK. However, medical, industrial and research facilities are unlikely to produce HLW. Following the rejection of the Nuclear Industry Radioactive Waste Executive's application at the public inquiry to set up the Rock Characterization Facility at Sellafield, it is highly unlikely that an intermediate level waste (ILW) disposal facility will be available in the UK within the next 30 or 40 years. Consequently the policy of the UK Government for ILW is to store it on the site. There is, however, a facility at Harwell for the disposal of small amounts of ILW. In small medical, industrial and research facilities low level waste (LLW) is the type of waste most likely to arise. There are some small LLW disposal facilities in the UK, but the largest one is at Drigg in Cumbria and is operated by BNFL on a commercial basis. If the activity concentration of waste is brought down to below 0.4 Bq/g then it can be released to a landfill site without regulatory control.

I.I–6. SUMMARY

The decommissioning operation, even for a small nuclear facility, involves a full range of activities, namely: project planning, the preparation of a decommissioning safety case, project management, cost estimates, regulatory interfaces, the demonstration of regulatory compliance, contractor control, waste management and arrangements for delicensing or a revocation notice. The experience gathered from decommissioning operations of small facilities can be summarized below.

- Early planning for decommissioning during the operational phase of a facility is essential for its smooth implementation. Detailed records of the design and construction of a facility, as well as any subsequent modifications to it, its operational history, records of any incidents or accidents, its radioactive inventory and composition, etc., would help its efficient and cost effective decommissioning.
- A dedicated project management team led by an experienced project manager needs to be appointed early in the decommissioning project. The selection of

the management team is crucial for the efficient implementation of the programme.

- A detailed decommissioning safety case that addresses radiological and non-radiological hazards, environmental issues and concerns, the waste management programme, etc., is a pre-requisite.
- Following the completion of the decommissioning work, a review of the operation in the form of a post-decommissioning report should be produced that highlights the problems, solutions found and lessons learned.

REFERENCES TO ANNEX I.I

[I.I–1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 1990 Recommendations of the International Commission on Radiological Protection, Publication 60, Pergamon Press, Oxford and New York (1991).

[I.I–2] EUROPEAN UNION, Council Directive 96/29/Euratom of 13 May 1996, Basic Safety Standards for the Protection of the Health of Workers and the General Public against the Dangers arising from Ionizing Radiation, Official Journal of the European Communities No. L 159/1–114, Luxembourg (1996).

Annex I.J

DECOMMISSIONING OF THE NRC LICENSED LABORATORY, HALLIBURTON NUS, PITTSBURGH, UNITED STATES OF AMERICA

After approximately 15 years of operation as an NRC licensed facility, the Tetra Tech NUS Laboratory facility was closed. At that time the facility held NRC licences for by-product material, source material and special nuclear material and a licence from the Commonwealth of Pennsylvania for naturally occurring and accelerator produced radioactive material [I.J–1].

On the basis of historical radiation surveys and operating records it was concluded that any necessary decontamination could be accomplished using typical maintenance methods, such as mopping or scrubbing, and that there would be no decontamination processes that would create significant potential for the exposure of personnel.

Based on this low risk, the NRC did not require the preparation and submission of a formal decontamination and decommissioning plan.

Tetra Tech NUS personnel, under the direction of the laboratory’s radiation safety officer, performed the activities necessary to terminate the licence. These licence termination activities included:

- The packaging and disposal of hazardous waste, radioactive waste and a small quantity of mixed waste;
- The removal of portions of the hood systems in two laboratory rooms;
- The removal of a fibreglass drain sump;
- The pressure cleaning of a portion of the sanitary sewerage system;
- A final radiological status survey.

Contamination surveys conducted during the year prior to the closure of the facility indicated that there would be minimal contamination present in the facility, so the final status survey was initiated without prior decontamination or screening surveys. The survey was conducted in accordance with the guidance given in Ref. [I.J–2]. All laboratory areas were surveyed, including administrative areas. During the survey, limited contamination was found in portions of the hood systems in two laboratory rooms, a fibreglass drain sump and sanitary sewerage lines leading from one room. The sewer lines were cleaned using a high pressure citric acid solution and the contaminated components were removed and disposed of as low level radioactive waste. A final radiological status report was submitted to the NRC and the Pennsylvania Department of Environmental Resources. The agencies reviewed the report and inspected the facility. Neither agency identified a need for further decontamination or surveys and approved the licence termination on the basis of the data contained in the report.

REFERENCES TO ANNEX I.J

[I.J–1] TETRA TECH NUS Inc., Consulting, Design and Remediation Services to Support Radiological Decommissioning, Tetra Tech NUS Inc., Pittsburgh (2000).

[I.J–2] NUCLEAR REGULATORY COMMISSION, Manual for Conducting Radiological Surveys in Support of Licence Termination, Rep. NUREG/CR-5849, NRC, Washington, DC (1992).

Annex II

PROBLEMS ENCOUNTERED AND LESSONS LEARNED FROM THE DECOMMISSIONING OF SMALL NUCLEAR FACILITIES

The following examples of lessons learned comprise brief technical information on the nuclear facilities involved and an outline of the problems encountered. The situations described are typical of the difficulties that can arise when planning or implementing decommissioning activities. Although the information presented is not intended to be exhaustive, the reader is encouraged to evaluate the applicability of the lessons learned to a specific decommissioning project.

II–1. CUBA

II–1.1. Problem: Contamination caused by a leaking source

A small research facility was used to store spent sealed sources, in which one ^{137}Cs source leaked. Serious contamination on some areas of the walls and floor was found in 1980. Owing to a lack of waste management expertise, infrastructure and financial resources, the contaminated areas were simply locked up and left. In 1986, in a decontamination attempt, the walls and floor were washed using pressurized water jets. This method was ineffective, however, since the contamination was only reduced by about 20% and the contaminated water spread the contamination, including to the drainpipes and the soil in the garden of the facility.

II–1.1.1. Solution

All the stored radioactive waste was removed from the facility. A decontamination of all surfaces, including a chemical decontamination, was performed. Specifically, K-Alum (PBK–HCl–Al) solutions, acidified with HCl (K–Al–HCl), were used for clays or cement floors and walls. Also, HCl plus Prussian blue plus K-Alum were used on soils, concrete, cement and asphalt. The waste solutions were collected and subsequently processed. Vacuum devices and polyurethane sponges were also used for recovering liquid waste. Part of the floor was removed from some rooms.

II–1.1.2. Lessons learned

The lessons learned were that:

- It is important that the interim storage of unconditioned radioactive waste be as short as possible and never exceed five years. During this time all waste should be gradually conditioned and transferred to a waste storage facility.
- The surfaces of walls and floors of small nuclear facilities should be easier to decontaminate.
- As the solubility of caesium in water is very high, it is recommended to contain and collect residual water as radioactive liquid waste.

II–1.2. Problem: Unexpected presence of radium sources

While decontaminating rooms in a Cuban hospital, a suitable low background area in the vicinity of the hospital was looked for in order to be able to measure contaminated items after removal. Unexpectedly high dose rates were discovered in the garden of the hospital and caused some concern.

II–1.2.1. Solution

Monitoring to find the highest dose rates (the hot spots) was carried out. In this way some spent radium sources were found deposited in the ground. The needles were collected and processed.

II–1.2.2. Lessons learned

The lessons learned were that:

- Rigorous control and registration of sources should be mandatory;
- A monitoring plan of all facility areas is crucial.

II–1.3. Cross-contamination caused by ants

During the decontamination of a small medical facility it was found that ants had scattered radioactive contamination.

II–1.3.1. Solution

The solution was to remove the anthill.

II–1.3.2. Lesson learned

The lesson learned was to avoid the presence of insects in controlled areas.

II–2. CZECH REPUBLIC

The examples described in Annex I.C have provided much experience and a number of lessons learned for the further decommissioning of small non-nuclear facilities in the Czech Republic; some of these are described below.

Legislation in the Czech Republic for decontamination was previously not fully developed in accordance with the latest international recommendations. It was found, however, that experienced staff were able to complete decommissioning activities within radiation protection criteria and requirements, and do this safely and without accidents, as the principles and criteria of radiation protection were well specified in the regulations valid at that time in the Czech Republic and because they were applied during decontamination to maintain full compliance.

Lesson 1: Good radiation protection legislation and practices are necessary for the safe performance of decommissioning activities.

Unpredictable problems may arise during decommissioning that must be solved flexibly and without delay, usually with an alteration of the planned procedures. The need for the decontamination of surfaces covered by other material may arise; some material to be released may require the use of more invasive and destructive techniques than planned and some chemicals applied during decontamination may cause difficulties in the treatment of generated radioactive waste if used in large amounts (e.g. surfactants and organic acids). The prevention and solution of these problems may be enhanced if the personnel involved have previously had to face similar non-conformities.

Lesson 2: Skilled and experienced personnel should be involved in planning and implementing a decommissioning. If national specialists with sufficient experience are not available, practical and theoretical training through international organizations should be considered prior to the start of a decommissioning.

The operation of older facilities is not usually well recorded. In particular, the consequences of accidents and incidents may be underestimated and not documented, which may complicate identifying the nature and extent of the contamination of exposed surfaces. Information about the position of radiation sources may sometimes not be available; this information may, for example, be lost when a company terminates the use of a radiation technique. In such cases only the memory of the original staff can provide valid and reliable information.

Lesson 3: Whenever possible a member of the original staff should be involved in planning and/or performing the decontamination and dismantling of a facility.

The technical condition of equipment and appliances deteriorates rapidly when they are taken out of service, particularly if they are not properly maintained. Corrosion processes and the loss of function of movable parts of equipment are the most typical reasons for later complications during a decommissioning procedure.

Lesson 4: The earlier the start of decommissioning after the termination of active operations the better the physical condition of the facility will be for decontamination and dismantling.

Construction features at some facilities may complicate later dismantling and decontamination (e.g. badly conceived designs of equipment, the use of improper material and complicated shapes). The consideration of decommissioning practices during the design of a facility that will employ radionuclides may positively influence the proposed construction solutions and, in this way, significantly ease its later decontamination and dismantling.

Lesson 5: The decommissioning of a facility should be considered in its design and development stages. The advice of a well informed designer may therefore simplify the procedures to be applied after the end of the operational life of a facility and the practical performance of the decommissioning procedures.

Documentation promoting both technical and administrative procedures are required in particular stages of a decommissioning. Obligatory legislative documents and records focus mainly on radiation safety and on keeping information about the final status of a facility. Nevertheless, a comparison of planned methods and procedures with actual ones and the respective lessons learned could be beneficial for solving future decommissioning cases.

Lesson 6: The more information is recorded and retained about the technical and radiological aspects of the decommissioning of a facility the better for managing future cases.

Documents produced during the preparation and implementation of a decommissioning can be divided into two basic groups. Those required by the regulator to issue a licence are kept in two different locations: at the premises of the regulator and of the operator. These documents usually tend to be summaries and principles rather than detailed descriptions of applied approaches. Documents describing the methods and procedures employed and a detailed characterization of the facilities cleaned are usually kept by both the implementer and the operator, which are usually private concerns. The probability of losing this technical information owing to a change in the status of such concerns (e.g. bankruptcy, the sale of the company or the termination of commercial activities) is significant. This may easily result in the disappearance of information concerning the radiation history of the facility.

Lesson 7: Do not rely on the continuous availability of information about the decommissioning of a facility held by the operator or owner, as the life of these companies may be limited. The more information provided to the regulator the better the chance of the later retrieval of that information.

The extent of decommissioning activities, their promptness and, as a consequence, the safety of a facility and of the procedures applied depends strongly on the availability of funding. In some cases a lack of funds may lead to postponing the start of the decommissioning procedure, which can result in consequences such as losing vital information about the facility, the loss of key personnel, losing control of radioactive contamination, failure to initiate the dismantling of equipment and increased migration of contaminants. There are two separate problems that need to be considered:

- At facilities in which work with radiation sources ceased it is recommended that State institutions be involved whenever the original owner is unable to cover the decommissioning costs;
- At facilities in operation a duty to set up a decommissioning reserve fund should be enforced on operators by legislation.

Lesson 8: Ensuring adequate financial resources through the obligatory creation of a decommissioning fund or through State intervention, where necessary, is a key condition for the early and safe decommissioning of a facility at the end of its service life.

II–3. HUNGARY

In Hungary the experience of decontamination and decommissioning has resulted from individual cases. It is important that a reliable company be engaged for these tasks. This company should have relevant expertise in the different fields of decontamination and decommissioning (e.g. handling both radioactive material and hazardous waste, transport, demolishing and decontamination techniques). If any of these are lacking, then qualified experts should be engaged.

At the beginning of a decommissioning project the type and amount of radioisotopes used in the given facility should be evaluated. In the past many institutions exchanged radioisotopes and some radioisotopes originated from abroad, mainly from the former socialist countries. The activities of these radioactive sources were mostly below the exemption level, but some were above it. The identification of unknown radioactive material is particularly difficult when in a liquid (non-solid) phase. A problem arises for sealed sources if there is no identification sign on the source. During the checking of an inventory of radioactive material, therefore, the register book of radioisotopes at the facility should be studied in detail. The Hungarian national register dates back to the 1960s, and in many cases this helped when the documents relating to a radioisotope were lost.

The characterization of contaminated areas must be performed carefully, because the planning and schedules are worked out based on it. However, the search for areas of contamination can meet with many difficulties. Firstly, the measurement of contamination is complicated for built-in systems such as ventilation and sewer pipes because most of these types of system are inaccessible and because their wall structures provide good radiation shielding. Thus special planning for exposing and demolishing these systems is needed.

At the beginning of a demolition task the classification of radioactive waste, together with a clearance procedure, must be established with the competent authority. It is better to overestimate the quantity of waste as it is rare that the quantity of waste is not found to be larger than expected, owing to unexpected contamination, and because the quantity of secondary waste from the decontamination process cannot be predicted accurately. The waste handling procedures should be prepared in advance to identify, for example, the storage location, type of packaging material, and treatment tools, equipment and material.

In all decontamination and decommissioning processes, in addition to measurements of radiation levels, sampling and laboratory analyses must also be carried out. Some radioisotopes (e.g. ^{3}H and ^{14}C) cannot be determined without special preparation and apparatus. During decontamination and decommissioning the best solution is for control measurements of radiation levels to be performed for every major step. The source should be searched for and removed as soon as possible when an increased or unexpected level is detected. However, instruments for the measurement of radiation must be suitable and calibrated for the given radioisotopes at the detection level and the energy response. Portable instruments are easily contaminated and must be decontaminated often in order to avoid inaccurate results. They should be calibrated at least daily.

The results of measurements, the major steps, logs, all relevant data and correspondence should be documented and kept by the owner and/or licensee. The collection and keeping of decontamination and decommissioning data is not currently covered by the legal system in Hungary and thus the competent authority can prescribe the data collection and procedures to be taken.

II–4. UNITED KINGDOM

II–4.1. Introduction

This section discusses some important nuclear and radiological safety management problems found, their solutions and the lessons learned during the decommissioning of the Jason Argonaut type low power training and research reactor in Greenwich, London, to unrestricted release status, between October 1996 and

December 1999 (Fig. II–1). These safety management problems encompass the safety case, independent peer review, safety management framework, control of contractors, continuity, records, control of work packages and permits to work. While these problems, solutions and lessons learned are derived from experiences gained during low power reactor decommissioning in a Ministry of Defence (MOD) licensed and regulated environment, they are generic in nature and would be equally applicable to the decommissioning of most other irradiation or nuclear installations and facilities.

II–4.2. The Jason safety case

The existing Jason operational safety case consisted of a single overarching safety document originally written in the early 1960s. This document had been extensively revised over the years but retained its original format and was supported by tiers of lower level documentation. The last major revision of the safety document occurred in 1994, when it was brought up to (the then) modern safety report standards. The document consisted of 19 sections covering all aspects of layout, structure, systems, power supplies, facilities, waste management, radiological protection, safety management, safety principles and safety criteria, hazard analysis, deterministic and probabilistic safety assessments, operation, commissioning, modification, quality assurance and decommissioning. Each section referred to other safety management or controlling documents, as necessary. The decommissioning section consisted of a basic plan derived from the IAEA, Nuclear Installations

FIG. II–1. Jason reactor site.

Inspectorate (NII) and Ministry of Defence (Navy) (MoD(N)) guidelines available in 1994, covering outline regulatory requirements, waste disposal, transport, environmental assessments, decommissioning plans, safety evaluation, administrative controls, quality, and emergency and security arrangements.

The decommissioning safety case consisted of a single preliminary safety report (PSR) and two sets of design substantiation reports (DSR), pre-decommissioning safety reports (PDSR), commissioning schedules (CS), commissioning reports (CR) and pre-operational safety reports (POSR) covering fuel removal and reactor dismantling activities. The final safety report was the post-decommissioning report (PDR).

II–4.2.1. Problem

It was clearly necessary to produce a new PSR that reflected the latest international, national and MoD(N) decommissioning guidelines and experiences prior to commencing the project.

II–4.2.2. Solution

A comprehensive PSR was established by the Jason project manager that included consideration of the latest IAEA, NII and MoD(N) decommissioning guidelines and industry experience gained from the previous decommissioning of low power reactors and other nuclear related facilities and sites. This process took a considerable amount of the Jason project manager’s and site licensee’s time and resources, which could have been considerably reduced had the existing safety case been more comprehensive and up to date.

II–4.2.3. Lesson learned

The site licensee should regularly review and update its existing site or facility safety case to ensure that it is consistent with the current IAEA, NII and MoD(N) guidelines on decommissioning strategies and activities and that it takes due cognisance of the experience gained during the decommissioning of similar sites or facilities.

II–4.3. Independent peer review

The tight project timescales during the decommissioning of Jason dictated a parallel production, review and assessment of both sets of safety reports associated with the fuel removal and reactor dismantling, written by separate contractors. As an example of parallel assessment, the later issues of the POSR (for fuel removal) would be considered in the same period as the middle issues of the design report (for reactor dismantling) and the early issues of the PDSR (for reactor dismantling).

II–4.3.1. Problem

Each individual report necessarily contained information from another report, which in itself was being amended and rewritten by a different contractor. Inevitably, some information was transferred over in a cut and paste form and often the text from an early issue would be carried over with mistakes that had been edited out of the current version of the document. In some cases the volume of out of date material was such that important safety principles could have become lost if great care was not taken by the reviewing authorities. The parallel production, review and assessment of individual safety reports and documents written by separate contractors to meet the overall decommissioning programme therefore required careful management and control.

II–4.3.2. Solution

This problem was effectively mitigated by appointing a single expert external independent peer review team under contract to assess all documentation during each phase of the decommissioning project, regardless of its source of origin. This process was greatly assisted by ensuring that both the site licensee and the Jason project manager organizations had robust document internal peer review arrangements and procedures.

II–4.3.3. Lesson learned

The appointment of a single expert external independent peer review team that covers all phases of a decommissioning project is recommended for future decommissioning activities.

II–4.4. Safety management framework

The site licensee was authorized to operate Jason in accordance with the Conditions of Operation (equivalent to a civil nuclear site licence) to meet the requirements of the safety document; however, the actual day to day nuclear safety management arrangements were detailed in lower level documents, such as the Jason standing orders. Other safety management documents included the Jason operating instructions, Jason maintenance procedures, radiological safety standing orders and emergency orders, quality assurance plans and programmes, health and safety plans and security plans of the facility. All safety reports and safety management documents were either written or amended to accommodate the decommissioning requirements, and, following due process, all had to be approved by the site licensee. The large number of safety reports and safety related documents could have caused confusion, particularly for the management arrangements for the decommissioning contractors.

II–4.4.1. Problem

The site licensee needed a single comprehensive document that addressed all its nuclear, radiological and conventional safety management responsibilities.

II–4.4.2. Solution

A single non-executive safety management framework document was produced to provide the site licensee with a dedicated safety management focus. This framework document covered the departmental safety policy, regulatory control, quality assurance, safety management organization, key personnel, safety groups and committees. The overall decommissioning policy, safety principles and criteria, facility area definitions, document peer review and approval, safety responsibilities, control of contractors, radiation protection, health and safety at work, audit and inspection, reporting and investigating incidents and accidents and emergency response arrangements were also outlined. Each section of the framework document referred to the applicable executive controlling document, as required.

II–4.4.3. Lesson learned

A non-executive nuclear safety management framework document should be produced prior to commencing any decommissioning project to provide the site licensee with a focus for its various nuclear safety, conventional safety and quality related responsibilities.

II–4.5. Control of contractors

Three main contractors and 13 subcontractors were employed in total on and off the site during the decommissioning of Jason, with the subcontractors working directly for the main contractors. In addition, other subcontractors where also employed at various times, working directly for the college property manager.

II–4.5.1. Problem

To maintain its mandated nuclear and radiological safety standards and responsibilities and to retain adequate control throughout the decommissioning project, it was clearly necessary for the site licensee to provide comprehensive and suitable safety management instructions for these various contractors and subcontractors.

II–4.5.2. Solution

Contractor control instructions (CCIs) were written early on in the decommissioning project to ensure that the site licensee maintained its nuclear safety responsibilities. CCIs were produced in two parts: the first part specified the arrangements for controlling the safety of work by contractors within the Jason decommissioning facility and the second part specified how these arrangements would be complied with. CCIs encompassed training, qualifications and experience for safety related post holders, written systems of work, personnel protective equipment, safety monitoring, health and safety records, visitor safety, operating rules, maintenance, inspection, testing, categorization, document clearance procedures, occurrences, incidents, accidents and emergency arrangements. Other CCIs addressed radioactive and non-radioactive waste management, the movement of radioactive and dangerous material, accounting for radioactive and nuclear material, radiation, criticality and general safety, the Control of Substances Hazardous to Health Regulations and the security of the facility. Twenty one individual CCIs were produced in total.

II–4.5.3. Lesson learned

To maintain its mandated nuclear and radiological safety management responsibilities the site licensee should ensure that comprehensive and suitable written instructions be available to control the activities of all contractors working within a decommissioning site or facility.

II–4.6. Continuity

At the same time as decommissioning Jason the site licensee was in the process of relocating its department to another site; this imposed additional management constraints over and above those needed to manage the decommissioning process safely. The actual move took place immediately following the removal of the fuel from the site, which was considered to be the highest risk based decommissioning activity. Prior to the fuel removal, the site licensee’s full management team was on the site.

II–4.6.1. Problem

The relocation and decommissioning had different management aims, considerations and constraints. Clearly there was an absolute requirement for the site licensee to maintain nuclear safety management standards and continuity throughout the decommissioning project.

II–4.6.2. Solution

The site licensee appointed the existing Jason reactor manager as the Jason decommissioning superintendent early on in the project. Furthermore, three additional dedicated full time chartered nuclear engineers were recruited and appointed to the resident Jason engineering team. Following the relocation, the Jason decommissioning superintendent and the resident team acted as the client overseeing team, which was fundamental in providing continuity and maintaining the site licensee's nuclear and radiological safety responsibilities during the decommissioning.

II–4.6.3. Lesson learned

The site licensee should ensure that a suitably qualified, experienced and dedicated in-house client overseeing team be appointed early on to prove continuity during a decommissioning project. Ideally, at least one member of the client overseeing staff should have long standing previous senior management experience of the operations of the facility.

II–4.7. Records

Following the installation of the Jason reactor in 1962, comprehensive records of facility operations, modifications and incidents had been kept and were available to the decommissioning contractors and the client overseeing team. However, the installation records, and in particular the records of previous nuclear and/or radiation related operations in the facility prior to the installation of Jason, were not so comprehensive.

II–4.7.1. Problem

The less than comprehensive installation and building previous use records had the potential to delay the completion of the decommissioning, particularly regarding meeting the final radiological site clearance criteria.

II–4.7.2. Solution

As decommissioning progressed the contractors made good use of the most recent records and of the Jason decommissioning superintendent's knowledge of the layout of the reactor. In addition, the whereabouts of retired Jason operational and engineering personnel became important, as they were able to provide the project with valuable insights and suggestions as the project progressed. This first hand knowledge became increasingly important during the final stages of the project, when unexpected extensive tritium contamination was found in the concrete floors outside the reactor hall, which lead directly to a two month delay to the anticipated early completion of the project. This tritium contamination was caused by previous neutron accelerator operations that predated Jason operations, which were discovered primarily by personal contacts with retired personnel, rather than through existing records. Had this matter been known about or considered at the beginning of the decommissioning, the project would have been completed about two months earlier.

II–4.7.3. Lesson learned

The site licensee should ensure that comprehensive and accurate previous building use, nuclear facility installation, through life operation, modification and incident records be kept and updated and made fully available at the start of any decommissioning project. In addition, the site licensee should also ensure that at least one member of its client overseeing staff has previous experience of facility operations to help to mitigate any shortfall in previous records.

II–4.8. Control of work packages

Traditional decommissioning work package reviews and hold points, produced by the Jason project manager, were detailed in the overall project plan. These reviews and hold points were mainly work and time–cost based project management points associated with discrete work packages, similar to what would be produced for any construction project.

II–4.8.1. Problem

The site licensee needed to have a workable safety management hold point and associated authority to proceed system in place to maintain nuclear and radiological safety standards during the sequencing of work, time and cost based project management hold points and work packages.

II–4.8.2. Solution

Early on in the decommissioning project the site licensee produced a nuclear safety management hold point strategy document. This divided up the major work items of the decommissioning project into manageable phases that maintained nuclear and radiological safety, as opposed to those work packages that addressed the standard project based time and cost considerations. Each hold point had its own particular site licensee completion criteria, which would have to be met before the authority to proceed was granted to commence the next work package and proceed to the next hold point. For example, verification of the non-active commissioning of the fuel removal equipment, including the completion of practice sessions and the approval of the POSR (for fuel removal) was required before the authority to proceed was granted to carry out the actual fuel removal activities. Within each major hold point there was a series of minor subhold points, each cleared by an acceptance group made up of senior site licensee personnel, who would issue an associated subhold point clearance certificate. For example, a subhold point clearance certificate would be issued following the correct installation of the fuel removal equipment, and another following its testing prior to carrying out non-active commissioning trials.

At each major hold point the site licensee would review the relevant subhold point clearance certificates and would judge whether the particular hold point completion criteria had been met. The site licensee would then issue its written authority to proceed to enable the project to proceed to the next major hold point, as appropriate. The clearance of the majority of the hold points and subhold points was primarily dictated by the progress of work on the ground, but also invariably coincided with the approval of the relevant safety case, design report or commissioning schedule. The key hold points were the completion of the post-operational clean out, commissioning of the fuel removal equipment, defuelling (Fig. II–2), dismantling of the fuel removal equipment, commissioning of the waste handling equipment, reactor dismantling (Fig. II–3) and waste removal (Fig. II–4), dismantling of the waste handling equipment, radiological clearance of the site (Fig. II–5) and final handover of the site.

II–4.8.3. Lesson learned

The site licensee should ensure that a comprehensive and suitable nuclear safety management hold point and authority to proceed control system is in force that addresses sequential and parallel decommissioning work packages and activities. This hold point and authority to proceed system is required to maintain nuclear, radiological and conventional safety responsibilities, in addition to the standard project management time and cost based hold points.

II–4.9. Permits to work

The large number of contractors working on the site necessitated the introduction of a robust and comprehensive permit to work and certificate of isolation system. This permit system was necessary to control the day to day work related

FIG. II–2. Removal of the fuel cask.

FIG. II–3. Reactor dismantling.

activities of all personnel working on the site, including the main contractors, subcontractors and other staff, and to maintain the site licensee's nuclear, radiological and conventional safety standards and responsibilities.

II–4.9.1. Problem

The site licensee's existing permit to work and certificate of isolation system, while adequate for normal Jason operations, had to be significantly upgraded to cover the decommissioning operations.

II–4.9.2. Solution

A new permit system, covering both nuclear and non-nuclear related activities and similar in format to that in use by the prime contractor at other civil nuclear sites, was established in conjunction with the contractors and approved by the site licensee. This new system was introduced over a trial period alongside the existing permit system until its satisfactory operation was proven. The overall permit to work system consisted of a contractor duly authorized person or suitably qualified and experienced person signed permit request form, combined with a nuclear work procedure or method statement, a hazard and risk assessment and a certificate of isolation, as required. The certificate of isolation identified the various electrical and mechanical services that needed to be isolated before the work commenced.

The permit had to be endorsed by the site licensee's health physicist and the health and safety officer before it was finally approved and formally issued to the contractor by the Jason decommissioning superintendent, acting on behalf of the site licensee. The site licensee's on-site health and safety officer was the single point of contact for all permits to work and certificates of isolation. The site licensee administered the day to day workings of the system, which included the receipt, final issue, monitoring and daily mustering of all permits and certificates in force.

All work undertaken within the Jason decommissioning facility required an approved permit to work and certificate of isolation before work could commence, regardless of whether that work was being undertaken by the decommissioning contractors or by site licensee staff.

II–4.9.3. Lesson learned

The site licensee should have a comprehensive and suitable permit to work and certificate of isolation system in force during a decommissioning project that effectively maintains its nuclear, radiological and conventional safety standards and responsibilities. This permit to work system should be used to control the day to day work activities of all decommissioning contractors and other personnel employed on the site.

FIG. II–4. Waste removal.

FIG. II–5. Site appearance after the completion of the decommissioning.

II–5. UNITED STATES OF AMERICA

II–5.1. Problem

During the preparations for the dismantling of some 60 plus plutonium gloveboxes in nine laboratories within Building 212 at the Argonne National Laboratory East Site from 1994 to 1995, an identified major item of concern was the release of residual material during the size reduction of the gloveboxes and even more so during the removal of the neoprene window gaskets of the gloveboxes. In order to ensure that the release of any airborne contamination was minimized and was adequately addressed in work planning, some proactive procedural contamination control measures were incorporated into the work plan.

II–5.2. Solution

Actions that were taken to maintain airborne contamination during the dismantling of the gloveboxes are described below.

- Tool effectiveness and the general operations were tested on a mock-up in a clean area;
- The glovebox cutting tool was vacuum cleaned after each use to minimize the generation of radioactive dust;
- The use of a clear plastic shield between the operator and cutter minimized the scattering of any contamination on to worker personal protective equipment;
- Local exhaust ventilation was used for radioactivity capture during the cutting of the gloveboxes;
- Aggressive decontamination techniques were used only where other techniques were ineffective in removing oily stained areas of radioactive contamination;
- Administrative hold points were incorporated into the dismantling work processes to provide the opportunity to improve upon future glovebox dismantling based on recent operational lessons learned.

All of the above solutions combined to allow for the project work to be completed with minimal airborne release to the size reduction containment enclosure and minimal contamination of protective clothing.

II–5.3. Lesson learned

Airborne contamination is a serious issue even in the dismantling of small nuclear facilities.

GLOSSARY

Definitions are taken from the IAEA Safety Glossary rev. April 2000, except those marked by an asterisk, which are additional definitions extracted from various sources and are only for the purposes of this report.

accelerator (particle accelerator). A device that increases the speed and kinetic energy of electrically charged atomic and subatomic particles.*

accident. Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

activation. The process of inducing radioactivity.

- Most commonly used to refer to the induction of radioactivity in moderators, coolants, structural and shielding materials, caused by irradiation with neutrons.

ALARA. As low as reasonably achievable. Taken in the context of the process of determining what level of protection and safety makes exposure and the probability and magnitude of potential exposure as low as reasonably achievable, economic and social factors being taking into account.*

brachytherapy. The use of small tubes or hollow wires containing small quantities of radioactive material to treat cancer tumours by direct contact or proximity.*

clearance. Removal of radioactive materials or radioactive objects within authorized practices from any further regulatory control by the regulatory body.

contamination. Radioactive substances on surfaces, or within solids, liquids or gases (including the human body), where their presence is unintended or undesirable, or the process giving rise to their presence in such places.

critical assembly. An assembly containing fissile material intended to sustain a controlled fission chain reaction at a low power level, used to investigate reactor core geometry and composition.

cyclotron. A device that accelerates charged atomic or subatomic particles in a constant magnetic field.*

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository which is closed and not decommissioned).

disposal. Emplacement of waste in an appropriate facility without the intention of retrieval.

exemption level. A value, established by a regulatory body and expressed in terms of activity concentration and/or total activity, at or below which a source of radiation may be granted exemption from regulatory control without further consideration.

notification. A document submitted to the regulatory body by a legal person to notify an intention to carry out a practice or other use of a source.

radiotherapy. The treatment of disease by means of X rays or other forms of ionizing radiation.*

registrant. See registration.

registration. A form of authorization for practices of low or moderate risks whereby the legal person responsible for the practice has, as appropriate, prepared and submitted a safety assessment of the facilities and equipment to the Regulatory Authority. The practice or use is authorized with conditions or limitations as appropriate. The requirements for safety assessment and the conditions or limitations applied to the practice should be less severe than those for licensing.

regulatory body. An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

source. Radioactive material used as a source of radiation.

disused source. A source no longer in use or intended to be used.

orphan source. A source which poses sufficient radiological hazard to warrant regulatory control, but which is not under regulatory control because it has never been so, or because it has been abandoned, lost, misplaced, stolen or otherwise transferred without proper authorization.

sealed source. Radioactive material that is (a) permanently sealed in a capsule, or (b) closely bonded and in a solid form.

spent fuel. Nuclear fuel removed from a reactor following irradiation, which is no longer usable in its present form because of depletion of fissile material, poison build-up or radiation damage.

spent fuel management. All activities that relate to the handling or storage of spent fuel, excluding off-site transportation.

spent source. A source that is no longer suitable for its intended purpose as a result of radioactive decay.

storage. The holding of spent fuel or of radioactive waste in a facility that provides for its containment, with the intention of retrieval.

teletherapy. Radiotherapy using a source of radiation at a distance from the patient.*

tomography. Method of radiography displaying details in a selected plane within the body.*

unsealed source. A source that does not meet the definition of a sealed source.

waste, high level (HLW). The radioactive liquid containing most of the fission products and actinides present in spent fuel — which forms the residue from the first solvent extraction cycle in reprocessing — and some of the associated waste streams; this material following solidification; spent fuel (if it is declared a waste); or any other waste with similar radiological characteristics.

waste, low and intermediate level (LILW). Radioactive waste with radiological characteristics between those of exempt waste and high level waste. These may be long lived waste (LILW–LL) or short lived waste (LILW–SL).

waste management, radioactive. All administrative and operational activities involved in the handling, pretreatment, treatment, conditioning, transport, storage and disposal of radioactive waste.

waste, primary. Waste unchanged from the form and quantity in which it was generated.*

waste, radioactive. For legal and regulatory purposes, waste that contains or is contaminated with, radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body.

waste, secondary. A form and quality of waste that results as a by-product from the processing of waste.*

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