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Considerations for Waste Minimization at the Design Stage of Nuclear Facilities

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CONSIDERATIONS FOR WASTE MINIMIZATION AT THE DESIGN STAGE OF NUCLEAR FACILITIES

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FOREWORD

The decommissioning of nuclear facilities designed and constructed many years ago has revealed problems associated with the dismantling of old equipment and the management of the material and waste generated. Extensive research, development and analysis of the techniques used during the decommissioning of old facilities have been undertaken, and the amounts and characteristics of both operational and decommissioning waste have been evaluated. Practical experience in the decommissioning and management of associated waste and material has also increased. Analysis shows that dismantling, decontamination and management of generated waste can be optimized if these steps are properly considered at the design stage of a nuclear facility. Consideration of waste minimization, waste management and decommissioning issues during the design stage influences the economic and safety aspects of facility operation as well as the facility's post-operation management.

Recognizing the growing importance of this subject, the IAEA has produced this report with the aim of identifying and outlining issues to be considered at the design stage of nuclear facilities to minimize future waste generation, facilitate future decommissioning and optimize management of decommissioning and operational waste and material. The IAEA is grateful to all those who participated in the various consultants and technical meetings and helped to prepare the report. Special thanks are extended to L. Teunckens (Belgium) and L. Valencia (Germany), who were involved in the process from the initial draft to the final version of this publication.

The IAEA officers responsible for this report were V. Efremenkov and Z. Drace of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

SUMMARY . . . 1

1. INTRODUCTION . . . 3

   1.1. Background . . . 3
   1.2. Objective . . . 4
   1.3. Scope . . . 5
   1.4. Structure . . . 5

2. TYPES AND QUANTITIES OF WASTE GENERATED DURING FACILITY OPERATION AND DECOMMISSIONING . . . 6

   2.1. Introduction . . . 6
   2.2. Generation of low and intermediate level radioactive waste in nuclear facilities . . . 10
      2.2.1. Waste generation from nuclear fuel facilities . . . 10
      2.2.2. Waste generation from operation of nuclear power plants . . . 14
      2.2.3. Waste from spent fuel reprocessing . . . 19
      2.2.4. Waste from decontamination and decommissioning of nuclear facilities . . . 21
      2.2.5. Waste from institutional and industrial facilities, including research laboratories . . . 26
   2.3. Mixed waste streams . . . 28

3. METHODOLOGIES FOR WASTE MINIMIZATION . . . 29

   3.1. Objectives of waste minimization . . . 29
   3.2. Waste minimization options . . . 29
      3.2.1. Control of radioactive waste generation . . . 30
      3.2.2. Prevention of activation/contamination . . . 31
      3.2.3. Reuse and recycling of materials . . . 32
      3.2.4. Reduction of radioactive waste volumes . . . 32

4. DESIGN OPTION CONSIDERATIONS FOR WASTE MINIMIZATION . . . 35

   4.1. Introduction . . . 35

4.2. Considerations for minimizing contamination problems 37
4.2.1. Building and equipment layout 38
4.2.2. Selection of components 44
4.2.3. Reducing surface contamination 49
4.2.4. Reducing leakage and crud traps 53
4.2.5. Quality control 53
4.2.6. Limiting corrosion 54
4.2.7. Minimizing the spread of contamination 54
4.3. Provisions to facilitate decontamination 55
4.4. Provisions to facilitate dismantling and segmentation 57
4.4.1. Minimum use of hazardous materials 57
4.4.2. Additional plant layout requirements to facilitate dismantling 57
4.4.3. Preplacement of dismantling aids 58
4.4.4. Shielding 59
4.4.5. Connectors 61
4.4.6. Intact removal of large components 61
4.4.7. Documentation 63
4.4.8. Planning 66
4.5. Development and improvement of techniques for decontamination and decommissioning 67
4.5.1. Development and improvement of decontamination techniques 67
4.5.2. Development and improvement of dismantling techniques 69
4.5.3. Development and improvement of measurement techniques 71
4.5.4. Developments to simplify waste management 73
4.5.5. Design for safe enclosure and deferred dismantling . . . 79
4.6. Development and improvement of the regulatory approach . . 82
4.7. Nuclear facility design basis guidelines for waste minimization 83

5. CONCLUSIONS 90

REFERENCES 95
ANNEX 103
GLOSSARY 141
CONTRIBUTORS TO DRAFTING AND REVIEW 143

SUMMARY

This publication identifies and outlines issues to be considered during the design stage of nuclear facilities to minimize future waste generation, facilitate future decommissioning and optimize management of decommissioning and operational waste and material. Extensive research, development and analysis of the techniques used during the decommissioning of old facilities have been undertaken, and the amounts and characteristics of both operational and decommissioning waste have been evaluated. Practical experience in the decommissioning and management of the associated waste and materials has also increased. Analysis shows that dismantling and decontamination of facilities and the management of the generated waste can be optimized if these aspects of the decommissioning process are taken into consideration during the facility design stage. Consideration of waste minimization, waste management and decommissioning issues during the design stage influences the economic and safety aspects of facility operation as well as post-operation management. Recognizing the growing importance of this subject in Member States, the IAEA has produced this report with the aim of identifying and outlining these issues.

APPROACH

This report discusses options for nuclear and other facilities handling radioactive material aimed at optimizing the management of their operational and decommissioning waste and facilitating their safe, effective and timely decommissioning. These options include consideration of the layout of the facility and its components, the selection of construction materials and components, maintenance and support activities, waste minimization measures, waste management options, materials recovery and reuse, and documentation and record keeping.

The principles discussed are applicable to the design and operation of all types and classes of nuclear facility dealing with radioactive material and to modifications of existing plants. Ideally, they will also be considered during the operational phase of any facility to facilitate future decommissioning and to avoid or mitigate the generation of waste in a form that complicates processing and disposal.

CONTENTS

Regarding the minimization of radioactive waste production, several options have been formulated for consideration when designing a new facility, modifying an existing plant or defining future decontamination and decommissioning (D&D) operations. These options are summarized as follows:

— Considerations to minimize contamination problems;
— Provisions to facilitate decontamination;
— Provisions to facilitate dismantling and segmentation;
— Documentation and design of a record keeping system;
— Decommissioning planning;
— Development and improvement of D&D techniques;
— Development and improvement of the regulatory approach;
— Guidelines for the design basis of a nuclear facility for waste minimization.

The Annex provides an overview of facilities on both the 'front end' and the 'back end' of the nuclear fuel cycle, including various power reactors, research reactors, critical assemblies, research laboratories and hot cells, as well as waste management facilities.

CONCLUSIONS

This publication provides conclusions derived from a review of the lessons learned from the operational and decommissioning experience gained by Member States to date. While plant designs will continue to mature and evolve, the waste minimization options identified here will remain relevant to all new facilities and can be used as a checklist during the design, licensing and operational phases of new plants or the modification of existing plants. A detailed set of conclusions is provided in Section 5 of the report.

1. INTRODUCTION

1.1. BACKGROUND

Nuclear and other facilities handling radioactive material are like any other industrial facility in that they produce by-products and waste materials in addition to their useful products. They also generate radioactive waste, which must be handled carefully in order to reduce potential harm to facility operators, the public and the environment. The amounts and nature of the by-products and waste produced by any facility depend on its design, including its layout, the processes employed and the materials used (e.g. in its construction, as process feeds). The generation of waste, particularly toxic and radioactive waste, needs to be minimized as far as is practicable, in accordance with Principle 7 of the Principles of Radioactive Waste Management [1], part of the IAEA Safety Fundamentals.

The concept of waste minimization is interpreted in various ways [2]. It is often taken to mean a reduction of the total quantity of waste and may or may not involve a reduction of the total activity in the waste. In either case, it leads to simplified waste management and to a reduction of the total costs, both of which are of interest to the operator. In contrast, the regulators are primarily concerned with controlling occupational exposure and the potential environmental impact. In practice, a trade-off is usually made between the benefits accruing from waste minimization and the costs of achieving those benefits.

There is extensive experience concerning the generation and management of operational radioactive waste from a diverse range of facilities handling nuclear and other radioactive material. Over the past 10–15 years, there has been a substantial decline in the volumes of waste generated, particularly at nuclear power plants. This decrease has resulted from a combination of technical improvements and the promotion of a safety and waste minimization culture.

Substantial experience has also been gained from the decommissioning of older nuclear facilities and the management of the associated waste and material. These decommissioning activities have revealed various problems associated with the management of this waste, some of which has required particular care because of its radioactive nature or owing to risks arising from its chemical toxicity or other hazardous properties.

These experiences have shown that, ideally, consideration will be given to waste management and decommissioning during the plant design and construction stages. Numerous reports and studies have highlighted the costs and complexities involved in the decommissioning of existing nuclear facilities.

Extensive research, development and analysis have been undertaken to assess the effectiveness of techniques used during the decommissioning and dismantling of such facilities and to evaluate the amount of decommissioning waste that has been or will be generated. The results show that taking the minimization of operational and decommissioning waste into consideration at the original design stage of the facility or prior to facility modification could have markedly reduced the difficulties associated with dismantling, decontamination and the management of the associated waste, particularly the most problematic waste.

Designers of new facilities and their customers need to be aware of these issues, including the strategies and techniques involved in decommissioning and waste management. Lessons learned from decommissioning projects are invaluable for creating an awareness of 'decommissioning friendly' features for future designs [3].

This experience will be of particular interest to Member States planning to construct or modify nuclear facilities. These facilities will benefit from much greater emphasis at the design stage on both operational and decommissioning waste minimization to ensure that timely, safe and effective decommissioning can be carried out.

1.2. OBJECTIVE

This report is aimed at a broad spectrum of the experts involved in the design and operation of new nuclear facilities, including design engineers and builders, owners, operators, regulators and authorities. Its objective is to identify options to be considered during the design and operation of nuclear and other facilities handling radioactive material to optimize the management of their operational and decommissioning waste and facilitate their safe, effective and timely decommissioning.

These options include consideration of:

— The layout of the facility and its components;
— The selection of construction materials and components;
— Maintenance and support activities;
— Waste minimization measures and waste management options;
— Materials recovery and reuse;
— Documentation and record keeping.

These options for waste minimization are derived from a review of the lessons learned from the operational and decommissioning experience gained by Member States to date. While plant designs will continue to evolve in the future, these waste minimization options will remain relevant to all new facilities and can be used as a checklist during the design, licensing and operational phases of new plants or the modification of existing plants.

1.3. SCOPE

The information in this report is applicable to the design and operation of all types and classes of nuclear facility dealing with radioactive material. The principles may also be applied to modifications of existing plants and can be considered during the operational phase of any facility to facilitate future decommissioning and to avoid or mitigate the generation of waste in a form that complicates processing and disposal.

The report also comments on mixed waste streams and waste arising from institutional, industrial, clinical, medical and research activities. Although no specific measures are identified for reducing future arisings at such facilities, the principles will be of value in reducing the waste that is generated.

The report does not go into detail concerning the methods for treating or characterizing wastes. They are the subject of separate publications [4, 5].

1.4. STRUCTURE

Section 2 of the report provides an overview of the typical types, quantities and origin of waste material currently generated from the nuclear fuel cycle during the operation and decommissioning of the major types of nuclear facility. Section 3 describes the principles of waste minimization, taking due account of the lessons learned during the actual decommissioning of certain installations. Section 4 provides a discussion of the options to be considered when designing new facilities and in developing future decontamination and decommissioning (D&D) practices. Section 5 summarizes the options and provides some conclusions from the review. In the Annex, a brief description is given of the typical processes used in the front end of the fuel cycle, at nuclear power plants and in the back end of the fuel cycle. A glossary of selected terms used in the report but not defined in the IAEA Radioactive Waste Management Glossary is provided at the end of the book.

2. TYPES AND QUANTITIES OF WASTE GENERATED DURING FACILITY OPERATION AND DECOMMISSIONING

2.1. INTRODUCTION

The generation of electricity by nuclear power plants involves the operation of a number of facilities, including: nuclear fuel cycle facilities to produce fuel (the ‘front end’ of the fuel cycle), which is then irradiated in a nuclear reactor, after which it is sent either to spent fuel storage or to facilities for the reprocessing of spent fuel (the ‘back end’ of the fuel cycle); facilities required for the management of operational radioactive waste; and facilities for the management of radioactive waste from decommissioning, including facilities for nuclear waste treatment and disposal. The radioactive waste streams are generated either by the operation of these facilities (including the nuclear power plant) or by their decommissioning at the end of their service life.

The front end of the fuel cycle encompasses uranium extraction, conversion, enrichment and fuel fabrication, and the fabrication of plutonium–uranium mixed oxide (MOX) fuels. The back end covers the storage and/or reprocessing of spent fuel and the management of the resulting operational waste [2, 4–8]. A general overview of the process material streams and routes of the entire fuel cycle is shown in Fig. 1.

The broad spectrum of non-reactor facilities includes some systems and processes similar to those found on reactor sites. These are mainly irradiated fuel storage facilities (wet or dry); radioactive waste handling, treatment and storage facilities; and ancillary facilities such as water purification circuits, ventilation plants, laboratories and maintenance facilities.

The management of radioactive waste produced during the operational period and during the decommissioning of the related facilities involves long timescales and, in many cases, different source terms and pathways. Waste management is to be carried out in such a way that human health and the environment are protected both now and in the future. Effects beyond national borders need to be taken into account, passing undue burdens to future generations is to be avoided, waste is to be minimized, appropriate legal frameworks are to be established and interdependencies among all these steps are to be taken into account [1]. These principles lead to requirements:

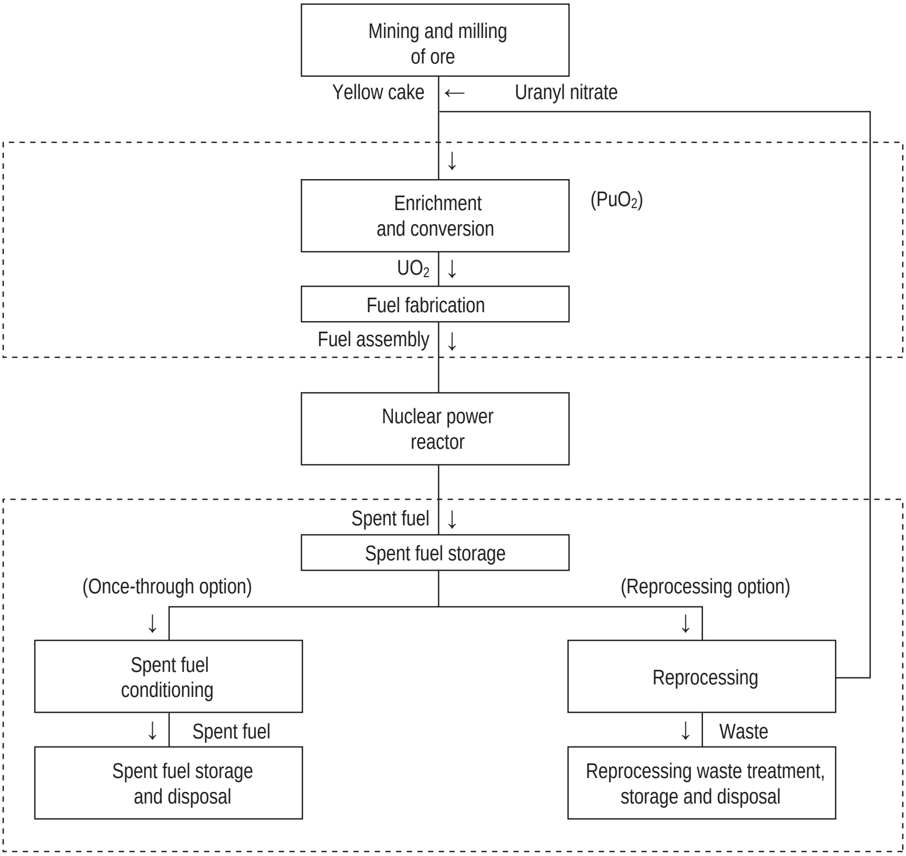

FIG. 1. Simplified nuclear fuel cycle.

— To specify the ultimate safe and satisfactory condition for all types of waste;
— To move waste to the end state as early as is practicable;
— To ensure that intermediate steps do not inhibit or complicate the achievement of the end state, and/or that the design of facilities and waste management practices can be optimized as part of the optimization of the overall system and its life cycle;
— To cover the costs of managing all waste in the life cycle;
— To cover the accumulated liability at all stages of the life cycle [9].

Much of the solid radioactive waste arising from the D&D of a nuclear facility is the same as the waste arising during the operational phase [7]. Depending on the nature of the facility, this waste comprises:

— High level waste, and low and intermediate level long lived waste in the form of spent fuel, the products of fuel reprocessing or material contaminated with long lived radionuclides. These types of waste generally are not associated with the actual dismantling of the facility.
— Low and intermediate level short lived waste in the form of irradiated items and material contaminated with short lived radionuclides. These may include nuclear facility components, equipment and building materials such as steel and concrete containing only small concentrations of radionuclides.

Liquid and gaseous effluents produced during D&D activities are generally similar to those produced during normal operation, except, perhaps, in cases where special chemicals are used during decontamination.

Most of the decommissioning waste is managed using the arrangements in place for dealing with similar waste arising during normal operation. Such arrangements are generally well developed, and their costs are known. Some of the waste, however, is unique to D&D activities, including:

— Very large items within the nuclear power plant, such as heat exchangers.
— Large quantities of graphite containing long lived radionuclides, in some cases constituting a possible fire hazard.
— Mixed waste containing toxic or hazardous material such as sodium, beryllium, lead or asbestos.
— Relatively large quantities of material having radionuclide concentrations close to the clearance levels at which it may be released conditionally or unconditionally upon its further use, depending on local regulations. This may include materials that have been decontaminated, such as steel, concrete or other useful materials.
— Large quantities of waste that is not radioactive but that is subject to regulatory control because it arises on a nuclear licensed site. This is sometimes treated as ‘suspect waste’, because of the possibility of its having become contaminated [7].

The costs of treating, storing and disposing of decommissioning waste may dominate the overall costs of decommissioning. Therefore, it is important to maximize the reuse or recycling of decontaminated and recovered material, and to minimize the amount of material that will require management as radioactive waste [7].

The principle of clearance has already been used successfully in some countries. Within the European Union (EU), guidance on its practical use is established by the European Commission. Member States of the EU are free to set their own clearance levels. Any inconsistency in the practical application may cause some difficulty for international trade or for transboundary shipment. It is also interesting to note that the maximum radionuclide levels set for clearance of material from sources under nuclear regulation are substantially lower than those for the unrestricted use or disposal of materials from conventional industrial sources containing technologically enhanced levels of naturally occurring radionuclides. The rate of production of these materials and their accumulated quantities are orders of magnitude greater than those of the low radionuclide concentration material arising from D&D activities [7].

In most OECD Nuclear Energy Agency (OECD/NEA) Member countries, consideration of D&D and waste management starts at the facility design stage, most often with the selection of appropriate materials and construction techniques. This approach reflects the first basic principle of waste management, namely that “generation of radioactive waste shall be kept to the minimum practicable” [1].

For example, in existing heavy water reactor systems, materials are selected and operating procedures are designed to enhance reactor efficiency and to limit radiological doses to workers and the public. This results in low radionuclide concentrations in the environment and good environmental performance. Many changes have been, or are being, implemented to improve reactors and to further reduce doses and discharges to the environment. Some of the existing heavy water reactors have been retrofitted to include these changes. Many of these improvements not only result in lower doses and reduced discharges of radioactivity, but also achieve reduced discharges of chemicals and products of metal corrosion. Improved reactor efficiency, dose reduction and environmental performance work ‘hand in hand’, and ideally they will be introduced at the planning stage. It is clear that, through design stage selection of appropriate materials and adoption of adequate waste management procedures, the environmental effects due to nuclear facilities can be mitigated to very low levels (to much lower levels) [10].

2.2. GENERATION OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE IN NUCLEAR FACILITIES

This section provides a brief description of the types, typical quantities and origin of waste generated in:

— Nuclear fuel fabrication;
— Operation of nuclear power plants;
— Spent fuel reprocessing;
— D&D of non-reactor nuclear facilities;
— Other institutional and industrial facilities, including research laboratories.

2.2.1. Waste generation from nuclear fuel facilities

The processes typically used during the refining, conversion, enrichment and fuel fabrication stages are given below, along with an overview of the types, quantities and origin of waste generated during these processes [2].

2.2.1.1. Refining

Refining is defined as the processing of uranium ore concentrates (UOCs) to produce uranium trioxide (UO_3) or uranium dioxide (UO_2). This process may be carried out at a single site or as part of an integrated process involving different sites. A general sequence of different processes resulting in UO_3 and UO_2 production considers the following:

— Purification;
— UO_3 and UO_2 production by:
  • The thermal denitration (TDN) process;
  • The ammonium diuranate (ADU) process;
  • The ammonium uranyl carbonates (AUC) process.

Table 1 lists the typical arisings from the refining processes of a hypothetical facility per 1000 t of uranium throughput. A major part of the waste generated during the refining process is associated with the purification stage or with the formation of uranyl nitrate liquor (UNL). After the purification stage, the UNL is quite pure, and a very small amount of waste is linked to the production of UO_3 or UO_2 (through the use of the TDN, ADU or AUC process, or during the filtration and calcination steps).

TABLE 1. TYPICAL ARISINGS FROM THE REFINING PROCESSES (PER 1000 t U)

Arising	Quantity	Classification	Comment
Drums	70 t	Material for recycling or waste	All processes
Insoluble waste and filter aid material	50 t	Waste	All processes (depends on the nature of UOC)
Liquid effluent	3 000–10 000 m^3	Waste	All processes (depends on the nature of UOC)
Sludge	300 t	Waste	All processes (depends on the nature of UOC)
Liquid nitrates	200 t	By-product	ADU and AUC processes

Note: ADU: ammonium diuranate; AUC: ammonium uranyl carbonates; UOC: uranium ore concentrate.

The quantities of insoluble waste and sludge that are generated are closely related to the type and quality of UOC. The waste arising from the TDN process is not directly comparable with that from the ADU and AUC processes, because the feeding of UOC for these processes is different.

2.2.1.2. Conversion

Conversion is defined as the processing of UO_3 or UO_2 to produce uranium hexafluoride (UF_6), although uranium tetrafluoride can also be used, for example, for the production of metallic uranium. Because of its thermal stability and relatively high volatility, UF_6 is the only uranium compound suitable for performing enrichment. All current enrichment processes are based on the use of UF_6. The production process has the following stages:

— Reduction of UO_3 to UO_2;
— Conversion of UO_2 to UF_4;
— Fluorination of UF_4 to UF_6.

In carrying out these processes, conversion plants handle some very aggressive chemicals (F, HF). They do not, however, produce significant amounts of radioactive effluents (those they do produce principally contain natural uranium (beta activity)).

Typical arisings from the conversion processes are given in Table 2.

TABLE 2. TYPICAL ARISINGS FROM THE CONVERSION PROCESS (PER 1000 t U)

Arising	Quantity (t)	Classification	Comment
Solid CaF_2	10	Material for treatment	Fluidized bed process
Sludge CaF_2, $Ca(OH)_2$, H_2O with small amounts of U	20–50	Material for treatment	Wet process
Sludge CaF_2, $Ca(OH)_2$, H_2O without U	30	Non-radioactive waste	Wet process

2.2.1.3. Enrichment

Enrichment involves increasing the proportion of ^{235}U in UF_6 from the natural level of 0.7% to an average level of 3–5%. This is primarily done using one of two different industrial methods: gaseous diffusion or centrifugation.

Gaseous diffusion enrichment is based on the different diffusion rates of gaseous $^{235}UF_6$ and $^{238}UF_6$ through membranes. In the centrifuge process, enrichment is achieved by differential centrifugation.

Centrifuge and gaseous diffusion processes produce only very small quantities of waste. This is because enrichment plants handle a single process medium (UF_6) that is completely contained in a high integrity system throughout the operation. Since the processes are physical, not chemical, there are no auxiliary inflows of material or rejects of intermediate or waste products in the accepted sense. The small quantities of waste that do arise result from the light gas, which is passed through a small scrubbing system to ensure that only clean exhaust is released to the atmosphere.

Very small quantities of uranium (^{234}U, ^{235}U, ^{238}U) are vented to the atmosphere from gaseous diffusion plants. The radioactive discharges from centrifuge enrichment facilities are even smaller. For instance, the atmospheric releases from EURODIF in 1997 were 3.3 kg of uranium, with a total activity of 0.16 GBq. The liquid releases were only 0.29 kg uranium, with a total activity of 0.0094 GBq [11].

Enrichment of 1000 t of uranium in the form of UF_6 leads to the generation of around 850 t of depleted uranium with a ^{235}U content of approximately 0.2%. This material may be classified as a by-product or as a waste. Laser technology now under development on a laboratory or a pilot scale level could potentially reduce the content of ^{235}U in depleted uranium by at least factor of five and eliminate the need for highly toxic fluoride in the enrichment process.

2.2.1.4. *Fuel fabrication*

During the fuel fabrication stage, fuel is produced for loading into a nuclear reactor. Uranium dioxide and metallic uranium are two products that are commonly used as starting materials for fuel fabrication. Natural uranium is only used for the production of metallic uranium fuel. When uranium dioxide is used, it can be either natural or enriched.

At the fuel fabrication stage, there is the potential to produce a significant quantity of material scrap. Most of this scrap is not considered to be waste because of its significant value and because the majority of materials can be recycled within the process. Low and intermediate level waste from fuel fabrication includes filter media from wash water cleanup, waste oils, spent acids and bases, spent analytical solutions, decontamination and cleaning solutions, and discarded scrap metal and equipment. Any of this waste may be contaminated with hazardous chemicals and uranium. Plutonium contamination is present in facilities manufacturing MOX fuel.

Table 3 lists typical arisings at a fuel fabrication facility per 1000 t of uranium throughput. Many of these arisings can be minimized through reductions at the source and through recycling of valuable materials from waste streams and/or their reuse after appropriate cleaning and control.

2.2.1.5. *General considerations relating to the origin of waste generation from nuclear fuel facilities*

Waste material with only naturally occurring radionuclides is produced during uranium purification, enrichment and fuel fabrication, although at significantly higher concentrations than occur in nature. Consequently, there are real prospects for cleaning up many of the waste materials to bring activity levels below clearance levels, if such an approach is acceptable according to local regulations. However, procedures for clearance must be safe, economically justifiable and environmentally acceptable.

The following general causes of radioactive waste generation can be attributed to the majority of the front end processes of the nuclear fuel cycle:

— Unnecessary contact between inactive and active materials;
— Limited or no recovery of valuable materials from waste streams for recycling and/or reuse;
— Limited or no segregation to ensure that waste is always in the lowest possible category, which would facilitate opportunities for decontamination and disposal;

TABLE 3. TYPICAL ARISINGS FROM FUEL FABRICATION ROUTES (PER 1000 t U THROUGHPUT)

Arising	Quantity	Classification	Process/origin
Ammonium fluoride solution	4000 m^3	By-product	AUC
Ammonium nitrate solution	5000 m^3	By-product	ADU and AUC
Extraction residues	10 m^3	Material for treatment	ADU and AUC
Sludge	1 m^3	Material for treatment	ADU and AUC
Hydrogen fluoride	1000 t	By-product	IDR
Magnesium fluoride	450 t	By-product	Magnox
Graphite	300 t	Material for treatment	Magnox
Zircaloy	1 t	Material for treatment	Water reactor fuel
Stainless steel	1 t	Material for treatment	Gas cooled reactor
Miscellaneous metal scrap	40 t	Material for treatment	All
Ventilation filters	100–200 m^3	Material for treatment	All
Mixed combustible material	300 m^3	Material for treatment	All

Note: ADU: ammonium diuranate; AUC: ammonium uranyl carbonates; UOC: uranium ore concentrate.

— Limited or no application of decontamination techniques to allow recycling, reuse, sale as by-products or disposal as inactive waste.

Waste minimization or even the elimination of some waste streams may result from the application of advanced concepts, approaches and technologies on an industrial scale. This might include, inter alia, the use of dry processes that reduce the environmental impact by eliminating liquid waste streams [2].

2.2.2. Waste generation from operation of nuclear power plants

2.2.2.1. Types and quantities of material

The main process waste streams derived from nuclear power plant operation are:

— Sludge and fine particulates from aqueous precipitation and filtration of liquid radioactive waste;
— Spent ion exchange resins used for purification of process water;
— Evaporator concentrates;
— Miscellaneous dry solid waste.

Sludges are generally composed of hydroxides of iron, magnesium, calcium and aluminium, along with mineral based materials, and are generally of a low toxicity. However, some sludges may contain toxic residual components such as chromium, copper or nickel. The ion exchange resins are likely to include toxic and non-toxic metals, for example, Fe, Cu, Zn, Mn or B. The most common toxic material in evaporator concentrates is boric acid. In addition to boric acid, evaporation concentrates generally contain a mixture of fission and activation products.

Nuclear power plant maintenance and repair operations produce discarded equipment, organic solvents used for degreasing and cleaning, and organic complexing agents from decontamination activities. In addition to the organic compounds, the waste from nuclear power plants may contain metals such as lead, mercury and barium [12].

The power plant cooling tower water generally contains chromium or other chemical anti-fouling materials. Therefore, cooling water blowdown or associated filtrate sludge will also contain these chemicals. Any radionuclides present in the plant coolant system may also appear in the cooling tower water.

The majority of the dry solid waste (i.e. miscellaneous refuse and organic and inorganic rubble) is cellulose materials (e.g. paper, rags, clothing and wood), rubber gloves and boots, plastic, steel and building debris. Such waste would not usually be regarded as hazardous; however, it may contain trace amounts of toxic elements. Information on chemically toxic substances present in such miscellaneous refuse is seldom available. The toxic metal content can sometimes be inferred by analysing the incinerator ash after treatment of dry solid waste. While the polymers in plastic waste packaging generally are non-toxic, a variety of hazardous materials may be present as surface contamination, since plastic sheeting is frequently used for isolating areas during decontamination, or as a packaging material.

Lead is widely used in nuclear power plant operation for shielding as lead blankets, lead sheets and lead bricks, and is sometimes used as a liner material in radioactive waste containers. Lead blankets and bricks may become contaminated with radionuclides and hazardous chemicals. However, the lead can be separated for treatment, recycling or disposal, as appropriate.

Tables 4 and 5 provide further information on organic and inorganic compounds in nuclear power plant waste.

TABLE 4. NON-RADIOACTIVE TOXIC SUBSTANCES IN TYPICAL WASTE STREAMS AT NUCLEAR POWER PLANTS
(life cycle estimate)

Waste	Waste volume (m^3)	Toxic substance	Concentration ($kg \cdot m^{-3}$)	Mass of substance (kg)
Operational low level waste	200 000	Cd	—	4 000
		Hg	—	4 000
		Be	—	20 000
		Se	—	8 000
Evaporator concentrates	3 300	B	45	150 000
Ion exchange resins	1 000	Cr	0.5	500
		Ni	0.7	800

TABLE 5. NON-RADIOACTIVE ORGANIC CONTAMINANTS IN TYPICAL OPERATIONAL WASTE FROM NUCLEAR POWER PLANTS

Waste volume (m^3)	Solvent	Concentration ($kg \cdot m^{-3}$)
200 000	Acetone	0.2
	Dichlorobenzene	0.4
	Ethanol	0.2
	Isopropyl alcohol	0.07
	Methylethyketone	0.2
	Toluene	0.08
	Trichloroethane	0.08

As Table 5 shows, organic solvent concentrations are not high, ranging from about 50 to 500 ppm. Other solvents and organic contaminants may be present in measurable quantities in waste arising from non-routine nuclear power plant operations (e.g. chemical cleaning of the secondary side of steam generators).

Abnormal events at nuclear power plants can potentially result in appreciable volumes of radioactive waste with chemically hazardous constituents [13]. Abnormal events can include events such as unplanned major modifications, process upsets and accidents of various kinds. The waste arising from abnormal events may include large quantities of miscellaneous refuse contaminated with decontamination chemicals, process chemicals and cleaning solvents, and adsorbents for organic liquids (e.g. contaminated pump oils and hydraulic fluids).

2.2.2.2. Origin of waste from nuclear power plants: General considerations

Two types of process lead to the generation of radionuclides in nuclear power plants: (1) the fission process, and (2) the activation process. Their impact on the contamination of reactor coolant and on the activity level of waste generated at the nuclear power plant is described below.

Fission process: In nuclear reactors, fission of ^{235}U, ^{239}Pu and/or other fissile materials generates fission products and transplutonium elements in the fuel. Although many of these radionuclides decay quickly to form stable elements, significant amounts of longer lived radionuclides, such as ^{99}Tc, ^{90}Sr and ^{137}Cs, are also produced. Most of these fission products remain within fuel elements and are managed with the spent fuel. However, a small fraction of fuel elements (usually less than 0.1%) may contain fabrication defects or develop defects during operation, so some fission products may be released to the reactor coolant system. Also, if a small amount of uranium contamination exists on the outside of the fuel element, it will undergo fission and be swept into the coolant.

Activation process: The sources of radionuclide production due to neutron activation are corrosion products, atoms of the reactor coolant, coolant impurities and chemical additives. The corrosion products can enter the reactor coolant as already activated nuclides, as with the erosion products of an irradiated reactor vessel or material from the reactor's internal components; alternatively, the corrosion products can be activated after entering the coolant, as with erosion film particles of plant primary system materials. Reactor coolant radionuclides typically generated during activation of corrosion products are ^{51}Cr, ^{57}Mn, ^{59}Ni, ^{63}Ni, ^{58}Co, ^{60}Co, ^{65}Zn and ^{94}Nb. The concentration of these radionuclides for any given plant depends on the construction materials used, the chemical regime of the reactor coolant, the power level and age of the reactor, and the oxygen content in the reactor coolant.

The atoms of the water used as a coolant (H and O) can also be activated while passing through a reactor core. The major radionuclides created in this process are ^{13}N, ^{16}N, ^{17}N, ^{18}N, ^{19}O and ^{3}H. Except for ^{3}H, all of these radionuclides are short lived and do not contribute to the volume or the activity of the radioactive waste to be disposed of.

Coolant impurities are naturally occurring nuclides that remain after processing/purification of coolant water for use in a reactor. The content of impurities varies from site to site and is strongly influenced by the minerals present in a given location. Because requirements for reactor coolant purity are extremely high, activation of such impurities occurs only infrequently. Radionuclides typically generated in the process of impurity activation are ^{24}Na, ^{27}Mg, ^{45}Ca, ^{49}Ca, ^{31}Si, ^{37}S and ^{38}Cl.

Reactor coolant additives serve two major purposes: (i) to maintain the system activity at the required level and (ii) to keep the water chemistry within specified boundaries. For example, in pressurized water reactors (PWRs), system reactivity is controlled by boron in the form of boric acid (H_3BO_4), while water chemistry is maintained mainly by a lithium base (LiOH). Both lithium and boron can produce tritium (^3H) through neutron capture following a nuclear reaction. Because of its relatively long half-life (12.3 a), ^3H can accumulate in reactor coolant systems, resulting in an increase of dose rates in the vicinity of systems containing reactor coolant. To alleviate this problem, the reactor coolant is periodically diluted, which leads to an increase in the volume of waste generated at the nuclear power plant.

As a result of contamination of the reactor coolant with radionuclides generated during plant operation and successive contamination of plant systems and plant areas, several types of waste are generated (see Section 2.2.2.1). The amount of waste material generated depends on the design of the reactor system, the nuclear fuel used and how the plant is operated. While the reactor system and the type of nuclear fuel determine the quantity of radionuclides generated in the plant and the volume of generated waste, the volume of waste for final disposal is primarily determined by how the plant is operated.

Based on the above, the following are identified as having a major influence on the generation of radioactive waste at nuclear power plants:

— The materials selected for the reactor vessel and its internal components;
— The materials selected for systems, equipment and components that are in contact with the reactor coolant;
— The chemical regime of the reactor coolant;
— The additives in the reactor system and their quality;
— The quality of the fuel cladding material.

Factors that may contribute to a reduction of radioactive waste generation in nuclear power plants are:

— Leak tightness of equipment and systems;
— Recycling and reuse of liquids and other materials;
— Strict segregation of liquids according to their radioactivity, their chemical composition, and their solids and oxygen content;

— Processing and immobilization of waste;
— Arrangement of equipment within service areas for better serviceability and maintenance;
— Appropriate layout of rooms and plant areas to minimize contamination and consequent decontamination;
— Proper maintenance procedures and suitable materials used during maintenance of equipment in radiologically controlled areas.

Consideration of these factors at the design stage is discussed in Section 4.

2.2.3. Waste from spent fuel reprocessing

The composition of the stored waste from spent fuel reprocessing, both intermediate and high level, may be very complex and not known with complete certainty. Many countries that reprocess spent fuel are still developing a disposal strategy for much of this waste.

Low and intermediate level radioactive waste containing hazardous or toxic contaminants from reprocessing of spent fuel ranges from water treatment filters to activated metal components or metal contaminated by fission products. The separated fission product waste streams are normally acidic and may contain erosion or corrosion products from plant process equipment. Equipment cleaning for maintenance includes the use of degreasing agents, which may include halogenated solvents, and decontamination and metal cleaning agents, which often contain strong acids and oxidizing and complexing agents. Any of these waste streams may become contaminated with fission products, uranium and/or plutonium.

In some countries, only the raffinate resulting from the first stage separation of uranium and plutonium in fuel reprocessing is considered to be a high level waste. All other waste streams from liquid reprocessing — such as fuel cladding (which is not dissolved in the fuel dissolution step), maintenance wastes, discarded equipment, laboratory analytical equipment and solutions — are usually considered to be low or intermediate level radioactive waste.

Activated and highly contaminated metals derived from the cladding of spent nuclear fuel generally have low or moderate toxicity, since metallic materials are either of a low toxicity, such as magnesium and iron, or of a low solubility, such as zirconium and stainless steels.

Nitrate is a significant component of a number of reprocessing waste streams, because nitric acid is used for spent fuel dissolution. Spent solvents can arise from the solvent extraction processes used for chemical separations. The most commonly used extraction solvent is tributylphosphate (TBP). The TBP is diluted for the extraction process, usually with a light saturated hydrocarbon such as dodecane or a mixture of paraffin hydrocarbons. Chemicals that can arise in waste from spent fuel reprocessing in low and intermediate level radioactive waste are:

- TBP and other organic extractants;
- Nitric acid and alkali metal nitrates;
- Organic solvents;
- Complexing agents;
- Metals such as zirconium, chromium, nickel, iron, aluminium and their nitrates;
- Alkali metal fluorides/chlorides;
- Mercury contaminated scrap metals (processing equipment);
- Uranium contaminated metals;
- Full cladding mills.

Table 6 provides further information on some of the inorganic compounds that may be present in reprocessing wastes.

Other factors that may contribute to a reduction of the generation of radioactive waste at reprocessing facilities are:

- Selection of appropriate materials for those components that are in contact with chemical solutions;
- Leak tightness of equipment and systems;
- Recycling and reuse of liquids and other materials;
- Strict segregation of liquids according to their radioactivity and their chemical composition;
- Processing and immobilization of waste;
- Arrangement of equipment and areas for better serviceability and maintenance of equipment;

TABLE 6. EXAMPLES OF ARISINGS FROM REPROCESSING FACILITIES

Waste stream	Volume (m^3)	Substance	Concentration ($kg \cdot m^{-3}$)	Total mass (kg)
Pu finishing low level waste	5 500	Be	3.3	18 000
Operational waste	86 000	Ni	1.3	110 000
Vitrification plant low level waste	26 000	Zn	1.6	41 000
Soil and rubble	69 000	Asbestos	26	1 800 000

— Appropriate layout of rooms and plant areas to minimize contamination and consequent decontamination;
— Optimized performance of plant components and equipment to minimize the downtime associated with replacement of worn components and the volume of waste from maintenance;
— Proper maintenance procedures and suitable materials used during maintenance of equipment in radiologically controlled areas.

2.2.4. Waste from decontamination and decommissioning of nuclear facilities

In general, the decommissioning of nuclear facilities, such as nuclear power plants, fuel fabrication and reprocessing plants, research facilities and other nuclear installations, results in large amounts of waste, including [14]:

— Concrete, bricks and other construction materials;
— Mild steel found in plant structures (e.g. rebar, equipment supports);
— Stainless steel, most commonly encountered in the construction of reprocessing plants;
— Aluminium, used in bulk in some processes, such as enrichment.

Some of the material arising from specific activities will be radioactive as a result of activation and/or contamination. However, a large proportion of the arisings will be inactive, which means that they may qualify for clearance.

In reactor facilities, most of the activated material is contained within the reactor vessel and its internal components, as well as in the biological shield surrounding the reactor vessel. Typically, these components contain materials such as steel, aluminium, reinforced concrete, graphite and zirconium alloys. The radionuclides associated with the activated materials can be calculated using analytical techniques [15–17].

The process equipment and components used to contain the process material, whether it is reactor coolant or reprocessing liquids, may become contaminated with fission products, activation products and/or transuranic isotopes. Other parts of the facility may be contaminated if there are any liquid, gaseous or particulate leaks.

Radioactively contaminated liquids can also arise from the decommissioning of a facility, for example, liquid waste arising from the decontamination or flushing of systems. The types of radioactive contaminant in the liquid depend on the type of facility being decommissioned and on the exact location in the plant and point in the process where the waste stream is generated.

Inactive solid materials and liquids also arise from the decommissioning of nuclear facilities. If appropriate segregation and decontamination processes are utilized, the volume of radioactive materials requiring treatment can be reduced significantly. Non-radioactive solid materials typically include items such as piping, pumps, tanks, duct work, and structural and electrical equipment. Inactive liquids and solid materials can be disposed of in accordance with applicable regulations and using conventional methods.

An accurate estimate of the volume of contaminated materials (activated, contaminated, alpha bearing versus non-alpha bearing) in relation to the methods and processes available for their treatment requires [18]:

— Classifying facility systems and structures with respect to activity (e.g. activated, contaminated, non-contaminated, alpha bearing versus non-alpha bearing).
— Characterizing the type of material that will be generated as well as its further treatment, handling, packaging and disposal requirements.
— Developing a detailed mass/volume inventory of facility systems and structures.
— Defining the quantities and volumes of materials that can be decontaminated and/or measured in view of clearance or recycling and reuse, including items generated during decommissioning.
— Defining the quantities and volumes of compactible and incinerable contaminated solid materials, including items generated during decommissioning.
— Defining the quantities and volumes of contaminated solid materials that cannot be processed further (neither compactible nor incinerable). As this category of materials can have a large impact on the technical equipment required for handling and conditioning, an accurate determination is necessary.
— Defining the volume of contaminated liquids. The volumes generated during decontamination and flushing operations will largely depend on the type of facility and the representative contaminants, the number of decontamination steps and their efficiency.
— Defining gaseous effluents and aerosols. Aerosols containing finely divided radioactive material are usually the result of cutting and abrasive surface cleaning methods. Some cutting and cleaning methods produce large volumes of toxic smoke and fumes. Filters and other cleaning devices in the ventilation systems used for contamination control must be adequate to collect and retain the particulate material and the other contaminants to minimize gaseous effluents release.

Table 7 lists the contaminated material generated from the decommissioning of an actual 250 MW(e) natural uranium graphite gas reactor [19], a 900–1300 MW(e) PWR [19] and a reference reprocessing plant with a capacity of 5 Mg/d [20].

The activity level of most of these materials is usually low. To a large extent, these materials will qualify for unconditional clearance after cleaning and/or adequate decontamination to the required clearance levels. Some quantities of tritiated water vapour may also arise during decommissioning operations. If necessary, it is possible to remove the tritiated water vapour from the ventilated air [21].

The quantity of radioactivity decreases with time after plant shutdown owing to the process of radioactive decay. As such, deliberately delaying the decommissioning and demolition of a plant, or conducting these activities in stages, will result in a decrease of the radioactive inventory over time by significantly reducing the quantities of material with higher radioactivity levels. Relevant calculations have been done and estimates are available for plants with various types of reactor.

TABLE 7. RADIOACTIVE MATERIAL GENERATION (Mg) FROM THE COMPLETE DECOMMISSIONING OF A REPRESENTATIVE NUCLEAR FACILITY

Radioactive material generated	250 MW(e) natural uranium graphite gas reactor	900–1300 MW(e) PWR	Reprocessing plant (5 Mg/d throughput)
Irradiated carbon steel	3000	—	—
Activated steel	—	650	—
Graphite	2500	—	—
Activated concrete	600	300	—
Contaminated ferritic steel	6000	2400	—
Steel likely to be contaminated	—	1100	3400
Contaminated concrete	150	600	1850
Contaminated lagging	150	150	400
Contaminated technological wastes	—	1000	300

As an example, Table 8 provides an estimate for a 1000 MW(e) PWR, giving the approximate masses and activities of steel from the active areas at various times post-shutdown. The table shows the decreasing proportion of beta–gamma emitters in low level radioactive steel as time progresses, which results from the decay of radionuclides such as ^{60}Co [22]. The table also shows the progressive reduction in the quantities of steel with radioactivity levels above 0.1 Bq/g (0.37 Bq/cm^2).

Calculations show that, between 5 and 25 a post-shutdown, the mass of steel contaminated to levels higher than 0.1 Bq/g (0.37 Bq/cm^2) decreases to 50% of the initial values. After 100 a, this proportion decreases to about 25% of the initial values.

From the radiological point of view, waste from power and experimental reactors can be divided into two broad groups:

— The constituent materials of the reactor (with the pressure vessel and its internal structure) and the biological shielding, which are primarily activated (as opposed to contaminated) and account for more than 90% of the total activity in the plant;
— The complete coolant circuits and secondary installations, which are primarily contaminated (as opposed to activated).

Fuel cycle installations, and in particular reprocessing plants, are usually contaminated by alpha emitters and fission products. For this reason there is little to justify delaying their decommissioning or demolition, as even after

TABLE 8. TYPICAL MASSES AND ACTIVITIES OF STEEL FROM A 1000 MW(e) PWR

		Time after reactor shutdown					
		5 a		25 a		100 a	
Surface activity (Bq/cm^2)	Average activity concentration (Bq/g)	Mass (Mg)	Total activity[a] (Bq)	Mass (Mg)	Total activity[b] (Bq)	Mass (Mg)	Total activity[c] (Bq)
37–370	10	800	8.0×10^9	440	4.4×10^9	240	2.4×10^9
3.7–37	1	1600	1.6×10^9	880	8.8×10^8	480	4.8×10^8
0.37–3.7	0.1	3200	3.2×10^8	1760	1.8×10^8	960	9.6×10^7

[a] 99.9% beta–gamma, 0.1% alpha.
[b] 99% beta–gamma, 1% alpha.
[c] 95% beta–gamma, 5% alpha.

several decades the resulting radioactive decay is not of significant benefit with respect to worker protection, radioactive material management or potential minimization of waste arising from the decommissioning of such facilities. In this case, the radioactive material is contained partly by dynamic sealing, which means that the ventilation systems must be kept running at all times. Also, a risk of corrosion from the chemicals used during operations remains. As a result, early dismantling is a desirable approach, since the annual cost of shutdown maintenance and surveillance can be substantial, leading to great expenditure with minimal benefit [23].

Because the situations of individual plants can vary greatly, general methodologies for making strategic or technical decisions concerning the decommissioning of nuclear facilities are not always appropriate. Analysis for decision making needs to be performed for specific decommissioning options and must be based on the results of individual evaluations from the operating period and the existing conditions of a specific installation. Approaches to waste management and waste minimization are important elements of these decisions.

An estimate of the quantity of decommissioning waste arising from different reactor types is provided in Ref. [24] and shown in Fig. 2. The data are presented in terms of the weight of radioactive material per reactor (or per unit) and provide an estimated maximum–minimum range for each reactor type, and thus give an indication of trends rather than the final values.

As shown in the figure, gas cooled reactors generate the greatest quantity of radioactive material during decommissioning. The variability of quantities within each reactor type may reflect differences in the extent of reactor support facilities and equipment, rather than in the reactor materials themselves.

The main areas that have been identified as offering opportunities for reducing the generation of radioactive waste during the decommissioning of nuclear facilities are:

— Areas identified for the various individual facilities, as mentioned above;
— Recycling and reuse of the materials produced;
— Strict segregation of materials and waste according to their radioactivity and their chemical and physical composition;
— Proper decommissioning and maintenance procedures, and the use of suitable materials during maintenance of equipment in radiologically controlled areas.

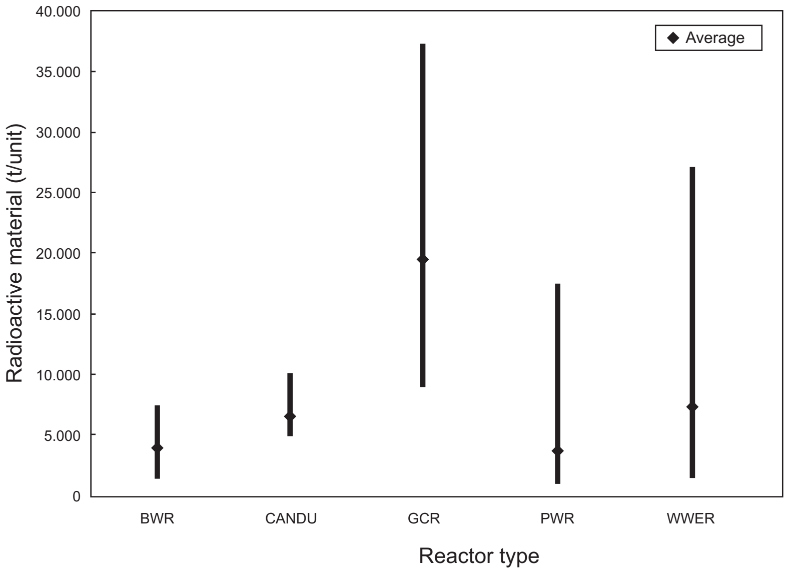

FIG. 2. Estimated maximum and minimum total weight of radioactive material by reactor type.

2.2.5. Waste from institutional and industrial facilities, including research laboratories

Institutional and industrial low and intermediate level radioactive waste varies widely in its composition [12]. Much of the waste is generated in small quantities from experiments or operations that change over time, and thus can have unique characteristics. As a result, it is difficult to define ‘typical’ waste streams for these processes, with the exception of scintillation liquids generated during the measurement of low level alpha and beta emitting radioisotopes. Historically, scintillation liquids have contained benzene, toluene and/or xylene. Some less toxic solvents have come into use in recent years [12]. Sources of institutional and industrial low and intermediate level radioactive waste include research activities, research reactor operation, and biomedical and industrial application of radionuclides [25].

Waste from research activities can contain a very large spectrum of organic and inorganic chemical compounds. Inorganic material may include toxic heavy metals, reactive metal salts, strong oxidizers, solutions of corrosive acids and bases.

The amount of hazardous constituents in waste arising from research reactor operation depends mostly on the irradiation programme of the particular reactor (activation analysis, irradiation of biomedical materials, etc.). The waste includes radioisotopes produced by activation analysis and nuclides used in tracer experiments. Because of the wide range of research activities, many different hazardous materials may appear in this waste. Associated organic liquids include oils, metal cleaning solvents with halogenated organics, flammable organic solvents, alcohols, aldehydes and possibly toxic materials such as organic phosphorus compounds. Various types of organic liquid waste have been described in more detail in other IAEA publications [26]. Metal waste may include lead shielding, mercury, discarded equipment, glove boxes and laboratory analysis apparatuses.

Liquid radioactive waste is generated during research reactor operations, isotope production and application of radioisotopes (e.g. medicine, research, education). The types of waste produced depend on the particular operation being conducted and can vary extensively in both chemical and radionuclide content. Most operations, particularly the larger ones, will also produce different types of radioactive liquid waste from activities such as showers, laundries, analytical laboratories and decontamination services [26].

Dry solid radioactive waste consists mainly of ‘general trash’, including protective clothing, plastic sheets and bags, rubber gloves, mats, overshoes, paper wipes, towels, metal and glass, hand tools and discarded equipment. This waste may also contain various process wastes from nuclear research centres, such as spent filter cartridges, spent resins and sludge from effluent treatment plants. Sealed radiation sources are a special category of radioactive material and are used in almost all fields. They are usually characterized, handled and disposed of in accordance with special regulations, policies and procedures.

A typical composition (by volume) of low activity dry solid radioactive waste generated by research centres is [27]:

— 70% compactible or combustible materials, subdivided into plastic (25%), paper and clothes (25%), small metallic or glass objects (15%) and miscellaneous (animal carcasses, wood, etc.) (5%);
— 20% heavy/hard materials (non-compactible) such as metal components, building materials and large items;
— 10% debris resulting from plant conversions and operational incidents (concrete, soil, etc.).

Radioactive waste arising from clinical, medical and biological research activities contains mainly short lived radionuclides. A notable exception is waste containing ^{14}C and ^{36}Cl. In addition, there is a potential for a wide range of spent sealed sources [27].

2.3. MIXED WASTE STREAMS

Some types of waste may raise specific issues relating to waste management and disposal options [28]. Although some of these materials may no longer be in common use in current nuclear facility design, they are considered here for completeness. These are:

— Highly heterogeneous wastes with divergent properties. Waste streams in this category are primarily those associated with facilities that have historically accepted waste from a number of different processes and sources. These streams are, therefore, highly heterogeneous and contain items with divergent properties. Such waste requires characterization facilities and, where necessary, segregation to enable production of packages with properties consistent with modern safety requirements.
— Materials where effective immobilization is difficult. For some types of waste, achieving immobilization has proved difficult. These include soft low density and/or absorbent waste such as plastics/cellulosics; waste with restricted access and/or low porosity, such as HEPA filters, filter beds and ion exchange columns; and waste that is wrapped or placed in containers, such as drummed vault waste and bagged waste items.
— Materials with inherent hazards, for example, reactive metals; waste containing pyrophoric materials such as uranium hydride, finely divided metals, sodium metal, etc.; waste containing accessible Wigner energy, such as low temperature irradiated graphite; and waste with high fissile content.
— Other specific materials and waste, including particular waste streams and derived waste forms like superplasticizers, chemical substances for effluent treatment (e.g. tetraphenyl phosphonium), toxic waste, and NORM and TENORM waste.

3. METHODOLOGIES FOR WASTE MINIMIZATION

3.1. OBJECTIVES OF WASTE MINIMIZATION

The concept of waste minimization is interpreted in various ways [2]. It is often taken to mean minimization of the total quantity (usually by volume but sometimes by mass) or of quantities of each individual waste stream. Waste minimization may or may not involve reduction of the total activity in the waste streams. In operational practice, it leads to a reduction of the total cost associated with waste processing, or of the cost of waste storage or disposal, which is of interest to the operator. On the other hand, the regulators are primarily concerned with minimization of activity and sometimes of the volume of waste for disposal, and hence of the potential environmental impact. Waste minimization usually requires a trade-off between the benefits accrued and the cost of achieving those benefits. In addition, implementation of a waste minimization strategy is always an optimization exercise that takes into consideration factors such as worker doses, the cost of recovering materials, the availability of disposal routes for specific types of waste, the quantities of waste generated in each category, and the duration and cost of interim storage of waste compared with the estimated ultimate disposal cost.

The objectives of waste minimization are to limit the generation and spread of radioactive contamination and the activation of materials and to reduce the volume of waste for storage and disposal, thereby limiting any consequent environmental impact, as well as the total costs associated with the management of such waste and of contaminated materials.

It is very important to conduct a periodic review of the effectiveness of the waste minimization programme in order to provide feedback and identify potential areas for improvement. Waste minimization assessments need to be made regularly to evaluate arisings, including all measures to prevent their generation.

3.2. WASTE MINIMIZATION OPTIONS

The main elements of a waste minimization strategy can be grouped into four areas:

— Source reduction;
— Prevention of activation of materials and spread of contamination;

— Recycling and reuse of valuable materials;
— Waste management optimization [4].

These areas define four fundamental principles to be considered when planning a nuclear facility. These principles can be summarized as follows:

— *Control of radioactive waste generation*: keep the generation of radioactive waste to the minimum possible or practicable level;
— *Prevention of activation and contamination*: minimize the spread of radioactivity leading to the creation of radioactive waste by containing it to the greatest extent possible;
— *Reuse and recycling of materials*: optimize the possibility of reusing and recycling valuable components from existing and potential waste streams;
— *Reduction of radioactive waste volumes*: minimize the amount of radioactive waste that has been created by applying adequate treatment technology.

3.2.1. Control of radioactive waste generation

Throughout the life cycle of a facility, the generation of radioactive waste is to be kept to the minimum practicable level in terms of both its activity and volume through appropriate design measures, facility operation and decommissioning practices. Source reduction, which is the most prominent component of a waste minimization strategy, involves the selection of appropriate processes and technologies, the selection of construction and operational materials, and the implementation of appropriate plant practices during operation and for modifications and improvements [1].

Significant savings during the operational life of a plant can be achieved if sufficient attention is paid to the waste minimization concept during the design stage. The financial implications of any proposals to minimize waste need to be examined to confirm the costs and benefits of each proposal [2]. Design features can help to reduce both the level of radioactivity within the plant and the quantity of waste generated.

One of the most important steps of a waste minimization programme is raising awareness of the need to minimize waste. Important operational components of waste minimization implementation lie in the education and training of employees, contamination control, quality control of materials, proper specification of the final product or product intermediates, process modification, etc.

Decommissioning is the final phase in the life cycle of a facility [16, 24, 29, 30]. It is advisable at the design stage to envisage the condition of the facility at the end of its useful life. Ideally, necessary features will be incorporated into the design to facilitate decommissioning and to minimize waste generation during decommissioning.

3.2.2. Prevention of activation/contamination

It is important to minimize the spread of radioactive contamination, which minimizes the creation of secondary waste and reduces the need for decontamination. All means of preventing contamination are to be used if they are economically justifiable and do not lead to additional complications in operation, maintenance and decommissioning [31–33].

Ideally, proper zoning of the facility will be considered at the design stage, as will provisions to avoid backward flow of contaminated material from areas of high activity to areas of low activity. During the design stage, consideration needs to be given to establishing working conditions that assist operators in performing their jobs; to contributing to the operational culture within the facility; and to minimizing radiation exposure, waste generation and the spread of contamination. These aims may be achieved by properly designing each working area and the instrumentation to ease operation and maintenance of equipment, etc.

Administrative controls, management initiatives and proper maintenance of documentation in facilities can contribute significantly to an adequate waste minimization strategy. An organizational structure needs to be established to ensure that the responsibilities for all aspects of contaminated material management are defined and that the best practices in waste minimization are encouraged. Provisions for the comprehensive education of operators are very important. Introductory courses and regular ‘refresher’ courses may provide a means to foster operator awareness of the need to keep the generation of waste to a minimum.

The generation of waste containing hazardous contaminants is undesirable but not always avoidable. An important waste management consideration is to minimize the generation of waste with toxic components. Adequate evaluation and selection of alternative materials can eliminate hazardous components in some waste streams. Selection of treatment processes and consideration of associated chemicals may help to avoid production of waste with chemically toxic constituents. The proper selection of decontamination processes can also serve to minimize secondary waste generation.

3.2.3. Reuse and recycling of materials

Reuse can be defined as the reutilization of materials for the original purpose in their original form or in a recovered state. Recycling is the utilization of valuable materials, tools and equipment for other than the original purposes, with or without treatment [7]. The reuse and recycle option is attractive during refurbishment and decommissioning of nuclear facilities, where large quantities of materials and equipment, and some buildings and sites, are released from any further regulatory control. The decision of whether or not to reuse and recycle components from nuclear facilities depends on many factors that are specific to a given facility or country. Implementation of reuse and recycling options requires the availability of suitable criteria, a suitable measurement methodology and suitable instrumentation.

Technologies for preparing materials for reuse and recycling are widely available [32, 34–36]. The main challenges, however, continue to be characterizing the material and establishing a coherent dialogue with legislators and authorities to gain acceptance of reuse and recycling of recovered materials. Additionally, there is a need to ensure public understanding and acceptance of the concept of exemption/clearance of such materials. An overview of some reuse and recycling applications in various countries can be found in Ref. [37].

3.2.4. Reduction of radioactive waste volumes

In addition to reducing the amount of radioactive waste generated at the source, it is also possible to minimize the volume of radioactive waste using appropriate treatment methods. The volume of radioactive waste may be reduced by use of volume reduction processes such as compaction, incineration, filtration and evaporation. These actions will extend the operating life of current disposal sites, limit the need for interim storage if disposal is not available and reduce the number of shipments of waste [38].

Proper characterization and segregation of waste are very important factors in waste processing. Characterization helps in developing a complete understanding of the physical, chemical and radiological characteristics of waste for sorting and shipping, either for selected processing or for storage/disposal. Sorting favours the maximization of unconditional clearance, reuse or recycling of materials and can reduce the volume of radioactive waste that does not meet clearance or recycling/reuse criteria. Proper sorting requires staff training and additional space and containers in the waste collection and storage areas.

The following information may be required as part of an appropriate waste characterization:

— Type of emitter (alpha, beta, gamma, X ray);
— Source of emission (loose or fixed contamination, induced radioactivity);
— Physical state and chemical composition;
— Geometry, surface area, level of radioactivity to be measured;
— Potential for interference from several sources of radiation.

Segregation of waste with different characteristics provides a number of advantages including:

— Simplified documentation requirements for shipment and handling;
— Simplified treatment and conditioning;
— Simplified disposal of waste segregated according to disposal waste categories and disposal acceptance criteria.

Measurement of initial and residual activity in materials after decontamination is important for characterization/segregation of materials and for providing proper control before clearance of materials [4]. Techniques for the measurement of radioactivity can be organized into three general groupings: direct measurement, indirect measurement and measurement by sampling.

Direct measurements are taken using a radiation detector positioned in near contact with the surface or object to be measured. Various detectors are available, and the choice of detector depends on a number of factors, including the type of radiation to be measured, the size and shape of the object or surface to be measured, and the anticipated level of radiation.

Indirect measurements are taken using a paper smear to swipe a surface area to assess whether loose contamination is present. The smear is evaluated for contamination by performing a near contact measurement of it and by using radiation detectors suitable for the range of radionuclides expected to be present.

When taking measurements by sampling, representative portions of materials are analysed by laboratory processes — that is, chemical separation and alpha, beta and gamma spectrometry — in order to identify the radionuclides present in the material and to determine their concentrations.

A considerable reduction of waste volumes can be achieved by appropriate treatment of primary radioactive waste. Radioactive waste needs to be treated according to the type of waste, the concentrations of radionuclides, and the requirements for waste storage and/or disposal [4]. The choice of treatment processes depends on a variety of parameters including physical, chemical and radiological properties of waste, the storage and disposal alternatives available, and economic considerations. Some typical volume reduction techniques are [5, 13, 39, 40]:

— Compaction and supercompaction;
— Size reduction;
— Incineration and thermal treatment;
— Vitrification;
— Non-thermal destruction technologies;
— Biological treatment.

4. DESIGN OPTION CONSIDERATIONS FOR WASTE MINIMIZATION

4.1. INTRODUCTION

Many of the nuclear facilities that have reached or are now approaching the end of their operational lifetime were not designed and constructed with enough consideration given to future decommissioning and waste minimization. Therefore, application of available dismantling techniques, proper characterization of facilities and equipment to be decontaminated, full scale application of recycling practices and implementation of adequate waste minimization approaches are not always easy to achieve at such facilities [4]. As a result of the lessons learned during the decommissioning of certain facilities, several difficult areas have been identified and options have been formulated for consideration when designing new facilities or plant modifications, or when defining future D&D operations, particularly with regard to minimizing radioactive waste arisings.

The requirements and costs of decommissioning are now better understood, and the nuclear industry is increasingly aware of the importance of including decommissioning considerations at the design stage of new nuclear facilities or as soon as possible for existing facilities. Decommissioning may be more difficult and costly if consideration is given to facility decommissioning only late in the facility's lifetime. The increased difficulty and cost may occur owing to a lack of adequate records and information, a need to install or modify equipment, or increased complexity of decommissioning activities. It may also be caused by the need to incur additional doses as a result of particular aspects of the design that complicate decommissioning activities [41].

The objectives during D&D activities are to reduce occupational exposure, to minimize waste generation and to simplify dismantling procedures, which will also result in cost savings. These objectives must not conflict with the primary objective of the facility, which is safe and efficient operation. However, some design and construction features to facilitate decommissioning may result in significant cost savings, especially if they also benefit plant operation and maintenance. Cost–benefit analysis and other optimization techniques can be of assistance in the selection of such features.

It must also be recognized that the provisions of some design features aimed at facilitating decommissioning can conflict with plant design and operation. For example, any extra equipment to facilitate D&D may reduce access spaces and may result in additional, possibly radioactive, waste [30].

In addition, there are regulatory/licensing requirements that demand consideration of decommissioning at the design stage, particularly with respect to minimizing waste arisings and facilitating access for dismantling. Moreover, in an increasingly environmentally conscious world, there is a need for continual assurance about the environmental impact of all aspects of nuclear power. Well established plans for decommissioning during the design stage will help to provide such assurance [6].

It is widely recognized that most of the features to be considered at the design stage to facilitate disassembly and removal of components for maintenance will also be beneficial for the decommissioning strategy. For example, selecting low activation materials may enable the use of less complex remotely operated equipment and may even facilitate hands-on intervention so that remotely operated equipment becomes unnecessary [30].

Although design and construction features that facilitate operation and maintenance are of higher priority, additional features designed specifically to facilitate decommissioning and minimize production of contaminated materials during D&D need to be incorporated. The goals in this context are to [19]:

— Limit the costs of maintenance and surveillance during waiting periods prior to dismantling;
— Minimize the additional facilities required to ensure safety and protection during the D&D operations;
— Reduce radiation exposure of the (decommissioning) workforce;
— Facilitate final dismantling, disassembly and cutting, and the associated operations of manipulation and handling;
— Reduce the costs of dismantling work and any additional equipment needed;
— Optimize the minimization of waste generation, reducing the amounts of radioactive material and effluents produced and making these compatible with the requirements for storage and transport;
— Completely dismantle installations and restore sites to the public domain.

Meeting these objectives requires that the design of structures and equipment consider the following:

— Activation of materials is to be limited as much as possible;
— Contamination of the plant and equipment is to be avoided as much as possible;
— Contaminated or active areas are to be easily separated from non-contaminated areas;

— Adequate space and access points are to be provided to allow the use of special tools and equipment for remote operation and handling, and to allow the installation of appropriate shielding;
— Plant and equipment items must be able to be easily dismantled, handled and transported, and adequate openings must be provided to allow for easy removal of components and materials from the active area;
— Equipment and buildings must be able to be easily decontaminated;
— Sampling and measurements taken for characterization during decommissioning must be facilitated;
— Remote monitoring of the radiation field in inaccessible areas is to be facilitated.

Moreover, it is necessary for design studies to incorporate an outline of the dismantling scenario. Although such a scenario will probably need to be reviewed as the technology develops and experience is gained, it will allow the proposal of upper limits on equipment size and weight to facilitate the handling of dismantled parts as well as provisions for the necessary access points, handling routes and lifting equipment. Most of these arrangements proposed for decommissioning will also facilitate operation of the nuclear installation, contribute to its safety and facilitate its maintenance. As such, additional capital expenditure is justifiable from the point of view of reducing both occupational exposure and the human resources involved.

In addition, from the design stage on, the managed system ideally will ensure that drawings and other relevant documents are issued, controlled and archived in such a way that all documentation is fully up-to-date when decommissioning starts.

With the experience in decommissioning gained to date, construction and operation methods can be outlined that aim at facilitating the decommissioning of future installations, with reference to three kinds of decommissioning activity: materials management, contamination and decontamination, and dismantling operations such as cutting, handling and transfer of materials.

4.2. CONSIDERATIONS FOR MINIMIZING CONTAMINATION PROBLEMS

A well-designed facility will incorporate features that minimize contamination problems arising during operation, the safe enclosure period and decommissioning [5, 6]. These features will take into account safe and efficient operation as the primary objective of the facility. Designs and techniques aimed at improving operation and maintenance will be beneficial during decommissioning. A variety of design features and techniques are available to reduce or prevent contamination of components and minimize associated problems.

4.2.1. Building and equipment layout

Building layout plays an important role in minimizing the waste generated during the operation and decommissioning of a plant. First, the process to be carried out in the plant needs to be broken down into sub-processes grouped according to their safety and radiological impacts. The plant layout needs to be zoned based on the radioactivity and contamination levels. To avoid cross-contamination, it is desirable to locate inactive services/equipment in a separate building or buildings. Routes for the movement of personnel/material need to be planned so that cross-contamination is avoided and necessary check points for decontamination are included. The access doors and openings need to be sized to reduce the amount of dismantling work on the equipment. Ideally, the plant layout will be optimized to give a compact design. National regulations must be met while designing the buildings. Some areas that require further consideration are identified in the following sections.

4.2.1.1. Layout of building

Optimization of the plant design, layout and access routes will facilitate [42]:

— Cell layout amenable to remote inspection;
— Segregation of process equipment based on radioactivity levels and unit operation;
— Access for removal of large components;
— Detachment and remote removal of significantly activated components;
— Future installation of decontamination and waste handling equipment;
— Decontamination or removal of embedded components such as pipes and drains;
— Control of radioactive material within the installation.

Considerable information on these topics is available in the literature, and examples of particular difficulties experienced during decommissioning may be especially useful [6].

In general, locating a plant with other nuclear fuel cycle facilities would offer some advantages with respect to decommissioning. Sharing the costs of services, utilities, surveillance and maintenance during the safe enclosure period following protective storage activities, combined with the greater flexibility and availability of on-site staff, will reduce the overall cost of and time required for decommissioning [6].

Ideally, plant layout will facilitate radiation and contamination control during operation and maintenance. The main objectives of this control are to minimize the radiation dose by appropriate segregation and shielding, and the spread of contamination by appropriate measures such as containment, zoning, ventilation and active drains. It is preferable that buildings for storage of radioactive waste be situated above the groundwater level, and they must not be located in flood plains. A subsurface facility must be designed and constructed with appropriate systems to protect against in-leakage of groundwater (i.e. the facility needs to be leaktight). A layout that enables controlled access during normal operation and decommissioning will facilitate decommissioning [30]. Boreholes around the facility need to be provided for detection of any subsoil contamination.

Special consideration must be given to the layout and location of plant equipment and components that are likely to become contaminated and/or activated during operation. These provisions will facilitate access to and removal of equipment during decommissioning. In the layout and construction, plant and personnel access routes are provided either as permanent operational and maintenance routes or as temporary construction routes. Ideally, this approach will be integrated with access and hatch opening requirements (sealed for reactor operation) to allow removal of the largest equipment either intact or with minimal segmentation prior to the dismantling/demolition of the main building structure. Therefore, access provisions for the replacement of large components during the plant lifetime could also be beneficial for decommissioning [30].

4.2.1.2. Layout of ventilation systems

Normal design practice for nuclear facilities separates active and inactive areas to make decommissioning easier. In addition, building ventilation systems are designed to move air from inactive to active areas to reduce the amount and severity of contamination of inactive areas.

Primary confinement barriers are to be provided between process material and any auxiliary systems (e.g. a cooling system) to minimize the risk of material transfer to an unsafe location or the introduction of an undesirable medium into the process area (cross-contamination). Differential pressure across the barrier(s) is to be used wherever appropriate. Each confinement barrier should be checked analytically against challenges to ensure that it will be able to withstand them without loss of function. This applies to any form of the hazardous material (gas, liquid or solid) and its carrying medium (air or liquid). To reduce migration of contamination, anti-backflow devices along with closure devices or permanent seals are to be provided on entrances to and exits from piping, ducts and conduits penetrating confinement barriers. More detailed overviews with descriptions and designs of confinement systems can be found in the literature [43].

Proper design of plant ventilation systems will help to maintain contamination control during plant operation and can potentially reduce the extent of decontamination required and related waste arisings at the time of decommissioning [30].

When designing ventilation systems, it is important that consideration be given to the requirements for ventilation during decommissioning operations, such as the need for increased or more frequent air changes and/or the use of local ventilation units. These provisions need to be compatible with the main ventilation requirements for operation. The design must also consider the provision of ventilation by natural circulation during safe enclosure with minimum surveillance requirements.

Deposition of radioactivity on the inner surface of the ventilation ducts during normal operation and during anticipated operational occurrences requires special attention. The installation of upstream filters in the supply air system would minimize the potential for deposition.

Isolation of sections of the process area ventilation system is often required for confinement of radioactivity during selected decommissioning activities. Decommissioning could be expedited by providing for changes to the ventilation system, especially in areas where the ventilation equipment is not readily accessible [6].

Ducts for cell ventilation must be made of corrosion resistant materials appropriate for the process conditions envisaged. For example, stainless steel is an appropriate choice of material for chemical reprocessing cells; the supply duct should be installed in the lower part and the exhaust in the upper part of the cells. It is advisable to put filters and anti-backflow devices in the ventilation system at points where the air is transferred from areas of lower activity to areas of higher activity to prevent contamination of ventilation ducts. In addition, the design of ventilation ducts ideally will have provisions for their cleaning. The configuration of the ventilation ducting must also allow sufficient access to the duct accessories and to the walls, ceiling and floor of the areas around the ducts.

Compartmentalization of process functions, with comparable unit operations and radioactivity levels in each compartment (i.e. process cell), is desirable for decommissioning. With such design features, a high radioactive contamination level in one set of process equipment does not affect the decommissioning of equipment with lower contamination levels. A potential disadvantage of compartmentalization is that having more walls increases the amount of surface area and piping [6].

4.2.1.3. Layout of piping

Reducing the amount of piping in primary and auxiliary systems and minimizing the number of valves facilitate plant operation and decommissioning. Pipe penetrations through walls, floors or ceilings need to be as straightforward as possible. Shielding to limit radiation shine paths should use lead, cast iron or heavy concrete blocks and should take into account decommissioning needs. Experience has shown that piping arrangements where pipes are embedded in concrete can make dismantling and contamination monitoring very difficult, especially for small diameter pipes or curved penetrations. Direct embedment of pipes, ducts or conduits in concrete is to be avoided. Rather, these are to be routed through metallic jackets embedded in concrete. Such embedment preferably will be of a simple construction that can be filled up with lead or an aggregate. Penetrations in the cell floor or roof are to be avoided.

Routing of pipelines carrying primary coolant and other process liquids must not create dead zones and low velocity areas. This approach will help to prevent deposition of crud in these systems [30].

It is advisable that the piping be routed above ground to the greatest extent possible. Piping that must be routed below ground requires two containment barriers — for example, waterproof trenches with sumps and inspection features — to prevent subsoil contamination in case of pipe leakage. Proper sloping of trench flooring will ensure passive drainage of any leakage to sumps. Chemical and mechanical decontamination of stainless steel liners may be difficult and cumbersome, particularly where pipe trenches are small enough to make human access difficult. Alternative approaches would be to use a larger trench with a liner or to leave the trench unlined but to place each pipe or a group of pipes inside a larger pipe for double containment. The latter, in combination with built-in means to section and extract the pipes from one or both ends of the trench, would expedite the decommissioning of these areas and reduce costs and occupational exposure [6].

It is desirable that thermal insulation around piping and equipment be easy to remove and designed not to absorb any liquid spillages to minimize accumulation of contamination. One of the recommended methods is to use mineral insulation enclosed in metallic jackets.

To reduce occupational exposure during maintenance, repairs and/or decommissioning, the piping layout must not pose undue obstacles to manned or robotic access to contaminated areas.

It is advisable that equipment and piping containing radioactive liquids, and especially equipment and piping having components that may leak or that require frequent maintenance, not be located in rooms/cubicles with drains leading to active collection tanks. Drip trays with connections to drains are to be placed under items that might leak, but these trays must not be located in special rooms. Collection systems that rely on drip trays must be designed to avoid stagnation or spillage of radioactive liquids.

4.2.1.4. Design of storage tanks and equipment

Pools and storage ponds are to be enclosed and equipped with stainless steel liners. It is advisable that stainless steel also be used for inserts, storage racks and baskets. Ideally, floors will facilitate the elimination of dust and fines and will be sloped towards a sump, from which the water can be routed to a water purification system. Mechanical filters are to be placed as close to the suction point as possible. A water skimming/cleaning system is highly advisable.

Piping penetrations through pool walls are to be minimized. Where such penetrations are necessary, it is advisable that conduits, sleeves or connectors be used to permit ready decommissioning of the associated systems without massive damage to pool structures.

Internal structures in tanks and equipment are to be avoided (except where necessary, as with the decontamination nozzle, fluid transfer jets, etc.), as these increase areas of potential and persistent contamination. Whenever possible, heating and cooling circuits, thermocouples and other measurement and control items are to be installed on the outside of the equipment. If this is not practicable, the design should ensure either that there is no connection to the bottom of the tank or to the equipment, or that the items are positioned in such a way that 'dead' zones or deposits do not occur. If necessary, features can be provided to allow material contents to be agitated in order to keep any solids suspended.

Ideally, there will also be provisions for installing deslugging devices. Where deposition of solids may occur, built-in jet nozzles or connections through which mobile jet lances can be introduced into tanks need to be provided to enable the removal of solids or residues from tank wall surfaces by using, for example, high pressure water jets. Equipment and tank design has to allow complete clear-out of the content.

Special consideration is required in the design of storage tanks for high and intermediate level liquid waste, where cooling is needed for heat removal during decay and where decommissioning has been found to be one of the most difficult tasks. Consequently, from the standpoint of decommissioning, interim storage of this liquid waste is undesirable. Such storage can be eliminated by waste solidification as it is generated in the mainline processes [6].

Alternatively, if such waste storage is incorporated into the plant design, the following needs to be taken into account:

— Waste storage tanks need to be located such that ready access is provided to the entire area above the tanks. This access could be obtained by locating the tanks below a canyon type structure.
— Within the tanks, means are to be provided for inserting nozzles for chemical decontamination sprays into as many locations as possible.
— Means are to be provided to remove all liquid in the tanks except for liquid films clinging to surfaces.
— A system for waste heat removal that eliminates or reduces the amount of piping inside the tanks is to be provided.
— The waste solidification process equipment is to be located in the main process building in remotely operated cells to provide for more effective use of existing service facilities. Waste solidification off-gas treatment equipment can be eliminated by routing off-gases to systems with comparable capabilities in the main process building, thus reducing the amount of process equipment and process cell space requiring decommissioning.
— The amount of irregular surface areas on the outside of the tanks needs be minimized (e.g. elimination of stiffener ribs on the sides and top, and the numerous support ribs under the tank bottom), leading to more effective chemical decontamination.

4.2.1.5. Additional opportunities for improving layout of building and services

Proper design of electrical distribution and lighting will keep most of the equipment in inactive areas. Equipment and/or lighting installed in an active area must be amenable to decontamination. Cables in active areas are to be laid in a sealed duct. Provisions for isolated low voltage supply are desirable for dismantling work in the cells. Provisions for undertaking remote inspection and monitoring of radiation fields in cells are also very helpful.

The means to maintain and manipulate equipment within a highly radioactive process cell without requiring entry by personnel is also useful for decommissioning. This capability reduces occupational exposure and requirements for decontamination before the other activities are undertaken [6]. Ideally, provisions will be made for a ready room near or within the process area for maintenance, operating and monitoring personnel. Such a room is to be located in an area with low background radiation [43].

In addition, tools required for interventions need to be available, as do workshops for equipment repairs and adaptation of contaminated tools.

Equipment requiring periodic inspection, maintenance and testing is to be located in areas with the lowest possible radiation and contamination levels. It is advisable to provide for in-place maintenance and removal to an area of low radiation for repair of equipment that may become contaminated during operation. Maintenance areas for repair of contaminated equipment need to provide for containment or confinement of radioactive material [43].

The capacity to treat, handle and package low level waste may not be required during production operations. However, provisions enabling such capabilities (e.g. volume reduction of combustible or compactable wastes, sectioning or compaction of removed materials and electropolishing for decontamination of contaminated metallic equipment [6]) during decommissioning are highly desirable. An alternative is to utilize mobile units with such capabilities.

4.2.2. Selection of components

The components of the system must be reliable and robust, as this will result in future minimization of waste generation. The construction material for components needs to be selected so as to avoid activation. Major considerations are discussed in the following subsections.

4.2.2.1. General considerations

Activation and contamination problems can be significantly reduced by judicious selection of materials and appropriate design of components [29]. Therefore, the selection of suitable materials for areas subject to neutron activation plays an important role in design, operation, maintenance and decommissioning, including any safe enclosure period. The best method is to select construction materials that contain only small amounts of elements susceptible to activation. Ideally, any candidate material will fulfil the following basic requirements to the greatest extent possible [30]:

— Low activation properties;
— High resistance to temperature and radiation fields;
— Good corrosion and erosion resistance;
— High resistance to nuclear heating effects;
— Ability to withstand static and dynamic loadings;
— Good machinability, chemical stability and structural integrity.

The overall activity at the time of dismantling is a result of the activation of irradiated structures, production, deposition and activation of crud, and leakage of fission products from fuel assemblies into the primary circuit. The timing, methodology and dose commitment associated with dismantling activities can depend on the inventory of the radionuclides in the plant at the end of its life [30]. Table 9 provides a list of important radionuclides in the context of decommissioning.

Some of these radionuclides can be more significant for decommissioning than for plant operation; for example, long lived radionuclides may become significant dose contributors during a safe enclosure period. Typical concentrations of elements of specific construction materials liable to undergo activation are widely available in the technical literature [44–46]. The radiological effect on decommissioning may be dominated by different radionuclides, depending on the timing of the actual dismantling. For example, ^{60}Co is usually the predominant radionuclide shortly after final shutdown and for several decades thereafter. However, after several decades ^{94}Nb and ^{108m}Ag may become predominant. The effects of these radionuclides can be significantly reduced by

TABLE 9. IMPORTANT RADIONUCLIDES IN THE CONTEXT OF DECOMMISSIONING

Product	Half-life[a]	Reaction	Percentage natural abundance or half-life
^{60}Co	5.27 a (10.48 m)	^{59}Co (n.γ) ^{60}Ni (n, p) ^{61}Ni (n, np) ^{63}Cu (n, α)	100 26.1 1.13 69.2
^{59}Ni	7.5×10^4 a	^{58}Ni (n.γ) ^{60}Ni (n, 2n)	68.3 26.1
^{63}Ni	100.1 a	^{62}Ni (n, γ) ^{64}Ni (n, 2n) ^{63}Cu (n, p) ^{64}Zn (n, 2p) ^{66}Zn (n, α)	3.6 0.91 69.2 48.6 27.9
^{91}Nb	680 a (62 d)	^{92}Nb (n, 2n) ^{92}Mo (n, np) ^{92}Mo (n, nα)	3.5×10^6 a 14.8 14.8
^{94}Nb	2.03×10^4 a (6.26 m)	^{93}Nb (n, γ) ^{95}Nb (n, 2n) ^{94}Mo (n, p) ^{95}Mo (n, np)	100 35 9.3 15.9

[a] Figures in parentheses refer to the half-life of the meta-stable form of the product isotope.

TABLE 9. IMPORTANT RADIONUCLIDES IN THE CONTEXT OF DECOMMISSIONING (cont.)

Product	Half-life[a]	Reaction	Percentage natural abundance or half-life
^{93}Mo	4×10^3 a	^{92}Mo (n, γ)	14.8
	(6.9 h)	^{94}Mo (n, 2n)	9.3
		^{95}Mo (n, 3n)	15.9
^{99}Mo	65.94 h	^{98}Mo (n, γ)	24.1
(^{99}Tc)	2.111×10^5 a	^{100}Mo (n, 2n)	9.6
^{108m}Ag	418 a	^{107}Ag (n, γ)	51.8
		^{109}Ag (n, 2n)	48.2
^{55}Fe	2.7 a	^{54}Fe (n, γ)	5.8
		^{56}Fe (n, 2n)	91.8
		^{58}Ni (n, α)	68.27
^{14}C	5726 a	^{14}N (n, p)	99.64
		^{17}O (n, α)	0.037
		^{13}C (n, γ)	1.11
^{10}Be	1.589×10^6 a	^{13}C (n, α)	1.11
		^{9}Be (n, γ)	100
^{107}Pd	6.5×10^6 a	^{106}Pd (n, γ)	95.548
	(21.9 s)	^{106}Pd (n, 2n)	100
		^{110}Cd (n, α)	12.4
		^{107}Ag (n, p)	51.83
^{150m}Eu	35.8 a	^{151}Eu (n, 2n)	47.8
^{152}Eu	13.54 a	^{151}Eu (n, γ)	47.8
	(96 m)	^{153}Eu (n, 2n)	52.2
^{154}Eu	8.6 a	^{153}Eu (n, γ)	52.2
^{158}Tb	180 a	^{159}Tb (n, 2n)	100
	(10.5 s)	^{158}Dy (n, p)	0.1
^{186m}Re	2×10^5 a	^{185}Re (n, γ)	37.4
		^{187}Re (n, 2n)	62.6
^{93m}Nb	16.4 a	^{94}Nb (n, n')	100
^{93}Zr	1.501×10^6 a	^{93}Nb (n, p)	100
		^{92}Zr (n, γ)	17.1
^{36}Cl	3×10^5 a	^{35}Cl (n, γ)	75.5
^{3}H	12.33 a	^{7}Li (n, α)	92.5
^{41}Ca	10^5 a	^{40}Ca (n, γ)	96.9

[a] Figures in parentheses refer to the half-life of the meta-stable form of the product isotope.

paying careful attention to the specifications of materials used in the construction of those reactor components that will become neutron activated. To achieve this, it is necessary to identify elements known to become activated to form short or long lived daughters during the design stage, and to control their levels as appropriate for operational and decommissioning purposes. Another important radionuclide in the long term is ^{63}Ni. However, as ^{63}Ni is a low energy pure beta emitter, it is less hazardous during dismantling. Also, it cannot be easily reduced, as nickel is a main component of stainless steel.

The composition of materials likely to be exposed to the neutron flux needs to be checked carefully, with particular attention being paid to impurities that could produce significant quantities of neutron induced radioactivity [30]. It is recognized that it is not feasible or economically viable to reduce the levels of various impurities to zero or to insignificant values; however, as a result of optimization procedures, low levels of these impurities can be achieved. Impurity control of materials facilitates decommissioning by reducing radiation levels for access and easing waste management. This is in addition to the benefits associated with maintenance and inspection activities during the lifetime of the plant.

4.2.2.2. Selection of metal component

From an activation viewpoint, stainless steel is not a preferred material for use in high neutron flux areas because of its significant cobalt concentration [30]. For operational reasons, stainless steel will continue to be used as cladding or as bulk material to achieve extended life in reactor vessels. In practice, stainless steel internals can be handled in a way that leads to very little radiation exposure during decommissioning.

The amount of contamination resulting from the activation of trace elements in core components and corrosion products in the primary coolant of a reactor can be minimized by reducing the levels of these elements. For example, stable ^{59}Co is present as a hardening agent in the alloys of stellite in order to impart its favourable characteristics — hardness/toughness, corrosion resistance, etc. — to the material. As such, it cannot be totally removed from the alloy. Stable ^{59}Co can be activated to form ^{60}Co, one of the major radionuclides of concern during decommissioning. Therefore, if possible, materials with low cobalt content are to be specified. In many countries, steels having low cobalt content are being used or proposed for in-core components [47, 48]. This principle is also applicable to the existing operating stations for component replacement or for systems retrofit.

A study of the impacts of stellite was undertaken on a number of PWRs constructed by the German Kraftwerk Union (KWU) [49]. The most recent reactors have eliminated stellite from the reactor pressure vessel and almost all other areas of the primary circuit. The only area where stellite has been used to any notable extent is in the control rod drive mechanisms. Elimination of stellite has resulted in a dramatic reduction in plant radiation fields and occupational exposure. Channel head radiation fields average less than 10 mSv/h, compared with levels of 50–100 mSv/h for reactors that have not undergone stellite reduction.

Given the expected low implementation costs, the use of low cobalt materials can be highly recommended for reactor internals and primary circuit components. The potential use of low cobalt carbon steel for reactor vessels needs to be evaluated carefully and judged on a case-by-case basis. Potentially suitable replacement materials for stellite containing cobalt are available and can be considered for new construction and for replacement actions [50, 51].

4.2.2.3. Selection of concrete components

Activated or contaminated concrete is another major potential source of radioactive waste during the decommissioning of many types of facility [52]. Such concrete includes the neutron activated biological shield surrounding the reactor vessel, and floors and walls contaminated by spills during operation. There are two different approaches to the design of biological shields aimed at minimizing neutron activated concrete in the waste.

The first approach consists of fabricating the entire biological shield from precast, steel reinforced, interlocking blocks held together with steel bands and bolts. In this design, only the activated blocks have to be removed for the controlled disposal. The block approach eliminates the need for blasting or other methods that generate dust, which can contaminate non-activated concrete, thereby increasing the waste volume.

The second approach consists of fabricating the inner part of the biological shield from a material similar to plaster (possibly applied in layers), which could then be easily demolished and would permit simplified removal of only the radioactive portions. The practicality and cost effectiveness of these approaches still need to be proven, especially for power reactors.

Other proposals have considered the avoidance of certain trace elements (Cs, Co, rare earth elements) in the concrete’s constituents and in the reinforcing bars [53, 54]. In addition, the neutron absorption capability of the innermost layers of the biological shield can be increased by adding an absorbent shield between the reactor vessel and the biological shield that can be easily dismantled.

4.2.2.4. Protection of component surfaces

Surface contamination on floors and walls can be minimized by using steel plates or gratings instead of concrete slabs. Steel flooring can be decontaminated to unconditional clearance levels more easily than concrete, thus reducing the volume of waste.

Furthermore, cracks in concrete floors may cause contamination of clean floor or wall areas, thereby increasing waste volumes. However, if the concrete floors and walls in active areas are prepared with a smooth surface finish and protected with an epoxy or similar coating, decontamination of the concrete will be much easier, and thus active waste volumes from this source can be reduced.

It is desirable that the design of rooms or cubicles for components containing contaminated fluids include drip trays and floor curbs with sufficient capacity to contain the maximum envisaged spill or leak resulting from a component rupture. The curbs should direct spills to floor drains with sufficient capacity to collect all waste. Special care needs to be taken to prevent oil spills from mixing with water based drainage.

The spread of contamination during decommissioning can also be reduced by designing components for easy dismantling. For example, the use of modular blocks to build shielding walls could, where feasible, reduce the spread of contaminated dust during demolition of these structures.

4.2.3. Reducing surface contamination

In addition to the decommissioning oriented provisions discussed in Section 4.2.2, other practical measures to be taken into account during design, construction and operation to minimize residual surface contamination are as follows [30]:

- Provisions to decontaminate internal and external surfaces of the equipment need to be in place to facilitate inspection and maintenance work. This will reduce occupational exposure during operation and decommissioning;
- It is important to ensure that surfaces that are likely to be exposed to contamination are specified to be easily decontaminated (e.g. building floors and walls);
- The surface conditions of primary equipment should be specified, for example, to ensure requirements for electropolishing or for repassivation of surfaces;

— The spread of contamination is to be minimized through appropriate containment provisions for liquid spills.

For example, corrosion products, which are found throughout the primary system, circulate with the water through the reactor core, where they become activated [30]. Despite the installed purification systems, the amount of radioactive material deposited on the various surfaces may vary greatly, since it depends primarily on the corrosion rate of the affected material, the amount of radionuclides in these corrosion products, the chemistry of the coolant, the surface geometry and operating procedures. This deposition on the surfaces of the primary circuit of a nuclear power plant is one of the major sources of occupational exposure. The activation of this deposition increases over time, starting from initial plant startup; it typically levels off after four to six cycles of plant operation. A feasible way to reduce crud is to decontaminate the entire primary system chemically.

Decontamination, whether it is intended to facilitate inspection and maintenance or dismantling after shutdown, can substantially reduce occupational exposure. Design provisions to allow full system decontamination — for example, connectors, sampling and drainage lines, or spare tanks — may be cost effective and reduce occupational exposure during decontamination. Potential drawbacks include reduced space, additional components to be dismantled and increased secondary waste to be managed. It should be noted that, following permanent shutdown, more aggressive decontamination techniques are allowable, since the integrity of the affected components is not as critical as during operation.

All surfaces likely to be exposed to contaminated fluids during operation need to have a finish that can be readily decontaminated [30]. This finish could take the form of protective paint or coatings, or polished metal surfaces and liners as used in refueling and fuel storage ponds in the areas most likely to be contaminated. Such surface preparations are essential to preventing penetration of contaminated liquids into concrete. This is particularly relevant for buildings or areas where liquid radioactive waste is handled and/or treated. Experience has shown that, in the absence of major cracking, properly prepared concrete surfaces experience minimal activity penetration in comparison with areas where attention has not been given to surface finish [55]. Protecting concrete surfaces from contamination would lead to a reduction of the amounts of contaminated concrete rubble requiring packaging, shipping and disposal. Reduction of contamination by pretreatment of concrete surfaces is still an active research area [56, 57].

A number of surface and strippable coatings that can be applied directly are available. Most of these products have a plastic base, which seals the surface and facilitates future cleaning or removal. In fact, these products can be used as loose contamination fixatives, as decontaminants to help to remove loose contamination or simply for mechanical protection.

There are essentially two types of coating: aqueous based and solvent based. Aqueous based coatings have the merit of being non-flammable during application but are slower in drying. Solvent based materials dry much faster; however, flammable solvents cannot be used in areas with a high fire hazard. These coatings can be also used to fix loose activity or as a precoat where contamination is likely to be spread (e.g. by dismantling or cutting).

Submerged concrete surfaces continously exposed to contaminated liquids (e.g. fuel pools and sumps) can be protected using steel liners. Plastic liners such as epoxies or phenols have also been used as surface coatings. Another design feature is a sandwich type layer of steel plates embedded in the concrete to prevent deep penetration of contamination.

An important requirement for liners is durability and resistance to damage from both normal operation conditions and misuse of equipment, such as dropped objects. Some materials are susceptible to damage; for example, high density polyethylene sheet can easily be penetrated by dropped objects. Although it is easily repaired, the continuing integrity of the liner depends on any damage being identified and repaired. Plastic materials used in industry for containment liners have good chemical resistance and elasticity, but their inability to withstand high temperatures is a drawback. The long term effect of radiation on some plastics needs to be taken into account.

Other promising techniques include pretreatment of metallic surfaces to reduce contamination [30]. A number of techniques are available, among which electropolishing deserves special attention [58–60]. Preoperational electropolishing is a technique intended to reduce occupational exposure by reducing the inventory of radioactive contamination on metal surfaces. This technique smoothes the surface and reduces the buildup of contaminants during plant operation.

Electropolishing uses an electrolyte (usually orthophosphoric acid with additions of either chromic or sulphuric acid) and a cathode suitably shaped for the component surface to be polished. The process reduces the general roughness of the metal surface, thereby reducing the propensity of oxide films forming on such surfaces to absorb radioactivity. Experiments have shown that a 10–30% reduction of exposure can be achieved by implementing this practice in various types of plant [61].

The inventory of radionuclides fixed on the interior surface of tanks, piping, valves, pumps and other process equipment can also be reduced by inhibiting or limiting the rate of surface contamination after cleaning [30]. To prevent the rapid buildup of a tough adherent contamination layer, the cleaned surface can be pretreated prior to being put back into service. The process, referred to as repassivation of the surface, consists of forming an adherent oxide layer on the metal surface, which then prevents the formation of a layer containing radioactive contaminants.

The simplest method of repassivation is to use pure water at high operational temperatures and with controlled water chemistry over an extended period of time to form a protective layer on the internal surface. This practice has been successfully applied in all types of water cooled reactor. The treatment of surfaces with mild chemicals to form a protective layer may be considered an alternative method of repassivation. However, while chemical treatment is faster, it generates chemical wastes that require disposal.

Repassivation and other surface pretreatment methods reduce the equilibrium contamination level of the inner surfaces of the primary circuit. However, the results of any such methods may be highly plant specific, and a thorough evaluation is required before such a practice is implemented.

Decontamination and removal of contaminated structures are a major part of dismantling a nuclear facility and a large contributor to the total waste volume from decommissioning [30]. During normal operation of a nuclear facility, some concrete surfaces may become contaminated by liquid spills. A preventive measure that can be easily implemented in the design and/or construction stage is to provide for the containment of liquid spills at the source. Incorporating secondary containment features in the facility design, such as curbs or double valves, may reduce the contaminated surface area potentially requiring decontamination during decommissioning. In addition, it is advisable that the floors inside the curbs have a slope facilitating drainage towards the sumps [62].

If this feature is implemented in combination with the use of smooth and coated concrete surfaces, it will effectively decrease the volume of waste arisings. Reducing the extent of contamination may allow a change from using mechanical methods for concrete decontamination (e.g. destructive blasting, chipping and grinding) to using simple non-destructive methods such as washing and stripping. Local connecting points for water supply must also be provided for decontaminating walls and floors [62].

It is worthwhile to consider the possibility of flooding of building basements because of the potential for contamination of subsurface soil. Soil contamination can extend a significant distance from perimeter foundations. The design, type and leak resistance of joints at the floor to wall interface are of particular concern. Sampling provisions for subsoil ideally will be determined at the design stage.

4.2.4. Reducing leakage and crud traps

Each component in a liquid process system can contribute to an increase of the contamination problem, either by providing traps where crud can build up or by providing a path along which the radioactive liquid can leak out and cause local or widespread contamination [29]. Contamination problems can potentially be reduced by minimizing the number of flanges, joints, elbows and other crud traps and by ensuring that pipes or tubes have a sufficient slope. Use of welded rather than flanged joints might also reduce leakage of contaminated liquids. However, this benefit must be weighed against the potentially greater difficulties during dismantling. The designers of new facilities need to consider these practices, provided that safety, quality assurance and operational requirements are met.

4.2.5. Quality control

Spread of fission products in primary piping may be an important consideration if a reactor has been operated with significant fuel defects for extended periods of time. Timely replacement of defective fuel assemblies is essential when high concentrations of fission products are found [30].

It is advisable that the use of fuel assemblies with high reliability and very low leakage of fission products from premature failure resulting from manufacturing defects be ensured at the design stage. Therefore, quality assurance programmes that are based on good quality control and compliance with standards are essential in the nuclear industry [29]. Normally, quality control actions arise from a number of technical requirements that are of a much higher priority than decommissioning alone. In some cases, these actions are also beneficial from a D&D viewpoint. For example, the following design and operational initiatives relating to quality have resulted in reductions in the number of contamination problems:

— Improvements in the quality of cladding and welds for fuel elements have resulted in fewer defects and less alpha and fission product contamination of the primary heat transport systems in reactors and fuel bays.
— The use of better seals in joints and valves for liquid filled systems has resulted in reduced leakage rates and reduced external contamination.
— Tighter specifications for surface conditions of steam generator materials and electropolishing of steam generator channel heads have reduced contamination by major radionuclides.

Chemical interference that might lead to the formation of undesirable, highly active third phases and crud, and the blocking of liquid transfer systems by uncontrolled precipitation, crystallization or solids formation can be avoided by taking appropriate measures in the design stage.

4.2.6. Limiting corrosion

Reductions of the transport and deposition of activated corrosion products are beneficial to operation and decommissioning [30]. The corrosion rate of the piping and components in process systems — for example, in the primary heat transport system of a water reactor — can be reduced considerably by maintaining water quality requirements and tight control over the chemistry of the system [29]. This approach reduces the amount of radioactive crud in the coolant as well as the amount that settles out in the oxide layer that forms on components.

A purification system using filters and ion exchangers is necessary to maintain control of the levels of crud and dislodged solids in the coolant. Additionally, purification capability can enhance the entrapment of activity during plant shutdown.

It is desirable that the selection of new materials for reprocessing plants take into account the acidic nature of the process fluid.

4.2.7. Minimizing the spread of contamination

The design considerations mentioned above can significantly improve the decommissioning process. However, proper control of operations is also important, so that spills, accidents and other events that could lead to significant contamination are minimized.

To minimize the accidental spread of contamination to clean areas, nuclear facilities are zoned into active and inactive areas, with control points for movement of materials, equipment and personnel between zones [19]. Careful maintenance of zones is required during decommissioning, because of the potential for the spread of contamination. Frequent changes of zone boundaries as decommissioning proceeds are to be expected. Careful checking for external contamination on waste packages and vehicles coming from active zones, especially when the flow of material is high, can help to reduce the spread of contamination. The design of facilities must provide space for adequate interim waste storage during operation and for waste movements and tracking.

It is often very difficult to maintain good ventilation as the D&D process proceeds. Auxiliary portable ventilation equipment may be required to supplement the regular system. It is important that the design and construction of the equipment or tools to be used during dismantling includes provisions for the use of local ventilation nearby or for connection to local operations. These design features are valuable for controlling and minimizing the spread of contamination. The use of ventilated containment enclosures around any decommissioning processes that create large quantities of dust also significantly reduces contamination spread.

Special attention needs to be paid to the handling of nuclear fuel. In any case, the loss of integrity is to be avoided. However, if this occurs during operation or decommissioning, steps must be taken to minimize the consequences.

4.3. PROVISIONS TO FACILITATE DECONTAMINATION

Ways to facilitate the cleanup of contamination ideally will be assessed during the design of nuclear facilities and, if possible, incorporated into the design [29]. One method that is widely used and very effective — especially if the potential for liquidborne contamination exists — is the scaling of concrete floors and walls. Even the highest quality concrete has pores into which water can penetrate, often to depths of 10–20 cm. Concrete floors and walls that are likely to be exposed to waterborne contamination should be protected by a wear resistant barrier such as epoxy paint or steel cladding. In addition, the concrete surface itself can be made less porous by use of a suitable mix and appropriate application techniques during installation/construction.

Conventional stainless steel liners (i.e. thin liners anchored to the concrete in many places) appear to have a net benefit for decommissioning, provided they remain undamaged [6]. Stainless steel can be decontaminated relatively effectively, thus reducing the decontamination efforts required to allow personnel entry into a process compartment to complete the decommissioning. Covering more wall areas in the process compartments with stainless steel liners (depending on process equipment location and function) will likely make decommissioning easier. Using this concept may also result in less radioactive concrete rubble as waste. However, the large number of anchors used for securing the liner to concrete complicates dismantling.

Protective coatings on concrete can reduce the amount of radioactive contamination absorbed in the concrete; they can also assist in obtaining good chemical cleanup of the concrete. Floor–wall interface joints should be covered to the greatest extent practical to facilitate decontamination. However, to be fully effective, these protective coatings must maintain high integrity over the life of the plant, and must resist deterioration caused by radiation and by process and decontamination chemicals. Recoating the exposed concrete surfaces periodically during the life of the plant may accomplish the same objective.

It is advisable that process systems that need to be chemically decontaminated at fairly frequent intervals — for example, the primary heat transport system in a reactor — be designed to enable easy filling and complete drainage after decontamination. Ideally, all systems carrying radioactive liquids will be designed with suitable connection points, vents and drains, provided that these do not affect operational safety. These systems should be decontaminated and drained with minimal expenditure. Provisions should be made for recycling decontamination solutions.

Where practical, components and equipment are to be fabricated with smooth surfaces and with the minimum number of crevices in order to reduce entrapment of contamination. Equipment that could be exposed to airborne or sprayborne activity should be protected to prevent contamination of parts. This approach is often used in hot cells, where, for example, manipulator arms are covered with sleeves that are sealed, thus permitting easy decontamination of equipment when an arm is removed either for repair or for disposal.

Tanks, pipes, components and systems likely to become highly radioactive need to be connected directly to the sumps and storage tanks of the liquid radioactive waste treatment system. Of course, direct connection of critical systems (e.g. the reactor's primary heat transport system) to the radioactive waste treatment system is not permitted for reactor safety reasons.

It is desirable that the design also provide for complete drainage of contaminated piping systems by including the installation of low point drains, pump drains, tank vent systems and drain systems, and by eliminating dead legs between valves in system designs.

Hazardous and flammable materials are not to be included in the design if they must be removed before the decontamination process can be started; this applies especially to areas likely to have high radiation fields. The presence of such materials could delay decontamination and increase occupational exposure, since the operators would have to take more care in removing them and thus spend more time in the radiation fields.

Adequate access to equipment requiring decontamination must be provided, along with suitable lay-down areas for portable decontamination equipment if necessary. Designers need to ensure that adequate space is left around equipment, and that access to cubicles and rooms is provided so that connections can be made to the system being decontaminated, and to enable the areas around the equipment to be cleaned up.

The removal of contaminated concrete in process areas is one of the most laborious and time consuming activities in decommissioning [6]. Current methods require drilling for use of explosives and rock splitters. These methods could be carried out much more quickly and at a lower cost if holes for placement of explosives were built into the concrete surfaces (but sealed from process materials). Built-in provisions for other techniques, such as spalling of the concrete by heat or electric current, might also be considered.

In addition, the plant owners and regulators must make sure that the operating team is fully trained to ensure that the plant will operate efficiently and with full implementation of the ALARA (as low as reasonably achievable) principle. A well trained and motivated operational staff can facilitate future decontamination by minimizing the spread of contamination during operation and maintenance work and by keeping records of spills and other untoward events.

4.4. PROVISIONS TO FACILITATE DISMANTLING AND SEGMENTATION

A well designed facility also needs to consider features that would facilitate dismantling and segmentation during operation and decommissioning [29]. These features ideally will take into account safe and efficient operation as the primary objective of the facility. A variety of concepts have been used or proposed for inclusion at the design and construction stages to facilitate the dismantling, removal or segmentation of components or equipment in a nuclear plant. A few of these concepts are discussed below.

4.4.1. Minimum use of hazardous materials

During the selection of materials at the design stage, the use of hazardous materials such as flammable coatings should be excluded if possible, as the operators would have to spend more time dismantling these materials because of requirements for special precautions [29]. These materials could also complicate waste management and disposal.

4.4.2. Additional plant layout requirements to facilitate dismantling

Appropriate component and piping layout, material handling provisions and other features can help to substantially reduce the time spent in active areas, occupational exposure and human resources requirements, and facilitate the management of decommissioning waste [30].

It is desirable for spacing around components to provide access for personnel and equipment in order to allow the components to be easily disconnected or dismantled [29]. The removal of equipment must be taken into account when the size of doors, hatches and hallways is decided, unless the walls around the item are designed to be easily demolished. Provisions for inserting radiation detector probes are useful at entries to cells.

Ready access to all contaminated process equipment is highly desirable [6]. One technique is to create a canyon type facility above process cells, with removable ceilings above the equipment. Access could also be provided from the side walls of the process cells. Ideally, this access will have some combination of large openings and ports for the passage of decontamination equipment or chemicals.

Suitable built-in ladders and walkways need to be included, especially in areas of high radiation, to minimize the amount of temporary scaffolding that may be needed during decommissioning [15]. Another useful feature is access to the process equipment for a remotely operated crane. Reducing the need for personnel entry into the cell would reduce occupational exposure and minimize the need for special equipment or innovative techniques to decommission the process cells.

Accessibility to contaminated and activated systems and components that may require special removal techniques during decommissioning needs to be considered at the design stage [30]. Particular attention must to be given to the possible use of removable panels, cubical covers, shield walls and access hatches. In addition, space for equipment replacement or removal, including lay-down areas, must be considered for segmentation and packaging of major components.

Appropriate arrangement of various plant components and access to them will facilitate maintenance and dismantling. It is important that consideration of features that will ease removal during maintenance and decommissioning — in particular for bulky components —be given during the design stage. This, combined with the broader use of unit construction techniques such as bolted precast concrete elements, structural blockwork or bolted structural steelwork connections, will allow better access for equipment removal.

The deep, inaccessible vaults or chambers that exist for reasons of construction or as an inherent result of layout may create a special problem. These vaults may be used during plant operation to accumulate active waste, which is then difficult to remove. It is advisable that they be either avoided or provided with means for retrieval of accumulated waste.

4.4.3. Preplacement of dismantling aids

Preplacement of selected dismantling aids or arrangements for their installation at a later date could facilitate the dismantling and segmentation of components and reduce occupational exposure [29]. It is much easier to install tracks for guidance of remotely operated cutting devices or manipulators, or for translatory movement of lifting equipment such as cranes, monorails and hoists, during construction than once radioactivity has been present in the facility. Also, major pieces of equipment such as pumps and tanks need to be equipped with attachment points to facilitate their removal. This approach will reduce dismantling time later. However, by the time these aids are used for decommissioning, the type of cutting equipment may have changed, rendering the aids obsolete.

If explosives are to be used for dismantling monolithic concrete structures, holes to hold the explosives could be arranged in these structures during their construction, with the holes being capped and positioned perpendicular to incident radiation to prevent streaming. The safety of this approach at a particular facility needs to be assessed on a case-by-case basis.

Giving consideration at the design stage to installing viewing and inspection aids can assist in conducting visual inspections and radiological surveys of areas or items [30]. Very complicated and costly remote viewing and manipulating equipment has had to be developed owing to the lack of access and provisions for inspection and viewing.

4.4.4. Shielding

If operational design requirements allow, shielding arrangements for equipment in the plant should not obstruct decommissioning and preferably will provide sufficient shielding during the dismantling of the equipment [30]. Reductions of occupational exposure during decommissioning can be achieved by extensive use of simplified shielding such as leaded blankets, lead sheets and lead brick in cases of extended exposure [15]. Temporary shielding can be provided rapidly by stacking 200 L drums and filling them remotely with water. The designer needs to weigh the benefits of installing extensive shielding against the exposure that may occur during dismantling.

Remotely controlled vehicles can be used to detect spots with high radiation fields. Occupational exposure can be reduced by installing shielding before human access is permitted.

The primary design function of biological shielding is radiation control during operation [30]. However, the biological shielding around the reactor vessel ideally will also be designed to reduce the level of activation of the material, thereby minimizing the decommissioning waste and potentially reducing occupational exposure during dismantling. This shielding, if properly designed for ease of dismantling, can reduce the radiological problems associated with dismantling and disposal of these structures.

At present, the biological shields in nuclear power plants are constructed of concrete and steel [30]. Two main factors influence the activation of these materials: the reactor neutron flux and the quantity of elements in the concrete and steel having the potential to become activated. In order to achieve low radiation fields, one or both of these factors needs to be reduced.

Regarding the first factor, to reduce the neutron flux, it would be necessary to incorporate into the design some neutron absorbing material, either in the space available between the reactor core and the vessel wall or between the vessel wall and the concrete shielding. The former option has the additional benefit of reducing activation in the reactor pressure vessel wall as well as in the concrete. Such material would have to be inserted into the available space and must not present difficulties at the dismantling stage.

There are two possibilities for improving the neutron shielding properties of concrete, thereby leading to a possible reduction in the thickness and overall volume of activated material. The first is to increase the hydrogen content of the mix, which could be achieved either by increasing the water content of the mix or by using hydrous aggregates. The second is to introduce boron into the concrete mix in the form of insoluble boron frits. Prior to the adoption of either approach, testing is required to confirm that implementation will give the anticipated shielding properties. It will then be necessary to establish procedures for producing the concrete mix and to check for any unacceptable effects on concrete workability, heat of hydration, creep, shrinkage, strength, seismic resistance or durability.

The second factor relates to the chemical composition of the concrete itself. The elements in concrete giving rise to radionuclides have been identified [30]. However, control of the quantities of such elements is difficult to achieve, since even very small quantities can lead to significant activation.

If reductions in activation levels can be achieved, then parts of the shield walls could be classified as non-radioactive at the start of the decommissioning phase, or at some convenient time during a safe enclosure period.

Large amounts of steel reinforcing bars, prestressing tendons and steel punching have been used in the construction of concrete bioshields. Reducing their number or size in the concrete will reduce the amount of induced radioactivity [63]. However, the layout of reinforcing bars and prestressing tendons is usually decided on the basis of structural stability requirements. Alternate designs utilizing steel liners and concrete blocks as the bioshield or introducing neutron absorbent materials such as water or borated steel may reduce dismantling times and may further reduce the induced activity associated with these shields. A comprehensive description of techniques to reduce activation of the biological shield can be found in the literature [64–66].

Several methods have been used or proposed to facilitate the removal of concrete shielding during decommissioning and to segregate active from inactive concrete [29]. However, application of these methods in a particular facility must be evaluated on a case-by-case basis.

If other design and safety requirements are met, shielding and dividing walls can be constructed from modular components, rather than from poured concrete, to facilitate easy dismantling. If necessary, the modules can be interlocked or given additional structural strength by use of an iron girder frame. The modular concept may be inappropriate where concrete structures have both shielding and structural strength functions.

If modular construction cannot be applied, it may be possible to exploit planes of weakness and thereby assist the decommissioning workforce during the dismantling of concrete structures.

As an alternative to modular construction, composite shield construction could, in some cases, reduce the dismantling time. For example, it might be possible to construct certain shielding walls with an inner and an outer steel wall (the inner space would be filled with shielding material). Such walls would be much easier to dismantle than reinforced concrete walls and would be just as suitable, unless the wall is required for structural reasons.

Implementation of these approaches assumes that the wall designs neither result in a loss of shielding function nor compromise any other design requirements.

4.4.5. Connectors

It is desirable for the design to make use of connectors, fasteners, hold-down devices and simple, plain shaped supports to ensure the avoidance of traps and dead holes, since these can be removed easily during dismantling [29]. However, the designs must ensure that the premature release or loosening of these devices does not occur inadvertently during operation. Anchor points, which are required to facilitate the removal of equipment, need to be considered during the design stage for installation during construction.

4.4.6. Intact removal of large components

The possible replacement during operation or removal for decommissioning of heavy components such as steam generators, reactor coolant pumps or any other large objects ideally will be taken into account at the design stage [30]. The plant design needs to include lay-down space and crane capability, facilitate the use of remotely operated equipment where required, and simplify the handling and removal of these components.

On the one hand, segmentation of the reactor vessel is considered in most decommissioning plans. On the other hand, intact (one piece) removal is identified as a technique to reduce exposure incurred through segmentation of a component prior to its removal. The necessary lifting and transport technique is basically a reverse installation technique. Such services are commercially available from various specialist contractors. The detailed performance specifications need to be defined at the time of dismantling. Further dismantling after removal will not be necessary if an acceptable storage/disposal site is available. If that is not the case, a separate facility for dismantling major components will be required. However, the intact removal may still reduce the overall time required for decommissioning, since it may be possible to conduct work in parallel.

It should be noted that implementation of the intact removal technique will require a thorough review of structural strength calculations, and that some reinforcement and other temporary measures may be necessary. As an example, removal of a steam generator can be achieved by removing obstructions from the area, erecting scaffolding and localized shielding, and severing the steam generator from the reactor coolant and secondary coolant piping. In terms of dose exposure, this technique may be preferable to one based on cutting the steam generator into several segments [61].

An important consideration for the removal of large components is the load bearing capacity of the flooring. Flooring that needs to be removed for access is to be segregated in the form of easily removable slabs/blocks fully within the reach of plant material handling units. Other properties of large components may also require special handling procedures owing to their irradiation and ageing (e.g. embrittlement).

Intact removal of large components like reactor vessels and steam generators may be beneficial from the point of reducing radiation exposure. However, if size reduction and packaging of the waste following intact removal are still necessary, then the total dose commitment may not be significantly different from that with size reduction performed in situ, especially if the primary contributor to dose is the component itself rather than the other sources local to the work area [67, 68]. Intact removal and disposal of these major components requires a large volume of burial space; this may increase the cost compared with segmentation, which may reduce the waste volume to a large extent. The use of appropriate remotely operated tools for segmentation of major equipment may also reduce occupational exposure.

4.4.7. Documentation

Dismantling and demolition will be facilitated if technical records are available [29]. Therefore, after a facility is commissioned, a records retention system needs to be established to provide the following [69]:

— Details of design and construction;
— Details of operating history;
— Details of modifications to the plant and maintenance experience.

An extensive amount of information is generated during the design, construction, operation and shutdown of a large nuclear facility [15, 42]. Some of these data could be extremely valuable in planning and executing the decommissioning of the facility. However, as a large percentage of the information is not pertinent, the primary tasks at the design and construction stages are to select the information required for decommissioning and to find a suitable, cost effective method of storing it so that it will be available when required.

A database for this purpose must contain an accurate and detailed description of the facility so that the most effective ways of removing equipment and structures can be determined. Ideally it will include items such as:

— Details of design and construction of the facility, including records of:
  - Construction material specifications and analyses;
  - Construction prints and as built drawings of the plant, and lists of these items;
  - Photographs taken during construction and installation;
  - Pre-operation environmental and radiological data;
  - Any data from a national database that are particularly relevant to the particular facility;
— Details of the operating history of the facility, including records of:
  - Fuel failures and fuel accounting, if applicable;
  - Incidents leading to spillage or inadvertent release of radioactive material;
  - Unusual occurrences during operation that might affect the decommissioning procedure;
  - Radiation and contamination survey data, particularly for plant areas that are rarely accessed or are especially difficult to access;
  - Releases that could potentially affect groundwater;
  - Radioactive inventory;

    - Details of decontamination carried out during operation and plant maintenance;
    - Details of radioactive/toxic waste stored;
- Details of modifications to the plant and maintenance experience including records of:
    - Updated as built drawings and photographs, including details of the materials used;
    - Special repair or maintenance activities and techniques (e.g. effective temporary shielding arrangements or techniques for the removal of large components);
    - Details of the design, material composition, and history and location of all temporary experiments and devices;
    - Information and drawings of specialized procedures and equipment used during maintenance that could be of value during decommissioning.

It is advisable that documentation systems be designed to ensure long term maintainability and readability.

A site specific database for a particular facility could save money and time during decommissioning. The concern is that creating and maintaining such a large database for decommissioning purposes alone probably would not be cost effective. The database that is used during the construction and operational phases could later be downgraded to eliminate data that are not relevant to decommissioning. Such a database would need to be designed to ensure intact data storage for long periods of time, perhaps for a long as a hundred years. It is imperative that at least two separate physical locations be established to maintain the documentation over these long periods of time to ensure that the backup information is not lost through events such as fires.

If computer databases are used to store information for long periods of time, the systems associated with these databases will require frequent updating to ensure compatibility with current computer operating systems. Data retrieval procedures will also require updating. Computer databases should be backed up by hard copies of the data to ensure data availability and retrievability.

Generic data outlining techniques or experience from previous decommissioning or decontamination may be stored in national or international databases, which can be then accessed by computer. One type of generic data that is not currently in such databases but that could eventually be of assistance in planning and costing future decommissioning operations is unit cost factors. These factors are used to describe the decommissioning process by breaking it down into elementary activities such as removing pumps and cutting pipes.

Experimental irradiation of specimens of selected materials (coupons) used in the construction of the facility may also assist in comparing the measured data with the calculated activation levels to provide better estimates of the final radioactive inventory. Therefore, such coupons should be obtained during construction and preserved for future experimental use.

Pre-operational environmental data are often unavailable for existing facilities [70]. Analysis of information from background locations (unaffected by past or current facility operations) and information provided by the characterization survey regarding the current state of the natural environment and the extent of contamination will help in the understanding of pre-operational environmental conditions. This will allow identification of changes to the environment arising from operations and waste management practices. However, it is important that the area to be used as a reference to establish background conditions be environmentally similar to the site, since environmental conditions can vary over even short distances. Characterization and baseline surveys use the same measurement and sample analysis techniques as those applied to the environmental media. Characterization surveys require that soil sampling be conducted at greater depths to assess contaminant leaching into the subsoil, and it is necessary to sample waste materials. The use of a non-uniform distribution of monitoring wells may be needed on the site, as well as off the site, to track possible existing contaminant groundwater plumes.

Similar requirements need to be considered for characterization surveys of buildings or selected materials.

The owners, designers, builders and operators are responsible for ensuring that all data are available to the decommissioner in a readily usable form, even if dismantling is not to begin until a much later date. The designers and owners need to establish a suitable database, which is to be maintained by the builders and operators. The data management system must ensure that:

— All relevant documents impacted by any modification are identified and updated, remain consistent with the plant specific design requirements and accurately reflect the modified plant configuration;
— All changes to the design over the lifetime of the plant are based on the actual status of the plant, as reflected in the current plant documentation;
— The modified plant configuration conforms fully to the conditions and supporting documentation of the operating licence.

It is recognized that the knowledge of operational and research staff is beneficial for decommissioning, particularly during the transition from operation to decommissioning [3]. A lack of action or unjustifiable delays in this transition are a serious concern, particularly owing to the ageing of operational staff in many facilities with research reactors. Mechanisms need to be in place to expedite the transition from operation to decommissioning, and to include the establishment of a comprehensive set of decommissioning records. Nevertheless, planning for decommissioning and for the plant final end state requires the development and maintenance of significant professional decommissioning expertise. Such expertise is specific to decommissioning and different from the experience base required during plant operation. Training or retraining of decommissioning operators is another important consideration, particularly for deferred decommissioning.

4.4.8. Planning

Decommissioning activities are greatly facilitated by appropriate strategies and plans [5]. Ideally, outlining of the decommissioning plan will begin early in the facility design stage. These plans need to be reviewed on a regular basis during the design stage and the subsequent operational lifetime, and/or as required by the regulatory body. The operators must ensure that an acceptable decommissioning strategy and detailed plans are developed and updated to facilitate decontamination and dismantling.

Planning for the actual decommissioning includes establishment of a well defined decommissioning programme, including:

— The purpose and status of the project;
— An assessment of alternatives;
— The organizations involved and their responsibilities;
— The overall cost, schedule and technical approach;
— The management, engineering and specialized decommissioning techniques to be used;
— Analyses of radiological and industrial safety aspects;
— An assessment of sociopolitical aspects [15].

Planning must also include the development of well trained work crews who are knowledgeable about the job and the radiological protection required to keep occupational exposure as low as reasonably achievable within actual social and economic conditions. Crews should be aware of their responsibility and be familiar with each task to be performed and with the equipment to be used. Workers not immediately involved in the task in progress should be trained to move out of the area of radiation fields until needed. The timing for startup of the dismantling operation and preselection of remotely operated equipment can have a significant impact on the doses received by operators. Usually, delaying decommissioning reduces occupational doses.

Decontamination of certain systems/facilities can also be a good method for reducing radiation fields and occupational doses. However, the value of the decontamination effort must be compared with the financial cost and dose budget (person–Sievert cost) for performing decontamination and for treating the waste arising from the decontamination process.

As the sequence of the dismantling activities to be carried out can have a large effect on the dose received by the staff, such sequences require careful planning. If the majority of the activity in a component is from relatively long lived radionuclides such as ^{137}Cs (half-life: 30 a), then planning for removal of the component early in the decommissioning sequence may reduce exposure. However, if the high activity comes from short lived radionuclides, then planning to delay component removal for one or two years could be beneficial to dose reduction. If several similar facilities are to be decommissioned, carefully planned sequential decommissioning of these facilities may reduce occupational doses, with the added benefit that the necessary equipment and experienced crews will be available.

A wide variety of planning techniques can be applied during the design and construction stages to reduce occupational exposure during decommissioning. Whether or not these are applied will depend on many factors, such as the cost versus the benefit and engineering practicality.

4.5. DEVELOPMENT AND IMPROVEMENT OF TECHNIQUES FOR DECONTAMINATION AND DECOMMISSIONING

Because of the importance of conditional or unconditional clearance of materials to the environment and the delicensing of sites, considerable work has been undertaken worldwide to develop new techniques essential to waste management. Details of some of the current techniques for simplifying the management of waste materials are given below.

4.5.1. Development and improvement of decontamination techniques

A large number of decontamination techniques are available for use in the decommissioning process [4]. Not all of them are capable of achieving the residual contamination levels needed to meet the established clearance criteria. In some cases, decontamination is carried out in stages, with a final step aimed at reaching the desired levels.

A number of initiatives are currently carried out to develop and demonstrate decontamination technologies for decommissioning and to achieve unconditional clearance of materials. Not all of them have reached the same state of maturity, and their status needs to be taken into account when making selections for application.

Improvement of decontamination techniques is envisaged with the development of new, innovative technologies and the further development of existing techniques. Two novel technologies that are being studied are the following:

— *Microbiological degradation:* Used to decontaminate concrete and steel. The technique uses microbes to penetrate surfaces and to degrade them in such a way that they and their contamination can be more readily removed. The decontamination concept [71, 72] has been proven at the laboratory scale [73]. Its potential advantages include minimization of radioactive waste, less intensive use of labour and the avoidance of capital expenditure for mechanical equipment.
— *Light ablation:* Uses the absorption of light energy and its conversion to heat to selectively remove surface coatings or contamination. Decontamination by light ablation is being tested in US Department of Energy (USDOE) demonstration projects. The results show that up to 6 mm layers of concrete can be removed [74–76]. Work is also under way in Europe using an ultraviolet laser for the decontamination of plastic and metal tanks or chambers [77].

Research and development aimed at improving a number of existing techniques continues. The following are techniques of current interest:

— *Aggressive chemical processes*: Processes operating in aqueous media and utilizing strong acids and bases that are generally used as reactants, as well as strong oxidation–reduction pairs such as Ce^{4+}/Ce^{3+}. Not surprisingly, these processes generate large quantities of liquid waste, although reagent regeneration systems with ion exchange cleanup can limit this volume. A process for the electrochemical decontamination of alpha wastes has also been successfully tested [78].
— *Foams:* A recent development, using recirculating foam, has been tested [79]. The most satisfactory foams use biodegradable surfactants in combination with strong acids and bases. One advantage is that performance can be enhanced by injecting ozone rich oxygen to reoxidize the redox agent. Foams, as well as laser systems, have the potential advantage of separating operators from the contamination. Both techniques eliminate the contamination trapped in the oxide layer and the substrate.

— *Carbon dioxide blasting:* A variation of grit blasting in which CO_2 pellets are used as the cleaning and decontamination medium [74, 80]. One advantage of the process is that most of the secondary waste is CO_2 gas, which is easy to treat [81]. There is a track record of successful applications [82].
— *Sponge blasting:* Another variant of the blasting technique is sponge blasting, in which sponges made of water based urethane are blasted onto a surface, which causes the sponges to expand and constrict, thereby creating a scrubbing effect. An 'aggressive' grade of sponge, impregnated with abrasives, can be used to erode material such as paints, protective coatings and rust [83].
— *Abrasive blasting:* The decontamination of metals by abrasive blasting (wet and dry) to achieve unconditional clearance levels has been successfully demonstrated [84]. A semi-industrial scale trial showed that the wet process was less efficient, had higher costs and produced more secondary waste compared with the dry system. On the basis of the results achieved in this demonstration programme, an industrial scale dry abrasive blasting unit was installed to decontaminate 1500 Mg of contaminated metal [85].

In addition to these decontamination techniques, some substantial programmes are being targeted to develop a range of techniques for reuse and recycling. The USDOE has a programme on technologies for decommissioning and recycling under its environmental restoration programme [86]. The United Kingdom has a significant R&D programme on the use of melting technology for gaining unconditional clearance [87]. Efforts are also being made to improve melting technology effectiveness for decontamination. Some countries such as Belgium, Germany, Sweden, the United Kingdom and the United States of America have successfully recycled metal by melting [37, 87, 88].

A method for separating the radioactive contamination from bulk concrete and soil is under development. The separation is achieved by thermal treatment followed by milling and sieving. Pilot scale testing on concrete derived from the decommissioning of the VAK plant in Germany was successful [89].

4.5.2. Development and improvement of dismantling techniques

A large number of methods are already available to meet the dismantling requirements of various nuclear installations. However, in many cases it is still necessary to improve their performance, broaden their field of application and properly control the impact of their use on the immediate environment. In some cases, automation and remote control would appear to be necessary, either to make the dismantling equipment more effective or to allow it to be used inside hazardous areas. Specifically, it is essential to:

- Improve the capacity of tools for cutting thick steels such as those used on vessels, flanges and lids from large reactors;
- Increase the operating speed of systems for breaking up concrete, while restricting the amount of debris produced;
- Adapt tools for underwater work that perform satisfactorily in a conventional industry and can be modified for nuclear application.

Some specific processes that are of interest but not yet commonly used are as follows:

- *Cutting steel by cracking:* Extraneous metal is deposited by an electrode on the part to be cut; the underlying metal is embrittled and cracked when cooled. This process is limited to use on low thickness metal (15–20 mm) and is difficult to use. However, its value lies in preventing the spread of contamination contained inside the vessel being cut up [90, 91].
- *Lasers:* Laser techniques are being developed for cutting steel and concrete [92–94]. The laser sources are too large to be brought into the active area or to be remotely handled. One approach is to route the beam from the source using a polyarticulated, remotely controlled arm fitted with mirrors. To be useful, this system will require improvements including: provision for higher power at the point of application, cooling of the mirrors, protection of the articulated arm as it moves around and a method of system controls for use under water.
  Cutting thicknesses using lasers with power ratings of 2–10 kW are limited to 30 mm for stainless steel. It must be stressed that thick steel can be cut with fixed, very high power lasers, provided the part can be moved in front of the beam (this method has been tested for cutting up PWR pipework by RANDEC, the Japanese utilities group).
- *Electrolytic cutting:* In this operation, an electrode (cathode) penetrates into the metal to be cut at the same time as an electrolyte flows around the cut. The part being cut forms the anode and is gradually destroyed; the electrolyte carries away the metal particles produced by the cut for recovery. This process has been proven in conventional industry and can cut through considerable thicknesses (up to 30 cm), make highly precise cuts and control secondary wastes, which can be easily treated.

The technique is fairly slow, and methods for using it remotely in an industrial framework have yet to be developed [95]. Interesting results have been obtained from cutting tests on unirradiated test pieces representing the walls of a PWR vessel (ferritic steel 22NiMoCr37 lined with stainless steel with a combined thickness of 143 mm).

— *Pyrotechnic cutting:* The effectiveness of the explosive cutting of pipes, concrete, etc., needs no further demonstration. The remaining problem is to make this process compatible with the working conditions of nuclear installation decommissioning (it must be possible to position the charges, avoid spreading contamination, avoid shaking nearby installations that are still functioning, recover debris without undue dispersion, etc.). This process has already proved its value in cutting pipes with wall thicknesses of up to 3 cm. Charge carriers make possible the positioning of explosives by remote control.

Tests have been carried out on cutting concrete from the biological shield, separating steel liners from cell walls and cutting up pressure vessels. The results of these tests are of considerable interest and suggest that these techniques are fully competitive. Work performed at full scale on the German HDR reactor has shown that it is possible to use this process without undue risks and under acceptable conditions [96, 97].

A number of other techniques are also worth mentioning, such as the use of microwaves to 'de-scale' concrete or even to cut it [97], the use of a portable arc saw for cutting steam generator pipes and the use of high pressure jets containing abrasives for underwater applications [98–100]. Care should be taken when thermal techniques for dismantling are used, since radioactivity might get incorporated into the molten base metal or be dispersed in the surrounding area. Further information on these various techniques and equipment can be found in the annual reports of the Commission of the European Communities [101].

4.5.3. Development and improvement of measurement techniques

A number of measurement systems exist that are applicable to waste management in general or to unconditional clearance in particular [4]. New developments are in progress, aimed at improving measurements for unconditional clearance. That these developments are not at the same stage of maturity needs to be considered prior to their selection for use.

Work on mass activity measurements has been carried out using an automated large scale radioactivity measurement facility to extend its application to measure more than 100 Mg of different materials from a WWER reactor [102]. The complex nuclide mixtures and the age of the materials made the measurements difficult, and only some of the material was able to be free released. However, the work demonstrated that, by adjusting the measurement and evaluation procedures to meet the specific requirements of a project, a modified model of such a large scale measurement facility can successfully be used for materials having a high concentration of nuclides that are difficult to measure.

Spectrometric radiation detectors such as NaI(Tl) and Ge(Li) detectors, and more recently high purity germanium detectors, have been used extensively to measure ground contamination and to estimate dose rates created by natural radioactivity in soils. During the later stages of decommissioning, the large surfaces of buildings, etc., need to be monitored to ensure that clearance levels have been achieved. Currently, measurements can be made either by using adequate strategies for analysing samples taken from the surface or by measuring the surface activity using large proportional counts. An alternative approach under development uses a collimated in situ gamma spectrometer [103]. Prototype equipment has been tested at seven facilities in Germany and France. Comparisons of the established method and the in situ technique have, in most cases, demonstrated the capability of the new device to meet the required clearance criteria.

The use of long range alpha detection (LRAD) is being developed. Long range alpha detection is sensitive to all forms of ionizing radiation, but it is particularly suited to the measurement of alpha particles [104]. Instead of detecting radiation directly, the LRAD technique detects ions created in the surrounding air. It has a potential advantage over existing techniques in situations where it is difficult to perform direct measurements. For example, air can be transported through contaminated piping to an ion detector. Similarly, an object can be placed in a chamber and the air can be ionized by passage over it. While the LRAD concept has been proven, full commercialization has not yet been achieved.

In view of the final demolition of buildings, an alternative unconditional clearance methodology has been proposed [36]. Application of the methodology based on surface measurements and core samples is complicated for the clearance and demolition of buildings. An alternative being developed considers at least one complete measurement of all concrete structures and the removal of all detected residual radioactivity. This monitoring sequence is followed by a controlled demolition of the concrete structures and crushing of the resulting concrete parts into smaller particles. Metal parts are separated from concrete first. Representative concrete samples are taken during the crushing operation. The concrete samples are milled and homogenized before a smaller fraction of concrete samples is sent for laboratory analysis. Based on the results of these laboratory analyses, the material is released for unconditional reuse and removed from the site for further use in conventional road construction.

4.5.4. Developments to simplify waste management

4.5.4.1. Minimization of waste volumes

Minimization of the volume of material requiring disposal as waste is an important consideration in decommissioning projects. It is important that, during the design of a nuclear plant/facility, waste minimization principles be incorporated to ensure reductions of liquid waste and the generation of secondary waste, to ensure segregation of radioactive material from material that can be free released and to facilitate isolation of contaminated areas from uncontaminated areas [30].

Selecting appropriate materials for construction at the design stage is the first step towards waste minimization. The use of hazardous materials such as oils, flammable coatings or fibrous materials must be minimized, as these can become contaminated during normal operation and may also complicate waste management and disposal. The decommissioning workforce will need additional time and will encounter unnecessary hazards when dismantling or removing such materials because of the special precautions that will be required.

Ideally, operational waste handling and treatment systems will be designed to avoid the accumulation of large amounts of operational waste that must then be dealt with at the end of operations or before decommissioning starts. Waste handling and treatment facilities need to be available and sized for use during decommissioning. Alternatively, provisions can be built into these facilities to augment the capacity for increased demand caused by D&D. The use of mobile waste management facilities can also be considered as an alternative.

4.5.4.2. Selection of adequate characterization technique

It is essential to ensure that hazardous constituents are identified and that the waste is treated within the appropriate regulatory framework. Characterization is necessary to define a waste management system and to support its successful operation through, for example, the collection of data to allow demonstration of compliance with acceptance criteria for the treatment/storage/disposal system [12].

Waste characterization should be designed to include but not be limited to the following main components, for which more detailed information can be found in the literature [105]:

— *Process knowledge*: Process knowledge about waste generation is a cost effective and reliable method for predicting the probable constituents of a waste stream. It also may provide a basis for precluding consideration of a wide range of contaminants. For instance, if the waste stream itself does not include and has never been exposed to toxic organic solvents or compounds, then it is not necessary to provide for removal or destruction of such constituents. Process knowledge is especially important for defining the presence of hazardous components in the initial waste streams and for controlling their content through selected options of the waste management process.

— *Radio-assay results:* Radioanalyses provide information used for safe handling of a waste stream as well as for defining the waste category for treatment, conditioning, storage and disposal. The radiological properties of the waste stream or package determine whether manual handling (contact) is possible or remote handling is required. In addition, the activity levels of long lived radioisotopes must be ascertained to help to define the disposal option and, ultimately, to select possible disposal sites. Typically, the determinations will include:
  • Beta–gamma survey (to determine the waste handling category);
  • Gamma spectroscopy (to identify particular radionuclides);
  • Passive or active neutron interrogation (to determine the transuranic element content).

— *Intrusive sampling and analysis:* If chemically toxic substances may exist in a waste stream, an analysis needs to be undertaken to confirm the range of their concentrations. Such information is required to select the treatment option. Sampling of common waste streams that are homogeneous is generally straightforward. Obtaining a representative sample of a heterogeneous mixture may be very difficult. Simple sorting of waste into specific components and weighing the various fractions may provide adequate information if the composition of the sorted waste is known. In practice, intrusive sampling requirements are statistically based to minimize both the analytical costs and the hazard to the sampling personnel.

  Information required from intrusive sampling is entirely dependent on the type of waste and the expected treatment process. Common determinations for hazardous constituents may include:

  - Standard tests for flammability or ignitability;
  - pH control (for corrosivity);
  - Standard tests for reactive chemicals;
  - Standard tests to identify and quantify solvents and organic compounds;
  - Standard tests to identify and quantify toxic metals and complexes.
- *Radiographic examination:* Radiography is used to image waste packages as a means to confirm inventory information and to facilitate safe handling. Identification of unacceptable items in waste packages, such as aerosol cans and free liquids, is important for safe handling and for compliance with the acceptance requirements.

4.5.4.3. Selection of appropriate waste management techniques

Considerable information is provided in various IAEA publications on radioactive waste treatment and disposal, including the waste from decommissioning [5, 31, 106]. Most of the methods to minimize radiation exposure and facilitate dismantling described in the previous sections would also be effective in reducing decommissioning waste.

The major technical factors to be taken into account in selecting appropriate waste management technologies are waste characteristics, the scale of technology application, expected future needs, the maturity of the technology, the robustness of the technology, the range of technology applications, characteristics of the treated products, complexity and maintainability, volume reduction, the state of R&D, secondary waste and compatibility with existing processes, safeguards and nuclear safety, site availability and location, and potential for intrusion [107].

In addition, it is important that consideration be given at the design stage to those parts of the waste management provisions that would be of use at the decommissioning stage. This may typically include:

- Waste volume reduction facilities (e.g. those for incineration, compaction, melting);
- Facilities to treat and condition liquid waste generated from decommissioning (e.g. decontamination solutions);
- Waste transfer routes.

In handling waste from decommissioning, preference is to be given to waste facilities used during the plant's operational lifetime. However, consideration must also be given to providing additional waste management capabilities as needed during decommissioning. This may, as a minimum, concern allocation of adequate space.

Since large amounts of solid, high density waste are generated during the dismantling of nuclear facilities, it may be necessary to increase the capacity of the waste management system to avoid the unnecessary accumulation of unprocessed waste during decommissioning.

4.5.4.4. Selection of appropriate decontamination techniques

It is advisable that consideration be given to providing on-site decontamination facilities that are also suitable for decommissioning purposes. As decontamination requirements may be much greater during plant decommissioning than during plant operation, consideration ideally will be given to the management of secondary waste resulting from decontamination.

Three different approaches are possible to manage decontamination activities for decommissioning purposes, all of which need to be considered at the design stage:

- Construction of a specialized decontamination workshop;
- Allocation of space for specialized contractor services;
- Refurbishment of existing buildings for the installation of decontamination facilities.

Melting can also be considered as an effective technique for decontaminating, characterizing and recovering radioactive scrap metal, and is likely to be an integral component of many reuse/recycling alternatives [37]. Melting inherently produces a 'decontamination' effect on ^{137}Cs, a gamma emitter of particular concern. Caesium-137 is almost entirely removed during the melting process, because it volatilizes from the metal and accumulates in the dust collected by ventilation filters. Additionally, uranium and other oxides can be removed from the metal in the slag, thereby reducing the quantity of alpha emitters. Cobalt-60 and other fission by-products dominate the remaining nuclides. The short half-life of these nuclides (5.3 a for ^{60}Co) permits consideration of alternatives that otherwise would be precluded by the presence of ^{137}Cs.

Melting also simplifies procedures for radioactive metal characterization and eliminates difficulties associated with inaccessible surfaces, since any remaining radioactivity content is homogenized over the total mass of the ingot. For this reason, melting can be the last step in the decontamination and clearance of components with complex geometries. Chemical methods that remove radionuclides that otherwise would remain in ingots after melting can be also used to decontaminate material prior to melting.

4.5.4.5. Selection of adequate waste management facilities

Effective radioactive waste management depends on the national waste management strategy [27]. Radioactive waste management operations may be carried out at the source (i.e. on the waste generator's premises (local waste management)), at a centralized facility (centralized waste management) or at both facilities.

Approval from the regulatory body is required before proceeding with local waste management operations. This will ensure consistency with the national waste management strategy, compliance with national laws and regulations, and the availability of adequate resources.

Local waste management operations are only to be implemented after personnel have been adequately trained, and once the operators and managers of the waste management facility are fully acquainted with their responsibilities. The extent of local waste management will depend on the national waste management strategy and on the particular application of radioactive material. It is important that the scope of local waste management be established for each waste generator. Local waste management can include a full range of operations, such as waste minimization, segregation, characterization, treatment, conditioning, storage and disposal. However, as a minimum, waste segregation, basic characterization, minimization and storage need to be applied. It is essential that full documentation be produced and retained (waste characteristics, origin, treatment and conditioning methods, etc.), and that adequate safety precautions be applied at all stages of local waste management.

The construction and operation of a centralized waste management facility are subject to national laws and regulations, and the facility needs to be subject to regular inspections by the regulatory body.

Waste generators should request from the operator of a centralized waste management facility the necessary information relating to requirements for waste segregation, treatment and packaging. Operations carried out by the centralized waste management facility will potentially include:

— Operational storage;
— Treatment and conditioning;
— Interim storage of conditioned waste.

If no disposal route for waste exists, interim storage of operational and decommissioning waste will be required until an adequate disposal route is available. In the interest of reducing the size and extent of residual structures left during the waste storage period, it is desirable to utilize or convert existing space or volumes within the existing buildings. Consideration of this reuse of space may begin during the design of the initial plant layout, when these areas should be identified. This approach is more desirable than the construction of new interim waste storage buildings. Another advantage of interim waste storage is that it allows radioactive material to decay to below clearance levels.

More technical details relating to the storage and processing options at a centralized waste management facility are presented in specific technical reports [108, 109].

4.5.4.6. Use of mobile waste treatment systems

The use of mobile pretreatment, treatment and conditioning systems may be incorporated as part of the plant design for new nuclear plants [110]. Such systems can be delivered to a site, operated on a campaign basis (e.g. to recover some of the storage capacity of a long term accumulation of drummed and stored waste) and then moved for use at another site.

The application of mobile waste treatment systems has the added advantage of flexibility in choosing an optimum technology and waste management approach according to the actual needs of a specific plant or country and adopting new technologies, as they emerge, without the high implementation cost. It may be of particular interest to those plants that will not produce sufficient volumes of waste to justify the large expenditures required for some high efficiency technologies, providing the opportunity to deploy specialist teams of operators from the waste treatment plants, rather than having to train operators at each plant.

Mobile waste treatment systems may also be of particular interest for optional evaporator use in PWRs for boron recovery, and for treatment of the charcoal masses from gaseous radioactive waste processing systems when incinerators are not available. In some countries, such a strategy may also deliver benefits in terms of ease of licensing, while in other countries, licensing and repeated licensing procedures may require time consuming and expensive planning activities. For some countries, mobile waste treatment systems may not be feasible, because current regulations require that the equipment involved belong only to a specific nuclear site.

Some waste treatment processes cannot be made mobile or transportable. Some equipment designs would have to be compromised in order to make them mobile.

An additional consideration is that the transport costs of mobile waste treatment systems can be high, depending on the size of that equipment.

Consideration also needs to be given to the decontamination costs of the transport equipment.

Existing facilities may have difficulty adopting mobile waste treatment systems owing to plant design restrictions. Efforts to accommodate a mobile treatment system to the requirements of a number of existing plants may result in costly modifications. Examples of some support requirements and considerations that impact the application of mobile systems are:

— Material interfaces, for example, for the removal of sludge, ion exchange resins, charcoal, total organic carbon, detergent wastes, sump and floor drains to the mobile unit or an external treatment facility;
— System interfaces, for example, high power demand, remote power demand, water requirements, instrument air requirements, drains, gaseous effluent controls and support, vehicle exhaust, communications;
— Adjacency factors, for example, the impact of elevated dose rates on nearby work or other mobile systems, required shielding, space required for input and output of waste, required remote handling techniques, crane requirements, crane overhead clearance;
— Other general access factors, for example, vehicle movement interference (e.g. nearby fences, buildings, power lines), available land space, radiation monitoring, security, floor loading (how much weight can be placed on the floor per square foot or square meter).

In any case, a central radioactive waste processing building will remain in common use for collection and/or pretreatment and handling of the produced waste materials.

In general, the selection of a strategy will be the result of a well developed global feasibility study including an overall cost–benefit analysis.

4.5.5. Design for safe enclosure and deferred dismantling

It is important to recognize that parts of the nuclear power plant will remain in situ for some years following the shutdown of the plant, regardless of the decommissioning strategy [30]. During this time, which may include a safe enclosure period, the plant components, systems and buildings or other structures will need to be retained in a safe condition, and their integrity and functional capability will need to be maintained at an appropriate level. The environmental conditions within a shutdown plant will be different from those associated with operational plants, and this will need to be taken into account when considering the post-shutdown requirements.

In addition, the decommissioning strategies of some Member States may include a period of safe enclosure prior to the completion of dismantling of a nuclear power plant [30]. The periods considered vary from a few years to several decades. If a safe enclosure period is likely to be included in the decommissioning strategy, it is then appropriate to consider this issue at the design stage, so that any special requirements can be addressed.

The primary requirement during a safe enclosure period is to preserve those buildings to be retained, and the plant they contain, in a safe and secure condition, and to maintain their integrity throughout the whole period. If a safe enclosure period is to be considered, the full design life of the residual plant and buildings, including the safe enclosure period, may be significantly longer than the operational period. Furthermore, the environmental conditions (e.g. temperature and humidity) during the safe enclosure period may be significantly different from those experienced during the operational period. This means that the degradation mechanisms that require consideration are different from those during the operational period; for example, corrosion may become a significant issue.

The residual radioactivity remaining on the site during a safe enclosure period will be predominantly contained within various plant components, for example, the reactor vessel and heat exchangers. Corrosion of these components will be the primary mechanism behind any potential reduction of their containment integrity. Corrosion could result from increased humidity created when the reactor is shutdown and cold. Therefore, a sufficient corrosion allowance should be included in their design to cover both the operational and the safe enclosure periods. Alternative approaches could be to maintain low humidity conditions and/or use protective coatings. Attention must also be paid to the long term integrity of the supports for items such as vessels, pipework and ducts to ensure that they do not fail during the safe enclosure period.

The capability to fully drain, flush and dry out the vessels and associated pipework also needs to be considered at the design stage to ensure that any liquids, chemicals or contaminants are removed before the start of the safe enclosure period. Similarly, insulating materials that may be hygroscopic and may result in enhanced corrosion during the safe enclosure period are to be avoided if possible, or designed to be readily removable before the start of the safe enclosure period. Means for performing surveillance of the critical components and features, where required, during the safe enclosure period need to be considered, for example, design of corrosion samples relevant specifically to this period.

In addition, it is preferable to avoid the need for plant systems to remain in service during the safe enclosure period. However, it may be necessary to operate some systems during this period and/or during the deferred dismantling. This requirement could be addressed by utilization of the existing systems or may require installation of a new service plant. It is also important that consideration be given at the design stage to possible long term requirements for any plant systems and for incorporation of helpful features, where appropriate.

Fire detection and suppression, radiological and environmental monitoring, compressed air, drainage and waste treatment facilities, lighting and power supply, ventilation and service water are some of plant systems whose availability the regulatory body may require during different decommissioning stages. Design of these systems needs to be reviewed accordingly, and any limitation on their operational lifetime is to be clearly specified.

One particular design criterion for plants subject to extended periods of safe enclosure is related to the electrical distribution system that will be needed for decommissioning. Some thought needs to be given to providing for a separate system for supply of the essential services needed during decommissioning — for example, cranes, sump pumps, ventilation, lighting and any other requirements. While the main electrical distribution systems of the plant are being removed, which could be as early as the commencement of decommissioning, the system identified for decommissioning will need to remain in service without any interference from the other systems being removed. Several shutdown nuclear plants have had to rebuild electrical distribution systems to accommodate decommissioning requirements.

It will also be necessary to maintain the building structures in a sound condition throughout the safe enclosure period. These structures are likely to incorporate various construction materials with a variety of potential long term degradation mechanisms (e.g. carbonation and chloride penetration of concrete as well as corrosion of steelwork). The structural components of buildings may be required to perform a variety of functions including structural support, containment, weather protection and protection against water infiltration. These functions will be required to continue throughout the safe enclosure period.

Consideration needs to be given at the design stage to the potential long term integrity of the building structures that will be required during the safe enclosure period. Structural weakening of the buildings needs to be avoided, radioactivity containment must be maintained and the buildup of water that could potentially result in the spread of contamination needs to be minimized. Water infiltration could result from rain or surface water and from groundwater in-leakage through floors and basements, all of which are to be avoided. Features for continued water collection and drainage must be included. It may be possible to prevent groundwater in-leakage into buildings by ensuring that any susceptible features are above the local groundwater table.

Periodic monitoring of all safety related components of the plant must be incorporated into the decommissioning plan [42].

4.6. DEVELOPMENT AND IMPROVEMENT OF THE REGULATORY APPROACH

Substantial quantities of contaminated materials (predominantly steel and concrete) are likely to be generated during the decommissioning and dismantling of nuclear facilities. Without an adequate waste minimization strategy, which includes having acceptable clearance standards, these potentially valuable materials cannot be systematically recovered from the radioactive waste through decontamination and/or reuse or recycling practices. A significant portion of this material is not or is only slightly contaminated with radioactivity. Disposal of radioactive scrap metals currently relies on disposal at licensed low level waste disposal facilities or, less commonly, on clearance on the basis of a detailed evaluation.

The availability of national and corporate policies, and global long term strategies in support of waste minimization principles, in which clearance of material and reuse and recycling options may play a major part, can have a profound impact on the efficiency and extent of waste minimization practices [4]. A coherent dialogue among legislators, competent authorities and the public must support these practices to gain acceptance for waste minimization through clearance practices, and to promote options for reuse and recycling of materials rather than for their restriction. In the absence of a national policy promoting reuse and recycling, practitioners need to take the initiative in providing input into policy development (i.e. using acceptable principles and the results from actual demonstration projects).

However, the practice of releasing materials varies, depending on a number of factors. Some present indications are that practices that take into consideration major environmental impacts and nonradiological health effects, in addition to radiological health risks, strongly support reuse and recycling options. A recent comparison of the relative merits of disposal and replacement versus reuse and recycling practices shows that the latter produce lower health risks to humans and reduce environmental impacts by more than a factor of two [37]. This approach also has the advantage of matching acceptable decommissioning strategies with proposed waste minimization options while keeping risk to the public at an appropriately low level.

In addition, a strong case can be made that waste minimization and material recycling standards need to be developed within the broad context of health risks due to radioactivity in the environment and the potential hazards posed by the relatively large amounts of unregulated, naturally occurring radioactive materials dealt with in several other industries.

A technological basis for implementing the criteria is also an integral component of the process. Measurement capability for surface activity on components depends on the contamination mechanism (e.g. wet or dry), on surface characteristics (roughness, chemistry and material), on decontamination methods and on the type of wipe test applied.

On the whole, a global waste minimization strategy supported by adequate reuse and recycling practices requires a set of acceptable international clearance standards. These standards should be based on realistic scenarios that make use of available data from actual examples. As such, further research is needed to calibrate/validate the models and calculations used to derive risk based clearance levels [111]. This should be based on data derived from existing practices, in order that excessive and costly conservatism can be avoided.

In addition, careful consideration should be given to public acceptance of the practice of recycling materials derived from D&D. Policies that bring public acceptance of reuse and recycling practices into line with public perceptions of risk related to products containing radioactive materials (e.g. smoke detectors) should be developed and supported. Public perceptions are influenced by familiarity with the product, the associated benefits and the extent to which the radioactive aspects of the product are publicized.

4.7. NUCLEAR FACILITY DESIGN BASIS GUIDELINES FOR WASTE MINIMIZATION

As indicated in the preceding sections, it is advisable that many process options be considered to improve the design of a nuclear facility with a view to minimizing waste. Table 10 provides a brief summary of some of the process options to be considered. Table 11 provides examples of factors to be considered in the evaluation of design options.

When evaluating the various influencing factors for a specific process option, a decision matrix approach can be adopted to allow simultaneous evaluation of several options and influencing factors. Using this method, a matrix can be constructed of the various options for waste minimization at the design stage of an installation versus the applicable influencing factors for the overall project. Moreover, a weighting value attributed to each factor can be used as a multiplier for the scores of individual factors to reflect the priorities identified in a specific project. Adopting various values for these weighting factors allows for sensitivity analyses to resolve the most critical influences.

TABLE 10. OPTIONS FOR PROMOTING WASTE MINIMIZATION TO BE CONSIDERED AT THE DESIGN STAGE OF NEW NUCLEAR FACILITIES OR PRIOR TO FACILITY MODIFICATIONS

Subject	Topic	Option
Design for decommissioning	Decommissioning plan	• Prepare detailed decommissioning plan at design stage • Other
	Timescales	• Plant and system lifespan • Plant and system replacement during facility life • Reliability • Safe enclosure (deferred decommissioning) and ALARA • Other
	Location	• Consider locating plant with other nuclear facilities • Geology, particularly groundwater • Environmental baseline • Equipment to measure environmental impact, boreholes • Other
	Building layout/ plant	• Provide sufficient space for future decommissioning plant • Size openings to suit largest plant • Consider plant removal when sizing structural members, cranes, etc. • Leave lifting attachments • Consider modular construction • Personnel access to consider decommissioning strategy, dose uptake • Layout in cells amenable to remote operation • Segregation of process equipment according to operation and radioactivity • Minimize quantities of systems, e.g. pipes, tanks, cables, etc. • Waste products, special waste • Interim waste storage areas, waste treatment • Provisions for decontamination or dismantling, e.g. holes in concrete for bursters • Preplacement of dismantling aids • Installed access for monitoring equipment, viewing systems and investigation for inaccessible areas • Other

TABLE 10. OPTIONS FOR PROMOTING WASTE MINIMIZATION TO BE CONSIDERED AT THE DESIGN STAGE OF NEW NUCLEAR FACILITIES OR PRIOR TO FACILITY MODIFICATIONS (cont.)

Subject	Topic	Option
	Layout of ventilation systems	• Anti-backflow gadgets • Upstream filters on the supply air system • Minimize contamination traps • Include access for cleaning • Decommissioning capability • Other
	Layout of piping/ tanks	• Pipe routing through embedded metal enclosure • Pipe within a pipe or in lined trench • Storage tanks and equipment • Bonded systems to contain spillage • Minimize contamination traps, avoid internal structures • Include connections for decontamination • Ability to drain and store contents • Purification systems • Other
	Records	• Maintain construction records, as built drawings, materials • Record all plant modifications • Record all plant operational history • Organize records for longevity and recovery • Knowledge capture from operations staff • Other
	Materials	• Minimize hazardous, flammable and porous materials • Consider activation properties of chemical components of materials • Suitability, radiation resistance, corrosion, durability, etc. • Maintain samples • Other

TABLE 10. OPTIONS FOR PROMOTING WASTE MINIMIZATION TO BE CONSIDERED AT THE DESIGN STAGE OF NEW NUCLEAR FACILITIES OR PRIOR TO FACILITY MODIFICATIONS (cont.)

Subject	Topic	Option
Design for decontamination		• Decontamination techniques • Decontaminable surfaces: stainless steel linings, coatings (epoxies, strippable coatings), steel floors, grillages, electropolishing, surface hardening of concrete, mix design • Ventilated containment • Segregated areas • Install decontamination systems, e.g. lances in tanks • Plan for operational decontamination of systems/plant • Equipment designed to be easily decontaminated • Other
Preparation of the operational culture		• Workplace design and working conditions • Training of staff to minimize operational waste • Training/culture to avoid spillages and incidents that spread contamination • Maintenance designed to reduce or remove contamination and lower dose rates • Other
Use of improved techniques for D&D	Use of improved decontamination techniques	• Microbiological degradation • Light ablation • Aggressive chemical process • Other
	Use of improved dismantling techniques	• Lasers • Electrolytic cutting • Other
	Use of improved measurement techniques	• (Insert specific options for consideration)
Simplification of waste management	Selection of adequate characterization techniques	• (Insert specific options for consideration)
	Selection of appropriate waste management techniques	• (Insert specific options for consideration)

TABLE 10. OPTIONS FOR PROMOTING WASTE MINIMIZATION TO BE CONSIDERED AT THE DESIGN STAGE OF NEW NUCLEAR FACILITIES OR PRIOR TO FACILITY MODIFICATIONS (cont.)

Subject	Topic	Option
Safe enclosure and deferred dismantling	(Insert specific topics for consideration)	• (Insert specific options for consideration)
Other	(Insert specific topics for consideration)	• (Insert specific options for consideration)

TABLE 11. FACTORS TO BE CONSIDERED IN THE EVALUATION OF DESIGN OPTIONS

Factor	Objective
Objectives	Maximize safety and demonstrate ALARP Minimize LLW packing Exercise basic project management in the processing of waste Cost effective solution Other
Criteria	Technical Environment Safety Cost Stakeholder (regulators, site owner/operator, local community, etc.) considerations Other
Technical	Feasibility Track record Reliability Project risk Compatibility with existing plant/processes Ease of implementation, including training Skill base/knowledge management Other

TABLE 11. FACTORS TO BE CONSIDERED IN THE EVALUATION OF DESIGN OPTIONS (cont.)

Factor	Objective
Environment	Volume of ILW generated Volume of LLW generated Volume/nature of hazardous wastes Volume of secondary wastes Consideration of flora and fauna Discharges to the environment (aqueous and gaseous) Recycling potential Utility usage (water, steam, compressed air, electricity, etc.) Impacts on local environment (accommodation, road use, etc.) Other
Safety	Worker safety — radiological Worker safety — conventional Worker safety — chemical, biological, hazardous (e.g. explosives, asbestos) Public safety Safety of other on-site workers Plant safety Other

The final result of such an analysis would be a relative, numerical ranking of the options (the score for each option).

A matrix evaluation of a reprocessing plant ventilation system is given in Table 12. The process options are technically feasible and result in increased safety and reduced environmental impacts, but implementation may result in increased costs for the facility.

TABLE 12. TYPICAL DECISION MATRIX WITH DESIGN OPTIONS

Process option	Technical feasibility		Safety evaluation		Ease of D&D		Environmental impact		Cost		Final score
	Weight	Score	Weight	Score	Weight	Score	Weight	Score	Weight	Score	
Construction materials											
– Stainless steel	V_1 %	C_1	W_1 %	F_1	X_1 %	R_1	Y_1 %	D_1	Z_1 %	I_1	Σ_1 %
– Carbon steel	V_2 %	C_2	W_2 %	F_2	X_2 %	R_2	Y_2 %	D_2	Z_2 %	I_2	Σ_2 %
– Fibre reinforced plastic	V_3 %	C_3	W_3 %	F_3	X_3 %	R_3	Y_3 %	D_3	Z_3 %	I_3	Σ_3 %
– Galvanized steel	V_4 %	C_4	W_4 %	F_4	X_4 %	R_4	Y_4 %	D_4	Z_4 %	I_4	Σ_4 %
Surface treatment											
– Epoxy painting	V_5 %	C_5	W_5 %	F_5	X_5 %	R_5	Y_5 %	D_5	Z_5 %	I_5	Σ_5 %
– Electropolishing	V_6 %	C_6	W_6 %	F_6	X_6 %	R_6	Y_6 %	D_6	Z_6 %	I_6	Σ_6 %
Routing through metallic embedment	V_7 %	C_7	W_7 %	F_7	X_7 %	R_7	Y_7 %	D_7	Z_7 %	I_7	Σ_7 %
Backflow prevention	V_8 %	C_8	W_8 %	F_8	X_8 %	R_8	Y_8 %	D_8	Z_8 %	I_8	Σ_8 %
Pre-filter and supply air	V_9 %	C_9	W_9 %	F_9	X_9 %	R_9	Y_9 %	D_9	Z_9 %	I_9	Σ_9 %
Filter flow duct pass from low active to high active areas	V_{10} %	C_{10}	W_{10} %	F_{10}	X_{10} %	R_{10}	Y_{10} %	D_{10}	Z_{10} %	I_{10}	Σ_{10} %
Provision of cleaning	V_{11} %	C_{11}	W_{11} %	F_{11}	X_{11} %	R_{11}	Y_{11} %	D_{11}	Z_{11} %	I_{11}	Σ_{11} %
Other	V_{12} %	C_{12}	W_{12} %	F_{12}	X_{12} %	R_{12}	Y_{12} %	D_{12}	Z_{12} %	I_{12}	Σ_{12} %

5. CONCLUSIONS

As the requirements for decommissioning and its costs have become better understood, the nuclear industry has grown increasingly aware of the importance of including decommissioning considerations at the design stage of new nuclear facilities. The objectives are to reduce worker exposure, to minimize waste generation and to simplify dismantling procedures for decommissioning. These objectives must not conflict with the primary objective of the facility, which is safe and efficient operation of the plant.

Some design and construction features to facilitate decommissioning may result in significant cost savings, especially if they also benefit plant operation and maintenance. However, it must also be recognized that the provisions of such design features can be mutually conflicting. A cost–benefit analysis can assist in the selection of these features.

In addition, there are regulatory/licensing requirements that demand that decommissioning be considered at the design stage, particularly with respect to minimizing waste arisings and facilitating access for dismantling. Well established plans for decommissioning at the design stage will also provide assurance to the public concerning the environmental impacts of all aspects of nuclear power.

This report is aimed at a broad spectrum of those involved in the definition, design and operation of new nuclear facilities, including design engineers, builders, owners, operators, regulators and authorities. It provides a set of options for consideration during the design and operation of nuclear facilities handling radioactive materials to optimize the management of their operational and decommissioning waste and to facilitate their safe, effective and timely decommissioning.

Several options regarding the minimization of radioactive waste production have been formulated for consideration when designing new facilities, modifying existing plants or defining future D&D operations. These options can be summarized as follows:

— *Considerations to minimize contamination problems:* A variety of design features and techniques to reduce or prevent contamination of components and minimize associated problems have been recognized. Among the aspects discussed are that the building layout and location need to be designed to prevent the buildup and spread of contamination; system components are to be of reliable and robust construction and made from low activation materials; systems are to be designed with filters and purifiers to remove radioactive corrosion products; surfaces are to be designed to be easily decontaminated and systems designed to limit the buildup of contamination; and quality control is very important.

- *Provisions to facilitate decontamination:* During the design of nuclear facilities, it is important that ways to facilitate the cleanup of contamination be assessed and, if possible, incorporated into the design. It is advisable that the building layout provide adequate access to equipment requiring decontamination, and there is a need for built-in provisions for decontamination equipment. Process systems are to be routinely chemically decontaminated at fairly frequent intervals under regular maintenance arrangements. Management arrangements need to ensure that the operating team is fully trained to minimize the spread of contamination during operation, maintenance and decommissioning work. Examples of methods to aid decontamination have been included, for example, the use of easily decontaminated liners and coatings in areas expected to become contaminated and recommendations concerning the design of tanks, pipes, components and systems.
- *Provisions to facilitate dismantling and segmentation:* Facility design needs to take into account features that would facilitate dismantling and segmentation during both operation and decommissioning. A variety of concepts have been proposed for inclusion at the design and construction stages to facilitate the dismantling, removal and/or segmentation of components or equipment.
- *Documentation:* It is noted that dismantling and demolition can be greatly facilitated if good records providing details of design and construction, operating history, plant modifications and maintenance experience are available. The knowledge of operational staff will be beneficial during decommissioning and should also be captured and retained. The records systems need to be designed for long term maintainability and readability, perhaps for as long as one hundred years.
- *Planning:* Outlining of the decommissioning plan ideally will begin early in the facility design stage, and decommissioning plans need to be reviewed on a regular basis during design and the subsequent operational lifetime, and/or as required by the regulatory body. The operators then must ensure that an acceptable decommissioning strategy and detailed plans are maintained and updated to facilitate decontamination and reduce occupational exposure during decommissioning. Whether or not these are applied will depend on many factors, such as the cost versus the benefits and engineering practicality.

- *Development and improvement of techniques for decontamination and decommissioning:* Many techniques for decontamination, dismantling, monitoring and waste management are mature and widely used today. However, the designer must be aware of the work being undertaken to develop and improve existing technologies, including the use of modular, mobile or external technological units. Considering the long design life of a new facility, current technical options might be substantially improved over the course of the facility's lifetime. Thus, such prospective options ideally will be examined and considered in the design process, as they may become available in the future.
- *Design for safe enclosure and deferred dismantling:* It is noted that parts of the nuclear facility will remain in situ for some years following shutdown, regardless of the decommissioning strategy, and that these periods may vary from a few years to several decades. During this time, which may include a safe enclosure period, the plant components, systems and buildings or other structures must be retained in a safe condition, and their integrity and functional capability must be maintained at an appropriate level. Thus the full design life of the residual plant and buildings may be significantly longer than the operational period and must be taken into account during the design process.
- *Development and improvement of the regulatory approach:* Substantial quantities of contaminated materials (predominantly steel and concrete) will be generated from the decommissioning of nuclear facilities. These valuable materials can be systematically recovered from the radioactive waste management system for reuse and recycling. However, national, corporate and global long term strategies in support of waste minimization principles are immature. These strategies will need to be developed through a coherent dialogue among designers, operators, legislators, regulators, competent authorities and the public in order to gain acceptance for waste minimization through release practices and to promote opportunities for the reuse or recycling of materials.
- *Nuclear facility design basis guidelines for waste minimization:* Three examples of tabular analysis techniques used to assess options during the design process are presented: (i) examples of process options to be considered at the design stage in order to minimize contamination problems, (ii) factors to be considered for the evaluation of design options and (iii) a generic decision matrix for analysis of design options.

These conclusions are derived from a review of the lessons learned from the operational and decommissioning experience gained by Member States to date. While actual plant designs will continue to mature and evolve in the future, the waste minimization options that have been identified will remain relevant to all new facilities and can be used as a checklist during the design, licensing and operational phases of new plants and the modification of the existing plants.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Principles of Radioactive Waste Management, Safety Fundamentals, Safety Series No.111-F, IAEA, Vienna (1995).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Minimization of Waste from Uranium Purification, Enrichment and Fuel Fabrication, IAEA-TECDOC-1115, IAEA, Vienna (1999).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning Techniques for Research Reactors, IAEA-TECDOC-1273, IAEA, Vienna (2002).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Methods for the Minimization of Radioactive Waste from Decontamination and Decommissioning of Nuclear Facilities, Technical Reports Series No. 401, IAEA, Vienna (2001).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Managing Low Radioactivity Material from the Decommissioning of Nuclear Facilities, Technical Reports Series No. 462, IAEA, Vienna (2007).

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

[7] OECD NUCLEAR ENERGY AGENCY, The Decommissioning and Dismantling of Nuclear Facilities: Status, Approaches, Challenges, OECD, Paris (2002).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Minimization of Radioactive Waste from Nuclear Power Plants and the Back End of the Nuclear Fuel Cycle, Technical Reports Series No. 377, IAEA, Vienna (1995).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles: Report of Phase 1A of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1362, IAEA, Vienna (2003).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Heavy Water Reactors: Status and Projected Development, Technical Reports Series No. 407, IAEA, Vienna (2002).

[11] Innovative Technologies for Nuclear Fuel Cycles and Power Reactors (Proc. Int. Conf. Vienna, 2003), IAEA, Vienna (2003) CD-ROM.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Low and Intermediate Level Radioactive Wastes with Regard to Their Chemical Toxicity, IAEA-TECDOC-1325, IAEA, Vienna (2002).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Abnormal Radioactive Wastes at Nuclear Power Plants, Technical Reports Series No. 307, IAEA, Vienna (1989).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Recycle and Reuse of Materials and Components from Waste Streams of Nuclear Fuel Cycle Facilities, IAEA-TECDOC-1130, IAEA, Vienna (2000).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology and Technology of Decommissioning Nuclear Facilities, Technical Reports Series No. 267, IAEA, Vienna (1986).
[16] OECD NUCLEAR ENERGY AGENCY, Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installations Decommissioning Projects: Achievements of the NEA Co-operative Programme 1985–1990, OECD, Paris (1992).
[17] COMMISSION OF THE EUROPEAN COMMUNITIES, Die Aktivierung des biologischen Schildes im stillgelegten Kernkraftwerk Lingen, Rep. EUR 8801, Office for Official Publications of the European Communities, Luxembourg (1985).
[18] COMMISSION OF THE EUROPEAN COMMUNITIES, Compendium on the State-of-the-Art in Decommissioning and Decontamination, EC Contract FIKWT-CT-2001-20134, Office for Official Publications of the European Communities, Luxembourg (2005) CD-ROM.
[19] CREGUT, A., ROGER, J., Inventory of Information for the Identification of Guiding Principles in the Decommissioning of Nuclear Installations, Rep. EUR 13642 EN, Office for Official Publications of the European Communities, Luxembourg (1991).
[20] SCHNEIDER, K.J., JENKINS, C.E., RHOADS, R.E., Technology, Safety and Costs of Decommissioning — a Reference Nuclear Fuel Reprocessing Plant, Rep. NUREG-0278, Battelle Pacific Northwest Lab., Richland, WA (1977) 7–45.
[21] SMITH, G.M., HEMMING, C.R., CLARK, M.J., CHAPUIS, A.M., GARBAY, H., Methodology for Evaluating Radiological Consequences of the Management of Very Low Level Solid Waste, Rep. EUR 10058, Office for Official Publications of the European Communities, Luxembourg (1985).
[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Handling of Tritium-Bearing Wastes, Technical Reports Series No. 203, IAEA, Vienna (1981).
[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Facilities: Decontamination, Disassembly and Waste Management, Technical Reports Series No. 230, IAEA, Vienna (1983).
[24] OECD NUCLEAR ENERGY AGENCY, Decommissioning Nuclear Power Plants: Policies, Strategies and Costs, OECD, Paris (2003).
[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Radioactive Waste from the Use of Radionuclides in Medicine, IAEA-TECDOC-1183, IAEA, Vienna (2000).
[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Minimization and Segregation of Radioactive Wastes, IAEA-TECDOC-652, IAEA, Vienna (1992).
[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Small Quantities of Radioactive Waste, IAEA-TECDOC-1041, IAEA, Vienna (1998).
[28] PALMER, J., "Acceptance issues for large items and difficult waste" in Working Party on Decommissioning and Dismantling (WPDD), Topical Session on Materials Management, NEA/RWM/WPDD(2002)7, OECD/NEA, Paris (2002).

[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning of Nuclear Facilities, Technical Reports Series No. 286, IAEA, Vienna (1988).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Design and Construction of Nuclear Power Plants to Facilitate Decommissioning, Technical Reports Series No. 382, IAEA, Vienna (1997).

[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Decontamination of Nuclear Facilities to Permit Operation Inspection, Maintenance, Modification or Plant Decommissioning, Technical Reports Series No. 249, IAEA, Vienna (1985).

[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Quality Assurance for Safety in Nuclear Power Plants and other Nuclear Installations, Code and Safety Guides Q1–Q14, Safety Series No. 50-C/SG-Q, IAEA, Vienna (1996).

[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for Optimizing Nuclear Power Plant Maintenance Programmes, IAEA-TECDOC-1383, IAEA, Vienna (2003).

[34] BALAKRISHNA, M., "Growth scenarios with thorium fuel cycles in pressurized heavy water reactors", Thorium Utilization (Proc. Indo-Japan Sem., Bombay, 1990), Indian Nuclear Society, Mumbai (1991) 62–66.

[35] HOBBS, J., Spent fuel: Sierra Nuclear Corporation–BNFL Transfer (TM), J. Nucl. Eng. Int. **41** (1996) 38.

[36] BAUMANN, S., TEUNCKENS, L., WALTHERY, R., LEWANDOWSKI, P., MILLEN, D., "Concrete crushing and sampling: A methodology and technology for the unconditional release of concrete material from decommissioning", paper presented at 8th Int. Conf. on Environ. Management, Bruges, 2001.

[37] OECD NUCLEAR ENERGY AGENCY, Recycling and Reuse of Scrap Metals, OECD, Paris (1996).

[38] THE LEAGUE OF WOMEN VOTERS EDUCATION FUND, The Nuclear Waste Primer: A Handbook for Citizens, Nick Lions Books, New York (1985).

[39] SCHWINKENDORF, W., et al., Alternatives to Incineration, Technical Area Status Report, DOE/MWIP-26, USDOE, Washington, DC (1995).

[40] PERKINS, B.L., Incineration Facilities for Treatment of Radioactive Wastes: A Review, Rep. LA-6252, Los Alamos National Laboratory, NM (1976).

[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. WS-G-2.4, IAEA, Vienna (2001).

[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

[43] UNITED STATES DEPARTMENT OF ENERGY, DOE Handbook: Design Considerations, DOE-HDBK-1132-99, USDOE, Washington, DC (1999).

[44] COMMISSION OF THE EUROPEAN COMMUNITIES, Trace Element Assessment of Low-Alloy and Stainless Steel with Reference to Gamma Activity, Rep. EUR 9264 en, Office for Official Publications of the European Communities, Luxembourg (1984).

[45] NORDIC LIAISON COMMITTEE FOR ATOMIC ENERGY, Some Studies Related to Decommissioning of Nuclear Reactors, NKA Project KAV 350, SSI, Stockholm (1990).
[46] NUCLEAR REGULATORY COMMISSION, Long-lived Activation Products in Reactor Materials, Rep. NUREG/CR-3474, NRC, Washington, DC (1984).
[47] WIELING, N., HOFMANN, P., Erosions- und Korrosionsversuche mit Kobalt freien Werkstoffen. Rep. EUR 9865, Office for Official Publications of the European Communities, Luxembourg (1985).
[48] BOOTHBY, R.M., WILLIAM, T.M., The control of cobalt content in reactor grade steels, Eur. Appl. Res. Rep., Nucl. Sci. Technol. Sect. 5, No. 2 (1983).
[49] GARBETT, K., “Overview of the impact of stellite removal on radiation fields in KWU PWRs”, Water Chemistry of Nuclear Reactor Systems (Proc. Int. Conf. Bournemouth, 1992), Vol. 2, British Nuclear Energy Society, Ashford Press, Southampton (1992) 31–38.
[50] DUBOURG, M., “Design improvements to facilitate decommissioning”, Decommissioning of Radioactive Facilities (Proc. Int. Conf. London, 1992), Mechanical Engineering Publications Ltd., London (1992) 143–147.
[51] OCKEN, H., WOOD, C.J., “Status report on radiation exposure reduction at US nuclear power plants”, Water Chemistry in Nuclear Reactor Systems (Proc. Int. Conf. Bournemouth, 1992), Vol. 1, British Nuclear Energy Society, Ashford Press, Southampton (1992) 1–8.
[52] DUBOURG, M., “Design features adopted to facilitate decommissioning”, Decommissioning of Nuclear Installations (Proc. Int. Conf. Brussels, 1989), Office for Official Publications of the European Communities, Luxembourg (1990).
[53] BITTNER, A., et al., Konzepte zur Minimierung der Aktivierung des Biologischen Schilds, Rep. EUR 9442, Office for Official Publications of the European Communities, Luxembourg (1985).
[54] VERRAL, S., FITZPATRICK, J., Design Concepts to Minimize the Activation of the Biological Shield of Light-Water Reactors, Rep. EUR 8804, Office for Official Publications of the European Communities, Luxembourg (1985).
[55] IRVING, B.A., “Relationship between effective decontamination, contamination control principles and decommissioning”, Nuclear Reactor Decommissioning Planning (Proc. Int. Conf. Bethesda, MD, 1985), Rep. NUREG/CP-0068, Nuclear Regulatory Commission, Washington, DC (1986) 432–435.
[56] COMMISSION OF THE EUROPEAN COMMUNITIES, Recherche et caractérisation de revêtements pour la protection des structures en béton, Rep. EUR 9595, Office for Official Publications of the European Communities, Luxembourg (1985).
[57] COMMISSION OF THE EUROPEAN COMMUNITIES, Rivestimenti rimovibili per la protezione di superfici in calcestruzzo dalla contaminazione, Rep. EUR 9463 it, Office for Official Publications of the European Communities, Luxembourg (1984).
[58] DUBOURG, M., “Design improvements to facilitate decommissioning”, Nuclear Power Option (Proc. Int. Conf. Vienna, 1994), IAEA, Vienna (1994).

[59] PICK, M.E., “PWR oxide characterization/surface finish improvement and development of decontamination processes for gas and water cooled reactors”, Decontamination and Decommissioning of Nuclear Facilities (Results of a Co-ordinated Research Programme, Phase II: 1989–1993), IAEA-TECDOC-716, IAEA, Vienna (1993) 185–192.
[60] PICK, M.E., YOUNG, M.A., “Effect of surface treatment on radioactivity deposition on stainless steel coupons exposed in DOEL 2”, Water Chemistry of Nuclear Reactor Systems (Proc. Int. Conf. Bournemouth, 1992), Vol. 2, British Nuclear Energy Society, Ashford Press, Southampton (1992) 74–79.
[61] NUCLEAR REGULATORY COMMISSION, Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors, Rep. NUREG/CR-3587, NRC, Washington, DC (1986).
[62] ENTE NAZIONALE PER L’ENERGIA ELECTTRICA, Criteria for Active Effluent Reduction of the Source, Design Guide No. 25, 20/2/1992, internal report, Enel, Milan (1992).
[63] TINSLEY, N., DEWIS, D., Decommissioning: The Industrial Lessons Being Learned from WAGR, BNFL Nuclear Observer (April 1985).
[64] COMMISSION OF THE EUROPEAN COMMUNITIES, Konzepte zur Minimierung der Aktivierung des biologischen Schilds, Rep. EUR 9442 de, Office for Official Publications of the European Communities, Luxembourg (1985).
[65] COMMISSION OF THE EUROPEAN COMMUNITIES, Design Concepts to Minimize the Activation of the Biological Shields of Light-Water Reactors, Rep. EUR 8804 en, Office for Official Publications of the European Communities, Luxembourg (1985).
[66] WOOLLAM, P.B., The Choice of Cement for the Manufacture of Concrete To Be Activated: The Potential for Reducing the Radiological Consequences of Reactor Decommissioning, Rep. TPRD/B/0651/N85-DECOM 85/1, CEGB, Berkeley, UK (1985).
[67] EICKELPASCH, N., STEINER, H., “Pilot dismantling of the KRBA boiling water reactor”, Decommissioning of Nuclear Installations (Proc. Int. Conf. Luxembourg, 1994), Luxembourg (1995).
[68] EICKELPASCH, N., STEINER, H., “Experiences with the dismantling of the NPP KRB Gundremmingen Unit A and Versuchatomkraftwerk Kahl (VAK)”, Decommissioning and Radioactive Waste Management (Residential Summer Course, Cambridge, UK, 1995), IBC Technical Services, London (1995).
[69] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Medical, Industrial and Research Facilities, Safety Guide, Safety Standards Series No. WS-G-2.2, IAEA, Vienna (1999).
[70] INTERNATIONAL ATOMIC ENERGY AGENCY, Monitoring and Surveillance of Residues from the Mining and Milling of Uranium and Thorium, Safety Reports Series No. 27, IAEA, Vienna (2002).

[71] BENSON, J., ECCLES, H., “A novel technique for the decontamination of concrete and steel”, Nuclear Decommissioning: The Strategic, Practical, and Environmental Considerations (Proc. Int. Conf. London, 1995), Mechanical Engineering Publications, London (1995) 363–370.
[72] MESERVEY, R.H., et al., Idaho National Engineering Laboratory Decontamination and Decommissioning Technology Logic Diagram, Rep. EGG-WTD-11104, Vol. III, USDOE, Washington, DC (1994).
[73] HAMILTON, M.A., et al., “Biodecontamination of concrete”, SPECTRUM ’96: Nuclear and Hazardous Waste Management (Proc. Int. Topical Mtg Seattle, 1996), American Nuclear Society, La Grange Park, IL (1996) 1977–1983.
[74] DERMER, R., et al., “A review of decontamination technologies under development and demonstration at the ICPP”, DDER ’94: Decommissioning, Decontamination and Environmental Remediation at Contaminated Nuclear Sites (Proc. Topical Symp. Washington, DC, 1994), American Nuclear Society, La Grange Park, IL (1994) 114–118.
[75] FLESHER, D.J., MILLER, R.D., Light Aided Decontamination 1996 (LAD), Surface Cleaning, Cutting and Welding with Light: A “NOW” Technology, Rep. WHC-SA-3110-S, Westinghouse Hanford, Richland, WA (1996).
[76] UNITED STATES DEPARTMENT OF ENERGY, Decontamination and Decommissioning Focus Area, Technology Summary, Rep. DOE/EM-0300, USDOE, Washington, DC (1996).
[77] COSTES, J.R., et al., “Decontamination by ultraviolet laser: The Lexdin Prototype”, SPECTRUM ’96: Nuclear and Hazardous Waste Management (Proc. Int. Topical Mtg Seattle, 1996), American Nuclear Society, La Grange Park, IL (1996) 1760–1764.
[78] COSTES, J., Développement et essais d’un procédé de décatégorisation par électroérosion en milieu nitrique et application aux déchets contaminés alpha de l’installation RM2, Rep. EUR 16940, Office for Official Publications of the European Communities, Luxembourg (1996).
[79] COLQUHOUN, A.P., personal communication, 1998.
[80] BOING, L.E., COFFEY, M.J., Waste Minimization Handbook, Vol. 1, Rep. ANL/D&D/M-96/1, USDOE, Washington, DC (1995).
[81] SANDERS, M.J., et al., “Decontamination of buildings, hot cells and similar facilities”, Decontamination and Decommissioning (Proc. Int. Symp. Knoxville, 1994), USDOE, Washington, DC (1994).
[82] SCOTT, S.M., et al., “Decontamination of the jet vacuum vessel using the CO_2 pellet blasting (cold jet) technique”, ibid.
[83] UNITED STATES DEPARTMENT OF ENERGY, Decommissioning Handbook, Rep. DOE/EM-0142P, USDOE, Washington, DC (1994).
[84] TEUNCKENS, L., et al., “Decontamination and melting of metallic material from the Eurochemic reprocessing plant”, Melting and Recycling of Metallic Materials from Decommissioning (Proc. 3rd Eur. Sem. Nyköping, 1997), Office for Official Publications of the European Communities, Luxembourg (1997).

[85] WALTHERY, R., TEUNCKENS, L., LEWANDOWSKI, P., MILLEN, D., BAUMANN, S., "Abrasive blasting: A technique for the industrial decontamination of metal components from decommissioning to unconditional release levels", Environmental Management (Proc. 8th Int. Conf. Bruges, 2001), Bruges 2001.
[86] WARREN, S., et al., "Technologies in use and in development for surplus facility decommissioning and recycling under environmental restoration", ICEM '95: Waste Management and Environmental Remediation (Proc. Int. Conf. Berlin, 1995), American Society of Mechanical Engineers, New York (1995) 1737–1741.
[87] CHRISTOPHER, A., "Free release R&D within BNFL", Melting and Recycling of Metallic Materials from Decommissioning (Proc. 3rd Eur. Sem. Nyköping, 1997), Office for Official Publications of the European Communities, Luxembourg (1997).
[88] PFLUGRAD, K., MENON, S., ANDERSSON, L., ibid.
[89] CORNELISSEN, H., Development of a Process for Separating Radioactive Constituents of Concrete, Including Active Pilot Plant Scale Testing, Rep. EUR 16917, Office for Official Publications of the European Communities, Luxembourg (1996).
[90] PERRIN, J., TANIS, G., "Fissuration cutting", Decommissioning of Nuclear Power Plants (Proc. Eur. Conf. Luxembourg, 1984), Office for Official Publications of the European Communities, Luxembourg (1984).
[91] CHAVAND, J., et al., "Découpage de composants métalliques par fissuration intergranulaire", ibid.
[92] MIGLIORATI, B., GAY, P., "Laser beam cutting", ibid.
[93] GARREC, P., "Polyjointed robot with integrated laser beam", Decommissioning of Nuclear Installations (Proc. Int. Conf. Brussels, 1989), Office for Official Publications of the European Communities, Luxembourg (1990).
[94] MIGLIORATI, B., MANASSERO, G., TARRONI, G., DI FINO, M., PICINI, P., "Investigation of laser cutting applications in decommissioning", ibid.
[95] STANG, W., FISCHER, A., "Electrochemical technique for the segmenting of activated steel components", ibid.
[96] HARISTOY, D., GUETAT, P., CHAPUIS, A.-M., Définition des autorisations de sortie ou 'clearance levels' pour les bétons venant du démantèlement, Rep. EUR 16004, Office for Official Publications of the European Communities, Luxembourg (1995).
[97] FLEISCHER, C.C., FREUND, H.U., "Explosive techniques for dismantling radioactive concrete structures", Decommissioning of Nuclear Installations (Proc. Int. Conf. Brussels, 1989), Office for Official Publications of the European Communities, Luxembourg (1990).
[98] HILLS, D.L., The Removal of Concrete Layers from Biological Shields by Microwaves, Rep. EUR 12185, Office for Official Publications of the European Communities, Luxembourg (1989).

[99] HAFERKAMP, H., LOUIS, H., MEIER, G., “Submerged cutting of steel by abrasive water jets”, Decommissioning of Nuclear Installations (Proc. Int. Conf. Brussels, 1989), Office for Official Publications of the European Communities, Luxembourg (1990).
[100] ROUVIERE, R., et al. Adaptation des jets d’eau haute pression aux abrasifs au démantèlement des installations nucléaires, Rep. EUR 12490, Office for Official Publications of the European Communities, Luxembourg (1989).
[101] COMMISSION OF THE EUROPEAN COMMUNITIES, Annual Progress Reports on the Community’s Research and Development Programme on Decommissioning of Nuclear Installations, Rep. EUR 10740 (1985), Rep. EUR 11112 (1987), Rep. EUR 11715 (1987), Rep. EUR 12338 (1988), Office for Official Publications of the European Communities, Luxembourg (1986–1989).
[102] KUCHNER, L., et al., Further Development and Operation of an Automated Large Scale Radioactivity Measurement Facility for Low Level Decommissioning Waste, Rep. EUR 1544, Office for Official Publications of the European Communities, Luxembourg (1995).
[103] GUGHÖR, P., et al., High Resolution In Situ Gamma-Spectrometer for Use on Contaminated Reactor Building Structures and Outdoor Grounds under Decommissioning, Rep. EUR 18349, Office for Official Publications of the European Communities, Luxembourg (1998).
[104] LUFF, C., DOCKRAY, T., MESSENGER, A., MACARTHUR, D., “Free release monitoring of pipes and ducts”, Waste Management ’97 (Proc. Int. Conf. Tucson, 1997), Waste Management Symposia, Tucson, AZ (1997).
[105] INTERNATIONAL ATOMIC ENERGY AGENCY, Characterization of Radioactive Waste Forms and Packages, Technical Reports Series No. 383, IAEA, Vienna (1997).
[106] INTERNATIONAL ATOMIC ENERGY AGENCY, Methods for Reducing Occupational Exposures During the Decommissioning of Nuclear Facilities, Technical Reports Series No. 278, IAEA, Vienna (1987)
[107] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of the Factors Affecting the Selection and Implementation of Waste Management Technologies, IAEA-TECDOC-1096, IAEA, Vienna (1999).
[108] INTERNATIONAL ATOMIC ENERGY AGENCY, Reference Design for a Centralized Waste Processing and Storage Facility, IAEA-TECDOC-776, Vienna (1994).
[109] INTERNATIONAL ATOMIC ENERGY AGENCY, Reference Design for a Centralized Spent Sealed Sources Facility, IAEA-TECDOC-806, Vienna (1995).
[110] INTERNATIONAL ATOMIC ENERGY AGENCY, Innovative Waste Treatment and Conditioning Technologies at Nuclear Power Plants, IAEA-TECDOC-1504, Vienna (2006).
[111] MENON, S., BRUN-YABA, C., YU, C., CHENG, J.J., BJERLER, J., WILLIAMS, A., Validation of Dose Calculation Programmes for Recycling, Rep. 2002:23, Swedish Radiation Protection Institute, Stockholm (2002).

Annex

OVERVIEW OF FACILITIES

Section 2 describes the types, quantities and origins of waste material generated in nuclear fuel fabrication and reprocessing, in power plants and other minor nuclear facilities, and during D&D activities. To fully understand the processes that create such waste material, and thus the requirements for designing new nuclear facilities to minimize its production, this Annex presents a brief description of typical nuclear facilities, from reactors to plants necessary for the entire fuel cycle.

A–1. FUEL CYCLE FACILITIES

The generation of electricity by nuclear power involves operations to produce fuel ('front end') and irradiation in a reactor, followed by storage and reprocessing or disposal of spent fuel and waste management including treatment and storage ('back end'). The front end of the cycle encompasses uranium extraction, conversion and enrichment, and fuel fabrication plants, supplemented by facilities for fabrication of plutonium–uranium mixed oxide (MOX) fuels. The back end covers storage and/or reprocessing of spent fuel and management of the resulting waste [A–1].

A brief description is given here of the typical processes used in the front end of the fuel cycle, at nuclear power plants and in the back end of the fuel cycle. A general overview of the process material streams and routes in the entire fuel cycle is shown in Fig. 1 in Section 2 of this publication.

The wide spectrum of non-reactor facilities includes some systems and processes similar to those found on reactor sites. These are mainly irradiated fuel stores (wet or dry), waste handling treatment and storage plants, and supporting ancillaries such as water purification circuits, ventilation plants, laboratories and maintenance facilities. In these cases, the nature of the facilities and radioactive material involved poses similar problems for their decommissioning and for managing decommissioning waste. These include complex radiation protection requirements, high levels of alpha contamination and different waste strategies influenced by a diversity of waste streams and their categorization.

A–1.1. Front end of the fuel cycle

A–1.1.1. Uranium extraction

The uranium ore required to produce a tonne of uranium depends on the average grade of the ore and typically amounts to about 10–1000 t ore/t uranium (grade 10–0.1% uranium) [A–2]. The higher grade deposits require a much lower rate of ore extraction, but they require more cautious radiation protection measures for the workers because of their higher radiation fields.

Mining of uranium ore is commonly carried out by either underground or open pit techniques. Compared with underground mining, the amount of waste material is larger for the open pit methods. A third method, the in situ leaching (ISL) technology, has a very small environmental impact, because no ore is brought to the surface during mining. However, its share is still limited to about 13% of the worldwide uranium production, because it requires some special conditions such as suitable sandstone type deposits.

At the extraction stage of the cycle, the primary environmental impact is largely limited to the mining and mill tailings. The radiological impact is related mainly to the release of radon during mining and especially from the mill tailings; this impact accounts for a collective dose of 0.8–1.0 man Sv per GW(e) year. Thus the main long term environmental issue is the effective isolation of the daughter products of natural uranium — mainly radon decay products — from the environment.

A–1.1.2. Uranium conversion facilities

Although uranium ore concentrate (yellow cake) is fairly pure, it requires further purification to reach the very high standards required for nuclear fuel. This is achieved by dissolving the yellow cake in nitric acid, filtering and treating the solution with chemical solvents. The product is the compound uranyl nitrate, which is usually more than 99.95% pure. The uranyl nitrate is reconverted to uranium oxide, and this, in turn, is converted to readily volatile uranium hexafluoride, which is used in the enrichment process. If enrichment is not required, uranium dioxide may be produced from the uranyl nitrate and shipped directly to a fuel element fabrication plant.

A simplified process flow diagram is shown in Fig. A–1 [A–1]. Uranium conversion facilities are similar to other chemical plants where solvents are used in the purification step and gaseous processes (hydrofluorination, fluorination) are used in the conversion step. Although only uranium isotopes and their daughter products are normally present and the radiation hazard is low, adverse health impacts can arise from non-radioactive materials, such as

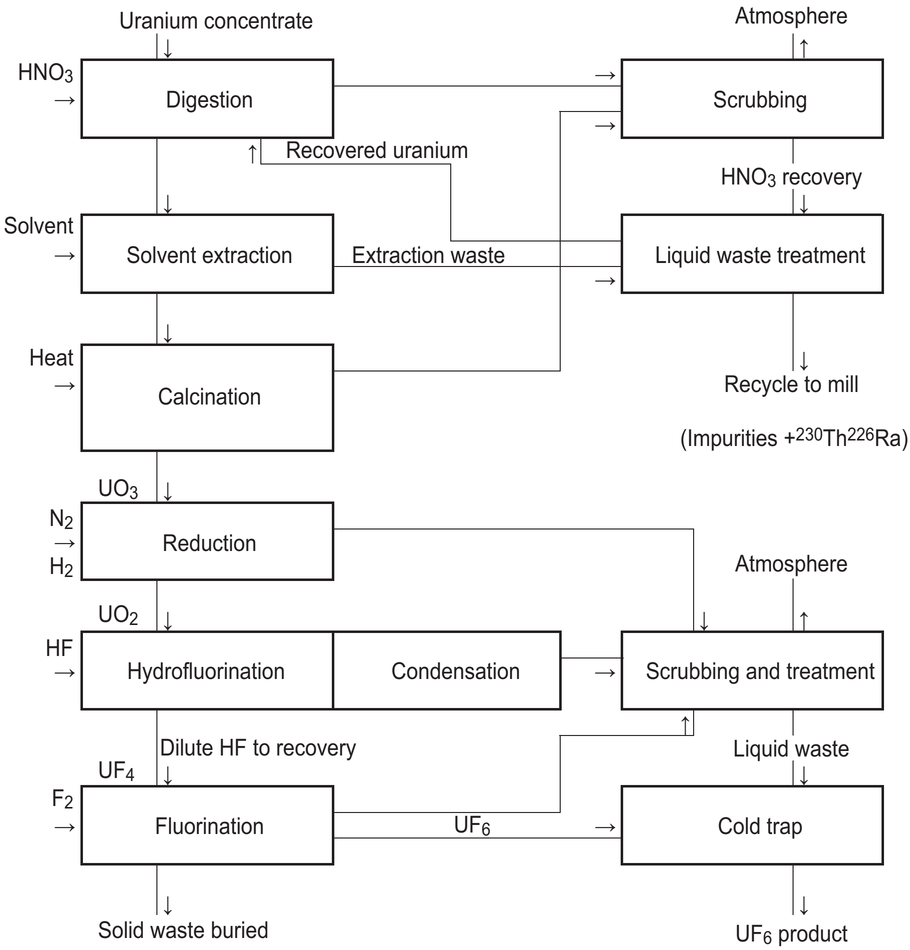

FIG. A–1. Simplified flow diagram of a wet solvent extraction/fluorination process.

fluorine and organic solvents. The size of a conversion plant depends on its production capacity and the technology used; plants typically range in size from a few to several tens of hectares. Usually, the plant is located in several buildings where individual processes are carried out; adjacent areas may contain basins, ponds and lagoons for sludge, extraction waste and sewage. Apart from the physical size and the presence of conventional hazards, the decommissioning of these facilities is usually more straightforward than in other parts of the nuclear fuel cycle. Complications may arise when reprocessed uranium is recycled. In this case, the radiological problems arising from the presence of contaminants such as ^{99}Tc and ^{232}U daughters need to be taken into account.

The conversion of uranium oxides (UO_3 and UO_2) to uranium hexafluoride includes the following processes:

- Reduction of uranium trioxide (UO_3), using hydrogen gas, to produce uranium dioxide.
- Wet route production of uranium tetrafluoride (UF_4) by precipitation following reaction of aqueous hydrogen fluoride (HF) and UO_2.
- Dry route production of UF_4 by reaction of UO_2 directly with anhydrous gaseous hydrogen fluoride.
- Production of uranium hexafluoride (UF_6) by reaction of UF_4 with fluorine gas. This can be achieved either in a flame reactor or in a fluidized bed. Excess fluorine may be scrubbed with potassium hydroxide to give fluoride. The potassium fluoride may be regenerated by reaction with lime (calcium hydroxide) to provide insoluble calcium fluoride.

Considering these processes, conversion plants handle some very aggressive chemicals (F, HF). They do not, however, produce significant amounts of radioactive effluents (principally containing natural uranium (beta activity)).

Very small quantities of uranium (^{234}U, ^{235}U, ^{238}U) are vented from the process and auxiliary systems of gaseous diffusion plants to the atmosphere, while the radioactive discharges from centrifuge enrichment facilities are even smaller. For instance, the atmospheric releases from EURODIF in 1997 were 3.3 kg uranium, with a total activity of 0.16 GBq, and the liquid releases were only 0.29 kg uranium, with a total activity of 0.0094 GBq [A–2].

A–1.1.3. Uranium enrichment facilities

Uranium enrichment involves a partial separation of natural uranium into its two isotopes, ^{235}U and ^{238}U, yielding the enriched fraction and a depleted portion of tails containing less than the natural value of ^{235}U. The enrichment technique involves separation in the gaseous phase using readily volatile uranium hexafluoride. Originally, gaseous diffusion through porous membranes was the most widely used technique, but a number of newer facilities are using gas centrifuges.

After separation, the portion enriched in the ^{235}U isotope is transferred to the fuel element fabrication plant, and the larger part (enrichment tails) is stored or processed. The enriched stream may be used for either uranium metal for Magnox type fuels, or uranium oxide for the predominant reactor types.

Uranium metal is produced by reaction of the UF_4 with an alkali earth metal using a thermal type process. In addition to uranium metal, this produces uranium contaminated slag of the alkali metal fluoride. Uranium dioxide can be produced by one of two routes: the ammonium uranyl carbonate (AUC) process or the integrated dry route (IDR) process, where UF_6 is 'burned' in steam under a reducing hydrogen atmosphere. Both processes give rise to hydrogen fluoride as a by-product.

Particular issues raised by decommissioning of isotopic enrichment plants include nuclear non-proliferation security requirements associated with the disposal of the separation units, the decontamination of process equipment, and the recovery and recycling of large quantities of materials. Dismantling tends to be largely repetitive for identical units installed inside the very large buildings. Similar considerations apply to the recycling of reprocessed uranium in enrichment facilities as described in Section A–1.1.2.

A–1.1.4. Fuel fabrication facilities

The generic oxide and MOX fuel fabrication process consists of the following steps:

— Preparation of the material suitable for pellet formation (correct morphology, blend, purity, etc.). This is achieved, for example, by granulation and grinding, with blending of uranium and plutonium oxide powders where necessary (MOX fuels).
— Compaction to form a pellet either with or without a binder to sustain the integrity of the pellet.
— Sintering followed by grinding to produce pellets of the required dimensions.
— Pellet assembly in pins (Zircaloy or stainless steel tubes) with appropriate spacers and springs. These pins are assembled into fuel elements using bracing, end fittings and caps appropriate to the fuel.

A–1.1.4.1. Uranium oxide fuel fabrication

A simplified process flow diagram for uranium oxide fabrication is shown in Fig. A–2 [A–1]. After conversion of UF_6 to oxide, the resulting product is pressed into pellets and fired in a kiln to produce a dense ceramic fuel capable of withstanding high temperatures and retaining gaseous waste products. The fuel pellets are stacked together and then sealed in tubes of zirconium alloy (for water reactors) or other alloys (e.g. stainless steel). These loaded tubes,

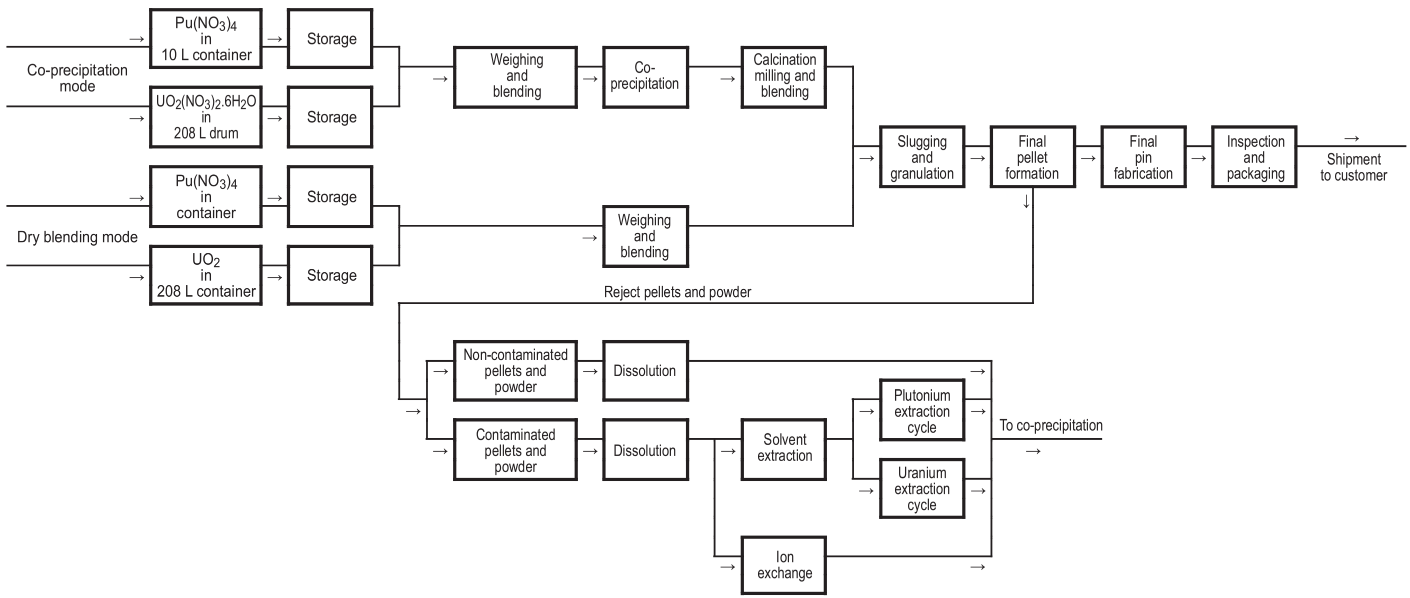

FIG. A–2. Process flow diagram for a MOX fuel fabrication plant.

called fuel pins, are put together in a lattice of fixed geometry to form a fuel assembly.

A typical uranium oxide fabrication plant with a capacity of 1000 t/a occupies an area of several hectares. The main building usually contains manufacturing, maintenance, decontamination and storage areas. Other buildings may house laboratories, a waste treatment facility, a waste recycling plant and other auxiliary facilities such as tanks and pumps, warehouses and storage areas.

Fuels with or without enrichment can be fabricated in a process line designed to handle the associated alpha radiation. This requires appropriate ventilation to prevent inhalation of fine particles of uranium dioxide by workers but does not require special shielding or the use of remote handling techniques.

Only very small quantities of uranium are emitted from the fuel fabrication process to the environment. For example, the atmospheric releases from the Romans UO_2 fabrication plant in 1997 were limited to 0.0156 GBq, and the liquid releases contained 2.644 GBq of uranium.

Radon is generated from the natural decay of uranium. The conversion process removes all uranium decay products, including radium, the direct parent of radon. As the radium is removed, and as uranium and its daughter products preceding radon in the decay chain have very long half-lives, radium will not be present in the fuel and no radon will be emitted, which is the same as in the enrichment process [A–2].

Decommissioning of uranium fuel fabrication plants may require special criticality precautions in addition to protection of personnel against alpha emitters. Efficient measurement devices to facilitate unrestricted release of materials from both uranium fuel fabrication plants and enrichment plants are important during decommissioning.

The chemical toxicity of uranium compounds must also be taken into account, especially for powders (UO_2 or UF_4) or soluble compounds (such as uranium nitrate). In facilities handling uranium metal, the pyrophoric nature of finely divided metal must be considered.

A–1.1.4.2. Fabrication of fuel containing plutonium (MOX fuel)

Some fabrication facilities have been used to produce plutonium metal for defence related purposes, or plutonium oxide for storage or use in fuels. Commercial scale fuel plants making plutonium oxide are in use, although older facilities were often on a laboratory or pilot plant scale. Compared with reprocessing plants, these facilities are relatively small; they are based on a building or a complex of buildings housing areas with glove boxes, ventilation ducts, filter banks, utility systems and associated equipment.

In the MOX fuel technology, plutonium is used as a raw material for MOX fuel for reactors. A MOX fuel fabrication plant is designed for the production of (Pu, U)O_2 fuel pellets and incorporation of these pellets into clad fuel rods. The plant may use a process involving blending of Pu and U solutions followed by co-precipitation and calcination to form MOX fuel. Recently, a mechanical blending process has become more common. The facilities for fabricating MOX fuels are also relatively small in size [A–1].

The overall technology may also include support processes such as solvent extraction, ion exchange or oxalate precipitation for recovery of effluents, and a liquid waste evaporation system followed by solidification of resulting concentrates. The facility generally uses criticality safe vessels located in glove boxes.

Major considerations in the decommissioning of MOX fuel fabrication plants arise from the presence of plutonium, particularly because PuO_2 and (Pu, U)O_2 powders will exist in some parts of the process. For this reason, the measurement of residual inventory to avoid criticality hazards is a major consideration, and strict control of containment, ventilation and means to restrict spread of contamination are essential. An accurate assay of waste arisings (with decontamination or segregation, as appropriate) is also needed. In plutonium plants, the possibility also arises of significant operator radiation exposure from inhalation or external irradiation from gamma or neutron emitters whenever residues exist in the plant.

A–1.1.4.3. Uranium Magnox fuel fabrication facilities

Reduction of uranium tetrafluoride to uranium metal is carried out using magnesium metal. After reaction, the uranium billet is separated from the magnesium fluoride slag and cleaned by pickling and shot blasting. This natural uranium billet, together with recycled clean scrap and essential alloying elements, is then remelted in a vacuum casting furnace. After casting, the uranium rods are heat treated, straightened, machined to diameter and cut to final length. The machined rods are cleaned and inserted into Magnox cans. End caps are then fitted and argon arc welded after helium charging of the cans [A–1].

Current facilities have a capacity of 1000–1500 t/a, the largest area being occupied by the automated machine shop and canning facilities. Appropriate ancillary facilities are provided, although laboratories and waste/effluent treatment facilities are shared with other manufacturing units on the same site.

Facilities for aluminium clad metal uranium slug fabrication could also be categorized under this type.

These plants largely resemble conventional metal working facilities in most respects. From the nuclear decommissioning point of view, the major considerations are:

— Size/scale, which is larger than for later stages in the fuel cycle;
— Chemical toxicity of uranium residues;
— Ensuring that any structure retained for reuse has been adequately decontaminated, in view of the low activity of natural (or low enriched) uranium and the difficulties of detection.

Similar considerations apply to the whole site if it is to be cleared for future unrestricted use.

A–1.1.4.4. Other metallic fuel fabrication facilities

Many countries have small research reactors or isotope production facilities that require non-standard fuel. Facilities used for the fabrication of this type of fuel often serve more than one purpose (research reactor fuel, isotope target production and provision of special services to R&D staff). These facilities are usually relatively small and similar to those outlined in Section A–1.1.4.2, but they do not deal with plutonium. The fabrication process is largely mechanical. The facility is designed to handle associated alpha activity, which requires appropriate ventilation to prevent inhalation of particles (fume hoods, glove box lines) but does not require special shielding or remote handling techniques [A–1].

In decommissioning such facilities, it is of great importance to ensure that:

— Appropriate protection measures for low level alpha radiation are undertaken to prevent inhalation/ingestion;
— The pyrophoric nature of the material is taken into account, where relevant;
— Criticality safety is observed and factored into decommissioning technique evaluations;
— Waste is segregated at the source to minimize the amount of alpha contaminated waste generated and to reduce overall project waste management costs;
— Metal (fume hood, etc.) is decontaminated to permit reuse or recycling to the metals market, thereby avoiding/reducing immediate waste costs and long term liabilities.

A–1.2. Back end of the fuel cycle

A–1.2.1. Nuclear fuel cycle and waste minimization options

There are two nuclear fuel cycle concepts. The first is the open fuel cycle, where the fuel material makes a once through passage from uranium ore to disposal of the irradiated fuel. The second is the closed fuel cycle in which fissile material is recovered from the irradiated fuel and reused for the fabrication of new fuel. The main difference between the open and the closed cycle lies in the reprocessing of irradiated fuel and the use of recovered fissile materials in new fuel.

There are two distinct fuel cycles in existence worldwide. The most important one is based on uranium and plutonium, which can be recovered and reused in either thermal or fast neutron reactors, while the second, much less common cycle is based on thorium. An example of the latter is a small experimental test reactor (MINI) in India based on ^{233}U fuel that recently went critical. The ^{233}U in this reactor has been recovered from the reprocessing of irradiated fertile thorium fuel elements [A–3].

A–1.2.2. Fuel and waste storage facilities

A–1.2.2.1. Interim storage and conditioning of spent fuel

Irradiated fuel assemblies are stored at reactor sites (AR) or away from reactors (AFR) at separate storage locations. Storage in water pools is a common practice for AR storage, while AFR storage has been implemented in several countries as wet storage in pools or as dry storage using concrete canisters, metal casks or concrete vaults.

Practical experience from AFR wet storage pools shows that discharges of radioactive substances to the environment are very small, and thus the radiological impacts of the discharges from AFR wet storage facilities to the public are negligible. Dry storage facilities for spent fuel assemblies show no or only very small discharges of radioactive substances to the environment.

The irradiated fuel assemblies that are not reprocessed have to be packed or conditioned prior to their disposal after a period of interim storage. The conditioning of spent fuel results in intermediate level waste (ILW) and high level waste (HLW). In general, 0.2 m^3 of ILW and 1.5 m^3 of HLW (i.e. the conditioned spent fuel) are created per tonne of spent fuel [A–2].

A–1.2.2.2. Fuel ponds

Besides AR storage of spent fuel and storage at reprocessing plants, fuel ponds may also provide for the storage of spent fuel in independent AFR facilities. Fuel ponds are all basically similar: they are rectangular, horizontal and 12–13 m deep for shielding purposes. Their walls and floor are constructed of reinforced concrete of sufficient thickness to meet structural requirements and to provide radiation shielding. Most ponds are lined with welded stainless steel plates; otherwise, the concrete walls and floor are coated with paint. The fuel assemblies are placed in storage racks, baskets or containers positioned in the pond.

Fuel ponds are provided with systems for:

— Fuel handling: to transfer fuel from the transport cask to the racks or from one position to another within the pond.
— Heat exchange: to remove heat produced by radioactive decay of the spent fuel.
— Water purification: ion exchange is used to control ionic impurities and filtration for particulates. In addition, skimmers take away impurities from the pond's water surface, while vacuum cleaners and scrubbers remove particles from racks, walls and the floor.

Water purification also controls the radioactivity in the pond, which arises from three types of radioactive species, generated while the fuel is inside the reactor: activation products, fission products and transuranics. Away from reactor ponds generally receive fuel after months or years of storage in AR pools; if the shipment is in wet casks, a small radioactive inventory from the AR pond will be carried by the cask and will mix with the AFR pond water during fuel discharge. The main transfer of radioactivity derives from the desorption (solubles) or spallation (particulates) of radioactive species from the fuel assemblies' surface. Transuranics will be significant where fuel with failed cladding is handled.

There is experience with refurbishing ponds in order to increase their storage capacity (compaction, using double tiers, etc.) or their volume and with decommissioning these facilities. During decommissioning, pond water is treated by means of any existing purification system until the concentration required for discharge is reached. Then, sludge and particulates on the racks and inside the pond are removed, followed by decontamination of racks and the pond's internal surfaces by various means (water jet, chemicals, mechanical brushes, etc.). The racks and the pond liner are subsequently dismantled and either reused or disposed of as waste. If the pond has no liner, it may be necessary to scarify the concrete surface, particularly if the radioactivity has penetrated through cracks in the paint. Finally, the auxiliary systems are dismantled [A–1].

A–1.2.2.3. Dry fuel storage facilities

In several countries, national decommissioning programmes have recognized the need to develop a facility to store spent fuel bundles removed from fuel bays as part of the decommissioning of reactor facilities. In Canada, concrete silos with the fuel bundles enclosed in welded stainless steel baskets are used. Although designed for a long life, these are only storage facilities and will eventually have to be decommissioned. Improvements have therefore been incorporated into the CANSTOR concrete module design.

Contamination of CANSTOR modules is expected to be insignificant when the fuel storage baskets are eventually extracted from the modules and shipped with their contents to a final repository at the end of their storage period. As the exterior surface of these fuel baskets may be slightly contaminated after leaving the storage pool, a minute amount of contamination could deposit on the internal surface of the cylindrical steel liners inside the modules. These liners form the second containment boundary for the spent fuel, the baskets being the first.

Once all fuel baskets have been removed from a module, a radiological inspection of the empty module can be performed, and any surface found to be contaminated can be decontaminated or the affected material removed. The radiological exposure associated with this task should be minimal because of the limited scope and the small amount of contamination.

After decontamination, the remainder of the module will be available for reuse or standard industrial demolition. This can be done by several means. First, the steel liners are cut into pieces and removed, and then diamond wire cutting of the monoliths can be undertaken. This is a proven technique that produces easily manageable blocks of concrete waste. Other available techniques are jack hammering, ball wrecking, sawing, stitch coring and controlled blasting. It is expected that the bulk of the waste will be radiologically clean industrial material; hence no radiological exposure will result from this work [A–1].

A–1.2.3. Nuclear material storage facilities

A–1.2.3.1. Plutonium storage facilities

Plutonium storage facilities (PSFs) typically contain strategic amounts of plutonium. The information contained in this section applies to facilities where strategic amounts of plutonium or significant quantities of other transuranic radionuclides, such as neptunium and californium, are stored. This section does not apply to ‘in process’ or ‘in use’ material, to material in assembly cells for use in weapons, or to material that is packaged in approved containers awaiting either transportation or disposition upon receipt. Plutonium-238 presents special design challenges because of its high specific activity; however, those considerations are not addressed here.

A–1.2.3.2. Unirradiated enriched uranium storage facilities

Unirradiated enriched uranium storage facilities (UEUSFs) are used to store unirradiated enriched uranium in a solid, liquid or gaseous form. Activities of UEUSFs may include shipping, receiving, handling, packaging and unpacking.

A–1.2.3.3. Irradiated fissile material storage facilities

Irradiated fissile material storage facilities (IFMSFs) are self-contained installations for storage of highly radioactive fissile material (spent fuel and target elements) that has been exposed to a neutron flux, usually in a nuclear reactor. The irradiated material must be properly clad or canned when received so that leakage from the assemblies is minimized and remains within specified limits.

A–1.2.4. Reprocessing plants

After spent fuel is discharged from a reactor, it is placed in storage ponds filled with water to allow short lived isotopes to decay prior to reprocessing. It may subsequently be placed in longer term, usually dry, storage to await disposal or reprocessing.

The fuel is conveyed to reprocessing plants in dedicated containers. Following receipt and temporary storage of irradiated Magnox fuel, reprocessing starts with stripping off cladding either under water or in dry shielded cells, or removing cladding by dissolution, and dissolving the fuel itself. In the case of fuel from light water reactors, fuel elements are sheared into small pieces prior to dissolution.

The solution of uranium, plutonium, other actinides and fission products is processed by solvent extraction in a series of stages, which are designed to produce separate solutions of plutonium and uranyl nitrates of high chemical purity. Some of the remainder (other actinides, fission products and impurities) is left as a highly radioactive solution, which is concentrated by evaporation and stored, typically in water cooled, double containment, high integrity stainless steel tanks, pending vitrification into glass blocks for decay storage and eventual disposal.

The separated solutions of uranyl nitrate and plutonium nitrate are further processed. The uranium can be converted to uranium dioxide for storage or for production of fresh fuel by blending with fissile material, or converted to uranium hexafluoride for return to the enrichment plant. The plutonium nitrate is usually converted to plutonium dioxide for storage or for incorporation into MOX fuels for thermal or fast breeder reactors.

Fuel cladding and other solid waste materials are typically stored, while medium active and low active liquid effluent streams undergo further treatment for storage or disposal, as appropriate. A simplified process flow diagram is shown in Fig. A–3 [A–3].

Contact with the radioactive materials results in severe contamination of the inner surfaces of the plant and equipment. After final shutdown, this residual contamination has a significant influence on dismantling and waste management.

In a reprocessing plant, many operations are conducted remotely, with heavy shielding to protect the personnel from the effects of radiation. The plants are ventilated and designed to ensure appropriate containment of radioactive material.

Typically, an industrial reprocessing plant covers a large area and is housed in several buildings where individual phases of the processes are carried out.

To support reprocessing, a complex of individual plants and equipment may involve:

— A fuel receipt facility;
— Ponds for fuel storage;
— Equipment for fuel decanning or shearing;
— Vessels, tanks and pipework for chemical treatment;
— Evaporators, condensers and storage tanks for liquids;
— Uranium and plutonium finishing plants;
— Furnaces and powder mixing equipment for solids;
— Ducting, filters and scrubbers for gases;
— Support facilities such as shielded pipe ducts, ventilation plants and laboratories;
— Waste treatment and conditioning plants.

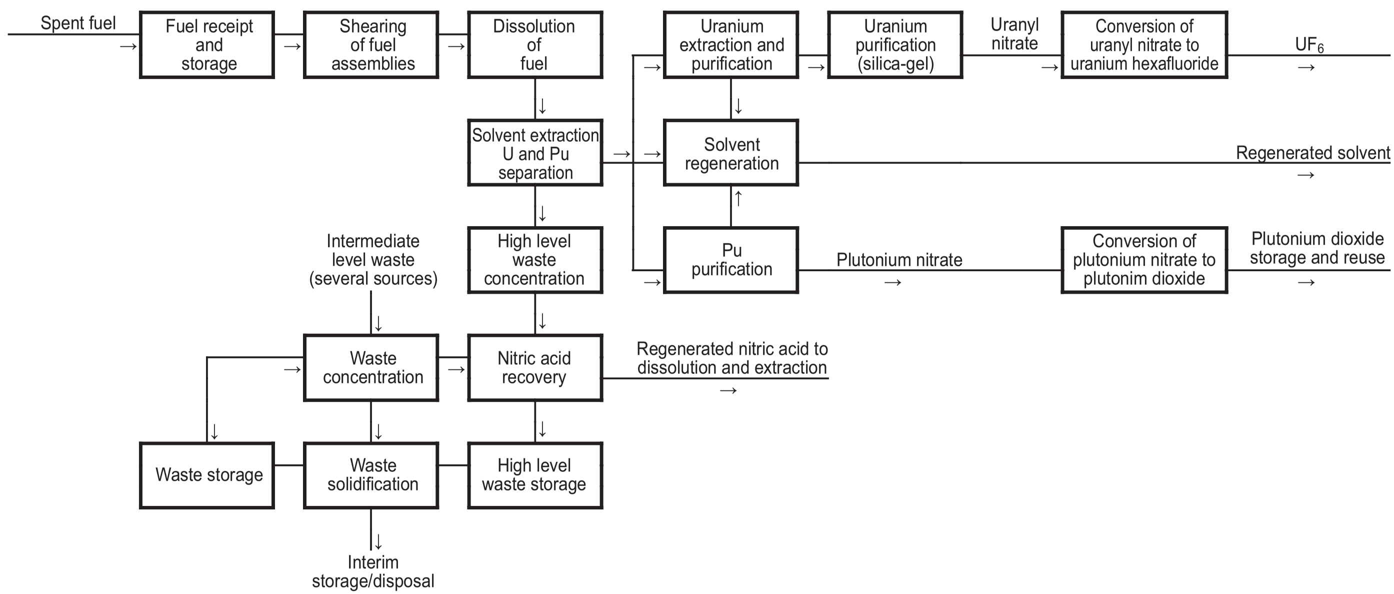

FIG. A–3. Simplified flow diagram of a typical fuel reprocessing plant.

Two of the principal challenges affecting decommissioning of these facilities are high radioactivity levels inside certain cells (due to fission products) and the presence of all types of contamination (alpha, beta and gamma emitting radionuclides). The criticality hazard potential may be increased during decontamination and dismantling of plutonium plants owing to changing concentrations and locations of fissile materials or the movement of dismantled components. The hidden presence of alpha emitters in contamination measurements (e.g. difficulties involved in alpha measurements in small diameter pipes) is also a major consideration. In certain cases, the primary radiological concerns may change over time, for example, because of the decay of ^{241}Pu to ^{241}Am [A–1].

A–2. POWER REACTORS

A–2.1. Types of nuclear power plant

A range of reactor types exists, the operation of which offers some opportunities for the reuse or recycling of materials and components resulting from operation, maintenance and decommissioning. These reactor types may be divided into two groups:

— Thermal neutron reactors, which can be differentiated on the basis of the applied moderator and coolant:
  - Light water moderated and cooled reactors, such as PWRs and boiling water reactors (BWRs);
  - Graphite moderated reactors, such as Magnox reactors, CO_2 cooled; advanced gas cooled reactors, CO_2 or He cooled; and RBMK reactors, a channel type water cooled reactor with high capacity;
  - Heavy water moderated and cooled reactors, such as CANDU reactors and pressurized heavy water reactors (PHWRs).

— Fast neutron reactors, most of which are at the prototype or demonstration stage; a common coolant in this reactor type is sodium or its alloys.

A–2.2. Light water moderated and cooled reactors

A–2.2.1. Pressurized water reactors

About 60% of the world’s commercial power reactors are PWRs; over 230 PWRs are in use for power generation and several hundred more are used

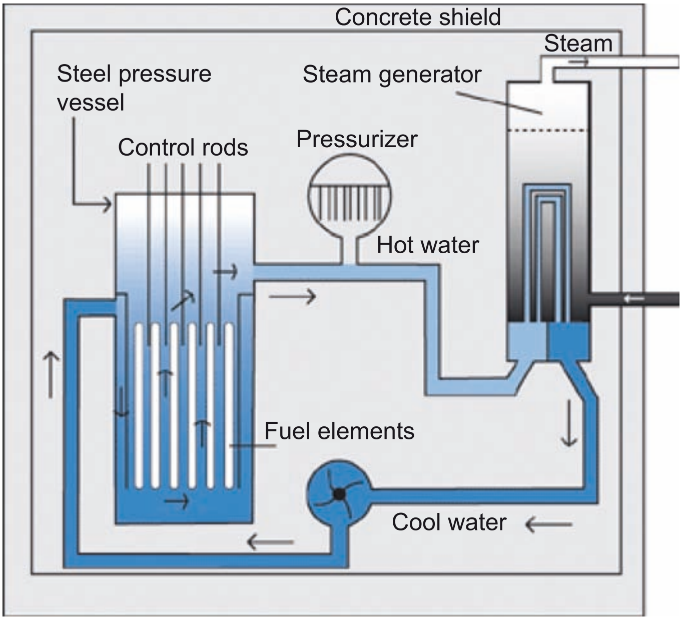

FIG. A–4. Schematic of a PWR.

in naval propulsion. A PWR consists of a compact core in a pressure vessel capable of containing ordinary water at high pressure and three separate cooling systems (Fig. A–4).

The core typically consists of about 200 fuel assemblies, each containing a similar number of fuel rods, holding 80–100 t of uranium. A fuel assembly contains 24 guide tubes in which control rods can slide in and out of the core. Each fuel rod comprises a stack of pellets of enriched uranium oxide (UO_2) cladded in a sealed Zircaloy (slightly alloyed zirconium) tube. The oxide is a ceramic that melts at about 2800°C. Water is able to flow freely between the fuel rods, while being directed through the fuel assembly in a prescribed fashion.

The control rods containing neutron absorbing material such as boron or cadmium are used to fine-tune the reactor operation and shut down the reactor in normal operation or in the event of a malfunction. Secondary shutdown systems involve adding other neutron absorbers such as boric acid, usually as a fluid, to the system. In PWRs, ordinary water is used both as a moderator and as a coolant.

The reactor coolant system consists of two, three or four cooling 'loops' connected to the reactor, each containing a reactor coolant pump and steam generator. The reactor heats the water that passes upward past the fuel assemblies. Boiling, other than minor bubbles called nucleate boiling, is not allowed to occur. Pressure is maintained through a heater and spray system in a pressurizer connected to the reactor coolant system. The water from the reactor is pumped to the steam generator and passes through tubes. The reactor cooling system is expected to be the only one with radioactive materials in it.

In a secondary cooling system (which includes the main steam system and the condensate/feedwater systems), cooler water is pumped from the feedwater system and passes on the outside of the steam generator tubes, where it is heated and converted to steam. The steam then passes through a main steam line to the turbine, which is connected to and turns the generator. The steam from the turbine condenses in a condenser. The condensed water is then pumped by condensate pumps through low pressure feedwater heaters to the feedwater pumps, to high pressure feedwater heaters, and back to the steam generators.

Vacuum is maintained in the condenser using either vacuum pumps or air ejectors. Cooling of the steam is provided by condenser cooling water pumped through the condenser by circulating water pumps, which take suction from water supplied from the ocean, sea, lake, river, or cooling tower.

A containment structure around the reactor core is designed to protect it from outside intrusion and to protect those outside from the effects of radiation or any malfunction inside. It is typically a one meter thick concrete and steel structure. The escape of products formed during fission is prevented by the high melting temperature ceramic pellets themselves, as fission products are trapped in small pores, and by the Zircaloy cladding, which is corrosion resistant with low neutron absorption. A small space at the top of the fuel rod accommodates any fission gas that escapes from the pellets.

A–2.2.2. WWER-440 reactors

WWER-440 nuclear power plants have six loops, isolation valves on each loop, horizontal steam generators, and rack and pinion type control rod drives; generally all have 220 MW(e) steam turbines. They use hexagonal fuel assemblies containing 126 fuel rod positions. Electrical power output of the units varies between 408 and 510 MW(e) after power upgrade.

A–2.2.2.1. Technical description of WWER-440 model 230 reactors

The WWER-440 model 230 reactor relies solely on local area compartmentalization to prevent the release of fission products (Fig. A–5). The design basis accident is a pipe rupture with an effective 100 mm diameter carrying

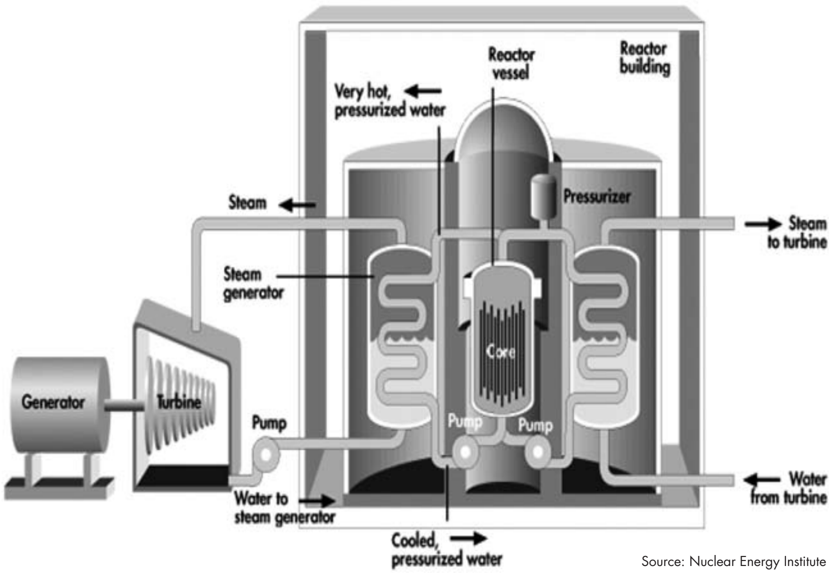

FIG. A–5. Schematic of a WWER-440 model 230 reactor.

a unidirectional flow reduced by special orifices. The WWER-440/230 comprises make-up coolant pumps with a limited capability for emergency core cooling, but has no emergency core cooling system (ECCS) as such. The sealed accident localization compartments contain pressure release valves intended to relieve overpressure and ensure subsequent closure after the pressure level has normalized. In some cases, the inside surface of the reactor pressure vessel is cladded. The model uses low inertia canned motor pumps.

A–2.2.2.2. Technical description of WWER-440 model 270 reactors

The design of the Armenian Nuclear Power Plant (ANPP) was based on the WWER-440/230. Taking into account the specific conditions of the plant location, design improvements were made to both the plant construction and the facility as a whole aimed at increasing the plant's resistance to a seismic hazard of magnitude 8 on the MSK-64 scale. These changes led to the WWER-440/230 type to be renamed as the WWER-440/270 model. The improvements in the plant's resistance to seismic hazards result in a higher material inventory and larger volumes of dismantled materials. Hence there also are consequences for the total decommissioning costs of the WWER-440/270 units.

A–2.2.2.3. Technical description of WWER-440 model 213 reactors

The WWER-440/213 reactor differs from the WWER-440/230 reactor in that it has an ECCS and connects a so-called ice condenser tower to the accident localization compartments of each unit in order to mitigate the effects of loss of coolant accidents. This model was designed to cope with a 500 mm pipe rupture. The inside surface of the reactor pressure vessel is cladded with stainless steel. Flywheels are incorporated into the primary coolant pumps in order to increase their coast-down time during an emergency situation.

The WWER-440/213 reactor has a variant that houses the nuclear steam supply system in a steel containment structure. There are two such units operating at Loviisa, Finland.

A–2.2.2.4. Material volumes of WWER-440/230 and WWER-440/213 reactors

The WWER-440/230 and WWER-440/213 units described above are basically the same. However, there are significant differences in technological solutions, material volumes used in construction and as built features that are important for both construction and decommissioning. Table A–1 summarizes the construction material volumes of the WWER-440/230 and WWER-440/213 nuclear power plants.

TABLE A–1. TYPICAL CONSTRUCTION MATERIAL VOLUMES OF THE WWER-440 TWIN UNITS

		Material (m^3)		
Model		Concrete	Stainless steel	Carbon steel
WWER-440/230	Bulgaria	110 000	8000	23 600
	Slovakia	117 000[a] 129 436[c]	5306[b]	25 294[b]
WWER-440/213	Hungary	160 000[a] 56 000[c]	9200[b]	41 000[b]
	Slovakia	220 579[a] 178 155[c]	5914[b]	42 913[b]

[a] Concrete above the level of –1 m.
[b] Inventory of technological equipment.
[c] Concrete below the level of –1 m (in some countries concrete structures at levels below 1 m are not to be removed after being monitored and cleared).

A–2.2.3. Boiling water reactors

A BWR has many similarities to a PWR, except that there is only a single circuit in which the water is at lower pressure (about 75.10^5 Pa) so that it boils in the core at about 285°C. The reactor is designed to operate with 12–15% of the water in the top part of the core as steam, which has a lower moderating effect. A BWR fuel assembly comprises 90–100 fuel rods, and there are up to 750 assemblies in a reactor core, holding up to 140 t of uranium.

Water is circulated through the reactor core, picking up heat as it moves past the fuel assemblies. The water is heated enough to be converted to steam. Steam separators in the upper part of the reactor remove the remaining water from the steam (Fig. A–6).

The steam then passes through the main steam lines to the turbine generators. The steam typically goes first to a smaller high pressure turbine, and then passes to moisture separators and to two or three larger low pressure turbines. Three low pressure turbines are common for a 1000 MW(e) plant. The turbines are connected to a generator.

After passing through the turbines, the steam condenses in a condenser, which is maintained at vacuum and is cooled by ocean, sea, lake or river water. The condensed steam is pumped to low pressure feedwater heaters. The water then passes to the feedwater pumps, which in turn pump the water to the reactor and start the cycle all over again.

As the turbines are part of the reactor circuit, and the water around the reactor core is always contaminated with traces of radionuclides, the turbine

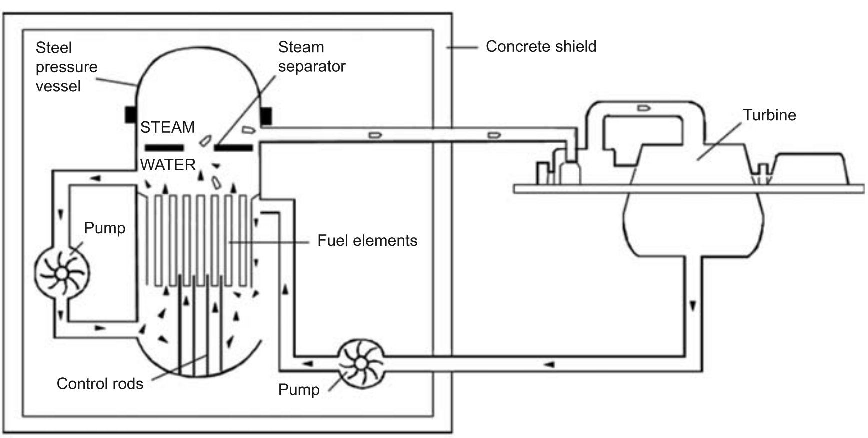

FIG. A–6. Schematic of a BWR.

must be shielded and radiological protection must be provided during maintenance. The cost of this shielding and protection tends to balance the savings due to the simpler design. However, as most of the radioactivity in the water is very short lived (mostly ^{16}N with a 7 s half-life), the turbine hall can be entered soon after the reactor is shut down.

The BWR is unique in that the control rods, used to shut down the reactor and maintain a uniform power distribution across it, are inserted from the bottom by a high pressure hydraulically operated system. Current BWRs have electrical outputs of 570–1300 MW(e).

A–2.3. Graphite moderated reactors

A–2.3.1. Gas cooled reactors

In a gas cooled reactor (GCR), the moderator is graphite and carbon dioxide gas is circulated through the core at a pressure of about 16×10^5 Pa in order to remove the heat from the fuel elements. There are two types of GCR, both of which were developed primarily in the United Kingdom.

Magnox reactors use natural uranium metal fuel cladded in a special magnesium alloy that absorbs very few neutrons, which is essential to achieving criticality with non-enriched fuel (or natural uranium). The specially developed magnesium alloy resists corrosion (no oxidation), hence the name ‘Magnox’. The volume of the core is about 40 times that of a PWR owing to the large distances needed to slow the neutrons to thermal energies.

The fuel sits in individual channels through the moderator. Large blowers drive the carbon dioxide coolant from bottom to top and then into the heat exchangers to raise steam as in PWRs. The control rods enter into holes in the moderator from above. Although the large size of the core makes the capital cost higher, it has the advantage that any fault involving the abnormal heating of the core proceeds very slowly. Also, single phase (gas) coolant is intrinsically safer, as it is devoid of void effects.

Advanced gas cooled reactors (AGRs) are second generation GCRs and use slightly enriched (2.5–3.5%) uranium dioxide fuel in stainless steel cladding. The UO_2 fuel has a higher melting point, allowing a higher fuel rating and higher thermal efficiency. The fuel pellets are similar to those used in a PWR but are larger in diameter and have a central hole. Clusters of fuel elements are joined together end to end in stringers placed in vertical holes in the graphite moderator.

The carbon dioxide circulates through the core, reaching 650°C, and then passes steam generator tubes outside the core but still within the concrete and steel pressure vessel. Control rods penetrate the moderator, and emergency

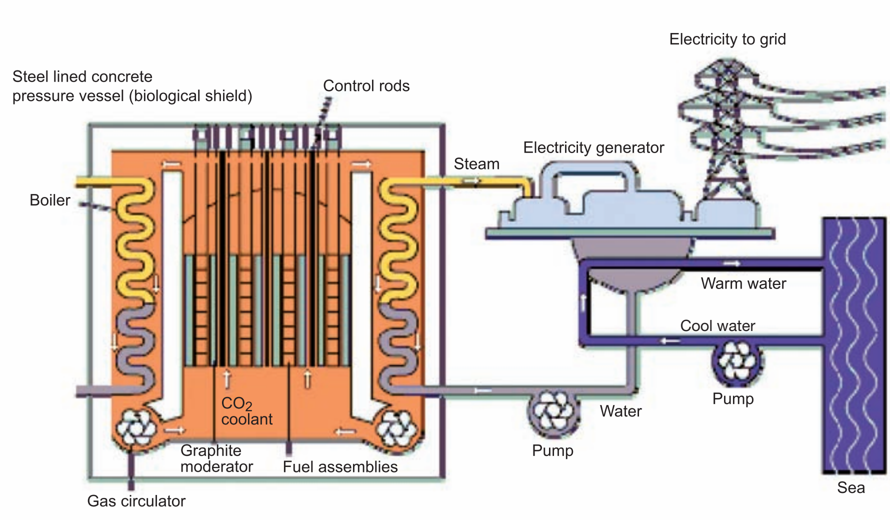

FIG. A–7. Schematic of a typical gas cooled reactor cycle.

shutdown can be accomplished in diverse situations by injection of nitrogen, which is a strong neutron absorber.

Two key advantages of this design are:

— The higher operating temperature with a higher thermal efficiency;
— The insusceptibility to accidents of the type possible with water cooled/moderated reactors.

Figure A–7 shows a typical GCR or AGR cycle.

A–2.3.2. RBMK reactors

The overall layout of RBMK reactors is similar to that of AGRs. RBMK reactors have a large graphite block structure as the moderator in order to slow down the neutrons produced by fission. Passing through the reactor core are about 1600 vertical tubes with a diameter of about 89 mm that circulate water as the coolant to remove the heat produced by two sets of long fuel assemblies (consisting of 18 rods lengthwise), which are also mounted in the vertical tubes. Fuel rods are about 12.7 mm in diameter and contain enriched uranium oxide. The total core height is more than 6.4 m. The graphite structure is contained in a steel vessel (with a diameter of approximately 12.8 m). A helium–nitrogen

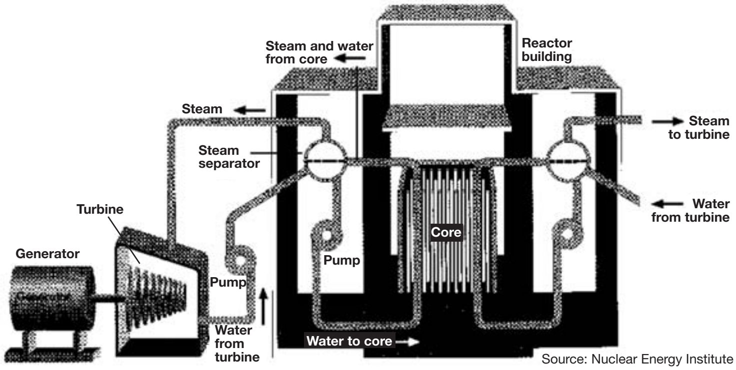

FIG. A–8. Schematic of an RBMK reactor.

mixture is used to improve the heat transfer from the graphite to the coolant channels and to reduce the likelihood of graphite oxidation. There are two horizontal steam generators and two reactor cooling loops, with headers feeding the pressure tubes in the reactor (Fig. A–8).

Boiling occurs in the RBMK design. The steam produced passes to the steam separator, which separates water from the steam. The steam then passes to the turbine as in the BWR design. Similar to the BWR case, the steam is radioactive. However, the steam separator introduces a delay time, so radiation levels near the turbine may not be as high as in the BWR case.

A–2.4. Heavy water moderated and cooled reactors

The dominant type of heavy water reactor (HWR) is the heavy water cooled, heavy water moderated pressure tube reactor, as defined for the CANDU and Indian HWRs. This type of reactor is designed to use natural uranium, but it can also use slightly enriched uranium or a variety of fuels.

Typically, the reactor core is contained in a cylindrical austenitic stainless steel tank (calandria), which holds the heavy water moderator at low temperatures (<80°C) and low pressure (10^5 Pa) (Fig. A–9). The ends of the cylinder are closed with two parallel end shields perforated with holes for the fuel channels, with the holes arranged in a square lattice pattern. Thin walled Zircaloy-2 tubes are fastened to each inner tube sheet and act as stays for the end shields in order to form a leaktight tank. The holes in each end shield are connected with stainless steel tubes (lattice tubes) (Fig. A–10) [A–4].

Each fuel channel consists of a Zr 2.5%Nb pressure tube joined to martensitic stainless steel end fittings, and occupies the tubular holes or lattice

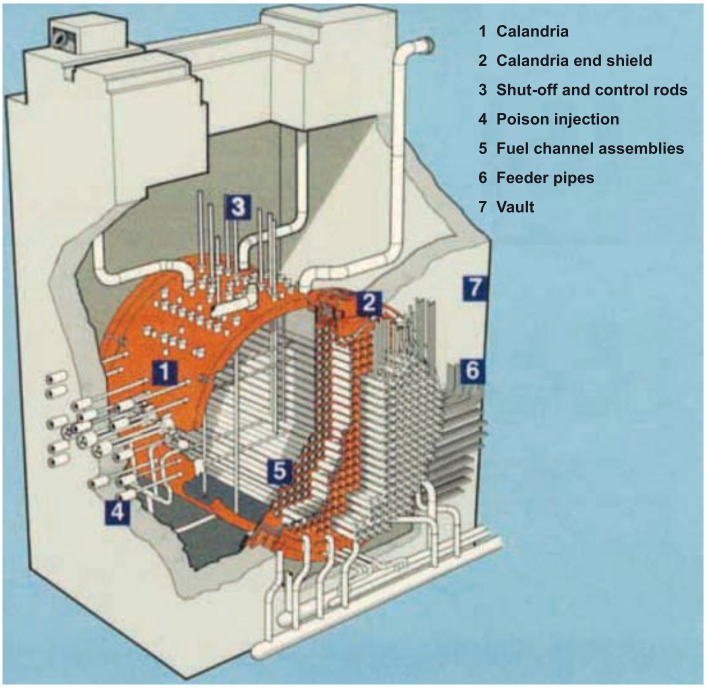

FIG. A–9. Illustration of a CANDU PHWR.

sites formed by each combined lattice tube and calandria tube. The fuel channel end fittings are supported on a pair of sliding bearings at each end, and the pressure tube is supported and separated from the calandria tube by annular spacers.

The end fittings have a closure plug at each end that can be removed by a fuelling machine to insert or remove 0.5 m long fuel bundles. The channel contains either 12 (CANDU 6) or 13 (Bruce/Darlington 800 MW reactors) bundles. The fuel channel is connected to feeder pipes at a side port on each end fitting. The coolant leaves each channel through carbon steel feeder pipes that transfer the heavy water coolant to the headers, from which it is sent to the steam generators before being pumped back to the channels. Control mechanisms operate in the cool moderator and are contained in tubular sheaths that penetrate the matrix of calandria tubes, either vertically or horizontally.

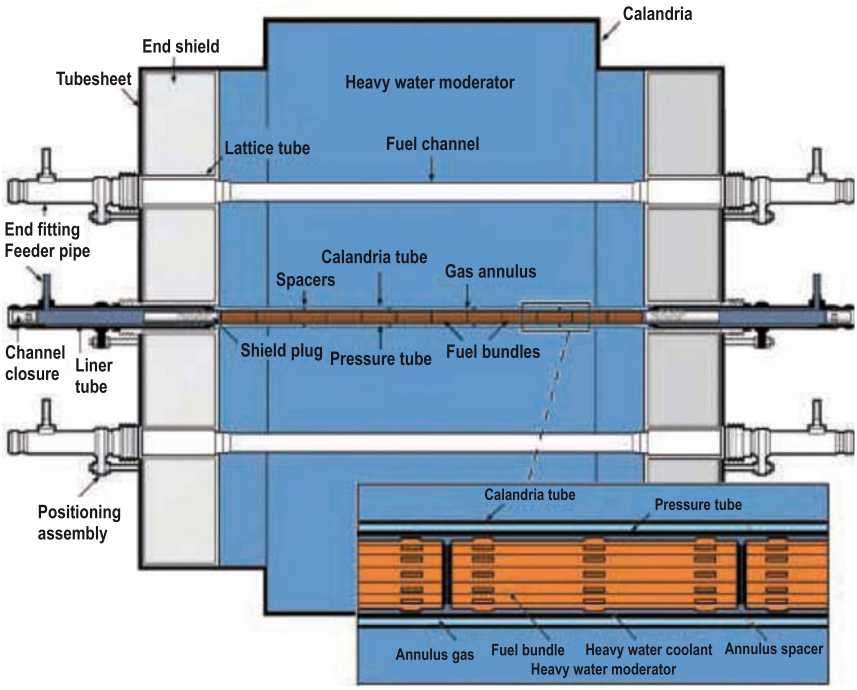

FIG. A–10. Cross-section of a CANDU calandria.

A–2.5. Fast reactors

A fast reactor is a category of nuclear reactors in which the fission chain reaction is sustained by fast neutrons. Such a reactor needs no neutron moderator, but must use fuel that is relatively rich in fissile material compared with that required for a thermal reactor.

In practice, this means that relatively highly enriched uranium or plutonium fuel is used. It is impossible to build a fast reactor that will use only natural uranium fuel. However, it is possible to build a fast reactor that will produce more fissile material than it consumes. The excess material thus generated is recycled in the fuel fabrication for further use in the reactor. This is the concept of the fast breeder reactor (FBR).

Because absorption in the moderator is a major loss of neutrons in a thermal reactor, a fast reactor has an inherently superior neutron economy, as there are neutrons in excess of what is required to sustain the chain reaction. These neutrons can be used to produce extra nuclear fuel, as in the FBR, or to transmute long lived elements produced during reactor operation to less troublesome elements.

Early fast reactors used mercury cooling and plutonium metal fuel. Later, NaK cooling was used and molten lead–bismuth alloy cooling was used for naval propulsion units. The most recent generation of fast reactors uses MOX fuel and molten sodium cooling.

Fast reactors include:

- The Prototype Fast Reactor (PFR), 250 MW(e), at Dounreay in the United Kingdom;
- Phénix, the first fast reactor built in France, currently used for experiments on transmutation of nuclear waste;
- Superphénix in France, 1200 MW(e), shut down in 1997 owing to the high operating costs compared with PWRs;
- The Monju reactor in Japan, 300 MW(e), the only FBR power station still in operation in 2004;
- BN-350, constructed by the Soviet Union near the Caspian Sea, 130 MW(e) plus 80 000 t of fresh water per day;
- BN-600, constructed by the Soviet Union, 600 MW(e);
- A 40 MW(th) fast breeder test reactor in India.

As of 2004, new FBRs were planned or under construction in China and India.

A–3. RESEARCH FACILITIES

Research facilities such as those associated with the nuclear industry (e.g. low and zero power reactors, critical and sub-critical 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 application of prepared compounds, using radioactive materials directly or indirectly, can be divided into three major groups: research reactors and critical assemblies, nuclear research laboratories and hot cells, and general research laboratories that use radionuclides.

A–3.1. Research reactors and critical assemblies

Research reactors play a significant role in the field of nuclear science and technology. Since early prototypes were designed and put into operation in the 1940s, the number of research reactors worldwide has increased rapidly as a result of developments in the nuclear industry in general and nuclear power programmes in particular. Research reactors have also contributed substantially in the area of non-power applications such as radioisotope production for applications in nuclear medicine, agriculture and industry; neutron beam research; training; and neutron activation analysis, material development and neutron radiography. In total, more than 600 research reactors have been built and operated worldwide.

The picture has changed considerably over the past 5–10 years with the reduced demand for many of the aforementioned programmes, maturity of the nuclear industry, increased competitiveness of the radioisotope market, increased competition for R&D funds and escalating operation and maintenance costs of ageing reactors. The number of redundant reactors has gradually increased to the point that the number of shutdown/decommissioned reactors is now comparable with that of operational ones. The trend has been clearly visible for a number of years, and there are no signs of a reversal of this trend. Thus, more attention needs to be given to decontamination and dismantling of these older research reactors.

According to comprehensive statistics, over 650 research reactors have been built or are in the construction or planning phase throughout the world. Of these reactors, over 350 have been shut down and decommissioned to various stages. Well over 200 research reactors operating today are already 30 years old and will become likely candidates for decommissioning in the near term. Many of these reactors are located in Member States where appropriate decommissioning experience may not readily be available. Research reactors are ubiquitous, making their decommissioning an important international issue.

Specific attention to the subject of research reactor decommissioning is considered necessary because of the unique characteristics of research reactor facilities compared with other nuclear facilities. Significant aspects of research reactors that make decommissioning activities distinctly different from those of other nuclear facilities include:

- The broad spectrum of research reactor types, including prototype reactors;
- The broad range and specificity of experimental work carried out in research reactors (based on the impact of government policy, direction of national programmes);
- The proximity of some research reactors to the public domain.

In particular, attention to and planning for the eventual decommissioning of research reactors was generally poor in most countries while reactors were being designed, constructed, operated and shut down. Plans for decommissioning were at best “a rough conceptual plan” in most countries, and proper infrastructure was either missing or inadequate. This included the lack of decommissioning oriented regulations, record keeping, waste management and disposal sites, expertise, training and technologies. None of these aspects received proper planning attention owing to the inaccurate perception that decommissioning could be accomplished quite readily with minimal planning and the available resources. Complacency in decommissioning planning and implementation has resulted in undue delays and a lack of funding availability and other resources, and has ultimately led to extra costs [A–6].

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. Small research reactors (e.g. having thermal powers of less than 250 kW) will have relatively low radiological inventories after the spent fuel is removed prior to the start of their decommissioning.

A–3.2. 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, glove boxes and/or hot cells in which 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. Material presenting a risk of inhalation for workers or a significant contamination problem for the environment is usually handled inside a glove box. Applications typically occurring inside glove boxes are the mechanical and chemical preparation of fissile material for the fabrication of fuel and the handling of very low activity radioactive material for radiological characterization. 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 include post-irradiation analyses of fuel rods, small scale reprocessing of spent fuel, analyses of reactor material, material and waste characterization, and packaging and sealing of radioactive sources recently produced in high energy neutron flux reactors. Hot cells can be located inside a research reactor facility.

Fume hoods, glove boxes 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 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 glove boxes is also a potential issue for decommissioning. For decommissioning purposes, it is important to record not only the type of contamination (alpha, beta and/or gamma), 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 glove box, the residual material and equipment may be more difficult to decontaminate, especially if a buildup of radioactive substances occurs over many years of operation [A–6].

A–4. PARTICLE ACCELERATORS

Particle accelerators include a variety of devices (Van de Graaff and linear accelerators, cyclotrons and synchrotrons) used for the production of radionuclides. 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 meters 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 1–2 T. The particles are accelerated by an alternating electric field driven by a radiofrequency wave. The energy of the accelerated particles ranges from a few to 300 MeV, depending on the type of 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 with walls typically 2–4.5 m thick 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 stand-alone, self-shielded units. Typical applications of cyclotrons are for nuclear cross-section measurements and the production of radionuclides for radiopharmaceuticals.

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

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 meters) of low active waste. The decay periods needed to achieve conditions allowing for release from regulatory control can be several decades. Large accelerator facilities can become activated by neutrons to levels many times higher than permitted by release criteria. For decommissioning purposes, it is important to stress that only charged particle accelerators delivering beams with energies higher than a few million electronvolts per nucleon and a beam power of at least 100 W are able to induce significant activation in building structure materials.

A–5. WASTE MANAGEMENT FACILITIES

Prior to disposal, all radioactive material from facility operations or decommissioning must be collected, stored, treated, conditioned and packaged as required for storage in dedicated facilities. In addition to local waste treatment facilities built for specific waste management activities at individual nuclear installations, there are a number of large centralized waste treatment facilities for radioactive waste that either are located with a production plant within nuclear licensed sites or are located at sites dedicated to receipt and treatment of waste from a number of sources or from other sites [A–3]. Examples include the waste treatment complex for treating transuranic wastes, the Site Ion Exchange Effluent Plant (SIXEP), the vitrification facilities and the grouting plant at Sellafield in the United Kingdom; the Marcoule and La Hague facilities in France; and the Waste Receiving and Processing (WRAP) facility at Hanford, in the United States of America. Other facilities that provide a central service to many waste producers include the Waste Acceptance Monitoring and Compaction (WAMAC) facility at Sellafield; the Manufacturing Sciences Corporation (MSC) and GTS 'Duratek', both located at Oak Ridge in the United States of America; ECN in Petten, the Netherlands; and the Belgoprocess waste management facilities in Belgium.

In general, the processes used in these central facilities for treatment and conditioning of waste are similar to those employed for single waste streams or in conjunction with other production facilities. Typically, these large scale or centralized facilities are designed with some flexibility to accommodate a range of wastes and frequently group a number of process steps together. Some of the generic treatment and conditioning processes found in central facilities include [A–3]:

— Liquid effluent treatment such as ion exchange, precipitation and evaporation;
— Immobilization of solids, sludge and liquid waste, for example, cementation, vitrification, bituminization and polymer encapsulation;
— Solid waste treatment such as compaction, decontamination, incineration and metal melting.

The equipment for treatment of the radioactive material includes incinerators and compactors for solid waste, and evaporators, ion exchange columns, precipitation tanks, cementation units, storage tanks and associated pipework for liquid waste. Secondary waste also includes filters from off-gas and ventilation air cleaning.

High level waste storage tanks and highly active waste processing and treatment facilities are of particular concern for decommissioning owing to their very high activity levels [A–1].

Centralized facilities could offer reuse and recycling opportunities not available at dedicated facilities. The high throughput of material consigned to central facilities offers economic incentives that do not exist at smaller or dedicated facilities. By exploiting larger volumes of waste, the centralized facilities can offer sophisticated treatment options and enhanced flexibility.

A–5.1. Radioactive liquid waste facilities

Radioactive liquid waste facilities (RLWFs) store, treat and condition for disposal liquid waste generated by facilities and activities. This waste includes low level, high level and transuranic contaminated waste including enriched uranium and ^{233}U. An RLWF may be a separate facility or an adjunct to another facility.

A–5.2. Radioactive solid waste facilities

Radioactive solid waste facilities (RSWFs) store, treat and condition for disposal solid waste. This waste contains low level, high level and transuranic contaminated waste including radioactive mixed waste. An RSWF may be a separate facility or an adjunct to another facility.

A–6. WASTE STORAGE FACILITIES

A–6.1. Storage facilities for non-conditioned waste

Radioactive materials that have been generated are to be segregated as far as possible to facilitate subsequent treatment and conditioning in order to reduce waste toxicity, mobility and volume. It is important that Member States take into account their present and planned national waste management systems in the generation of radioactive waste. If no disposal facility will be available in the foreseeable future, storage facilities for non-conditioned waste materials have to be provided and/or the storage period of waste packages must be prolonged. Provisions for this situation must be planned well in advance. Waste characterization and implementation of an adequate quality assurance programme need to begin at the point of waste generation by establishing a file containing detailed information on the waste [A–6]. In addition, the following specific measures may be considered relating to storage of non-conditioned wastes:

— Ideally, interim storage facilities will not accept any new waste that is not adequately characterized. Every package must comply with the acceptance requirements of the storage facility.
— Non-immobilized wastes placed in containers for storage are to be segregated in the storage facility to the greatest extent practical. The packages also are to be segregated by waste type and storage duration.
— Facilities for the storage of waste need to have a design life that is independent of the packages placed in them. The safety assessment must support the suitability of the facility for the duration of its intended use.
— Storage of low contact dose rate low and intermediate level waste (LILW) needs to allow for visual inspection of packages. For prolonged storage, surveillance requirements may need to be modified during the period of storage. For the storage of high contact dose rate LILW, high level waste (HLW) and spent fuel, indirect controls may be put in place to inform the operators in the event of a failure.
— If the waste is characterized and conditioned properly within an adequate quality assurance programme, failure of the package is not expected. But in the event of package failure, for any reason, provisions for retrieval of a package and repacking must exist in the storage facility. If a package failure occurs during storage owing to the waste form or widespread external corrosion of the container, further action to eliminate the hazard for all packages of the same type needs to be evaluated. Packages that fail during storage and cannot be retrieved and remediated are to be isolated

in place, whenever possible, to protect personnel, the environment and neighbouring waste packages from contamination.

— Packages might be damaged if they fall during handling in a storage facility; in this case, the storage facility must provide for retrieval of the package as well as for decontamination of the area, if necessary.

It is important that waste retrieval be carefully planned and, as far as practical, preceded by a detailed survey and inspection of those waste packages to be retrieved. Retrieval activities ideally will follow a careful review of the manifest of all waste in storage. The sequence of retrieval of waste packages needs to take safety aspects into account, including the condition of the packages. In some cases, high dose rate or other hazardous waste may need to be given priority for attention. Each package in the interim storage facility is to be registered, and all records are to be kept up-to-date [A–6].

A–6.2. Storage facilities for conditioned waste

The main functions of a storage facility for conditioned radioactive waste are to provide safe custody of the waste packages and to protect both operators and the general public from any radiological hazards associated with radioactive waste. In this context, the facility must be capable of maintaining the 'as received' integrity of the waste package until it is retrieved for disposal. The storage facility must protect the waste from environmental conditions, including extremes of humidity, heat and cold, or any other environmental condition that might degrade the waste form or container. Local climatic conditions may result in the need to cool or dehumidify the store atmosphere to avoid possible deterioration of the waste packages [A–6].

Storage requirements mandate external dose rate and contamination limits for waste packages to be accepted by the facility in order to protect personnel. A maximum allowable dose rate at the surface of each package is to be defined for specific interim storage facilities or parts of facilities. In other respects, the storage facility usually adheres to the waste acceptance requirements of the disposal facility. The storage facility needs to minimize radiation exposure to on-site personnel through appropriate siting and shielding [A–6].

The storage facility may be associated with an area for inspection (including sorting and/or non-destructive examination), certification and labelling of waste packages. The storage facility is usually divided into areas where low contact dose rate packages are stored, areas where packages not meeting waste acceptance criteria are stored and a shielded area where high contact dose rate packages are kept secure. The design of the facility usually permits package stacking, sorting and visual inspection [A–6].

Normally, the design will include provisions for maintaining a database of the chain of custody of each waste package in storage. Key information about the waste package includes the total radionuclide content, the waste matrix used for immobilization, the treatment and/or conditioning method (as applicable) and the unique package designator. Ideally, a hard copy file will follow the waste package from conditioning to its final disposal [A–6].

Storage facilities are to be of modular design to cater to future expansion and must satisfy the code requirements for fire rating, seismicity, etc. Material handling facilities of adequate capacity also need to be provided.

Storage facilities normally will be designed to allow control of any contamination from gaseous or liquid releases. Adequate ventilation is to be available to deal with any gas generated during normal operation or possible accident conditions. Provision must be made for fire protection, radiation monitoring, and decontaminating individual containers and facility surfaces. Arrangements must be made to treat (or transport to a processing facility) potential accidental releases [A–6].

Storage facilities are often built in anticipation of a need and have inherent limitations concerning the types and quantities of waste package they might receive. Of necessity, storage facilities for non-immobilized waste (NIW) may also provide space for immobilized and fully conditioned waste. Also, the initial design often needs to be changed in terms of space required, floor loadings and type of waste storage required. Because of these uncertainties, design criteria for storage facilities ideally will take into account the following considerations [A–6]:

— Adequate segregated storage is to be provided for NIW and/or conditioned radioactive waste, taking into account anticipated future storage needs if several types of package are stored in the same facility. These needs are, in turn, determined by the waste processing requirements and capabilities, and the availability of specific treatment or disposal facilities, as well as package storage life and conditions. Non-immobilized waste should be stored in a form and manner that limit the risk of dispersion. Such waste must be segregated according to its hazard level. Waste containing short lived radionuclides that is to be held for decay must be segregated in a way that permits discharge as cleared waste when clearance levels are attained, as authorized by the regulatory authority.
— Emplacement, storage and retrieval of waste packages must be designed to keep exposure of personnel as low as reasonably achievable.
— The storage capacity of the facility must be designed to accept the maximum operational holdings expected from the system. The store

needs to contain enough spare capacity to accept the contents of another storage unit whose integrity may be breached or suspect. Appropriate equipment for transfer between operational and spare units must be available.
- In the design of storage facilities for conditioned radioactive waste, consideration must also be given to waste package handling; clear identification of stored waste packages and record keeping; provisions for inspection and monitoring of stored waste; provisions for preventing possible degradation of waste packages during storage; provisions for adequate environmental conditions (heating, cooling, humidity control) to ensure proper conservation of waste packages during their storage in the facility; provisions for cooling heat generating waste; provisions for fire protection where combustible waste is present; provisions for gas dissipation if gas generation is anticipated; provisions for criticality control where a considerable amount of fissile material is present in the waste; provisions for prevention of unauthorized access; retrieval of the waste for further treatment, immobilization or disposal, or in the event of an accident that requires relocation of the waste; and maintenance.

Buildings to be used for storage of radioactive waste preferably will be situated above the groundwater level, and must not be located in a flood plain. In cases where a subsurface storage facility is designed, this facility is to be constructed with appropriate systems to protect against in-leakage of groundwater [A–6].

A–7. SUPPORT AND DEVELOPMENT FACILITIES

Support and development units encompass different types of facility, including:

- Support facilities, which are usually located on large sites to provide general services to several other buildings. Typical examples include analytical laboratories, laundries for contaminated clothing, and maintenance and decontamination facilities for equipment or transport casks.
- Research laboratories, which vary in type, size and purpose, and include fuel fabrication laboratories, post irradiation fuel examination laboratories, chemical processing laboratories and material laboratories.
- Pilot plants, which are usually facilities where processes that have been developed in laboratories are tested on a larger scale. A large variety of

such plants exist, covering most of the processes used in the nuclear fuel cycle.

It is not possible to provide descriptions of all these facilities here, as they differ widely in size, structure, contamination (total activity content and radioisotopes), purpose and types of equipment. However, they can be categorized into the following two types:

— Facilities handling predominantly highly irradiating beta/gamma materials;
— Facilities handling alpha contaminated materials.

Beta/gamma facilities usually comprise several ventilated concrete/structural cells. The walls and floors may be covered with stainless steel, epoxy resin or paint, and the floors are often provided with drip trays/sumps. Cells may also be equipped with shielded windows, remote handling equipment (master/slave manipulators) or overhead cranes, etc. Such cells may house different types of equipment for processing, mechanical operation, monitoring, off-gas, services, etc.

In alpha handling facilities, if the materials are not highly irradiating, unshielded glove boxes are often used. These glove boxes are ventilated and may be connected to each other, and generally house equipment of a much smaller scale than in beta/gamma cells. If external radiation must be dealt with, glove boxes may be equipped with shielding or semi-remote handling equipment.

As noted above, features of these facilities are highly variable. Consequently, decommissioning requirements may be quite different, ranging from those applicable to facilities having minor radiological hazards to requirements applicable to reprocessing plants [A–1].

A–8. OTHER INDUSTRIAL FACILITIES

There are two aspects of the industrial uses of radioisotopes to be considered: first, the facilities in which sources and equipment containing radioactive material are manufactured, and, second, the diverse applications of the radioisotopes based on absorption and scattering phenomena (e.g. radiography, irradiation, sterilization, in the measurement of parameters like thickness, density, surface levels, humidity, in detection of smoke and static electricity, well logging, radiotherapy and nuclear medicine). Although the second topic is outside the scope of this document, it must be noted that improper management of sealed or unsealed sources can constitute a significant exposure risk.

The production of radionuclides is generally undertaken using high power reactors or multipurpose research reactors. The radioactive material generated is then shipped to the source manufacturer, where the radionuclide content generally determines how the material is to be handled. For gamma emitters and some beta emitters, all work with the radioactive material is conducted inside hot cells, whereas either a glove box or a fume hood is normally used for low energy beta emitters and alpha emitters. The ultimate form of the source for gamma sources and volumetric beta sources is a sealed capsule, while the most common form of a low energy beta and alpha emitting source is a plate or disc, typically on a stainless steel backing material.

Radioactive contamination at source manufacturing facilities 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 glove box, that contamination may be spread outside the containment system and into the enclosures of the facility. This risk must be taken into consideration when planning decommissioning.

REFERENCES TO THE ANNEX

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

[A–2] Innovative Technologies for Nuclear Fuel Cycles and Power Reactors (Proc. Int. Conf. Vienna, 2003), CRS Papers Series No. 24/P, IAEA, Vienna (2004).

[A–3] INTERNATIONAL ATOMIC ENERGY AGENCY, Recycle and Reuse of Materials and Components from Waste Streams of Nuclear Fuel Cycle Facilities, IAEA-TECDOC-1130, IAEA, Vienna (2000).

[A–4] INTERNATIONAL ATOMIC ENERGY AGENCY, Heavy Water Reactors — Status and Projected Development, Technical Reports Series No. 407, IAEA, Vienna (2002).

[A–5] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning Techniques for Research Reactors, IAEA-TECDOC-1273, IAEA, Vienna (2002).

[A–6] INTERNATIONAL ATOMIC ENERGY AGENCY, Interim Storage of Radioactive Waste Packages, Technical Reports Series No. 390, IAEA, Vienna (1998).

GLOSSARY

All terms and general definitions used in this report can be found in the IAEA Radioactive Waste Management Glossary[1]*, except for the following.*

ablation. The removal of material from a surface by vaporization, dripping or other erosive processes.

blasting. The process of propelling abrasive or dry ice particles (grid, pellets) to supersonic speed to impact, smooth, shape and clean a surface by removing thin layers of it.

cost–benefit analysis. A systematic economic evaluation of the positive effects (benefits) and negative effects of undertaking an action. Cost–benefit analysis may be used for optimization studies in radiation protection evaluations.

disposition. Consignment of decommissioning waste to some specified (interim or final) destination, for example, disposal, recycling, reuse or storage.

minimization. A concept that embodies the reduction of waste with regard to its quantity and activity to a level as low as reasonably achievable. Waste minimization begins with nuclear facility design and ends with decommissioning. Minimization as a practice includes source reduction, reuse and recycling, and treatment with due consideration for secondary as well as primary waste material.

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radioactive Waste Management Glossary, 2003 edn, IAEA, Vienna (2003).

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This report identifies and outlines issues for consideration during the design and operation of nuclear facilities to minimize waste generation, facilitate future decommissioning and optimize management of operational and decommissioning waste and material. It is aimed at the broad range of experts involved in the planning, design, construction and operation of new nuclear facilities or the modification of existing facilities. The principles discussed are applicable to all types and classes of nuclear facility dealing with radioactive material. While plant designs will continue to mature and evolve, the waste minimization options identified here will remain relevant to all new facilities and can be used as a checklist during the design, licensing and operational phases of new plants or the modification of existing plants.

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