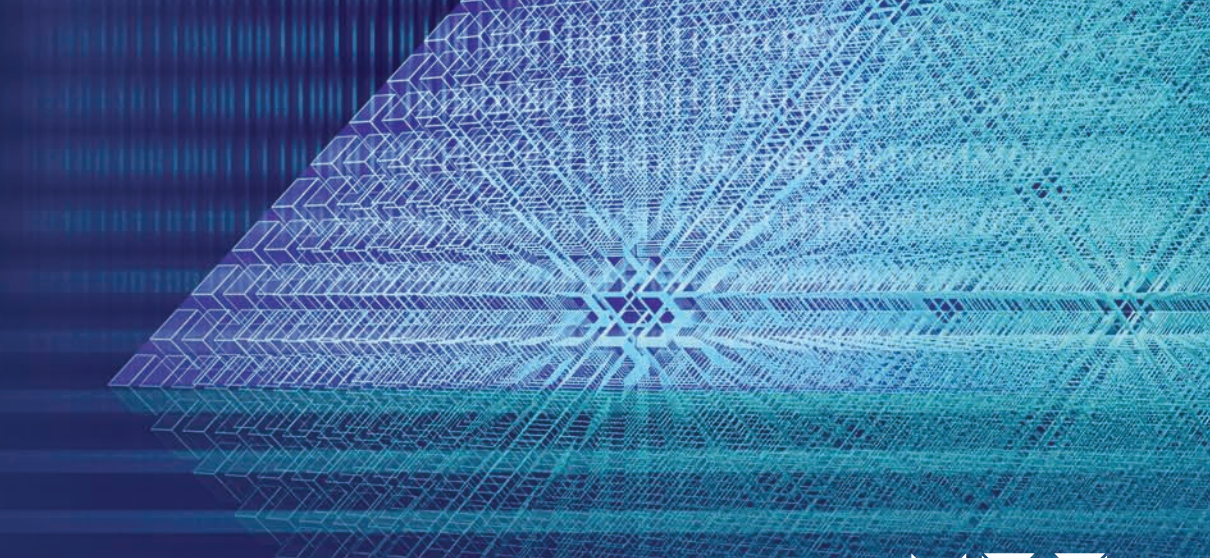

TECHNICAL REPORTS SERIES NO. 462

Managing Low Radioactivity Material from the Decommissioning of Nuclear Facilities

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MANAGING LOW RADIOACTIVITY MATERIAL FROM THE DECOMMISSIONING OF NUCLEAR FACILITIES

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2008

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1400 Vienna, Austria
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tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
March 2008
STI/DOC/010/462

IAEA Library Cataloguing in Publication Data

Managing low radioactivity material from the decommissioning of nuclear facilities. — Vienna : International Atomic Energy Agency, 2008.
p. ; 24 cm. — (Technical reports series, ISSN 0074–1914) ; no. 462
STI/DOC/010/462
ISBN 978–92–0–109907–5
Includes bibliographical references.

1. Radioactive substances — Management. 2. Radioactive wastes. 3. Nuclear facilities — Decommissioning. I. International Atomic Energy Agency. II. Series: Technical reports series (International Atomic Energy Agency) ; 462.

IAEAL 08–00507

FOREWORD

The IAEA's programme includes measures to support Member States in the planning and implementation of safe, timely and effective decommissioning of their nuclear facilities. The peculiar feature of nuclear as opposed to non-nuclear facility decommissioning is the presence of radioactivity. This hazard affects how dismantling and related activities are performed in order to ensure that personnel radiation doses are as low as reasonably achievable. In addition, it is necessary to manage the disposition of materials arising from decommissioning, recognizing the potential for these to produce a radiological hazard to the public and the environment.

Some of the decommissioning materials will be activated or contaminated, some not. Some continue to have an economic value and/or are in a form that can be recycled or reused; others with little or no economic value will need to be disposed of as wastes, possibly after storage if no appropriate disposal route is currently available. Much of the material arising from decommissioning will contain, at most, only small amounts of radioisotopes. For these materials there are substantial environmental and economic incentives to maximize the use of the principles of clearance from further regulatory control.

Options are presented in this report for the management of decommissioning material in order to inform the production of a materials disposition strategy consistent with current IAEA guidance. It includes a review of the relevant safety, regulatory, technological, economic, social and administrative factors influencing these options. The subject is examined in the context of the value, practicality and viability of the various disposition options, and the availability of suitable tools, techniques and instrumentation to monitor compliance with release criteria. The focus is at the lower range of radioactive concentrations in materials and wastes; high and intermediate level wastes are outside the scope of the report as disposal in dedicated repositories is assumed. Each of the range of disposition options discussed is feasible in principle and successful applications in Member States are described.

This report is intended to contribute to the systematic coverage of the entire range of decommissioning issues within the IAEA's decommissioning programme. It is hoped that this publication will assist in the flexible use of a wider range of disposition options than has sometimes been the case to date.

The IAEA wishes to express its appreciation to all those who took part in the preparation of this report. The IAEA officers responsible for this publication were M. Laraia, V. Efremenkov and P.J. McIntyre of the Division of Nuclear Fuel Cycle and Waste Technology.

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CONTENTS

SUMMARY

This report was written to assist in planning and implementing options for disposition strategies for the large amounts of lower activity material arising in the decommissioning of a commercially sized nuclear facility. It presents options for the management of the disposition of decommissioning material in order to inform the production of a materials disposition strategy consistent with current IAEA guidance. It includes a review of the relevant safety, regulatory, technological, economic, social and administrative factors influencing these options. The subject is examined in the context of the value, practicality and viability of the various disposition options, and the availability of suitable tools, techniques and instrumentation to monitor compliance with release criteria. The focus is at the lower range of radioactive concentrations in materials and wastes; high and intermediate level wastes are outside the scope of the report as disposal in dedicated repositories is assumed.

The nature of the materials arising from the decommissioning of facilities from different parts of the nuclear cycle is described. A range of disposal and recycling/reuse options are identified — disposal as low level waste; as very low level waste; or as radiologically cleared or clean waste; recycling/reuse in the nuclear industry; in nominated non-nuclear applications; or released for any use.

Factors that influence whether an option should form part of a site disposition strategy are discussed. These cover issues such as the quantity of material involved and the availability of radiological clearance criteria and technical solutions to achieve them. In addition, the future use of the site and the economics of the option in that context are also important. Finally, there needs to be a policy framework utilizing issues such as sustainability and environmental impact that encourage the appropriate utilization of all options and are likely to provide the necessary public acceptance. Guidance is provided on the selection of practical techniques and instrumentation to support the selection and utilization of the options discussed, within the constraints of existing clearance criteria.

It is concluded that each of the options examined is feasible in principle as demonstrated through a description of successful applications in Member States. A flexible approach is recommended that allows the selection of the optimum approach when considering safety, environmental impact, cost and other factors.

1. INTRODUCTION

1.1. BACKGROUND

As a rising number of nuclear facilities approach or reach the end of their operating lives, decommissioning planning and implementation have become increasingly important activities in Member States. An important aspect of such activities is the management of materials arising from decommissioning, some of which are activated or contaminated, some not. Some continue to have an economic value and/or are in a form that can be recycled or reused; others with little or no economic value will need to be disposed of as wastes, possibly after storage if no appropriate disposal route is currently available.

The purpose of decommissioning is the safe removal or treatment of facilities and materials from a nuclear site to allow the site, facilities or materials to be conditionally or unconditionally released for reuse or other purposes. In so doing, a subsidiary aim should be to maximize such reuse or recycling (in nuclear or non-nuclear applications). Materials that cannot be conditionally or unconditionally released or reused have to be treated as radioactive wastes.

The IAEA has published fundamental safety principles that apply to material that has been defined as radioactive waste by the appropriate national authorities, and to the facilities used for the management of this waste from generation to disposal [1]. The seventh principle states that "…people and the environment, present and future, must be protected against radiation risks." In particular, "...the generation of radioactive waste must be kept to the minimum practicable level by means of appropriate design measures and procedures, such as the recycling and reuse of material."

Materials management is a critical aspect of the decommissioning process, especially of commercial scale facilities. Managing tens of thousands of tonnes of decommissioning material is a significant task and requires a dedicated organization. Moreover, the costs of radioactive waste management are a significant element of the overall decommissioning costs and may even dominate in some cases. In Germany, it has been estimated that about 60% of the costs of decommissioning are attributable to waste management, including the costs of storage for 30 years. This is so even though only 3% of the materials arising from decommissioning are declared as radioactive waste [2]. This emphasizes the need for an accurate radiological characterization of materials and for seeking to maximize opportunities for the reuse or recycling of materials so as to minimize the amounts requiring treatment, storage and disposal as radioactive waste [3].

Much of the material arising will contain only small amounts of radioisotopes, if any. For these, there are substantial incentives to maximize the use of the principles of clearance, defined as the removal of radioactive materials or radioactive objects within authorized practices from any further regulatory control by the regulatory body. Environmental and sustainable development considerations encourage reutilization of non-renewable resources by way of direct reuse of equipment or buildings and by recycling usable materials. Furthermore, there may be worthwhile intrinsic value in recycled materials such as metals, or crushed concrete for construction [3]. Furthermore, the cost of disposal of conventional (non-radioactive) waste is generally much lower than that of radioactive waste.

The principle of clearance has already been utilized successfully in several countries. However, Member States are free to set their own clearance levels, and any inconsistencies have the potential to cause some difficulty for international trade and for shipments across international boundaries. The promulgation of IAEA guidance in this field [4] aims to promote harmonization in this area and should also encourage more usage of the flexibility available than is currently achieved in some countries or by some operators.

Despite the inherent benefits of a policy of utilizing clearance, there can be significant costs in reclaiming scrap, equipment and other materials. These include among others the costs (and extra man-sieverts) of labour resulting from decontamination and monitoring to ensure compliance with clearance criteria, the costs of administering the recycling programme, and the costs and other implications of managing the secondary wastes associated with these activities. In addition, these costs increase as the clearance levels decrease due to the need for a more demanding monitoring schedule when making measurements at low activity levels as well as the potentially greater decontamination effort required to achieve these levels [5].

Clearance for any future use may or may not be profitably applied to the disposition of one or more waste streams from the decommissioning of any given nuclear facility. Pursuit of a path of unrestricted release requires systematic assessment of the pros and cons as it is not always the optimum route. Other options may be more cost effective in some circumstances. These could include release restricted to specific applications (either nuclear or non-nuclear) or disposal as radioactive waste. In many countries, the decommissioning implementer is currently limited to a choice of unconditional release or disposal as low level waste (LLW). The implementer should be given access to a wider variety of options, thereby maximizing the flexibility and cost effectiveness of the overall strategy for disposition.

1.2. OBJECTIVE

This report has been written to assist in planning and implementing options for disposition strategies for the large amounts of lower activity materials arising in the decommissioning of a commercially sized nuclear facility. Specifically, it seeks to:

— Identify the range and source of materials arising from the decommissioning of nuclear facilities, particularly where volumes are expected to be large (Section 2);
— Describe options that could form part of a disposition strategy (Section 3);
— Propose, characterize and evaluate the factors that need to be considered in selecting strategic options for implementation (Section 4);
— Provide practical guidance on the methodologies for the flexible implementation of the chosen strategy based on experience in Member States (Section 5);
— Illustrate the issues by means of relevant experiences and lessons learned (annexes).

1.3. SCOPE

This publication presents options for the management of the disposition of decommissioning materials in order to inform the production of a materials disposition strategy consistent with current IAEA guidance. It includes a review of the relevant safety, regulatory, technological, economic, social and administrative factors influencing these options. The subject is examined in the context of the value, practicality and viability of the various disposition options, and the availability of suitable tools, techniques and instrumentation to monitor compliance with release criteria. The focus is at the lower range of radioactive concentrations in materials and wastes; high and intermediate level wastes are outside the scope of the report as disposal in dedicated repositories is assumed.

The information is intended primarily for nuclear operators, decommissioning contractors, waste managers and all those who are evaluating, planning or implementing the decommissioning of nuclear facilities and the associated management of potential decommissioning wastes. It should also be of interest to those who are developing the regulations and radiological criteria relevant to this work as well as those inspecting decommissioning activities.

In principle, the report is relevant to all types of nuclear facilities, but it is particularly relevant to those major nuclear facilities (e.g. fuel cycle and nuclear power plants) where decommissioning generates very large amounts of material requiring careful management.

2. MATERIALS ARISING IN LARGE QUANTITIES DURING DECOMMISSIONING

2.1. OVERVIEW

Potentially radioactive material and wastes may arise throughout the lifetime of a nuclear facility. The arisings broadly fall into three types:

- Operational wastes in the form of solids, liquids and gases.
- Life expired or failed plant components arising as a result of maintenance, modification or life extension work (e.g. steam generators, pumps, valves, control rods, spent filters, etc.) and potentially including local arisings of contaminated material associated with the failure or replacement of the components.
- Materials from the structure of the facility (e.g. steel, concrete, aluminium, graphite, etc.) and the plant, and equipment and services housed within. Most were expected to last the lifetime of the facility and generally only arise in large quantities upon decommissioning.

Wastes in the first category are normally dealt with as they arise, and the facility will generally have treatment and conditioning processes, and a plant to deal with these wastes. However, in older facilities in particular, the design approach may have been to store such materials in an unprocessed form so that a significant challenge in decommissioning is the retrieval, processing and disposal of these wastes.

Some of the items in the second category may be managed within the existing operational waste treatment facilities or it may have been planned that they would be stored until final decommissioning of the facility.

Materials in the third category would only be expected to arise during decommissioning. Given the large volumes of materials, they will require careful management to ensure their disposition for reuse, recycling, storage or

disposal is both environmentally and economically appropriate. This publication focuses on these materials.

2.2. RANGE OF DECOMMISSIONING MATERIALS AND WASTES

Some of the materials and wastes arising from decommissioning may differ from the wastes generated during normal operations or routine maintenance of the facility in terms of their mass, volume, chemical, physical, radiological and toxic characteristics. Due to these differences, some of these materials may be considered problematic in that the methods of treatment, conditioning and disposal routinely used at a facility during operation may not be adequate for decommissioning. Wastes similar to those produced during operation will often be able to be managed using established treatment facilities, and storage and disposal arrangements. Further capacity or even techniques may need to be developed if larger quantities are likely to arise in decommissioning. Leaving aside such operational wastes, much of the material arising from the decommissioning of a nuclear facility will be either not radioactive or at most only slightly so. For the remainder, i.e. only a limited proportion of the whole, disposal as radioactive waste may be the only option. Most of the inactive material will be concrete and brick rubble from building demolition, site cleanup material (soil, roads), material from secondary plant pipework and other auxiliary facilities. Thus, a major effort in material/waste management is concerned with segregation, measurement for verification of release limits and decontamination to reach those same limits.

Other materials of higher radioactivity content may arise in larger volumes, including very large items of a plant such as heat exchangers and graphite cores. These may also include so-called ‘exotic’ or special wastes containing toxic or hazardous materials, such as sodium, beryllium, lead, asbestos, etc. In addition, there are large quantities of material which are not radioactive but which, because they arise on a nuclear licensed site, are also potentially subject to regulatory control. If there is any concern that they may have become contaminated by other materials on the site, then these are sometimes termed ‘suspect wastes’ requiring assessment and possibly monitoring before release [3].

Section 2.3 describes the nature of the materials that may arise in the decommissioning of specific types of facility.

2.3. TYPICAL MATERIALS ARISING IN THE DECOMMISSIONING OF SPECIFIC NUCLEAR FACILITIES

This section describes the nature of materials arising from the decommissioning of facilities from different parts of the nuclear cycle. The typical processes used in the refining, conversion, enrichment and fuel fabrication stages are given, as well as an overview of the origin, types and quantities of waste generated during these processes [6]. The management of materials from facilities housing irradiated fuel is covered in terms of reactors and spent fuel reprocessing plants, and some coverage is also provided for smaller, more specialized facilities.

2.3.1. Refining and conversion

Refining is the processing of uranium ore concentrates (UOCs) to produce uranium trioxide (UO_3) or uranium dioxide (UO_2). This process may be carried out on a single site or as part of an integrated process involving more than one site.

Conversion is the processing of UO_3 or UO_2 to produce uranium hexafluoride (UF_6), although uranium tetrafluoride (UF_4) may also be prepared for the production of metallic uranium. UF_6 is the only uranium compound which is suitable for enrichment due to its thermal stability and relatively high volatility. All current enrichment processes are based on the use of UF_6.

Uranium conversion facilities are similar to many chemical plants. The processes are generally carried out in closed systems within vessels and linking pipework, and involve the handling of wet solids, solutions, solvents and gaseous products. As such, the plant components become contaminated by process fluids both on internal surfaces (especially in dead legs, crevices, etc.) and externally as a result of leakages and spills. Normally, only uranium isotopes and their daughters are present, and the radiation hazards are consequently low. Health risks can arise from the heavy metal nature of uranium and from the properties of other substances used in the process, such as acids, organic solvents, fluorine and hydrogen fluoride.

The size of a conversion plant depends on the technology used and the production capacity. It may be very large, occupying an area of several hectares. Usually, the process takes place across several buildings, where individual process steps are carried out. Adjacent areas may contain basins, ponds and lagoons for sludges, extraction wastes and sewage. Apart from their 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, being analogous to that of a conventional chemical plant. Complications may arise if reprocessed uranium is recycled. In this case, the possible presence of contaminants, such as ^{99}Tc and ^{232}U daughters, has to be taken into account.

2.3.2. Enrichment

Enrichment involves increasing the proportion of ^{235}U, from the natural level of 0.7% to an average level of 3–5%, in UF_6. This is done predominantly using two different industrial methods — by gaseous diffusion or by the use of centrifuges.

The decommissioning of these facilities also parallels that of many chemical plants. The radiological hazards are similar in nature to those in the later stages of conversion due to the gaseous forms (UF_4 and UF_6) of uranium used and the potential for criticality incidents; the latter issue being increased due to the presence of enriched uranium. In addition, given the adaptability of enrichment processes to the development of uranium based nuclear weapons, the decommissioning of such a plant introduces special security issues. Steps may be taken to ensure that sensitive components are destroyed. The usual recycling process for the many aluminium components, of melting into ingots, is itself destructive.

The decommissioning of such facilities tends to be largely repetitive and involves the dismantling of a very large number of identical units, installed inside very large buildings.

2.3.3. Fuel fabrication

Fuel fabrication is the manufacture of the fuel assemblies for loading into the nuclear reactor. For fuel fabrication, two products, uranium dioxide and metallic uranium are used as starting materials. Only natural uranium is used for production of metallic uranium fuel. Natural or enriched uranium dioxide can be used for oxide fuel manufacture.

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 house laboratories, waste treatment facilities, a waste recycling plant and other auxiliary facilities, such as tanks and pumps, warehouse and storage areas [3].

Fuel fabrication plant decommissioning may require special criticality precautions in addition to personnel protection against alpha emitters. The chemical toxicity of uranium compounds also requires consideration, especially for powders (UO_2 or UF_4) or soluble compounds, e.g. uranyl nitrate. The

pyrophoric nature of finely divided metal and turnings is a further important consideration for facilities which handle and machine uranium metal.

In the mixed oxide (MOX) fuel technology, plutonium is used as a raw material for mixed oxide fuel for reactors. A mixed oxide fuel fabrication plant is designed to produce (Pu-U)O_2 fuel pellets and to incorporate these pellets into clad fuel rods. The plant may use a process involving blending of plutonium and uranium solutions, followed by co-precipitation and calcination to form MOX fuel. Recently, a mechanical blending process has become more common.

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 critically-safe vessels located in glove boxes. The facilities for fabricating MOX fuels are relatively small in size.

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

2.3.4. Nuclear power plants

There are a large number of nuclear power plants in operation across the globe. Much work has already been done on the planning and, to a lesser extent, implementation of decommissioning. The literature on decommissioning of these facilities is considerable [7].

From the radiological point of view, reactor components may be divided into two separate groups:

(a) The reactor itself with the pressure vessel, internal structures and biological shielding, the constituent materials of which are primarily activated and account for more than 90% of the total activity in the installation;
(b) The complete coolant circuits and secondary installations, which are primarily contaminated.

Reflecting these differences, consideration may be given to preserving the reactor for a long period but rapidly dismantling the coolant circuits and auxiliary plant after decontamination in order to reduce the annual cost of surveillance and maintenance of the plant.

Table 1 provides quantities of potentially radioactive material generated from the complete decommissioning of typical gas cooled reactors (GCRs) and PWRs [8]. (A reference reprocessing plant with a capacity of 5 t/d [9] is also presented — see Section 2.3.5.)

The radioactivity level of most of these materials is usually low. Generally, they should be available for unconditional release after cleaning and/or decontamination to the required release levels. The quantities in this illustrative table are dominated by steel and, in the case of a GCR, also by graphite, the latter being discussed further later in this subsection. A major task in decommissioning will be the management of large quantities of concrete, not shown in this table, which may not be significantly activated but will have to be shown not to be contaminated, particularly along joints and other potential leak paths.

Because of the process of radioactive decay, the quantity of radioactivity decreases with time after plant shutdown, particularly for reactor components where ^{60}Co is dominant. As such, deliberately delaying the decommissioning and demolition of a plant, or conducting it in time separated stages, will result

TABLE 1. TYPICAL RADIOACTIVE MATERIAL GENERATED FROM DECOMMISSIONING

Radioactive material generation	250 MW(e) GCR (t)	900–1300 MW(e) PWR (t)	Reprocessing plant 5 t/d throughput (t)
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

in a subsequent decrease in the radioactive inventory over time and can significantly reduce the quantities of materials with higher radioactivity levels.

The approximate masses and activities of steels from the active areas at various times after shutdown in a 1000 MW(e) PWR are given in Table 2. The main benefit from radioactive decay is usually from a reduction in gamma radiation levels. The table demonstrates the decreasing proportion of beta–gamma emitters in low level radioactive steels resulting from the decay of radionuclides, such as ^{60}Co [10].

When comparing 5 and 25 years after shutdown, the amount of steel contaminated to levels higher than 0.1 Bq/g or 0.37 Bq/cm^2 decreases by about 50%. After 100 years, this proportion decreases by about 75%. Further reductions may be made by decontamination. Some quantities of tritiated water vapour may arise during decommissioning or melting operations. If necessary, removal of the tritiated water vapour from the ventilated air can be accomplished [11].

In a survey by the OECD Nuclear Energy Agency (OECD/NEA) of the variability of total weight of radioactive materials within and between reactor types, the data (Fig. 2.5 of Ref. [12]) shows that GCRs have the largest quantity of radioactive material, with substantial variability within the data for each reactor type. It is noted in Ref. [12] that this feature of GCRs is even more marked when the weight of materials is normalized by electrical power output (MW(e)).

A significant additional component of the total materials in a GCR is graphite. The great majority of the radioactive graphite is associated with the bulk moderator and reflector graphite, together with shield-wall graphite (or

TABLE 2. EFFECT OF DECAY ON MASSES AND ACTIVITY OF STEELS FROM A 1000 MW(e) PWR

		Time after reactor shutdown					
		5 years of decay		25 years of decay		100 years of decay	
Surface activity (Bq/cm^2)	Average activity (Bq/g)	Mass (t)	Total activity (Bq)	Mass (t)	Total activity (Bq)	Mass (t)	Total activity (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
		99.9% beta–gamma, 0.1% alpha		99% beta–gamma, 1% alpha		95% beta–gamma, 5% alpha	

other carbon bearing material) in certain cases. Permanent moderator and reflector blocks are also present in some other reactor designs.

The radioactive graphite coming from nuclear installations has different characteristics to other radioactive waste due to its physical and chemical properties, and also because of the presence of significant amounts of tritium and ^{14}C. Even after many years of irradiation, graphite retains most of its good mechanical properties, is relatively insoluble and is not otherwise particularly chemically reactive. It therefore appears to fulfil most of the general requirements for a solid radioactive waste suitable for disposal. However, the evaluation of the radioactivity inventory of graphite moderators and other irradiated graphite components shows that this graphite cannot be accepted by existing disposal sites without particular conditioning.

Different options have been studied for the management of radioactive graphite, but generally accepted solutions for its conditioning and/or disposal have not yet been decided. In practice, the main option is for a period of long term storage before final disposal. Three basic solutions are often proposed for disposal of waste graphite [13]:

— Direct disposal after suitable packaging;
— Disposal after incineration with consequent ash conditioning and with efficient filtration system of the off gas;
— Disposal after chemical treatment (liquid and/or gaseous extraction), conditioning (impregnation) and proper packaging.

2.3.5. Spent fuel reprocessing

A number of reprocessing plants have now been decommissioned and hence data are available on the waste arisings from these operations. Table 3 lists the arisings and the disposition of materials from the decommissioning of one of these plants, the WAK reprocessing plant at Karlsruhe, which had an annual throughput of ~30 t U (see also column 3 of Table 1). The high level liquid waste arisings were operational waste, which had been stored to await conditioning by vitrification upon final plant decommissioning. The table illustrates the domination of the decommissioning wastes by structural materials; 98% of these were cleared for recycling/reuse.

Fuel cycle installations and, more particularly, reprocessing plants are usually contaminated by alpha emitters and fission products. Even after several decades, the resulting radioactive decay is not of significant benefit for worker protection, radioactive material management or potential minimization of decommissioning waste. In this case, the radioactive material is contained partly by dynamic sealing, which means that the ventilation systems must be

TABLE 3. WASTE ARISINGS FROM THE DECOMMISSIONING OF THE WAK REPROCESSING PLANT

Material	Quantity (m^3)	Nature	Fate
High level liquid waste	52 (vitrified)	Radioactive waste 130 of 400 L casks	Storage
Contaminated plant, decontamination, wastes, etc.	1681	Conditioned radioactive waste (3360 m^3)	Disposal
	2840	Cleared	Recycling/reuse
Building rubble, structural materials, etc.	2279	Conditioned radioactive waste (4560 m^3)	Disposal
	110 000	Cleared	Recycling/reuse

kept running. Also, in spite of washing down and various forms of decontamination, a risk of corrosion from the chemicals used during operations remains. As a result, the annual cost of maintenance and surveillance can be substantial, leading to great expenditure with minimal benefit. This makes early dismantling desirable [14]. A key issue for plants with a plutonium inventory is that radiation levels rise with time due to ^{241}Am grow-in; therefore, early cleanout is vital.

2.3.6. Research, institutional and industrial facilities

Research, institutional and industrial facilities vary widely in the nature of the work undertaken and therefore so also do their operational wastes [15]. Materials arising from the decommissioning of research reactors will have much in common with those from nuclear power plants, being dominated by concrete, steel (or aluminium) and often graphite, although there may be special considerations depending on the experimental use and operational history of the reactor.

Decommissioning materials may include stored operational wastes generated in small quantities from experiments or operations, the scope of which changed over time. They therefore can have unique characteristics. As a result, it is difficult to define generic waste streams for these processes and solutions on disposition are likely to be needed on a case by case basis. However, the structural materials generated by decommissioning will be similar to those for other types of facility but are likely to be relatively low in

volume and, except for research reactors, without activation products. The issues related to disposition of these materials are reasonably bounded by the considerations for other facilities.

3. STRATEGIES FOR THE DISPOSITION OF LARGE AMOUNTS OF DECOMMISSIONING MATERIALS

3.1. GENERAL CONSIDERATIONS

As described in Section 2, the decommissioning of nuclear installations produces a wide range of materials. Of those which are classed as radioactive, most only show low levels of surface contamination. Some reactor components will also be activated and this will have penetrated some distance into the material. Having been used or having been brought for a while into a controlled area marks them as 'suspect material'. They can only be withdrawn from the radioactive waste management system by a thorough and intensive demonstration that any possible residual contamination is below defined criteria.

Depending on national policy, these materials can either be [5]:

— High and intermediate level wastes which, as stated in Section 1.2, are outside the scope of this report.
— Radioactive waste, which has no economic or practical value. This would be sent for disposal. Depending on its contamination levels, this waste could be sent for disposal as LLW or very low level waste (VLLW), depending on the categorization of radioactive waste, available infrastructure, and other factors and constraints.
— Components that are radiologically clean or have been decontaminated to bring their activity to below clearance levels. These items can be released for unrestricted use if it is economical and practical to do so. Indeed, some components such as tanks, pumps, motors and valves can be reused in industry largely as they are. If no economic use is available, then the items will be sent for disposal as conventional industrial waste (e.g. in a municipal landfill site) appropriate to their physical, chemical or toxic characteristics.
— Components whose activity levels can be reduced to levels acceptable for restricted nuclear use or defined non-nuclear applications (e.g. for

smelting, for recycling under predetermined conditions or for controlled disposal in a municipal landfill).
— Materials and wastes kept in storage to allow decay to bring radioactivity levels to below clearance levels or from LLW to VLLW. This interim approach can also be pursued pending the availability of one or more of the three above mentioned situations.

As referred to in Section 1, the IAEA Fundamental Safety Principles [1] stipulate that "…the generation of radioactive waste must be kept to the minimum practicable level by means of appropriate design measures and procedures, such as the recycling and reuse of material." This includes the selection of appropriate technology, the selection and control of construction and operational materials, the recycling and reuse of materials, and the implementation of appropriate procedures. Emphasis should be placed on the segregation of different types of material in order to reduce the volume of radioactive waste and facilitate its management. The environmental goal of waste minimization discourages the production of these wastes.

In addition, it is important to minimize the spread of radioactive contamination with a view to reducing to the strict minimum the need for decontamination, and hence also minimize the creation of secondary waste. It is desirable that use be made of all means of preventing contamination, to the extent that they are economically justified and do not lead to additional risks and complications in decommissioning operations [16]. As such, waste minimization can be considered as a key part of a strategy for avoiding, as much as possible, the production of these undesirable by-products. Where by-products arise, steps are required to minimize their volumes [17].

There are strong incentives to minimize the generation and release of decommissioning wastes, and their associated costs and hazards. Sustainability, environmental and economic considerations are major driving forces when considering recycling and reuse rather than disposal for both radioactive and clean materials. Nevertheless, other factors such as the likelihood of regulatory approval and stakeholder acceptance also need to be taken into account. Consequently, some level of optimization is an inherent part of determining whether recycling and reuse practices could be applied on a larger scale in a particular case or at a particular facility in the nuclear industry [18]. The factors and constraints that may influence disposition options are discussed in detail in Section 4.

3.2. POTENTIAL APPROACHES FOR THE DISPOSITION OF DECOMMISSIONING MATERIALS/WASTE

As discussed above, a range of options is available for disposition of decommissioning materials. An important element of the decision making process is the methodology for evaluating and characterizing the use of an option for a given material or waste stream with respect to cost effectiveness, viability and safe management.

Figure 1 provides a simplified overview of the options available for different decommissioning materials. It covers the segregation and routing of suspect or low level radioactive materials arising from decontamination and decommissioning activities through to the final disposition options. For clarity, not all possible paths are shown.

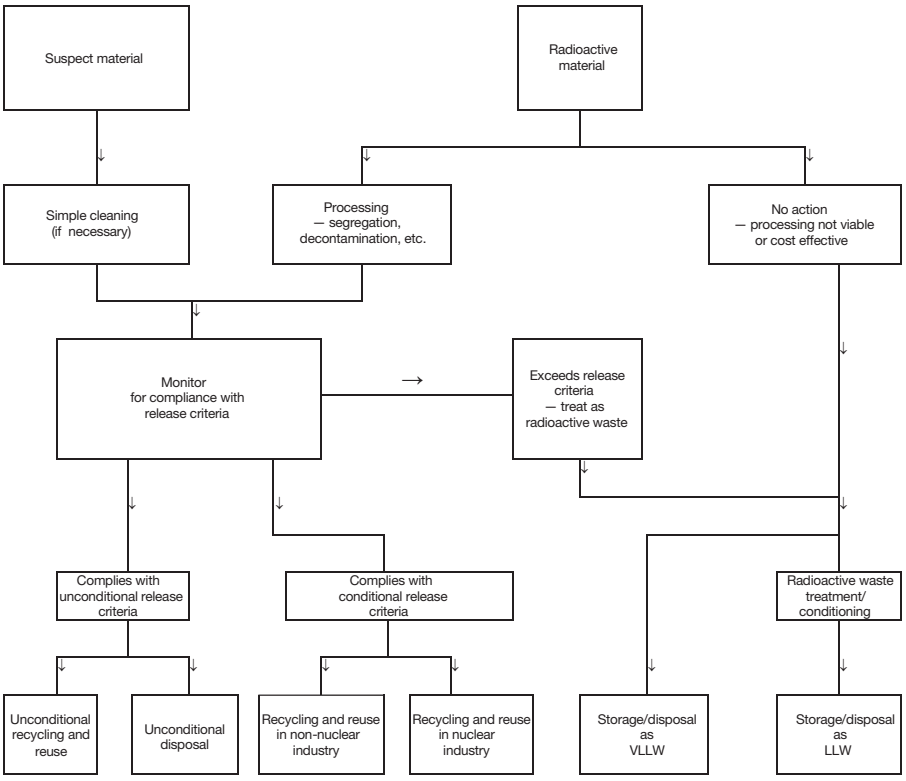

FIG. 1. Disposition options for the segregation and characterization of suspect and radioactive material.

By referring to Fig. 1, six disposition options may be identified:

— Unconditionally released materials for recycling or reuse (Section 3.3);
— Unconditionally released materials for disposal (Section 3.4);
— Radioactive material for conditional recycling or reuse within the non-nuclear industry (Section 3.5);
— Radioactive material for conditional recycling or reuse within the nuclear industry (Section 3.6);
— LLW disposed of to an engineered repository or storage facility (Section 3.7);
— VLLW disposed of to an appropriate storage or disposal facility (Section 3.8).

These options range from disposition methods implemented in all countries to those so far utilized in only a minority of countries or situations. A fully comprehensive disposition strategy for a facility or site should preferably be open to the appropriate use of all alternative options or a combination of options depending on the materials arising.

These disposition strategies largely correspond to the IAEA classification of radioactive wastes (Table II of Ref. [19]). At the lowest end of the activity spectrum, unconditionally released materials are those cleared from nuclear regulatory control in that their concentration levels are below clearance levels. These wastes can be safely disposed of, applying conventional techniques and systems, without specifically considering their radioactive properties. In parallel, reusable materials below clearance levels may be reused or recycled in an unrestricted manner.

Generic clearance levels have been recently promulgated by the IAEA [4]. These criteria offer the advantage of being simple to use, and avoid case by case recalculations and negotiations. However, the current situation is that the approach to clearance and related criteria vary from country to country, with the primary requirement on the decommissioning implementer being to comply with existing national legislation. Where criteria are generic and/or based on conservative assumptions, their indiscriminate use may lead to significantly higher costs compared to establishing criteria for a specific material or waste in a specific reuse, recycling or disposal application.

Such higher costs can be short term or long term in nature. The short term costs result from increased instrumentation and monitoring requirements, increased volumes consigned as radioactive waste and the consequent disposal costs. Longer term costs (possibly of a considerable scale) may result from the cost of building and operation of engineered storage facilities in order to take advantage of decay or in lieu of disposal facilities. Ultimately, there could also

be additional costs from the need to increase the size of disposal facilities to cope with larger volumes of radioactive waste.

In the IAEA classification of radioactive waste, the category of waste after cleared material is LLW. There are two subcategories defined in Ref. [19], short lived waste and long lived wastes. Depending on their specific characteristics, these wastes can be disposed of into near surface or geological disposal facilities. Neither disposal facilities are inexpensive to build and operate. Some countries have established dedicated facilities for VLLW where the design criteria to be met, and related cost, need not be as demanding as those for general LLW, due to the lower radioactivity levels present.

The following sections explore these disposition options in more detail with the objective of offering the decommissioning implementer the opportunity of considering a more flexible approach to their material and waste issues.

3.3. UNCONDITIONALLY RELEASED MATERIALS FOR RECYCLING OR REUSE

During the decommissioning of nuclear facilities, quantities of valuable metals and equipment may become available for unconditional recycling or reuse, provided that the radioactivity on or in them can be reduced to acceptable levels. Work has been performed by the IAEA, OECD and EC to define the basis for establishing suitable criteria for unconditional release, and for applying these criteria to actual waste management and decommissioning cases [20, 21]. The challenge is to define economic processes that adequately demonstrate that all of the material has been cleared in accordance with these criteria.

Again, it must be emphasized that national laws define the process and criteria. To assist in effective and efficient release of materials, the United Kingdom nuclear industry has jointly developed a code of practice on clearance and exemption that is likely to be adequate when making demonstrations to regulators in support of waste management proposals [22]. In Spain, an agreement has been established between the regulatory body, the decommissioning organization and metal recyclers to accept materials released for unrestricted use from nuclear sites [23].

Some countries have already implemented unconditional release of materials on a case by case basis. For example, 900 t of metal scrap from the operation of the Würgassen nuclear power station is to be decontaminated and unconditionally reused as scrap metal [20]. The nuclear ship Otto Hahn (Germany) was fully decommissioned by removing all the nuclear parts and

cleaning up any residual activity. The vessel can now be used as a conventional ship [20]. From the decommissioning of the Niederaichbach power station KKN, a total of about 3500 t of metal was unconditionally released [24]. Examples of materials (carbon steel, stainless steel, concrete, soil, gravel and others) that have been disposed of with radiological restrictions or various metals that have been conditionally recycled through melting in a specific melter are cited in Ref. [21].

Remelting of metals is a method of decontamination that guarantees homogeneity of the final product, and facilitates sampling and proving acceptability for unconditional release. In this case, the reduction in activity of the final product results firstly from the decontamination processes before melting and then by partition of the radionuclides to the melt, slag and dust during the melting operation. The EC has investigated melting and concluded that it is a promising method for conditioning steel waste with the purpose of volume reduction, immobilization of radioactivity and possible recycling of the steel [20].

Suitable criteria, measurement methodologies and instrumentation must be available to facilitate any release practice. This is particularly important for material for unconditional release, which usually represents a large proportion of all decommissioning materials. Release criteria should be established in a manner that achieves safety requirements but also encourages waste minimization through recycling and reuse. Achieving international consistency in criteria definition can improve the prospects for wider reuse of a cleared material. In addition, care needs to be taken in the development of these criteria, as current instrumentation may be incapable of measuring these standards on an economic industrial scale, leading to the potentially unnecessary consignment of materials as radioactive waste.

3.4. UNCONDITIONALLY RELEASED MATERIALS FOR DISPOSAL

If it is neither practical nor economical to reuse components or materials that are radiologically clean or have been decontaminated to bring their activity to below clearance levels, they may be sent for disposal as conventional waste (e.g. in a municipal landfill site), usually a lower cost option than disposal as radioactive waste. The wastes may be subject to special disposal provisions depending on their physical, chemical and toxic properties. Such provisions should not differ from the provisions defined for the disposal of other similar industrial or municipal waste materials. This means these wastes can be safely disposed of, applying conventional techniques and systems, without special

measures related to their radioactive status, probably again at lower cost than if treated as radioactive waste.

In order to dispose of clean or cleared waste materials, on-site or off-site disposal options may be considered. A particular example of on-site disposal of unconditionally released materials is 'rubblization', whereby above ground structures are partially decontaminated, demolished and disposed of in the below ground portions of the structures.

3.5. RADIOACTIVE MATERIAL FOR CONDITIONAL RECYCLING/ REUSE WITHIN THE NON-NUCLEAR INDUSTRY

The intent of providing limits for the clearance of materials with residual quantities of radioactivity is to ensure the protection to the public from exposure to the radioactivity, i.e. to meet statutory health protection objectives. Ref. [25] describes alternative approaches that meet these health protection objectives. These alternatives include material or application specific criteria (such as the release of metals for melting or concrete for use as aggregate within specified applications) and site specific criteria. This approach utilizes good knowledge of the practice being considered so that it becomes possible to limit the number of exposure scenarios that need to be considered and to introduce application specific data into the dose calculations. These considerations may be expected in general to lead to higher release criteria than in the case of unconditional release. However, the same level of protection to the public is being achieved as in unconditional release. The practice that only unrestricted criteria are permitted to be used is counter to the risk based approach normally accepted for these types of decision making processes.

These considerations are in accordance with an IAEA Safety Standards publication [4], which stipulates that a graded approach consistent with the optimization principle can be taken when activity concentrations exceed the levels given in the Safety Guide. This states that such an approach

"…shall be commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures and shall also conform to any requirements specified by the [regulatory body] or, whenever applicable, by the relevant Sponsoring Organizations."

There are numerous examples of national regulations and international recommendations allowing the use of conditional or restricted release approaches. The balance between cost and safety, and other factors (see Section 4) can be optimized by the use of site specific release criteria and the provision for this approach is recognized in Canada in CNSC Regulatory Guide G-219 [26] and R-85 [27]. In addition to unrestricted release criteria,

Germany also has release criteria based on the specific planned destination of the material [28] as follows:

— Solid materials for disposal (in landfills or by incineration);
— Liquids for disposal by incineration;
— Buildings for demolition;
— Scrap metal for melting.

Such levels can be replaced by a site specific radiological analysis on the basis of a 10 μSv/a dose criterion.

Ad hoc criteria have been enacted in Sweden for the smelting of contaminated metals (see Sections 4.2.4 and 4.2.5) using such an approach.

The EC has recommended criteria for general clearance [29]. In addition, it has recommended conditional clearance levels for scrap metal [30], and for buildings and building rubble [31]. As explained in Ref. [28], the generic clearance levels in Ref. [29] were developed by combining the already existing recommendations given in Refs [30, 31] by taking the lowest level for each radionuclide so that the result is suited to all quantities and destinations of materials.

In general, criteria for restricted release can be established on a case by case basis by making use of the most appropriate country, site, material or destination specific scenarios and excluding those scenarios that are irrelevant to the case in question. IAEA publications (such as Refs [32–34]) provide generic guidance on a selection of environmental scenarios and 'default' values of certain parameters. Other scenarios and parameters could be similarly developed for specific circumstances. Such case by case approaches normally demand an agreement to be reached between the implementers and the regulatory body. However, it should be noted that such approaches may cause problems if the resulting material is moved across international borders. Generic, internationally accepted clearance levels could remove this problem, albeit with the risk of some loss of flexibility.

3.6. RADIOACTIVE MATERIAL FOR CONDITIONAL RECYCLING/REUSE WITHIN THE NUCLEAR INDUSTRY

The recycling and reuse of decommissioned components and/or scrap materials within the nuclear sector has the potential to significantly reduce either the amounts of waste requiring disposal or large scale unrestricted release campaigns, thereby offering significant savings in management and disposal costs.

Rubblization may also present a low cost alternative that could potentially meet dose based cleanup standards. Controls are still necessary in its implementation and care will be needed to define how to measure the contamination, how to model expected doses and how to manage policy issues associated with leaving the radioactive material on-site (p. 32 of Ref. [35]). One interesting example of structures released on-site (below ground foundations and slabs) is the Maine Yankee nuclear power plant decommissioning project in the USA (Fig. 2) [36].

Other typical candidates for nuclear recycling/reuse include:

— Fabrication of steel ISOs, other containers and overpacks for radioactive waste;
— Cementitious grout and backfills to infill intermediate level waste (ILW) and LLW waste packages;
— Incorporation into the reinforced concrete structures of waste repositories and storage facilities;
— Construction of waste processing equipment such as supercompactors and cementation plants [37];
— Backfilling materials for waste repositories.

FIG. 2. Maine Yankee nuclear power plant during dismantling.

To implement these practices, the decommissioning implementer needs to identify viable opportunities for reuse and recycling. Although a nuclear site may not meet all of its construction material requirements from processing and recycling its own wastes, more opportunities may come from a national or international recycling market.

A review by the EC provided an estimate of the amounts of LLW concrete, steel, copper and aluminium which may arise from routine operations and decommissioning of nuclear facilities throughout the EC, and the timescales on which this material would arise [38]. With respect to the recycling of radioactive steels, the study concluded that this is an already well researched area which requires no further development with regard to the melting and refining of steel arising from nuclear facilities. Work within the nuclear sector has been driven by efforts to allow unrestricted release of this material into the conventional marketplace and, in general, any restricted release into the nuclear sector would be a straightforward extension, but may need regulatory agreement. Further development of the controlled recycling of steels into nuclear sector products is likely to be driven by market forces since considerable investment in a manufacturing facility will be required. The possibility of combined plants for unrestricted release of materials, controlled release of materials and possibly volume reduction would provide further cost savings for the operation of a steel melting and manufacturing plant.

Figures 3 and 4 show, respectively, copper and aluminium scrap being sold during decommissioning of the Trino nuclear power plant, Italy.

There are considerable arisings of concrete waste that will require management. The prospect of utilizing recycled concrete as an aggregate, and for manufacture into concrete disposal boxes and within grout would provide a sink for a significant proportion of this material. The cost drivers fostering nuclear sector concrete recycling are the avoidance of nuclear disposal costs for large amounts of waste, rather than for the release of material to conventional recycling markets. As an example, a review of the factors controlling the reuse and recycling of radiologically clean and lightly contaminated concrete in the United Kingdom is provided in Ref. [37].

3.7. LLW DISPOSED OF IN A DEDICATED REPOSITORY OR STORAGE FACILITY

The radioactive waste produced during operational and decommissioning activities in different facilities varies considerably by activity level, half-life, volume and physical and chemical nature. The treatment and final disposal

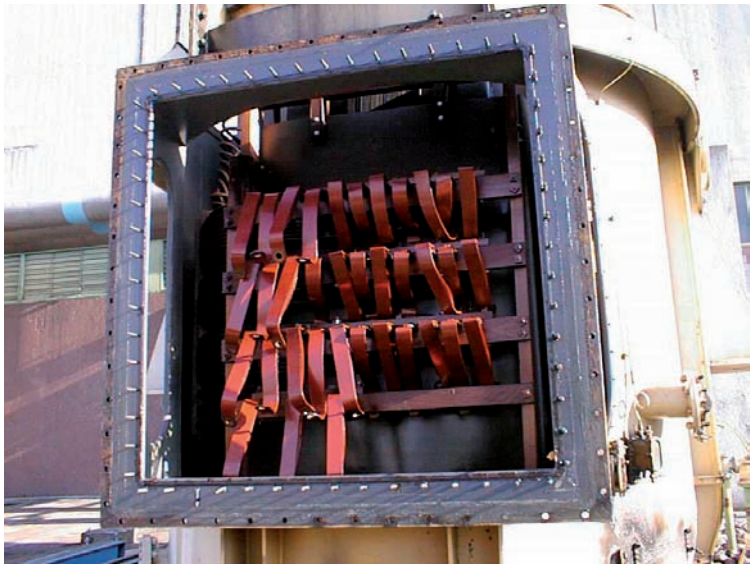

FIG. 3. Copper from main transformers sold at the Trino nuclear power plant, Italy.

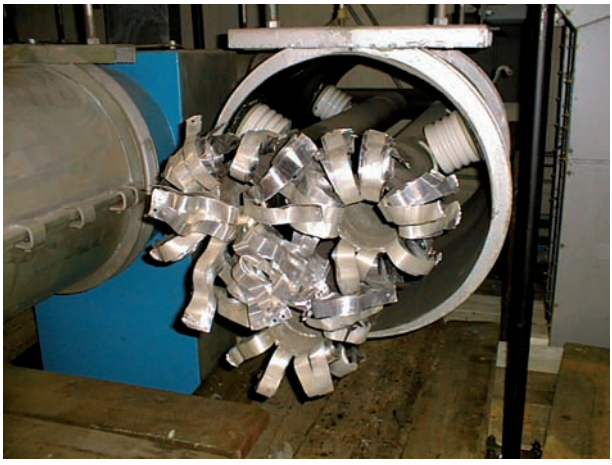

FIG. 4. Aluminium from electric bars sold at the Trino nuclear power plant, Italy.

solution must be adapted to the type of waste considered in order to manage it safely.

The IAEA waste classification system [19] distinguishes between, on the one hand, short lived and long lived waste and, on the other hand, between low,

intermediate and high level waste. Safety assessment is based on the radiological risks from existing or expected waste management pathways.

The methods for storage and disposal of radioactive wastes are governed by applicable national and international regulations, by the availability of appropriate storage and disposal facilities, and by the need to achieve an optimum cost–benefit balance for the disposal. The type and specific activity of the radioactive material present in the waste are the two most important factors in selecting the storage and disposal method. Other factors include the size of the package and the difficulty in handling the package during disposal.

The principal methods for disposal of LLW are near surface and surface disposal in an engineered facility [39–43]. Disposal in deeper geological repositories is generally envisaged only for higher level wastes with significant quantities of long lived radionuclides. The choice of the disposal method is dependent on conditions in the country and on many other factors specific to the disposal system to be developed.

Disposal facilities for LLW can be located on the nuclear site at which they were generated or elsewhere. The advantage of on-site disposal is that it avoids the environmental impacts and costs associated with transport but, in order to be followed, on-site disposal needs to be consistent with the planned end state of the site. In principle, this approach is possible for a variety of decommissioning wastes and is discussed further in Section 3.8.

3.8. VLLW DISPOSED OF IN AN APPROPRIATE STORAGE OR DISPOSAL FACILITY

VLLW does not feature in the IAEA classification of radioactive waste [19] but has been adopted as an operational waste designation by a number of countries (e.g. France, Spain and the USA). In practice, VLLW disposal is considered to be a special case of LLW disposal. The VLLW designation includes wastes with a bulk activity equivalent to the lower one or two orders of magnitude of the LLW activity range but above the unrestricted release level. VLLW will have essentially the same material characteristics as LLW.

VLLW arises mainly from the dismantling of nuclear facilities or from conventional industrial sites using some slightly radioactive substances. The quantities involved will increase considerably when the time comes for the large scale complete dismantling of the power reactors currently in operation.

The benefit of designating VLLW separately from LLW is that it can then be segregated and disposed of to dedicated facilities that do not need to meet design criteria as demanding as those for LLW. Moreover, the conditioning and

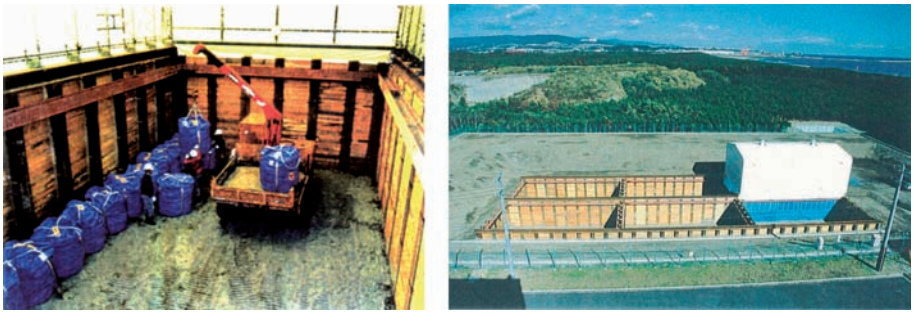

FIG. 5. VLLW JPDR disposal facility (left: during disposal; right: buried status).

packaging requirements of VLLW are likely to be simpler and more inexpensive.

Disposal facilities for VLLW can be located on the nuclear site at which it was generated or elsewhere. The advantages of on-site disposal are as for on-site disposal of LLW. A comprehensive discussion on factors relevant to on-site disposal is given in Ref. [44]. In Japan, VLLW is permitted to be disposed of on-site in open cut trenches. These wastes do not need any special solidification or packaging. A VLLW facility is situated at the former Japan Power Demonstration Reactor (JPDR) site (Fig. 5) and criteria for the associated management and disposal of decommissioning waste are given in Ref. [45].

Connecticut Yankee nuclear power plant in the USA is being dismantled and provides an example of off-site disposal of VLLW. The operator was faced with disposing of approximately 50 000 t of slightly contaminated concrete generated from the demolition of site buildings that were located in the radiologically controlled area at the plant. A waste disposal procedure was developed by the operator to dispose of these wastes as VLLW to a hazardous disposal site [46]. This disposal will be one of the first where a large scale disposal of radioactive material from a commercial nuclear plant has been approved by the Nuclear Regulatory Commission (NRC) to a facility not licensed by the NRC and where the surfaces of the building being demolished had not met the radiological unrestricted release criteria for surface contamination.

Another example from the USA is US Ecology's Grand View disposal facility in Idaho, which offers the opportunity to safely treat and dispose of a broad range of hazardous wastes and certain radioactive materials. More information, including radioactive waste specifications, is given in Ref. [47] and Annex I–8.

FIG. 6. Morvilliers VLLW site disposal of the first packages.

A French VLLW repository began operation at Morvilliers in 2003 following authorizations based on a radiological impact study and a public inquiry (Fig. 6). In France, recycling of lightly radioactive material is only permitted within the nuclear industry. The French system is supported by their 'zoning' approach to site decommissioning, in which parts of a facility are designated as likely to contain radiologically clean, VLLW or other radioactive wastes based on a functional analysis and review of the history of the facility. After decommissioning, a site-wide safety evaluation will be made to define possible use restrictions as well as surveillance schemes to preclude future unwanted practices on the site. In all cases, minimum restrictions will be put into enforcement in urbanization plans in order to enforce sufficient precautions when planning future uses of the ground or buildings. The French approach is described in more detail in Annex I–6.

One of the more challenging problems in site cleanup is how to deal with large volumes of soils, or similar materials, containing low concentrations of radioactive materials which could potentially be designated as VLLW. The traditional cleanup approach is to excavate the contaminated soil and dispose of it in a licensed radioactive waste disposal facility. Many operating organizations, however, do not view this approach as necessarily the best cost–benefit outcome for their sites. First, the large volumes mean the cost of off-site disposal is high. Second, some of the isotopes can be difficult to detect in the field, making excavations and final surveys a problem. Third, schemes for segregating the contaminated materials can be difficult, costly and may not yield the desired results. The owners of two US sites constructed on-site cells in which they disposed of contaminated and potentially contaminated materials

[48]. In Canada, a similar facility for the disposal of contaminated soil is operational at Port Hope. Another similar undertaking has been established in Sweden [49].

4. FACTORS AND CONSTRAINTS INFLUENCING OPTIONS FOR THE DISPOSITION OF MATERIALS IN DECOMMISSIONING

There are many factors that may influence decisions in the area of materials and waste management. The factors affecting the decision as to what disposition strategy should be adopted include (based on Ref. [17]):

— The quantities of materials;
— The technical feasibility, and the availability of technology and infrastructure;
— Costs and economic considerations;
— Radiological factors and the application of clearance criteria;
— National policy, regulatory frameworks, public acceptance and legal issues;
— The anticipated final end point of the decommissioning activities;
— Hazards and risks to people and the environment;
— Quality assurance (QA) and documentation issues;
— Environmental and ethical issues.

These factors are discussed in Sections 4.1 to 4.9. Not all factors will apply in every case but several of them are likely to be relevant. The inter-relationship between these factors may be complex. Some may be deemed to be more important than others and may have a larger influence on decisions. Consequently, some level of optimization is likely to form an inherent part of the determination of which segregation, release, recycling, reuse and disposal practices will be applied within particular decommissioning projects. Section 4.10 presents a brief summary of a number of different decision making tools that can be used to make decisions and optimize strategies.

The projects presented in Annex I demonstrate how various factors have influenced the decisions made for the disposition of materials.

When considering what disposition strategy to follow, flexibility is generally to be preferred. A flexible approach should enable a decommissioning

implementer access to as many disposition options as allowed by the national infrastructure and other conditions. In this way, optimal use of resources can be achieved. However, an extremely flexible approach may require more complex administration (e.g. from separate management of numerous waste categories or demanding QA procedures) and this complexity may increase costs compared to a simpler generic approach. For example, the extensive use of generic clearance levels offers the advantage of simplicity, albeit with the possible disadvantages of an overly conservative outcome.

Overall project funding and schedule are important. A decommissioning project heavily focused on research and development may be able to take advantage of longer timescales to explore and fully optimize materials disposition. Conversely, a project conducted under demanding time and funding constraints may follow a low risk approach of utilizing straightforward proven methodologies. Implementers should consider such viability aspects in tailoring the disposition strategy for their decommissioning materials and wastes.

4.1. QUANTITIES OF MATERIALS

Large quantities of materials will be generated during decommissioning and dismantling. A significant proportion of these materials will only be slightly contaminated with radioactivity, if at all. Due to economies of scale, recycling and reuse options are more likely to be cost effective for such large quantities of materials than for the relatively smaller quantities arising during operation.

A study has been performed on whether the EC recommendations on clearance of metals, buildings and building rubble from the dismantling of nuclear facilities are applicable in Sweden [50]. Estimates were provided of the quantities of wastes that would be released from dismantling Swedish nuclear power plants and what the impact on costs would be. As shown elsewhere in this report, clearance criteria have a strong impact on the amounts of materials that may be consigned as radioactive waste. The total amount of non-radioactive waste from system dismantling at the Swedish Oskarshamn 3 power plant was estimated to be 7830 t. In comparison, the total amount of radioactive waste from system dismantling was estimated to be 6040 t. A clearance level of 1 Bq/g for ^{60}Co would allow about 200 of the 6040 t to be released, while a clearance level of 20 Bq/g for ^{60}Co would allow another 3400 t to be released (3250 t arising from turbine systems). Moreover, about 7900 t of the systems are estimated to have a contamination of less than 1 Bq/cm^2 of ^{60}Co. If this contamination could be reduced by a factor of two, another 520 t would fall into this category. If the contamination could be reduced further by

a factor of 50 (a total decontamination factor of 100), only another 110 t would fall into this category, probably making it uneconomic to pursue.

Studies by the EC [10, 51] show the masses and activities of contaminated/activated materials for various reactor types, and times from reactor shutdown. For example, it can be inferred (Table A4 of Ref. [51]) that 3200 t of contaminated PWR steel could be eligible for unrestricted release if the criterion is 0.1 Bq/g (for a decommissioning project carried out five years after shutdown). Under the same conditions, another 1600 t might be released if the criterion was 1 Bq/g. Experience from Gundremmingen shows (Table A15 of Ref. [51]) that around 49% by weight of metallic materials generated from decommissioning were below 0.037 Bq/g. About 40% were between 0.037 and 0.37 Bq/g, and about 11% between 0.37 and 3.7 Bq/g (with a small residue above 3.7 Bq/g). This shows the strong effect that clearance criteria may have on amounts of radioactive wastes from decommissioning.

4.2. TECHNICAL FEASIBILITY AND AVAILABILITY OF TECHNOLOGY

4.2.1. Infrastructure

The availability of technically and economically proven techniques for the dismantling, segmentation, decontamination, monitoring and processing of components and materials is essential to any nuclear facility decommissioning programme. In addition, the availability of technically and economically proven means for disposition of materials from the decontamination and decommissioning of nuclear facilities is essential to any waste management strategy. Treatment methods should not give rise to large quantities of secondary waste generation, the further processing and disposal of which would involve substantial additional cost, impact on workers, the public and environment.

A number of technically feasible methods such as crushing and segregation of concrete have been successfully adopted from the conventional demolition industry. Other technologies are being developed and implemented on an industrial scale, such as metal melting. Currently, there are only a few commercial metal melting facilities available for the processing of radioactive materials (e.g. Studsvik [52–54], Siempelkamp [55], SOCODEI/CENTRACO [56, 57] and Duratek, Inc., in the USA). There are numerous other technologies to support recycling and reuse options that are at laboratory or pilot scale but these will require additional time, resources and efforts for further development to prove viability on an industrial scale [17, 18]. Regulatory

approval will be required for the selected technology and as public concerns have in some cases impeded such projects from becoming commercial, these potential concerns will also need to be addressed.

If legal or other mechanisms were used to limit disposal of material with the potential to be recycled or reused, then the latter practices would be promoted. If disposal is not available, there will be more incentive to develop reuse and recycling options. However, the capital cost of fully developed treatment and disposal routes can be substantial. Utilization rates will need to be high to ensure that prices for these routes do not become prohibitive. Wider national or international use for facilities would, in principle, help to spread the fixed cost base.

4.2.2. Transportation

Wide use of high capital cost facilities requires the availability of suitable radioactive waste transportation systems and arrangements (regulations, procedures, approved containers, etc.). Available routes (roads, waterways, railways) may favour transportation of radioactive waste off-site, or on-site storage may be preferable. In the latter case, a decision might be taken not to dismantle certain buildings but to reuse them for new purposes. Another related factor to be taken into account is the feasibility of transporting large components directly to a disposal site or to interim storage/treatment. An example of how transportation may affect disposition options in the United Kingdom is given in Ref. [35].

4.2.3. Characteristics of radiological contamination

The decision on whether to proceed with any disposition option will be affected by the characteristics of the material, the type and level of contamination (alpha, beta–gamma, loose or fixed, depth of penetration, absence or degree of activation), the nature and duration of (decay) storage, the accessibility of surfaces for decontamination and measurement, and the compatibility of materials with processes (e.g. potential for explosion or fire).

In addition, appropriate methodologies and monitoring techniques (procedures and instrumentation) for the radiological characterization of materials are essential for the implementation of disposition options. Matters of particular relevance to this subject are:

— Type and composition of the material to be characterized, its physical properties and geometry, and the quantities to be measured;
— Degree of surface surveying required;

— Accessibility of material parts;
— Natural and ambient background level, and the natural radionuclide content of the material each having an impact on the limit of detection;
— Radioactivity distribution on or within the material;
— Types of radionuclide to be measured, and the presence and significance of difficult to measure radionuclides;
— Required confidence level;
— Performance levels of available detection devices;
— Costs of technologies required.

All these factors are important in the selection and implementation of disposition options (see Section 5 for more detail).

4.2.4. Dilution with radiologically clean materials

It is realistic to assume that dilution with radiologically clean materials will occur at different steps in a recycling chain. For example, steel released from a nuclear facility is likely to be mixed with steel from other sources at a melting facility. This would, as an unintended by-product, further reduce the activity concentration of the steel in final form. However, the deliberate dilution of material to meet clearance criteria should not be permitted without the prior approval of the regulatory body. This is a different situation to the use of dilution in order to meet specific disposal criteria on the content of waste packages and also does not cover a policy of averaging radioactivity over the bulk of a material where the levels of radioactivity vary through the material.

If the option is selected to release material after demolition of a building, dilution should not be used as a means to release relatively high specific activity materials by the *deliberate* mixing of contaminated and uncontaminated building rubble in order to meet clearance levels. Buildings are typically only contaminated or activated on or near the surface with the interior of the structure being practically activity free. The calculations used to derive clearance levels assume that any highly contaminated surface layers are removed before demolition and disposed of as radioactive waste. It would not be appropriate to mix the activated part of the biological shield of a nuclear reactor with the underlying non-radioactive building structure with the intent of meeting mass specific clearance criteria. Similarly, documented contamination zones should be decontaminated to agreed release levels before demolition. Records should be kept of the dismantling operations in order to demonstrate that such zones of contamination have been decontaminated, and that highly activated and contaminated materials have been kept separate (see Section 4.8).

FIG. 7. Temporary ingot storage, Studsvik.

Dilution needs to be used sensitively in order to demonstrate implementer credibility and ethics in the management of radioactive waste and thereby maintain public acceptance. Nevertheless, it is a potentially valuable technique in appropriate situations and has been used successfully.

As an illustration, two alternatives are allowed in Sweden for free release and reuse of melted metal waste from nuclear facilities [54]:

— Release for unrestricted use based on general rules;
— Release for remelting in a non-controlled facility.

The maximum radioactivity content allowed for unrestricted reuse is 0.5 Bq/g total activity including a maximum of 0.1 Bq/g total alpha.

The present licence for clearance of ingots from the melting facility is based on the levels in EC recommendation RP 89 [30], with the condition that the ingots have to be sent for remelting in a non-controlled facility. Before release, Studsvik has to make sure that the ingots will be mixed in the way that is assumed in RP 89, which means a mixing of up to a factor of ten. The temporary ingot storage is shown in Fig. 7. This approach reduces radioactive waste for disposal and incurs no significant radiological risk to the public.

4.2.5. Melting

Much of the material from decommissioning consists of bulky equipment (e.g. heat exchangers, moisture separators, steam generators, motors, pumps, etc.) that, if disposed of without further treatment in appropriate repositories, would consume considerable volumes of the available space. A description of

Studsvik's methods for treatment of large contaminated components for volume reduction and recycling is given in Ref. [53]. Moreover, in many cases, this equipment contains valuable material that can be recycled, including pressure vessel steels, stainless steels, inconel, copper, lead and aluminium. By melting slightly contaminated scrap, it is possible to recover many of these valuable metals while simultaneously conserving valuable space at final disposal facilities. Pieces of equipment frequently also have complex geometries, making it extremely difficult, time consuming and expensive to determine the exact location and level of radioactivity on the internal surfaces. After melting, however, the radioactivity can be precisely determined from samples of each ingot. Moreover, an ingot can be released for restricted or unrestricted reuse, stored for decay to appropriate limits or disposed of as radioactive waste having achieved maximum density.

Melting completely destroys physical components and, as a decontamination technique, is effective only for contaminants that are volatile or that partition to the slag or dross (e.g. plutonium) rather than to the molten metal. In other uses, it would amount to dilution as discussed in the previous subsection. The decontamination efficiency varies widely depending on the radioisotope present. The radionuclides remaining in the molten material are distributed homogeneously and are effectively immobilized, thus reducing the possibility of the spread of contamination. In some cases, when ingots are found to be so active that they must be sent to a final repository, melting will have achieved significant volume reduction and thus preserved valuable repository capacity. As an alternative, some ingots with activity levels higher than freely releasable can be remelted to make shielding blocks or cold-rolled to fabricate containers for radioactive waste, and therefore can be recycled within the nuclear industry.

A particularly advantageous consequence of melting is its decontamination effect on ^{137}Cs, a volatile element that has a half-life of 30 years. During melting, ^{137}Cs accumulates in the dust collected by ventilation filters and is removed. The dominant remaining nuclide in the ingots (for most reactor scrap) is ^{60}Co. This element has a half-life of only 5.3 years. Other remaining nuclides have even shorter half-lives. Consequently, ingots with reasonably low activity concentrations can be stored for release in the foreseeable future. The secondary waste consists of the slag from segmenting and melting, as well as dust from the ventilation filters. This secondary waste only comprises between 1 and 4% of the weight of the melted scrap.

Melting may provide an essential step when releasing components with complex geometries, simplifying monitoring procedures for radioactive metal characterization. In addition to its decontamination effects, the problem with inaccessible surfaces is eliminated and the remaining radioactivity content is

FIG. 8. CIEMAT melting facility.

homogenized over the total mass of the ingot. Thus, melting can be a last step in the decontamination and release of components with complex geometries after these pieces have been decontaminated. As one example, Fig. 8 shows the CIEMAT melting facility used during dismantling of the JEN-1 reactor.

4.2.6. Chemical toxicity of radioactive materials

All materials arising during decommissioning activities, including chemically toxic and other hazardous materials, could be activated or radioactively contaminated depending on the nature of the nuclear facility in which the material originated and/or the purpose for which the material was employed. Therefore, their treatment, conditioning and disposal should consider both radiological and non-radiological hazards associated with these materials and wastes. An overview of the management options for relevant wastes is given in Ref. [58].

In the absence of clear regulatory guidance on chemotoxic components, many countries have introduced waste acceptance criteria that relate to the non-radioactive as well as the radioactive components of waste. In addition, disposal operators have undertaken assessments with the aim of addressing both sets of regulations.

Normal, dual purpose technologies for radioactive waste treatment combined with special methods and processes for destruction or stabilization of chemically toxic materials provide a promising and demonstrated technical

basis for the processing of toxic constituents in waste. Few individual technologies can provide a combination of organic destruction, radionuclide immobilization and other toxic materials immobilization after transferring them into a less toxic form. Therefore, a waste treatment system, using a process train of technologies, is required to address the full range of toxic constituents.

Disposal of LLW and VLLW is currently undertaken in near surface repositories. When considering the appropriateness of different disposal options for these wastes, consideration must be given to their chemically toxic components. There is also a need to ensure that the substances in the wastes do not compromise the performance of the engineered and natural barriers.

4.3. COST AND ECONOMICS

Whereas the choice of disposition option will be on the basis of an optimization of various factors, the economics of the choices will be the dominant factor in many situations. Economic aspects to be considered include:

- The economics of recycling or reuse versus those of storage and disposal;
- The cost of processing materials, including removal, characterization, decontamination, monitoring, size reduction, melting, transport and licensing;
- Price and marketability of the material, in comparison to similar materials from non-nuclear sources;
- The availability of adequate funds to pursue the preferred option;
- Contingency funding required to offset financial risk due to unforeseen events from causes such as legislative aspects, technical issues and public acceptance requirements;
- Any subsidies based on national policies promoting recycling and reuse practices, or conversely penalties due to the nuclear origin of the material.

The overall costs of characterization and monitoring programmes largely depend on the path being taken. The more handling and the lower the target activity levels, the higher these costs will tend to be. In general, the cost of clearance increases with decreasing permissible residual activity levels and the cost can be very high for a survey near state of the art detection limits at high confidence level. On p. 64 of Ref. [59], it is assumed that changes to clearance levels by factors in the range of three to ten in either direction create cost changes of a similar order of magnitude. A study conducted in Belgium for decommissioning costs [60] shows the following: choosing a clearance level of

0.1 instead of 0.3 Bq/g increases the decommissioning cost by 5% and the low level radioactive waste volume by 17%. It should be noted that the dose to the workers is also 7% higher, raising doubts on the overall safety benefit of the lower clearance level.

The costs of characterization and monitoring programmes are highly variable, depending on the number of measurements, the number of samples requiring analysis and the time needed to perform the work. To reduce costs, a statistical methodology to reduce the number of measurements is usually adopted (see Section 5).

In principle, reuse and recycling options can offer the lowest costs as long as clearance levels are not too low. This is because there are no disposal costs for the materials (other than for secondary wastes) and the scrap value of the item or its component materials can be realized.

Both of these factors depend on the circumstances in a particular country. For example, the cost of the least expensive disposal method can range from one to several thousand dollars per cubic metre. Even at the lower value, the cost savings can be significant. Similarly, the scrap value of an item depends on the nature both of the individual item and the intrinsic value of the materials from which it is made. The latter is dependent on the current market price for the material, which can be volatile.

The cost of reclaiming the scrap can also be substantial, and may include the labour, material costs and radiation dose to:

— Decontaminate the materials;
— Treat and dispose of any secondary wastes arising;
— Undertake the extra monitoring to select the items for recycling and to ensure that they are below the release limits;
— Decontaminate the treatment process equipment.

As an illustrative example,[1] total costs of €9–15/kg have been estimated in order to achieve unconditional release of scrap material in Germany including decontamination, release measurements and personnel [35]. It was further estimated that costs for reuse or recycling in the nuclear field (e.g. smelting under radiological control) may amount to around €12–15/kg. Costs for final disposal in a future deep geological repository are estimated in the range of €50–250/kg including conditioning, waste package, interim storage

[1] These figures should only be considered as examples denoting a trend and supporting generic statements given in this publication, but should not be used for comparison, funding estimates or other purposes.

and final disposal [35]. These costs depend to a large extent on disposal assumptions, the activity contents and the type of waste container chosen. For comparison, another German paper [2] mentions clearance costs of €5–9/kg, and low and intermediate level waste (LILW) disposal costs of €45/kg. In both illustrations, the estimated costs in Germany for recycling is significantly lower than those for disposal.

Notwithstanding the above analysis, the economic case for recycling will vary from country to country depending on the costs of disposal. For example, in France the 2005 indicative disposal costs of the French waste agency (ANDRA) were much lower than the German examples at €2500/t for LLW and (short lived) ILW at the Aube Centre and €270/m^3 for VLLW at the Morvilliers Centre. In Japan, the disposal costs for such wastes lay between the above examples [61]. Case by case economic assessment will be needed if cost is the driver for recycling.

4.4. RADIOLOGICAL FACTORS AND THE APPLICATION OF CLEARANCE LEVELS

The clearance levels in use vary between Member States. In some cases, these criteria were based on prescriptive national regulations, while in others they were based on a case by case evaluation. Two recent reviews of national clearance criteria are given in Refs [28, 62]. The variability in criteria applied in projects and in plants in various countries has also shown that release criteria are a significant factor in determining whether recycling and reuse can be applied on a larger scale.

The difference of up to two orders of magnitude in release limits applied in different countries is likely to be unacceptable in an open international trade market. For understandable reasons, this situation has been an obstacle to public acceptance of recycling. In several publications, it has been stated that it is important to arrive at internationally accepted criteria for the release and recycling of material from nuclear installations [32]. The recent promulgation of IAEA guidance in this field is intended to promote harmonization [4]. The difficulty in Member States adopting generic clearance levels as mass or surface concentrations was emphasized by a recent decision in the USA by the NRC to indefinitely postpone a decision in this regard [63].

It should be noted that clearance levels promulgated by the IAEA are given in mass concentrations, leaving the determination of surface contamination levels to national authorities. This may require further harmonization.

In parallel, the EC has also issued clearance levels [29–31]. One should note, however, that IAEA and EC criteria, though similar, are not identical and this may pose a dilemma for the Members States of both organizations. Work is in progress to seek harmonization of the levels.

Recently recommended clearance criteria remain to be tested in large scale decommissioning projects. For example, many derived clearance levels are close to, or below, current limits of detection for practical field instrumentation. Consequently, instrumentation, operational procedures and extensive sampling that are expensive in terms of both time and cost are required. Where this is not feasible, materials must be deemed to be above the clearance level and treated accordingly as radioactive waste, again with significant cost and potential environmental impacts.

Conditional release levels have been applied on a case by case basis, depending on the end use of the materials, and in certain cases specific formulas have been defined for restricted release, or specific values applied for the products of metal melting in designated melting facilities (Germany, Sweden).

Non-nuclear hazards for consignments to controlled disposal grounds are, in most cases, limited by national environmental protection acts (e.g. in Belgium, Germany, Sweden and the United Kingdom), although additional limitations have sometimes been imposed on the basis of the type of material (Canada), the destination of disposal (France) or the conditions specified by inspectors (United Kingdom) [17].

Matters such as waste averaging and finger printing for clearance purposes have to be defined by the operator and may need to be approved by the regulator as may the use of a statistical approach to reduce the number of measurements and samples. A contentious aspect can be the applicability of historical data to characterization and monitoring programmes. This is an area where regulatory positions widely differ worldwide. The use of environmental scenarios to justify conditional release or reuse options is also not fully standardized and left to case by case negotiations. In such cases, it is essential that an early dialogue be initiated between the operator and the regulator.

4.5. NATIONAL POLICY, PUBLIC ACCEPTANCE AND LEGAL LIABILITY

4.5.1. National policy

National policy will provide a constraint over which disposition strategies can be followed in any particular circumstance or may provide an indication as

to which strategy may be preferred, all else being equal. Given that waste managers have no direct control over national policy, it is important that they understand at the outset of a decommissioning programme which options are open to them, and be aware of any potential changes to policy which may arise causing costly amendments to the disposition strategy and potentially the decommissioning programme as a whole.

Germany's approach is an example of a national policy affecting the disposition of materials and waste where recycling and reuse are mandated by law (p. 20 in Ref. [64]).

4.5.2. Public acceptance

Depending on national policy, it may be necessary to gain local or national support for the preferred disposition strategy through a public consultation process. Even in situations where formal public acceptance is not required, waste managers need to be aware of the risks of adopting a disposition strategy that could be opposed by the public, particularly if the level of information provided to them is insufficient to fully explain the nature, risks and benefits of the strategy. In the extreme, public opposition can stop a project and lead to a loss of support for the decommissioning programme as a whole.

Certain disposition strategies may be more likely to achieve public acceptance than others, and the level of acceptance is likely to vary between Member States. Recycling and reuse outside the nuclear industry presents different public acceptance issues to disposal. Gaining public acceptance for the practice of recycling materials containing traces of radionuclides may be challenging because of the stigma associated with the nuclear industry in many countries. Recycling for restricted use should not, however, experience the same level of opposition [5]. Indeed, an agreement has been reached in Spain between the government, the regulatory body, the decommissioning organization (ENRESA) and metal recyclers to accept materials released for unrestricted use from nuclear sites [23]. Figure 9 shows the interest of the public in the Vandellos decommissioning project, in particular waste packaging activities.

Repositories for disposal of LLW and VLLW may be subject to similar public scrutiny and heightened sensitivity. Siting and licensing of radioactive waste disposal facilities has been the subject of intense political opposition in some countries.

While there is no universal answer to securing public acceptance, the following considerations may help to ease public concerns and aid understanding:

FIG. 9. Visitors viewing waste containers at Vandellos.

— A simple unified clearance system for deciding whether material is subject to regulatory requirements that adopts clearance levels that all Member States adopt and apply;
— Clear separation of political judgements (including cautionary conservatisms) from technical assessments — if political inputs into the clearance regime are necessary they should be explicit;
— Maximum use of recycling within the nuclear industry and of disposals on nuclear sites, although on the basis of standard clearance criteria (pp. 69–70 in Ref. [59]).

The following example from Germany illustrates the potential for public concern to affect a disposition strategy. All recommendations and regulations on clearance, as well as all supporting documents describing their derivation, had been published. However, it seems that for many years the fact that clearance was taking place (on a large scale) was not widely known to the general public. Clearance only became an issue when the draft of the Radiation Protection Ordinance was presented for public discussion in 1999–2000. Some environmental groups then developed a number of highly improbable scenarios and from them calculated very large doses. Endeavours by the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU) to bring the discussion back to a sound scientific basis were fruitful. A working group dedicated to clearance was set up under

the German Commission on Radiological Protection (Strahlenschutzkommission, SSK) in September 2000. This working group, as well as other discussion groups, now provide a valuable information exchange mechanism among relevant stakeholders [35].

Another example comes from the United Kingdom and relates to the Steam Generating Heavy Water Reactor (SGHWR) decommissioning project [65]. Although not used in the United Kingdom's nuclear industry, VLLW could in principle be sent for burial on existing land burial sites. The drawbacks identified were the anticipated costs and potential public relations difficulties. On the latter, nuclear power station and nuclear research wastes are perceived as 'bad' as opposed to hospital wastes which are not, and for which this route is currently open without controversy. The two drawbacks are inter-related in that the operators of the sites may charge a premium as compensation for any anticipated adverse publicity. In the event, the potential difficulties of securing agreement and achieving a satisfactory price convinced the team that this was not a satisfactory route to follow.

Ultimately, public perceptions regarding the acceptability of disposition options will significantly influence their implementation. Consequently, provision of information on the relative risks and benefits (economic, environmental and others) would assist in the achievement of informed public opinion and in the decision making process.

4.5.3. Legal liability

When evaluating disposition strategies for radioactive materials, a waste manager should be aware of the potential legal consequences associated with a strategy in terms of their own actions (e.g. maintenance of storage systems) and those of any third parties (e.g. a smelting company if materials are sold to be recycled). They should also ensure that whatever strategy is adopted, it is implemented in accordance with the relevant regulations. In some cases, it may be more challenging to meet the regulatory requirements associated with one strategy than another (e.g. tighter environmental constraints may apply) and this could be a factor in the decision. In most Member States, failure to adhere to the regulatory and legal framework could result in legal action by a regulator or other party, and a court may be the final arbiter of legal and financial liabilities. Failure to meet regulations may attract a financial penalty but may also damage stakeholder confidence in the decommissioning process.

A clear risk is that should end users of recycled material have to withdraw stocks of their product due to concerns over the quality of the clearance process at the nuclear site, then they are likely to seek damages from the

nuclear operator, and these could be substantial if a loss in the reputation of the users' brand were demonstrated.

4.6. FINAL STATUS OF DECOMMISSIONING ACTIVITIES

4.6.1. Site end state

The expected end state of the decommissioning site has a profound impact on the disposition strategy. Leaving buildings on-site for further use will obviate the generation of some waste. If the site is intended for further nuclear operation, it may be possible to store radioactive material on-site to allow decay. Alternatively, if the site is to be released for unrestricted use, there will be an interest in early removal of all materials and wastes off-site. It could even be possible to convert the decommissioning site into a national waste storage or disposal facility. Indeed, the decommissioning strategy could be the entombment of the nuclear facility with the waste retained on-site, perhaps in situ.

Disposition strategy options will need to be tested against the requirements of the future use of the site.

4.6.2. Disposal options: Availability and limitations

The availability or the access to fully developed treatment and disposal routes for large volumes of radioactive waste on a national or international basis may encourage the use of these routes rather than pursuing recycling and reuse. Indeed, once the capital cost of a disposal facility has been sunk, there may be financial imperatives for its use. On the other hand, acceptance criteria for disposal may exclude disposal of materials with a potential for recycling on environmental grounds. The unavailability of disposal facilities will promote the development of recycling, particularly if large costs for store construction and operation can be avoided.

In some countries, there are no radioactive waste acceptance criteria. If so, identifying a proper disposal route can be difficult and reconditioning or repackaging of radioactive waste may be required in the future.

4.6.3. Storage options: Availability and limitations

Storage can be an interim solution for the management of decommissioning materials and waste, where it fits into the overall decommissioning strategy. This may be pending the availability of disposal

routes or may be to allow later release for disposal or recycling as conventional material/waste following radioactive decay. This approach was used commercially at Studsvik, Sweden, to allow the release of metal ingots from the smelting of contaminated components and scrap. Permits were granted to Studsvik for decay storage of up to 20 years but on a case by case basis only [52]. The relevant Swedish regulatory body changed its policy in 2004 so that decay storage of foreign material is only accepted for up to ten years if there are special circumstances, such as if the activity in the ingots is higher than expected from earlier measurements of the scrap [66].

Storage may prove costly in the long run, unless disposal or other costs are thereby avoided. The following are examples of approaches that are routinely used to reduce the amounts of waste for on-site storage:

— Contaminated equipment can be offered as surplus materials for use by others;
— Contaminated metals can be melted and recycled for use as shielding material, to fabricate containers for LILW or to be stored on-site as billets;
— Contaminated waste oils may be incinerated on-site;
— Off-site volume reduction facilities may be used to decontaminate, compact or incinerate LLW.

Instead of building an interim on-site LLW storage facility, existing buildings may be used provided that such an approach is consistent with the overall site decommissioning timescale and programme [67].

4.7. HAZARDS AND RISKS TO PEOPLE AND THE ENVIRONMENT

Decommissioning on a nuclear site may involve both radiological and other more conventional hazards. All activities must be undertaken in accordance with the appropriate health and safety regulations. The radiological health risks from disposition options for the materials being discussed in this report will often be relatively low, whereas risks from other hazards may be potentially high. These hazards may arise from cutting and moving operations on complex, partially dismantled components. They may also be associated with working at height or using corrosive or toxic decontamination chemicals. The overall hazards and risks need to be considered when selecting a disposition strategy, not simply focusing on radiological issues.

4.8. QA AND DOCUMENTATION

All decommissioning must be performed under a QA programme [68] and that programme will provide a framework within which materials disposition must comply. Documentation of the process is a major feature, particularly where material may be released from further regulatory control. It will need to demonstrate that applicable regulations, codes and standards were met in disposition activities [69]. The purpose of a QA programme on monitoring for compliance with release criteria is to ensure and demonstrate that regulatory requirements have been met. This may cover sampling, analysis, monitoring, documentation, interpretation and use of data generated for this purpose. Such a QA programme should start with the programme design and be maintained at each significant step of the decommissioning process, up to the unrestricted release of the site.

Proper and accurate documentation is required for an operator or its contractors to demonstrate the acceptability of clearance and release, reuse or disposal of material. Important documents in this context include results of dose rate and surface contamination measurements as well as other evidence of the correct implementation of procedures. However, the lack of accurate construction and historical records can be an important factor influencing disposition options, because it will generally force the implementers to compensate for the missing information by a higher number of measurements. If doubts persist on the radioactive content of plant items, safety and economic considerations may lead to their disposal as radioactive waste. This can particularly be the case with complex geometries that are difficult or expensive to monitor.

Transparency and traceability, two key components of a QA programme, are crucial for clearance as QA procedures are the safeguard before decommissioning materials and wastes are released into the public domain, often untraceably. In practice, either no traceability is requested after unrestricted release or identification of the first recipient is required. Although unrestricted release should not require the tracking of the material, other reasons (e.g. information for use in case of later litigation) may still require follow up identification. In cases where the radioactive materials remain under regulatory control, the need for accurate records is important for any future intervention [70].

It is self-evident that traceability is an essential component of clearance where the material is released for a specific use or destination without further follow-up. In this case, the use of the specific release criteria applies to the release of the material from the regulatory regime where only the first use of the released material is controlled in order to ensure that it is indeed used in

the prescribed application. Regulatory control does not extend beyond this because the need for further control would be inconsistent with the very concept of clearance which is release from regulatory requirements. Thus, traceability is limited to the first use, e.g. disposing of material at a landfill, mixing fly ash into concrete under certain conditions or preparing the material in such a way that only a specific use is possible (e.g. cutting metal items into pieces so that they can only be recycled as scrap and not be reused). In general, it would also be difficult to demonstrate traceability beyond the first recipient [71].

A constraint on the disposition strategy may be the availability of sufficient numbers of suitably qualified and experienced personnel, particularly at times of peak demand. In these cases, it may be necessary to store materials until more resources are available or to consign materials to a route that is not the theoretically optimum one, to allow the overall decommissioning programme to proceed while meeting required safety standards. All personnel must be adequately qualified, experienced and trained. The minimum qualification and training requirements of the personnel should be clearly specified in the QA manual. Staff, procedures and training programmes inherited from the operating phase may continue to be useful but they need to be reviewed and, if found suitable, continued or adapted for the decommissioning phase [69].

4.9. ENVIRONMENTAL AND ETHICAL ISSUES

4.9.1. Sustainability

Policies for the long term protection of the environment stem from ethical considerations, in particular that the current generation should protect the environment for future generations. This is the basic concept behind sustainable development and combines environmental issues and socioeconomic priorities. The present generation, in benefiting financially and achieving higher standards of living from nuclear power, must do so without harming the environment or leaving a legacy of technical mismanagement for future generations. These objectives are reconciled within the 'sustainable development' principle. Briefly, this principle can be described as development that meets the needs of the present generation without compromising the ability of future generations to meet their own needs. Recycling or reusing decommissioning materials promotes sustainability by conserving natural resources for future generations. In its application, it is important to ensure that

this is not achieved by incurring excessive costs or producing secondary wastes that are themselves environmentally harmful.

Minimization of radioactive and clean wastes is a regulatory requirement in many countries. It can be achieved in many ways. For example, the surfaces of building structures, glove boxes, etc. may be made of non-absorbent or low absorbent materials so that during decommissioning, simple skimming of the surface will decontaminate the structure. Moreover, during decommissioning, spreading of contamination to nearby clean areas should be avoided.

The polluter pays principle requires that the producers and owners of radioactive waste are responsible for managing wastes safely and responsibly. These responsibilities are not limited to bearing the costs of managing and disposing of the waste by themselves, but also research and development costs undertaken by themselves and by the regulatory bodies.

All of the options for disposition considered in Section 3 need to be assessed for compliance with sustainability goals. The disposal routes available for waste tend to reflect their effects on the environment via the price that has to be paid to use them, so in this sense the polluter is indeed paying. However, the use of LLW repositories for material that could be disposed of as a lower category is arguably not consistent with sustainability and should be avoided where practical.

4.9.2. Global optimization: Full cycle impact

In general, a disposition strategy should present a net benefit when considering the health and safety of workers, the public and the environment, regardless of local or national boundaries; that is, it should be globally optimized.

When considering global optimization, it is important to consider all of the costs and benefits (financial and others) of the individual contributions to obtain a net benefit for any disposition option. A full analysis should not only include radiological impacts, but also the risks and potential environmental impacts such as:

— Non-radiological detriments, e.g. health risks from chemical exposures, industrial accidents and transport activities;
— Non-radiological environmental impacts on land, air, water, energy and other resources;
— Social and economic impacts, e.g. public acceptance, market factors and equity issues.

An OECD/NEA study [21] compared the impact of recycling or disposal of radioactive scrap material, taking account of both radiological and other factors. Based on considerable evidence, this study found that for total health risk estimates and overall environmental consequences there was a significant balance in favour of recycling, and that radiological risks are generally insignificant compared to other effects. Hence, it is important not to define the problem too narrowly when deriving criteria for acceptability of any practice.

Overall, ecological and sustainable development considerations promote the reutilization of non-renewable resources by way of direct reuse of equipment or buildings and by recycling of useful materials [3].

4.10. METHODOLOGY FOR DECISION MAKING FOR SELECTING DISPOSITION OPTIONS BASED ON SPECIFIC INFLUENCING FACTORS

As discussed above, various factors can influence the decision as to which disposition strategy should be adopted and some level of optimization is required as, on a case by case basis, the relevance and relative importance of these factors may differ.

When a specific disposition option has already been identified as preferred, a linear decision tree approach could be adopted to test and confirm this option's acceptability in a particular application. This is illustrated in Fig. 10 for testing a preference to recycling and reuse but with concerns about technical feasibility, economics, regulatory issues and public acceptability. Using this simple approach, the factors are evaluated one at a time and so cannot be considered in combination and no weightings can be applied to reflect relative importance.

A more sophisticated alternative to the linear decision tree method is a multi-attribute analysis or decision matrix approach which allows the simultaneous evaluation of several alternative options and influencing factors. Using this method, the various options for the disposition of materials are placed in a matrix against the relevant influencing factors for the decommissioning project. This method does allow a weighting to be applied to each factor which can be used as a multiplier for the scores of individual factors in order to reflect the priorities identified in a specific project. Adopting various values for these weighting factors allows some sensitivity analyses to be carried out to resolve the most critical influences. The final result of this analysis is a relative, numerical ranking of the options based on the score for

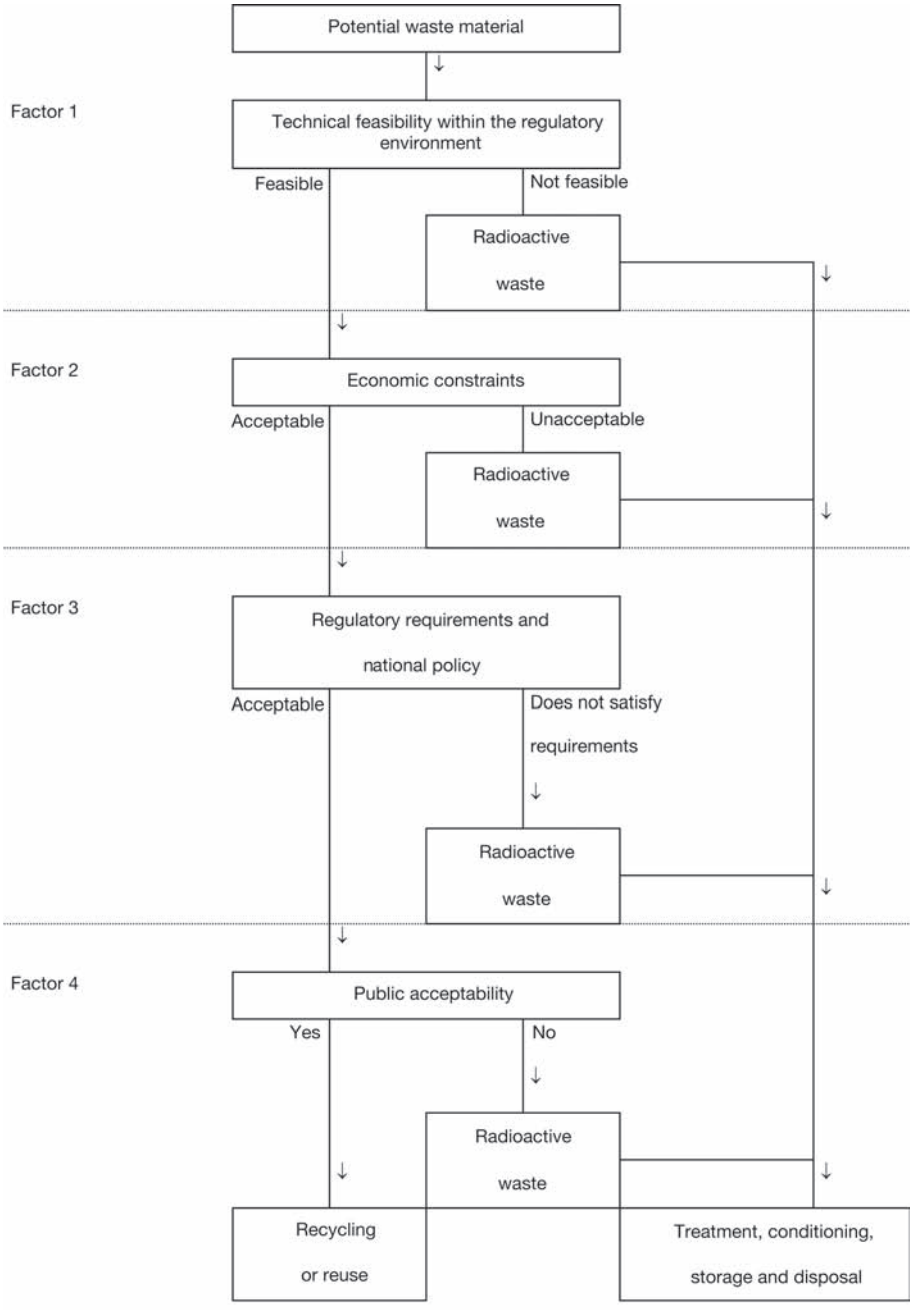

FIG. 10. Linear decision-tree approach for a recycling and reuse application.

each option. An example of a decision matrix is given in Table 4 and a recent comprehensive description of multi-attribute analysis is given in Ref. [72].

An interesting example of optimization aimed at waste reduction is given in Ref. [60]. The paper reviews any extra measures that might be taken to further reduce waste volumes, taking into account today's competitive environment and the current pressure on production costs. In Ref. [73], a cost–benefit analysis is performed weighing the cost of decontaminating and radiologically surveying the building media for release against the cost of disposing of the material as radioactive waste.

5. METHODOLOGIES TO EVALUATE AND CHARACTERIZE MATERIALS AND WASTES FOR DISPOSITION

Implementing a strategy for the disposition of decommissioning materials is performed in three basic steps:

— Perform a facility/site characterization;
— Select a material management strategy;
— Develop a monitoring strategy.

In so doing, both radiological and non-radiological hazards must be considered. Facility or site characterization is an ongoing process, and the initial data that are collected to support developing the material management and monitoring strategies must be continually updated and revalidated as decommissioning progresses. In addition, material management and monitoring strategies are inter-related. As a result, the methods used for material segregation and monitoring usually evolve during the course of a project.

The following sections provide an overview of the purpose of each of these tasks and some guidance on their performance. Section 5.4 provides a summary of the most frequently used monitoring systems, and the factors influencing the quality and applicability of their use for various monitoring methodologies.

TABLE 4. EXAMPLE OF A GENERIC DECISION MATRIX WITH DISPOSITION OPTIONS

Option	Cost		Technical feasibility		Risk		Availability of disposal		Full cycle impact		Final score
	Weight	Score	Weight	Score	Weight	Score	Weight	Score	Weight	Score	
Unconditional recycling and reuse	V_1 %	C_1	W_1 %	F_1	X_1 %	R_1	Y_1 %	D_1	Z_1 %	I_1	Σ_1 %
Disposal after unconditional release	V_2 %	C_2	W_2 %	F_2	X_2 %	R_2	Y_2 %	D_2	Z_2 %	I_2	Σ_2 %
Recycling and reuse in the non-nuclear industry	V_3 %	C_3	W_3 %	F_3	X_3 %	R_3	Y_3 %	D_3	Z_3 %	I_3	Σ_3 %
Recycling and reuse in the nuclear industry	V_4 %	C_4	W_4 %	F_4	X_4 %	R_4	Y_4 %	D_4	Z_4 %	I_4	Σ_4 %
Storage/disposal as LLW	V_5 %	C_5	W_5 %	F_5	X_5 %	R_5	Y_5 %	D_5	Z_5 %	I_5	Σ_5 %
Storage/disposal as VLLW	V_6 %	C_6	W_6 %	F_6	X_6 %	R_6	Y_6 %	D_6	Z_6 %	I_6	Σ_6 %

5.1. FACILITY/SITE CHARACTERIZATION

During the initial phase of planning a decommissioning project, radiological characterization is conducted to evaluate the quantity and type of radionuclides present, their distribution and their physical and chemical states. The objective of a characterization campaign is to obtain the information necessary to evaluate various decommissioning options and their associated costs, and to assist with finalizing a decommissioning strategy and planning the overall decommissioning programme. Characterization results provide the data necessary to evaluate the following factors:

— Distribution (or variance) of radionuclides and relative levels of contamination;
— Presence of difficult to measure radionuclides;
— Possible approaches to segregation (e.g. presumed clean, from known contamination areas, areas with different source terms, etc.);
— Instrumentation requirements and options;
— Personnel health and safety requirements;
— Potential waste categorization and waste management strategies.

Characterization data is also used during decommissioning implementation to support health and safety assessments and calculations for clearance (whether conditional or unconditional release), instrumentation efficiencies, scaling factors (or vectors) and radionuclide inventories for waste packages.

A comprehensive characterization programme includes the following components:

— Review of historical information;
— Implementation of calculation methods;
— Development of a characterization survey plan;
— Performance of in situ measurements;
— Performance of sampling and analysis;
— Validation and verification of data, and preparation of a characterization survey report.

These are discussed in more detail in the following subsections.

5.1.1. Historical information

A thorough review of historical information is necessary to ensure that the full range and locations of possible contaminants are understood. This

information may also assist in the identification of non-radiological hazards. Some examples include:

— Operational records of the facility (e.g. effluent releases, routine radiological monitoring, etc.);
— Documented area work controls (e.g. procedures, laboratory operating protocols, etc.);
— Radiological control area or zone categorizations (zoning);
— Hot cell/glove box records;
— Fissile material management records;
— Waste management records;
— Unplanned events and incident reports;
— History of hazardous material use (e.g. mercury, asbestos, lead, etc.);
— History of failed fuel in a reactor (indicating the possible presence of actinides);
— Records of system leakages (e.g. primary to secondary leaks);
— Previous contamination immobilization efforts (e.g. application of paint or epoxy over surfaces, replacement of floor or ceiling tiles, etc.);
— Interviews with staff and retired employees, with a long history at the site/facility.

Historical information assists with determining where the characterization surveys should be focused. If the historical records are reasonably complete, the characterization survey can be limited to verifying the historical information is still valid. If the records are incomplete or suspect, a more detailed survey will have to be implemented.

5.1.2. Calculation methods

Computer codes are available for estimating the neutron activation of materials in a nuclear reactor and to predict the radionuclide distribution as a result of normal operations, accident scenarios and transport of mobile contamination. Some examples at the time of writing include:

— Radionuclide inventories of irradiated fuel, such as: CANIGEN, ORIGEN2, ORIGEN-S, WIMS, DARWIN-PEPIN-2;
— Neutron activation of reactor components, such as: ONEDANT, ORIGEN-S;
— Transport modelling, such as: MCNP, TRIPOLI-4;
— Radionuclide inventories in process and technical nuclear power plant waste, such as: the LLWAA (Low Level Waste Activity Assessment) code

developed by Tractebel, Belgium, which uses the design characteristics and operating conditions of the nuclear power plant.

The calculation of material activation around a reactor is a two step process. First, the spatial and energy distribution of the neutron flux throughout the reactor is modelled based on fuel type and average reactor power (burnup), and then the induced radioactivity distribution is estimated based on the materials used in the reactor components.

Shielding codes such as MicroShield and Mercurad can also be useful for estimating inventory levels based on gamma field readings when the radionuclide mixture has already been identified. This is accomplished by modelling the specific contamination or activation scenario and determining the gamma field (exposure rate or dose rate) to activity ratio.

5.1.3. Characterization of the survey plan

Based on the historical data, and expected and potential contaminants, a site characterization plan should be developed that states the data quality objectives and details the radiological measurement, sampling, analysis and QA requirements necessary to achieve the objectives. This should include defining the details and rationale for any statistical approaches that are utilized, and the sample geometries to be used for laboratory analysis to ensure the results are representative. Reference [74] provides details of characterization recommendations, options and methodologies for radiological appraisal of shut down nuclear reactors, which are also applicable for other nuclear facilities, such as research laboratories.

A similar plan should also be developed for non-radiological hazards. In some cases, sampling and analysis for non-radiological hazards may have to be done by a specialized third party if the work is outside the normal competence of facility staff. Processing historical data of the facility might make it easier to assess non-radiological hazards.

5.1.4. In situ measurements

In situ measurements include:

— Alpha, beta and gamma measurements using portable gross (non-radionuclide specific) counting instruments to assess relative total contamination levels;
— Gamma imaging systems to assess the distribution and/or localization of radioactive material;

— Gamma spectrometry using a NaI(Tl), CsI(Tl), CdZnTe or high purity germanium (HPGe) detector to obtain radionuclide specific measurements.

Portable alpha and beta contamination monitors are used to quickly assess surface contamination levels. These include gas proportional counters, Geiger–Müller (GM) counters and scintillation detectors in hand held, floor monitor and robotic (e.g. wall crawler) configurations. They are used to scan large surfaces to identify areas with elevated levels as well as to obtain fixed measurements to quantify contamination levels, often in conjunction with data loggers to simplify the data collection process. Gamma dose rate instruments are used to identify the location and relative levels of inaccessible radioactive materials such as within piping and ventilation ducting.

Gamma imaging systems are used to locate radioactivity on surfaces and inside systems, and to provide a visual representation of the results overlaid on a visual image. A gamma detector, such as a CsI(Tl) scintillator, is usually coupled with an intensified CCD (charge coupled device) camera. As a result, gamma and visible images are obtained in the same detection time. An integrated collimator makes it possible to obtain angular resolution. Examples of configuration and performances of gamma imaging systems are provided in Refs [75–77].

Portable high resolution gamma spectrometers (HPGe detectors) are also used for in situ measurements to obtain the isotopic distribution of the gamma emitters. Powerful mathematical calibration and modelling tools are available for obtaining quantitative estimates of radionuclide inventories in complex geometries based on gamma spectrometry measurements. These techniques have been developed in order to reduce or replace sampling and subsequent laboratory analyses that are costly and time consuming. At the time of writing, the most widely used are the ISOCS calibration software by Canberra, and ISOTOPIC-32 and M-1 by ORTEC/AMETEK. This software makes it possible to eliminate the need for traditional calibration sources during the efficiency calibration process. By combining the detector characteristics modelled by a calculation code such as MCNP, mathematical geometry templates and a few physical sample parameters, the software provide the ability to produce qualitative and quantitative gamma assays of objects of most types and sizes. Custom efficiency calibration curves are specifically tailored for detector and object characteristics.

Typical geometries that can be modelled include:

— Contamination covered by paint or new concrete layers;
— Pipes and ventilation ducts including those hidden inside walls or under floors;

— Soil (in situ or in containers);
— Full and partially filled tanks;
— Waste containers (drums and boxes).

Mathematical calibration and modelling tools can also be applied to gamma spectrometry measurements to assess the depth of activation or penetration of contaminants by evaluating the peak ratios or the Compton front:

— The peak ratio method works for multi-energetic isotopes. It is based on an iterative efficiency curve calculation using the Canberra ISOCS calibration software and has been developed as a method of evaluating deeply activated concrete (e.g. the reactor bioshield). The main activation products are ^{152}Eu, ^{154}Eu and ^{133}Ba, each of which emits several gammas within the operating range of a HPGe detector. The depth of activation is estimated by comparing the relative level of each radionuclide's gammas at different depths and therefore with different attenuations;
— The measurement of the penetration depth of ^{137}Cs with the multi-energetic approach is more difficult because all but one of the peaks are of low abundance and low energy (≤36 keV) which is out of the detection range of most detectors. Using a method that applies Monte Carlo modelling of the Compton front of the 661 keV peak is more reliable but this method is still under validation.

Care should be taken to ensure other radiation sources do not affect the measurement. Their presence would increase the detection limit and might lead to overestimation and false interpretation of the activity level of the object being investigated. This problem can be resolved by shielding the radiation source, shielding the detector or by moving the object to a low background area.

Additional information on monitoring methodologies is provided in Section 5.4 of this report as well as in Refs [17, 78].

5.1.5. Sampling and analysis

Representative samples are obtained to identify the radionuclides that are present, their relative abundance and the depth of penetration in the case of porous or activated materials. What constitutes representative sampling of heterogeneous materials is usually assessed on a case by case basis.

Annexes I–1 and I–6 provide examples of satisfactory sampling techniques applied to concrete rubble generated from the decommissioning of

the former Eurochemic Reprocessing Plant in Belgium and the Triton Research Reactor in France, respectively.

Each area where the radionuclide mixture may vary should have detailed analysis performed. Sample analysis can be expensive and time consuming but provides a means of verifying assumptions based on historical data and validating calculated estimates.

Sampling methods to consider include the following:

— Core samples (or 'carrots'): Used to develop depth profiles for activation or contamination migration, for identifying self-shielding characteristics of the material and for comparison to computer code results for validation purposes.
— Swipe (or smear) samples: Used for obtaining loose contamination samples for evaluating relative contamination levels (fingerprints) and for radionuclide specific laboratory analysis (e.g. for tritium).
— Material samples (or coupons): Samples of a material for radionuclide specific laboratory analysis. They are usually obtained from higher activity areas to improve the possibility of identifying contaminants that are present in low concentrations or are difficult to detect (e.g. low energy gamma emitters).
— Removal of surface material (e.g. paint, epoxy, floor tiles) to access historical layers: Necessary to identify surfaces with previously immobilized contamination underneath. This is very important where potential alpha contamination exists.
— Bulk sampling: Bulk sampling is often used for samples with low activity to improve the detection limit of the analysis. It is primarily suitable for analysing soil, water and materials with known homogeneous contamination.
— Surface scarification: Removal of a surface layer (e.g. concrete) for radionuclide specific laboratory analysis. This is often used in areas of low activity to improve the detection limit of the analysis.

In each of these cases, typical laboratory analysis may include:

— Radionuclide specific analysis (e.g. alpha, gamma and mass spectroscopy, liquid scintillation counting, etc.);
— Chemical separation for radionuclide specific analysis (e.g. ^{90}Sr, Pu, etc.);
— Chemical analysis for measurement of spikes used for trace radionuclide determination.

5.1.6. Data evaluation and verification

A characterization survey report should be prepared to summarize the review of historical documentation, the measured data and the results from any calculated estimates. A comprehensive review of the results should be conducted and documented to demonstrate that the data quality objectives stated in the survey plan have been met; if not, then additional survey data may be required.

Where multiple characterization methods have been used for an area or system, the results should be compared to confirm that they correlate, thereby providing some validation. For example, measured data can be used to verify conclusions made from the historical information and assumptions used as inputs to software modelling.

The characterization report should make particularly clear:

— How representative the sampling was with respect to the waste or material stream (in terms of radionuclide composition, activity level and physicochemical composition);
— Measurement uncertainties;
— Any caveats associated with the data.

5.2. MATERIAL MANAGEMENT STRATEGY

Once the initial characterization is complete, the materials that will be produced during decommissioning are evaluated for potential disposition paths. Based on the material origin, radiological history, characterization results, chemical and physical properties, monitoring requirements and available disposition paths, the materials can be segregated into specific categories.

The possible disposition outcomes for decommissioning material were described in Section 3. A more detailed overview of the main options for the segregation and routing of suspected or known radioactive materials arising from decommissioning activities is illustrated in the flow sheet shown in Fig. 11.

Nuclear facilities often include buildings and infrastructure that have been limited to conventional non-nuclear use. Materials resulting from the decommissioning of these parts comprise of office furniture, tools or structural components that should not be radioactive. If no further confirmation requirements have been defined, these materials may be:

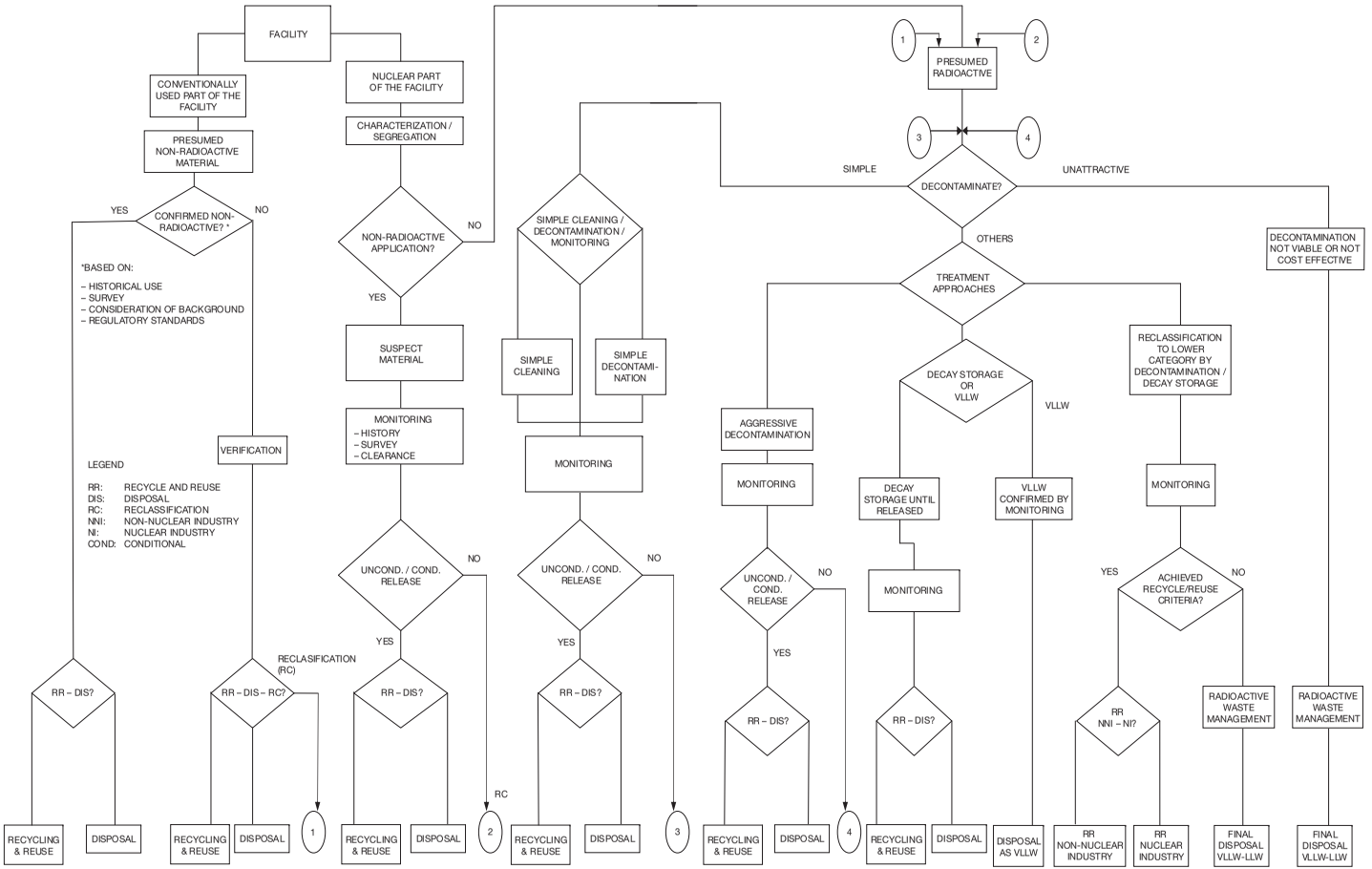

FIG. 11. Flowsheet from generation to release or disposal of decommissioning material/waste.

— Reused;
— Recycled;
— Disposed of in the same way as conventionally sourced waste at a municipal or industrial disposal site (following any special disposal provisions associated with their chemical/toxic content).

If confirmation that there has been no impact of nuclear operations is required, a survey may be implemented to verify that the materials are suitable for release. Any materials with detectable radioactivity above background levels, which could be due to site usage, would be removed for further consideration as potentially radioactive material, joining those arising from the nuclear part of the facility.

Based on history of use, some of the materials from the nuclear part of the facility may be presumed to be non-radioactive (e.g. those from office areas). However, having been used or brought for a period into a controlled area marks them as 'suspect material' and they can only be withdrawn from the radioactive waste management system by a thorough demonstration that any possible residual contamination is below specific clearance levels. Components or materials such as office furniture, tools or structural components, that after dismantling and simple cleaning activities meet clearance levels, can be released for unconditional or conditional recycling, reuse or disposal as described above for clean materials.

Materials with activity above the clearance levels are removed for further consideration as presumed radioactive materials along with materials from known contaminated areas of the facility. All decommissioning materials that are presumed radioactive should be adequately segregated and characterized in order to define further options. Materials or components that can easily be cleaned or decontaminated can be released for unconditional or conditional recycling, reuse or disposal once monitoring has confirmed they meet clearance levels.

The remaining materials may be subject to the following measures:

— Implementation of aggressive decontamination and monitoring of the materials in order to meet clearance levels;
— Storage to allow radiological decay of materials that have been contaminated with short lived radioisotopes in order to meet clearance levels;
— A combination of the above two methods to achieve clearance levels;
— Disposal of materials without further treatment for those materials classifiable as LLW;

— Disposal as LLW of materials that are radioactive and cannot be monitored against the clearance levels (e.g. due to their shape) or cannot be sufficiently decontaminated in a cost effective way;
— Recategorization as VLLW may still be achieved in the above cases depending on case specific factors and costs for decontamination, waste conditioning, packaging, shipment, storage and/or disposal.

Further segregation may also be required in each category based on the potential presence of chemical or other non-radiological hazards.

Different monitoring strategies are usually required for each of these cases. In general, it is more efficient and cost effective to monitor structures destined for demolition in situ. All areas where potential contamination may reside will have to be exposed prior to monitoring. If it can be performed safely, decontamination should also be performed while the structure is still standing. If reuse of the building is not an option, then it can be demolished and removed once the building has been cleaned to release levels, taking care to prevent recontamination from adjacent areas. The available monitoring methodologies will also influence the approach to segregation of materials.

A clearance monitoring facility should be established for loose materials and items that require removal prior to monitoring. It should be in a low background area, preferably protected from the weather and having sufficient space for safe staging and handling of the materials.

A low background area should also be established away from the clearance facility for characterizing material being evaluated for possible recategorization or destined for storage or disposal as radioactive waste.

5.3. MONITORING STRATEGY

Based on the strategy adopted for management of the decommissioning materials, the monitoring strategy can then be finalized. For each of the general categories, the materials may also be segregated into subcategories according to their physical and chemical properties, the type of contaminants present and the capability of monitoring instruments that will be used. The segregation of materials and the selection of monitoring instrumentation are inter-related. Monitoring protocols are then developed for each group of materials with the scope for which the protocol is valid clearly defined. The scope should include the following elements:

— Material physical/chemical composition;
— Type of contamination;

— Material and monitoring geometry, accessibility to contamination;
— Surface conditions;
— Scaling factors (vectors) to be used;
— Homogeneity versus inhomogeneity of contamination;
— Maximum averaging measurement unit (i.e. surface area, mass);
— Maximum background level of radiation;
— Determination of efficiency.

These elements will also have an influence on the methodology validation, the conversion of regulatory criteria or requirements to in-field or laboratory measurements, and the estimation of the measurement error. The main items are discussed in the following sections with further advice available elsewhere, e.g. Refs [79, 80].

5.3.1. Scaling factors (vectors)

Scaling factors provide a mechanism for evaluation of activity of difficult to measure radionuclides, such as ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni and ^{129}I, based on the activity of key radionuclides, mainly ^{60}Co and ^{137}Cs. Typically, characterization of contamination associated with reactor decommissioning waste uses ^{60}Co for the corrosion products, ^{137}Cs for the fission products and some alpha emitters. ^{241}Am is used for actinides when they are present in appreciable quantities. The key radionuclide is measured by radionuclide specific analysis (e.g. gamma or alpha spectrometry). The scaling factor is applied to the results to ensure all relevant radionuclides are accounted for when calculating total activity. The isotopic ratio is also used to determine reference efficiencies for gross beta or gamma counting instruments. Guidance on this method is provided in Refs [81, 82].

Corrosion product radionuclides show only small differences in production and transportation behaviour due to being produced by the activation of reactor material and having low solubility. The solubility of fission products and alpha emitters differs depending on the radionuclide considered. If ^{137}Cs is selected as the key radionuclide, there can be differences in radionuclide composition ratios between typical homogeneous (e.g. resins) and inhomogeneous waste because of differences in solubility and chemical properties. Therefore, it is necessary to differentiate waste into at least these two categories, and scaling factors need to be determined for each waste stream and key radionuclide.

If ^{60}Co is selected as the key radionuclide for fission product radionuclides and/or for alpha emitters, it is important to categorize a reactor according to the fuel history and fission product concentrations in the primary

circuit. This is because the generation mechanism of the key radionuclide is different from that of fission product radionuclides. Within each category, a unified scaling factor can be applied, though further subcategorization of scaling factors is required for more accurate estimation.

The use of scaling factors is ideal in situations where radionuclide mixtures are relatively constant (e.g. in an operating nuclear plant where source mixtures and transport mechanisms are known and consistent). They are less appropriate for situations where radionuclide mixtures are known to vary (e.g. laboratories where new radionuclides are introduced routinely), often moved to different locations or where contamination incidents are rarely similar.

Scaling factors often vary within a facility. Developing scaling factors for different areas of the facility provides greater accuracy and less conservatism than an all-bounding generic factor. If the most restrictive scaling factor is applied to every measurement, then the difficult to measure radionuclides may be overestimated, ending up in an overestimation of the actual total activity. It can be difficult to identify scaling factors when materials cannot be traced to their origin or in facilities where a large range of radionuclides has been handled as mentioned above.

Scaling factors can also change due to decontamination processes. For example, chemical decontamination is usually more effective for soluble radionuclides (e.g. ^{90}Sr and ^{137}Cs) than for others and scaling factors may be very different before and after decontamination steps.

Additional measurement data (from all sources such as clearance surveys, air sample data, etc.) should be evaluated for confirmation of the continued validity of scaling factors and may be utilized for improving scaling factor accuracy.

Some radionuclides that are important for waste characterization may be trivial in clearance measurements. The key determining factors in the former application are the half-life and radiotoxicity of the radionuclide. To evaluate how significant a radionuclide is for a clearance measurement, its scaling factor and clearance level should be compared with the other radioisotopes present. For example, consider a case where the scaling factor of ^{63}Ni to ^{60}Co is two. The clearance level recommended for ^{60}Co is 0.1 Bq/g [4]. Having a lower radiotoxicity, the recommended level for ^{63}Ni is 100 Bq/g. The following summation formula for the mixture of radionuclides of artificial origin present is applied to clearance measurement assessments:

$$\sum_{i=1}^{i=n} \frac{C_i}{CL_i} \leq 1$$

where C_i is the activity concentration (Bq/g) of the ith radionuclide of artificial origin in the material; CL_i is the clearance level value for the radionuclide i; n is the number of radionuclides present.

^{63}Ni will contribute 500 times less per gram to reaching the clearance limit than ^{60}Co and is therefore trivial for clearance monitoring in this scenario. Due to its long half-life, ^{63}Ni is significant for waste disposal and has to be evaluated and reported in the radionuclide inventory for waste packages. Conversely, accurately measuring radionuclides with short half-lives (less than some months to a few years depending on national practices) may not be considered important when characterizing waste.

5.3.2. Conversion of regulatory criteria into in-field or laboratory measurement

An essential part of implementing the proposed disposition strategy is to validate that the preferred monitoring approach will reliably deliver the correct sentencing of material against clearance criteria. A key part of that validation is to convert the regulatory criteria to reliable in-field or laboratory measurement techniques and criteria.

Regulatory criteria may include one or more of the following:

— A set of radionuclide specific clearance levels in Bq/g and/or in Bq/cm^2;
— A global clearance level per emitter type (e.g. alpha or beta/gamma), mostly applied to surface contamination;
— Specific activity limits for the different waste categories;
— Separate regulatory criteria for specific types of materials and disposition paths (e.g. copper destined for recycling, concrete for use in road beds, etc.).

A validity range should be defined for each monitoring protocol used to assess conformance to these criteria. This could include amongst others: measurement range, type of waste and matrices to be measured, and the efficiency curves to be applied. If a measurement were made outside the validated range, then the measurement itself would not be valid.

5.3.3. Main factors affecting measurement methodology

The main factors that could affect a monitoring methodology and therefore influence practical clearance measurement criteria are discussed in the following sections.

5.3.3.1. Type of material and its physicochemical composition

Building materials such as concrete may contain naturally occurring radioactive materials.

5.3.3.2. Geometry of the material and accessibility to remaining contamination

Flat surfaces and smoothly curved surfaces of large diameter can be measured with hand held monitors or preferably with gamma spectrometry if the source term can be modelled. Particular care is needed with items having a complex geometry that contain difficult to access surfaces. These might be monitored with hand held monitors using appropriate probes.

Materials with inaccessible surfaces may also be monitored with gross gamma counting or in situ gamma spectrometry, when the contamination includes sufficient gamma emitters as reliable vectors. It will be necessary to apply more or less simplified models for gamma spectrometry by recognizing any shielding effects that limit accessibility to the contamination.

One common problem encountered is that paint or resin may have fixed contamination in place. This makes the direct measurement of alpha emitters with surface contamination probes impossible. Where pure alpha contamination is suspected, the paint has to be removed before the measurements can be performed. When it is technically feasible and cost effective, pipes may be cut to expose their inner surfaces and then monitored using hand held detectors or smear tests.

Whatever the method, scaling factors have to be applied to evaluate the total activity including difficult to measure radionuclides. However, when dealing with pure alpha emitters or low energy beta emitters, the only realistic option for clearance may be to expose and monitor all surfaces.

Materials that are believed to be suitable for unrestricted release but have internal surfaces that are difficult to monitor can be processed in a manner that allows final sampling for clearance after processing, where allowed by the regulatory body. Examples include crushing or pulverizing concrete and melting metal items destined for recycling.

Regarding measurement of surface contamination using hand held detectors, the following ISO standards may help in implementing measurement and calibration methods [83–85].

5.3.3.3. Surface condition

Most surface monitoring instrument efficiencies are determined with plated sources and are only valid for smooth, clean surfaces. When measuring

rough, porous or rusty surfaces, as well as surfaces with a layer of oil or dirt present, a correction factor due to radiation absorption may need to be taken into account. Some guidance is provided in Ref. [86] on activity measurements of solid materials considered for recycling, reuse or disposal as non-radioactive waste and on how to take surface absorption into account.

5.3.3.4. Type of contamination

Selection of the detection system depends on the radionuclides to be measured. In most cases, the detector only measures a part of the isotopic mixture and the alarm value then has to be selected to take into account the radionuclides that cannot be detected by the instrument. If multiple scaling factors are used for different areas of the facility, the areas where each applies have to be clearly defined.

5.3.3.5. Homogeneity versus inhomogeneity of contamination

The homogeneity of the distribution of the activity of a material should be assessed when monitoring a large quantity of material. This assessment determines the averaging allowed for the monitoring methodology by setting the maximum area or mass of material that can be measured at one time. Averaging over a large area or mass is only acceptable if the distribution of the activity throughout the material is sufficiently homogeneous. This would be the case when measuring resins or materials homogenized by crushing. Materials with significant inhomogeneities in the distribution of activity should be measured over smaller areas or masses. Considerations and guidelines regarding limitations due to inhomogeneous distribution of activity in solid materials and characterization of expected distribution of the contamination have been provided in the United Kingdom [22].

5.3.3.6. Averaging in monitoring procedures

Some degree of averaging is an inevitable part of the monitoring methodology. The definition of averaging value should not lead to mass dilution (see Section 4.2.4). Averaging over a large quantity of material cannot guarantee that a fraction of that material is not significantly contaminated, potentially above the clearance level. The averaging used should ensure that the monitoring procedure cannot be used to intentionally clear material that is in part above the clearance level. The importance of this point may mean that the procedure requires agreement from a regulatory body.

Practical values for averaging include a surface of 1 m^2 and 1 t for concrete [31]. A metal average over a few hundred kilograms and an averaging area from several hundred cm^2 up to 1 m^2 is given in Ref. [30]. Discussion of related issues is referenced in Refs [80, 87].

5.3.3.7. Background level of radiation

The detection limit of an instrument or monitoring system depends on several factors including detector efficiency, scanning speed, counting time, desired confidence level, acceptable false alarm rate and background level. For any given set of conditions, there is a maximum background level above which the desired detection limit cannot be achieved. This background level should be determined for each monitoring method.

Background radiation impacts monitoring capabilities in two ways:

— As ambient background increases, longer count times are required to achieve the required detection limit of the instrument;
— Concentrations of naturally occurring radionuclides vary with material type and can be present at levels near or above generic (non-radionuclide specific) clearance levels. For example, concrete walls and ceramic and other construction materials contain, to different degrees, natural isotopes such as ^{40}K which contribute to the material background. The concentration of the natural radionuclides varies according to the origin of the material and therefore may or may not be significant. When radionuclide specific measurements are not used, the typical method of determining the natural radionuclide contribution is to obtain the background reading from clean materials of the same type as those to be measured.

5.3.3.8. Efficiency determination

There are two main approaches to determine the efficiency of the detection system under the monitoring conditions. Firstly, the efficiency in a particular monitoring situation can be calculated using codes (see Section 5.1.4). The conditions affecting the response include geometry of measurement, self absorption by the material to be measured, buildup factors and attenuation in the air. The other alternative is to produce calibration sources that, as far as possible, reproduce the conditions defined in the monitoring protocol. This could be a package filled with material of consistent density with a source placed at various known points in the package to simulate a homogeneous distribution. Alternatively, various materials could be packed

into a drum in a known way and a central source measured through the materials to simulate an inhomogeneous distribution.

When measuring by gross counting, the isotopic ratio applies at two levels to convert the regulatory criteria into a measurement value applicable to the instrument:

(a) To calculate the efficiency of the instrument either:

(i) select a reference isotope; or
(ii) calculate it by weighting the specific isotope efficiencies by the percentage of each isotope in the mixture.

This method involves either determining the efficiency based on a representative but usually conservative radionuclide, or calculating the effective efficiency based on the relative abundance of each radionuclide in the waste stream and their individual efficiencies. When instruments used for scanning surveys only provide count rate results, the clearance level has to be converted for use in the field. Once the efficiency for each emitter type of concern (e.g. alpha, beta, tritium, etc.) has been determined, the clearance levels are converted to count rates for use by the operator.

(b) When the regulatory limit is isotope specific, then the isotopic ratio and the isotope specific regulatory limit have to be taken into account in the conversion factor.

Validation of a monitoring methodology should be done taking into account the above mentioned factors in converting regulatory criteria to in-field measurement performance. Monitoring methodologies should only apply to materials that have been segregated into categories that meet the field of application of these methodologies.

5.3.4. Measurement uncertainties and their acceptability

Measurements are normally expressed along with an uncertainty at a given confidence level (quoted as either a percentage or a number of standard deviations). For example, a result of 10 ± 3 Bq/kg at the 95% confidence level indicates that there is 95% confidence that the true result is somewhere between 7 and 13 Bq/kg.

Uncertainties are often estimated based on counting statistics alone. However, in the case of direct monitoring methods, besides these random

sources of error, there are also systematic components to the global uncertainty attached to a measurement due to:

— Distribution of radioactivity in the material;
— Homogeneity of density inside the material (affecting self absorption of radiation);
— Variation in the radionuclide composition of the contamination;
— Modelling assumptions;
— Human error.

Furthermore, for laboratory measurements there will be potential for error associated with how representative sampling has been.

A clear distinction has to be made between systematic error due to a false assumption and random counting errors, particularly when using hand held detectors and gross monitoring systems. The error due to a false assumption is due to a bias between the measurement conditions and the conditions that were assumed in defining the protocol. This occurs, for instance, when assuming that the activity is homogeneously distributed whereas in reality the activity is concentrated in the centre of the material being monitored. In many cases, the systematic error may be larger than the random counting error. Such errors could lead to inappropriate material being cleared or material being consigned unnecessarily as radioactive waste. Errors may be reduced by checking for inhomogeneities, averaging over a smaller area and ensuring that the monitoring technique is well adapted to the conditions found in the item being measured.

Evaluation of random counting error is more straightforward and formulas such as those given in Refs [87–89] are typically used. In determining the outcome of clearance measurements, it might be agreed that the measured value, plus one or two standard deviations, must be below the relevant limit. Additional samples or longer counting times will reduce the standard deviation associated with the results.

An alternative approach is to justify that the measurement method is conservative enough to guarantee that the actual level of activity is lower than the relevant limit. This can be done by calibrating the device in the most restrictive conditions relating to the field of application of the monitoring technique.

In any case, the monitoring protocol must define the approach taken to determine the confidence level of the clearance measurement. This may be of regulatory interest and, where appropriate, should be subject to regulatory oversight and agreement.

5.3.5. Determination of the detection limit

The detection limit of a monitoring method is based on several factors, including:

— Background radiation level;
— Background and monitoring counting intervals;
— Instrument detection efficiency;
— Surface condition;
— Attenuation effects of overlaying material (self absorption);
— Backscatter effects;
— Desired confidence level and acceptable false alarm rate;
— Detector area for surface area measurements or material mass and volume for volumetric measurements;
— Human factors for scanning surveys.

Section 5.3.3 discusses several of these factors in detail. Guidance on the factors to consider and methods of calculating detection limits are provided in Refs [87–89].

In most cases, the measuring time is the key factor that is adjusted in order to reach a desired detection limit. It is considered good practice to strive for a detection limit that is 10–50% of the clearance level.

5.4. REVIEW OF TYPICAL MONITORING METHODOLOGIES

This section provides a summary of the most frequently used monitoring systems and the factors influencing the quality and applicability of their use for various monitoring methodologies. It identifies their advantages and limiting factors with respect to performing characterization and release surveys.

There are two basic types of measurements, direct and indirect. Direct measurements are performed either in situ or in a dedicated low background area. Indirect measurements are usually performed in a laboratory after sampling.

Many devices are available for performing waste characterization and monitoring materials for release. A detailed review is available in Refs [17 (Appendix III), 78].

5.4.1. Direct measurements

Direct measurements include both the scanning of surfaces and obtaining integrated counts at fixed data points. Both are usually required for release surveys. Instruments used for performing direct measurements can be divided into four general categories:

— Portable instruments;
— Large object gross beta instruments;
— Large object gross gamma instruments;
— Gamma spectrometry instruments.

Portable instruments

Portable instruments in hand held, floor monitor and robotic configurations are used to assess the presence or level of residual contamination on material surfaces, either in situ or in a low background area. For release surveys, the most common instruments utilize either gas proportional, plastic scintillation or GM detectors. Typically, these detectors are configured to measure either alpha or beta radiation; however, several instruments are available that can measure both simultaneously. They are available with gross count rate readouts as well as with software that provides activity levels from the measurements.

Gas proportional detectors typically require a constant purge of a counting gas but many can be disconnected from the purge and operated in the sealed mode for a few hours before fresh gas is required. Some permanently sealed detectors are available for beta only detection with more robust windows but at the cost of detection efficiency.

The key advantages of gas proportional detectors include:

— Relatively high efficiencies for both alpha and beta radiation;
— Low background levels for the same detector size;
— Good detection limits;
— Can be configured to discriminate between alpha and beta radiation with no change in detector efficiency;
— Can be manufactured to sizes equal to the allowed averaging areas (e.g. 100 cm^2, 300 cm^2, etc.).

The disadvantages of gas proportional detectors include:

— Detector surface is easily punctured;

— Not available below 100 cm^2 and therefore unsuitable for internal surfaces of small objects;
— Requires a suitable counting gas (e.g. P-10, propane, etc.);
— Fails if air is introduced into the detector;
— Larger detectors are fairly heavy.

Scintillation detectors have the advantage of not requiring a counting gas and are often built directly into the instrument.

The key advantages of scintillation detectors include:

— Relatively high efficiencies for both alpha and beta radiation;
— Low background levels for alpha detectors;
— Good detection limits;
— Can be configured to discriminate between alpha and beta radiation;
— Lightweight.

The disadvantages of scintillation detectors include:

— Detector surface is easily damaged;
— When configured for simultaneous alpha and beta monitoring, the low energy beta efficiencies drop significantly;
— Rarely available in sizes above 50–100 cm^2.

GM beta detectors are very robust but their use for characterization and release surveys is usually limited to assessing the internal surfaces of materials because of their lack of versatility when compared to other detector types.

The key advantages of GM beta detectors include:

— Relatively high efficiency for beta radiation;
— Robust construction;
— Available in small sizes for monitoring internal surfaces;
— Lightweight.

The disadvantages of GM beta detectors include:

— Not being able to discriminate between alpha and beta radiation;
— More sensitive to gamma radiation than other detector types, therefore higher background for a given detector size;
— Typically only available in sizes ≤ 20 cm^2.

All of these detector types can be used for either scanning surfaces or obtaining integrated counts of survey locations.

Scanning surveys may be conducted on building surfaces as part of the characterization to identify slightly elevated areas on which to focus the clearance survey (i.e. to identify biased survey locations when using a statistical approach) or to directly evaluate materials for clearance. With scanning surveys, the detection limits can be difficult to define depending on how the instrument processes the results. Because release surveys monitor for low levels of contamination, smoothing algorithms are often used to reduce the variability in the reading. As a result, the reading has a slower response time to a change in level and the detection limit is dependent on the user's ability to discern a change in the reading. Some scanning instruments are available that integrate counts for short periods of time (1–5 s), and then calculate and display activity levels from each integration cycle. This method has the advantage of providing accurate readings with no response time issues and allows for accurate determination of detection limits.

When instruments used for scanning only provide count rate results, the clearance level has to be converted for use in the field. This is calculated by applying the detector size and efficiency for the radionuclides of concern to the appropriate clearance level to obtain the associated net count rate (level above background).

Integrated counting with portable instruments (i.e. acquiring counts over a specified count period) provides improved detection limits over scanning and is primarily used for release surveys where statistical measuring approaches (i.e. <100% surveying) are incorporated into the monitoring strategy, often in conjunction with instruments capable of recording the results (i.e. data logging) for later analysis and reporting purposes. Many instruments can be configured to report the results in units of total counts, net counts, count rate, total activity or activity per area (based on the detector area). When data logging is available, the most common approach is to record the background and survey results as total counts to simplify detection limit and error propagation calculations (e.g. using a spreadsheet) for inclusion in the report.

Large object gross gamma instruments

Typical large object gross gamma counting instruments include bag, box, drum or vehicle type monitors. These can be true 4 π instruments (measuring all six sides) as in the case of typical bag monitors, 2 π instruments (measuring from one side) as in many box and drum monitors or pass through monitors where the material is passed through a set of detectors (such as vehicle

monitors). The detectors are usually plastic scintillators and often incorporate shielding to reduce background levels and improve detection limits.

The monitoring results are reported as total activity and/or specific activity when a weigh scale is incorporated into the device. These types of monitors do not provide nuclide specific results and are rarely accepted alone for clearance. Typical uses are for waste characterization, when the waste stream is well known and appropriate scaling factors have been developed, or for a quality control check after clearance monitoring has been completed by other means, e.g. in monitoring cleared scrap metal prior to release from the site using a similar system to the receipt monitor at the recipient's location.

Low energy gamma emitters (e.g. ^{241}Am) may be difficult to detect when monitoring high density materials (e.g. metals).

Large object gross beta instruments

Large object gross beta counting instruments are usually sorting table or conveyor based systems used for clearance monitoring. They can be configured to automate monitoring of large quantities of like material such as scaffolding, plywood, lead bricks, pulverized concrete, etc. Most systems are of the pass-through type (two detector arrays) but some only monitor one side of the material. The pass-through type usually incorporates at least one movable detector array to allow close monitoring of the material surfaces. This requires materials to be batched monitored by segregating them into groups of similar geometry, the monitor being reconfigured for each unique material type. Sorting table type of instruments often incorporate integrated count times for improved sensitivity.

Monitoring of low energy beta emitters (e.g. ^{14}C, ^{63}Ni) may not be practical with this system.

Gamma spectrometry instruments

Specialized gamma spectrometry systems are available for assessing waste packages for characterization or clearance monitoring and for in situ assessment of activity levels of objects, walls, pipes, etc. These systems include:

— Drum, large bag and box monitors for assessing waste packages or monitoring groups of materials for clearance;
— Portable systems that use calculation codes to model the effects of geometry, attenuation and activity distribution on detector efficiency.

Drum, bag and box monitors are available with single or multiple detectors in either single or two sided configurations. A motorized turntable is often incorporated into single sided systems to minimize the error associated with inhomogeneous distribution of contamination or material in a container.

Portable systems are usually mounted on a cart that provides a means of aiming the detector at the object to be monitored. Collimators are often incorporated to limit the detector field of view for specific applications.

The results are nuclide specific and therefore scaling factors can be applied to estimate the difficult to measure radionuclides. Often, conservative assumptions have to be made on the distribution of contamination within the packages or items. Obtaining measurements from more than one side can help to reduce the degree of conservatism associated with the results.

5.4.2. Indirect measurements

Indirect measurements include assessing removable contamination levels for characterization and clearance surveys, using swipes/smears, and obtaining samples from materials for laboratory analysis. Nuclide specific analysis is performed when the samples are for characterization purposes.

When used for clearance surveys, swipes/smears are usually analysed with a low background alpha/beta counter to obtain the low detection limits required. Samples from bulk materials are usually analysed for both general activity levels, using the low background alpha/beta counter, and for radionuclide specific concentrations using a gamma spectrometer.

For inhomogeneous materials, statistical tools are used to determine the sampling requirements (i.e. number and location of samples) necessary to evaluate and demonstrate how representative the samples are. Known homogeneous materials (e.g. pulverized concrete, melted metal, etc.) require a limited number of samples to characterize the material or demonstrate that clearance levels have been met.

Samples should be analysed and/or measured using appropriate equipment and procedures in a well established laboratory. Measurement techniques and equipment types used for direct measurements are also used for laboratory analyses and/or measurement, but usually under more controlled conditions which allow lower detection limits to be achieved and greater discrimination to be made between radionuclides.

Laboratory methods also often involve the combined use of both chemical and instrumental techniques to quantify the low levels expected to be present in samples analysed for clearance surveys. Knowledge of the radionuclides present, along with knowledge of their chemical and physical forms, and their relative abundance, are prerequisites for selecting the

appropriate laboratory methods. Those responsible for the survey should be aware that chemical analyses require lead times which will vary according to the nature and complexity of the request. For example, a laboratory may provide a fairly quick turnaround for gamma spectrometry analysis because computer based systems are available for the interpretation of gamma spectra. On the other hand, soil samples that must be dried and homogenized require a longer turnaround time.

Analytical and/or measurement methods should be capable of measuring levels below those of the established criteria.

5.4.3. Factors influencing the quality of monitoring methodologies

Various factors impact on the quality and the effectiveness of monitoring methodologies. These include:

— Procedures and monitoring protocols;
— QA documentation and records;
— Human factors and training;
— Complexity of the process;
— Regulatory interaction.

Procedures and monitoring protocols help ensure the monitoring process and requirements are clearly defined, approved and understood. These include:

— Characterization protocols;
— Monitoring protocols for conditional or unconditional release;
— Calibration procedures for instrumentation;
— Operation procedures for instrumentation including routine quality control checks.

Routine quality control checks using a check source are good practice and are required in many regulatory environments to verify that the instrument is still operating correctly and that the results are still valid. This is done by either measuring a certified source or a non-certified check source that was measured shortly after calibration to determine a reference value. The acceptance criteria are typically set at 10–20%.

A strong QA programme is required to ensure traceability of the material and the measurement results (e.g. that the instrument is properly calibrated and validated for the application). This includes:

— Maintaining instrumentation calibration and quality control records;
— Maintaining material handling records;
— Standardizing the recording measurement results;
— Ensuring retrievability of the records.

Human factors, safety culture and involvement of the personnel in the facility play a clear role in the quality and effectiveness of monitoring methodologies.

Monitoring materials for release is a time consuming and laborious task, especially if done by hand. For this type of monitoring, it is important to ensure that staff understand the monitoring requirements, e.g. clearance measurements can take several seconds and that scanning of the surfaces alone may not be sufficient to demonstrate compliance with clearance levels. This is the case when long monitoring times are required to meet detection limits. Adequate training of personnel is important to ensure that all requirements are clearly understood and agreed.

In order to minimize the potential for error, monitoring protocols should be as simple as possible. This concept applies at all levels of the process including material segregation, measurement and handling.

For material segregation, the scope of each protocol in use should be clear so that the operator can easily segregate the material in accordance with the applicable methodology. The process should require minimum interpretation by the operator. Interpretation introduces the risk of inconsistency; therefore, expert interpretation may be necessary to validate the results.

It is important to initiate an early dialogue with the regulator to obtain feedback and agreement on the proposed approach for releasing structures and materials prior to implementation. It is also important to maintain an open dialogue during decommissioning to ensure that any changes can be addressed in a timely manner.

The regulator may require independent (third party) verification of release survey results (structures, lands and/or materials). When this is the case, it is important that the methodologies and radionuclides used to calibrate the release survey instrumentation are clearly understood by the organization performing the verification to ensure that their results are not biased by selecting a different method.

6. CONCLUSIONS

This report has discussed the disposition of large amounts of low activity materials arising in decommissioning, much of which risks being disposed of as LLW in the absence of the consideration of other options which may be more environmentally or economically advantageous. The issues to be addressed in finding and implementing the most appropriate strategy for disposition of decommissioning materials have been presented as options, factors and methodologies, based where possible on actual project experience in Member States.

A range of disposal and recycling/reuse options has been identified: disposal as LLW, as VLLW or as radiologically cleared or clean waste; recycling/reuse in the nuclear industry, in nominated non-nuclear applications or released for any use.

Factors that influence whether an option should form part of a site disposition strategy have been discussed. These cover issues such as the quantity of material involved (large volumes are more amenable to industrial scale processes and the economies of scale that follow), and the availability of radiological clearance criteria and technical solutions to achieve them, in a properly controlled and safe manner. In addition, the future use of the site and the economics of the option in that context are also important. Finally, there needs to be a policy framework utilizing issues such as sustainability and environmental impact that encourage the appropriate utilization of all options and are likely to provide the necessary public acceptance.

Successful implementation of the chosen strategy will rely on quality assured processes for the characterization of materials, their management and reliable and economic monitoring to allow the material to undergo disposition by the preferred route. These processes have to be supported by appropriate and practical methodologies, and the report describes such methodologies either directly or by reference to industry experience. Guidance is provided on the selection of techniques and instrumentation to support the selection and utilization of the options discussed, within the constraints of existing clearance criteria.

Each of the options referred to above is feasible in principle and successful applications in Member States have been described. The aim should be to achieve a flexible approach that allows the selection of the optimum approach when considering environmental impact, cost and other factors. Although the IAEA has issued clearance criteria, Member States currently continue to vary in the clearance criteria that they apply, leading to differences in the availability of options or in clearance practices. Material has

undoubtedly been consigned as LLW, with the related environmental and economic consequences because better options were not yet provided for in regulations or had not been pursued by decommissioning implementers. It is hoped that this report will assist in the use of a wider range of disposition options with environmental and cost benefits, while ensuring that all safety requirements are fully achieved.

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Annex I

NATIONAL AND PROJECT CASE STUDIES

The examples provided in this annex address a wide range of disposition situations in decommissioning projects in several countries. This material is intended to illustrate how the issues described in the main sections of this report have been managed in actual project conditions. The aim is to provide general practical guidance of approaches that have met with some success rather than to present the experience as necessarily best practices for replication elsewhere as they each have elements that will be specific to the project, site and national context in which they were undertaken. Further examples of lessons learned from project experience are presented in Annex II.

I–1. DISPOSITION OF CONCRETE FROM THE DECOMMISSIONING OF THE FORMER EUROCHEMIC REPROCESSING PLANT, BELGIUM

I–1.1. INTRODUCTION

Belgoprocess started the industrial decommissioning of the main process building of the former Eurochemic reprocessing plant in 1990, after completion of a pilot project to dismantle two small storage buildings for final products from reprocessing. This was to verify the assumptions made in a previous paper study on decommissioning, to demonstrate and develop dismantling techniques and to train personnel. Both buildings were emptied and decontaminated to background levels. They were demolished and the remaining concrete debris was disposed of as industrial waste and green field conditions were restored [I–1.1].

The main process building was a large rectangular construction, about 80 m long, 27 m wide and 30 m high (Fig. I–1.1). 106 cell structures had to be dismantled, involving removal and decontamination of equipment from each cell; decontamination of cell walls, ceilings and floors; dismantling of the ventilation system and complete monitoring for unconditional release of the remaining structures. About 1500 t of metal and 12 500 m^3 of concrete with 55 000 m^2 of concrete surfaces had to be removed and/or decontaminated. Most of the work involved hands-on operations under protective clothing tailored to

FIG. I–1.1. Main process building of the former Eurochemic reprocessing plant.

each task. Tool automation and automated positioning systems were successfully applied [I–1.2].

I–1.2. FUNDAMENTAL PRINCIPLES OF THE BELGOPROCESS STRATEGY

The final shutdown of a nuclear facility requires that a series of strategic, tactical and technical decisions is taken, emerging from an iterative process of study and discussion. Depending on the selected strategy, different kinds and amounts of contaminated material may be produced during the decommissioning activities. For each option, the minimization of radioactivity and volumes of waste material for storage and disposal should be considered as well as the consequent environmental impact and the costs associated with the management of the contaminated material. The availability of strategies and techniques for the minimization of decommissioning waste has a large impact on the selection of adequate options.

Some fundamental principles were considered for the management of materials, equipment and/or components resulting from the decommissioning of nuclear installations, which are mainly based on the guidelines in the IAEA

Fundamental Safety Principles with respect to radioactive waste management [I–1.3]. Two of the fundamental principles dealing with the strategy for the management of materials from decommissioning are:

— The generation of radioactive waste shall be kept to the minimum practicable (seventh principle);
— Radioactive waste shall be managed in such a way that it will not impose undue burdens on future generations (fifth principle).

The fifth fundamental principle is based on the ethical consideration that generations receiving the benefits of a practice should bear the responsibility for managing the resulting waste. An additional ethical principle may be added, stipulating the conservation of primary resources for future generations.

Under the seventh fundamental principle, it is indicated that "This [set of measures] includes the selection and control of materials, the recycling and reuse of materials…" Advanced decontamination techniques may help to achieve this objective as decontaminated materials may be removed from the radioactive waste management system, minimizing the amount of remaining radioactive waste material for disposal.

In a broader context, recycling of materials may be considered to be a first order ecological priority in order to limit the quantities of radioactive wastes to be disposed of, to reduce the technical and economic problems involved with the management of radioactive wastes, to make economic use of primary material and to conserve natural resources of basic material for future generations. When analysing disadvantages and risks involved with a specific practice, the full cycle of the practice should be considered, taking into account, e.g. the disadvantages and the risks involved with the use of recycled metal, as well as the disadvantages and the risks involved with mining and conversion of new metal materials.

Based on these fundamental principles, Belgoprocess made a straightforward choice for a strategy with minimization of the amount of material to be managed as radioactive waste [I–1.4]. Aggressive decontamination techniques and unconditional release of decontaminated materials would enable the objectives to be achieved. Unconditionally released materials would be recycled (i.e. metal material sent to conventional melting facilities) or removed to conventional industrial disposal sites if they had no remaining value.

The specific Belgoprocess approach should be highlighted. The decommissioning activities were carried out on an industrial scale with particular special emphasis on cost minimization, commitment to results within

an overall plan and use of state of the art technology on an industrial scale [I–1.2].

Decommissioning activities had to deal with the specific radiological characteristics of the facilities. While the decommissioning of a nuclear power plant is mainly characterized by radiation risks due to the presence of in depth activation products, the alpha contamination of equipment and building surfaces in a reprocessing plant required the decommissioning work to be performed with adequate protective clothing. Specific breathing and air cooling systems had to be provided to enable the operators to carry out the decommissioning tasks in acceptable working conditions [I–1.5].

I–1.3. PRACTICAL IMPLEMENTATION OF THE BELGOPROCESS STRATEGY

To cope with problems of increased waste processing and disposal costs, and to meet the proposed planning for the decommissioning activities, Belgoprocess put a lot of emphasis on waste minimization. Practical implementation of the fundamental principles was developed based on the following considerations:

— Keep the generation of radioactive waste to a minimum;
— Minimize the spread of radionuclides as much as possible;
— Optimize the possibilities for recycling and reuse of valuable components from existing and potential waste streams;
— Minimize the volume of produced radioactive wastes by means of adequate processing technology.

Some specific actions were defined in order to achieve these principles and to increase work efficiency.

I–1.4. FINAL DEMOLITION OF STORAGE BUILDINGS FOR END PRODUCTS OF REPROCESSING

As indicated above, Belgoprocess started its decommissioning activities with the dismantling and decontamination of two small storage buildings for end products from reprocessing. Both buildings were emptied and after decontamination of the concrete structures down to a level of 0.04 Bq/cm^2 for alpha and 0.4 Bq/cm^2 for beta–gamma emitters, two independent measurements of all building surfaces were carried out by the in-house health

physics department in order to confirm the above mentioned contamination levels. A third random control measurement was performed by an officially approved radiation protection control organization. All three measurements gave the same results.

Core samples were taken on the previously most contaminated spots. The specific activities of these samples proved to be well below 1 Bq/g. Measurements and analyses of these samples confirmed the presence of only natural radioisotopes. Consequently, the buildings were able to be withdrawn from the controlled area.

The final steps in the pilot decommissioning project were the demolition of the two buildings, the removal of the demolition waste to an industrial dumping ground for inert wastes and restoration of green field conditions. To enable this last step in the objectives of the pilot project, it was necessary to provide sufficient evidence to justify such action to the public. Other than the qualitative definition of radioactive waste, there were no regulations in Belgium for the unlimited reuse or radiologically uncontrolled disposal of suspected and/or decontaminated materials.

The evidence to be supplied to support the implementation of the disposition plan had to be evaluated against criteria based on a number of regulations (which had been applied in other decommissioning projects) and recommendations issued by radiation protection experts of the EC for recycling of materials resulting from the decommissioning of nuclear installations [I–1.5]:

— Removable surface contamination in alpha = 0.04 Bq/cm^2;
— Removable surface contamination in beta–gamma = 0.4 Bq/cm^2;
— Total specific beta–gamma activity = 1 Bq/g, mean value over an arbitrary mass of 1000 kg with an individual maximum of 10 Bq/g.

The results of the multiple 100% surface measurements in the two buildings and the additional controls on selective core samples (gamma spectrometry, and total alpha and beta measurements) showed that the requirements of the first two criteria were met, and that the third criterion, limited to the core samples taken, was also complied with. The only thing that remained to be demonstrated was that possible alpha contamination was also characterized by sufficiently low specific activities.

Based on experiences from the operational period of the plant (localized contamination) and on contamination located during decommissioning operations, it was decided to take concrete core samples not at random, but at the previously most contaminated spots, increasing the probability of detection of any remaining contamination.

It was conservatively assumed that the collected core samples were representative of all concrete volumes, as if all had the same history, and acquired their characteristics in a similar manner. This also meant that it was assumed that every part of both buildings was originally contaminated to the same degree as those parts where core samples were taken, and where before decontamination the highest degree of contamination was found, which in reality was certainly not so. This assumption allowed a conservative extrapolation of the results of the analyses on the core samples to all of the building material.

The gamma spectrometry analysis of the core samples by means of a 305 mm × 102 mm NaI(Tl) detector in a shielded bunker with low background radiation, revealed no indication of artificial contamination via low level activity measurements. The resulting spectrum offered a qualitative view with high sensitivity with only natural nuclides being found.

Gamma spectrometric monitoring (Ge detector) of the individual core samples was carried out to obtain a quantitative measurement of the gamma emitters, and the results were compared to the results of similar measurements on a reference, non-contaminated core sample. This method had limited sensitivity as, in essence, a surface measurement is carried out which can only detect radioisotopes in the specimen at low depths. The detected radionuclides were:

— Natural radionuclides: ^{40}K, ^{226}Ra and daughters, which can normally be found in elements of concrete and other construction materials. The mean values of the detected activities for both nuclides were comparable to the values found in a non-contaminated reference core sample (x = 0.4 mBq/g, s = 0.3 mBq/g and x = 0.27 Bq/g, s = 0.34 Bq/g, respectively). The operational history of the installations, and the fact that no peaks for ^{238}U or ^{235}U were found in the spectrum, supported the view that the installations had never been artificially contaminated with ^{40}K or ^{226}Ra. In any event, this would only have been an important concern if the detected values for ^{40}K or ^{226}Ra had been inexplicably high.
— Nuclides due to artificial radioactivity: ^{137}Cs. Compared to the activity detected in a non-contaminated reference core sample, the resulting total contamination in the core samples (2 mBq/g), for a homogeneous distribution of the specific activity, was not higher than 1 Bq/g, mean value over an arbitrary mass of 1000 kg, taking into account the gamma penetration into concrete.

The adopted gamma spectrometry did not find any ^{241}Am. Considering the operational history of the installations, this supported the absence of alpha

contamination through plutonium. If there had still been some remaining contamination in the decontaminated floors or on other surfaces of either building, then parts close to the surface would normally have shown the highest concentration of contamination, if the underlying material of the construction was still in its original condition. Some cross-sections of the upper part of the core samples were therefore analysed for alpha and beta activity control. The analyses carried out on six cross-sections of the core samples taken resulted in a mean value of $x = 16.7$ mBq/cm² and a standard deviation of $s = 8.2$ mBq/cm².

Given the conservative assumption of a uniformly spread maximum amount of contamination in the remaining structures, probability theory states that for a random sampling of n values from a normal population with calculated mean value x and standard deviation s, the most probable value for the mean will be somewhere between $(x \pm t)s/n^{0.5}$ where t represents the Student factor, depending on one single parameter, the degree of freedom of the sampling problem. For a degree of freedom $n - 1 = 5$, and for a confidence interval of 99%, it could be stated with 99% certainty that the most probable mean value for the measurements carried out, would not be higher than 30.2 mBq/cm². In practice, considering the adopted hypotheses it would even have been lower.

It was furthermore assumed that each of the fission products ^{90}Sr, ^{90}Y and ^{137}Cs was detected with equal activity (resulting mass absorption coefficient for the medium 0.026 cm²/mg) and homogeneous distribution in a concrete layer of 1 mm thickness. It could then be calculated that the transmission due to self-absorption would be limited to 17.4% [I–1.6]. In this way, the mean specific beta activity of the analysed concrete layers proved to be 0.44 Bq/g and the maximum for the mean value 0.79 Bq/g.

The evaluation of alpha measurements was only strictly necessary if alpha activity of artificial origin was suspected. In any evaluation, it had therefore to be recognized that an important part of any alpha contamination was coming from the radionuclide ^{226}Ra, due to natural radioactivity.

To identify any remaining alpha contamination in the core samples, the same six cross-sections that were analysed before were analysed for alpha radiation at both sides, resulting in a mean value of $x = 0.96$ mBq/cm² and a standard deviation of $s = 0.54$ mBq/cm².

A similar probabilistic approach was used to that described above for non-alpha contamination. For a degree of freedom $n - 1 = 11$, and for a confidence interval of 99%, it could again be stated with 99% certainty that the most probable mean value for the measurements carried out, would not be higher than 1.44 mBq/cm². In the case of a homogeneous distribution of the activity in a concrete layer with a thickness equal to the range R, it could be calculated that the transmission due to self-absorption, for 2 Π measurements,

would be limited to 50% [I–1.7]. With a value for R = 6.5 mg/cm² [I–1.8], the mean specific alpha activity in the analysed concrete layers proved to be 0.30 Bq/g, and the maximum for the mean value 0.44 Bq/g.

To check the indicated evaluations in reality, destructive analyses were carried out. By means of alpha–gamma spectrometry, the total alpha–beta activity in a cross-section of the core sample with the highest detected activity was determined. The results of these direct measurements revealed a beta activity of 0.78 Bq/g and an alpha activity of 0.64 Bq/g. These practical analyses confirmed the evaluations carried out in calculating the specific values for possible alpha or beta activity in the core samples taken.

Thus, conservative practical measurements and evaluations (confirmed by the results of destructive alpha and beta analyses) demonstrated that sufficiently low surface activities and specific activities were present in the remaining structure of both buildings.

In this way, sufficient evidence was delivered to carry out the last part of the decommissioning project of the two buildings, i.e. the final demolition of the remaining structures and the removal of the demolition waste to an industrial disposal ground as indicated in Figs I–1.2 and I–1.3.

FIG. I–1.2. Demolition of buildings 6A/6B.

FIG. I–1.3. Green field conditions after demolition of buildings 6A/6B.

I–1.5. FINAL DEMOLITION OF THE MAIN PROCESS BUILDING

At the beginning of 2000, the decommissioning operations carried out at the main process building had made substantial progress. They were executed on an industrial scale and it was expected that the first demolition works could be started at the end of 2005. With the approaching final demolition of the building, a clearance methodology had to be proposed. Application of the methodology applied to the storage buildings of the pilot project was problematic for several reasons, the most important being:

— The type and spread of contamination: at the end of the reprocessing activities, all cells were cleaned using a high pressure water jet technique, which caused subsurface penetration of contamination.
— The total surface was large, which would require extensive manpower if all surfaces had to be monitored twice for unconditional release.
— Taking core samples at the previously most contaminated places would result in a large number of samples to be taken and analysed. It would be very difficult to prove that these samples were representative of the remaining structures of the building.

— Maintaining the structural stability of the building would prevent the removal of all pipe penetrations prior to demolition.

There were uncertainties as to whether the authorities would accept that a building could be released before all pipe penetrations had been removed and whether a controlled demolition of a building would be acceptable once it had been released, but without additional monitoring during the demolition process itself.

Although application of the methodology used for the two buildings in the pilot project was not rejected as such, an alternative was thoroughly studied, considering at least one complete measurement of all concrete structures and the removal of all detected residual radioactivity. This monitoring sequence would be followed by a controlled demolition of the concrete structures and crushing of the resulting concrete. Concrete blocks containing pipe penetrations would be sent to a controlled area in order to separate the tubes from the concrete.

During the crushing operations, metal parts would be separated from concrete and representative concrete samples taken, the sampling frequency meeting prevailing standards. In a next step, the concrete samples would be milled and homogenized, a smaller fraction being sent to the laboratory for analyses. Both methodologies were discussed with an independent radiation protection control organization prior to submission to the authorities.

A research and development programme was carried out in order to crush, mill, sample and monitor concrete dust similar to the procedure adopted for the melting of metal material. Discussions were organized with the independent radiation protection control organization in order to install the adequate crushing and milling technology such that the resulting concrete material could be reused in road construction. A final report was prepared and agreement was obtained, recognizing the technical as well as the financial issues. The licensing documents were prepared and approved.

The research and development programme resulted in a set of achievable goals that had to be met during the technical design. The most important goals and the related achievements were:

— The design of a representative sampling technique, based on prevailing standards from the mineral processing industry. A specific sampling unit was developed taking approximately 75 partial samples of 2 kg per processed batch of 7000 kg of concrete blocks. This comprised a crusher to bring the granulate dimensions to the requested level for measurement and a sample divider to split the total sample into a reduced sample and a reference sample.

— The design of a crushing technique in order to separate reinforcement bars from concrete and also to provide the right granulate dimension to both the sampling unit and the concrete processor. A typical electrically powered jaw crusher was installed with automated feed rate control. A remote controlled hammering unit could be activated in case of obstructions on the inside of the crusher.
— The design of a technique for removing reinforcement bars in order to prevent these bars blocking either the sampling unit or the sample crusher;
— The design of the transport devices to and from the various components in order to smooth the overall process. A tilting device was used to load the installation. Vibration systems and conveyor belts were used for the internal transportation of the material.
— The design of a ventilation system in order to prevent the release of dust into the environment. The installation was encapsulated and extracted via self-cleaning prefilters and absolute filters. An additional dust sampling unit was provided in both the extraction circuits upstream and downstream of the crusher.

The orders for the installation were placed in the first trimester of 2000 with the various parts of the equipment to be installed in an existing building. The building was partially dismantled and the area for the equipment prepared. Concrete works for the supporting structures were finalized in September 2000. The entire crushing installation, metal separator, transport and filter systems were delivered at the beginning of September 2000, and installation of all the systems was finalized in the middle of December 2000 (Fig. I–1.4). The complete installation was 48 m long, 10 m wide and 9 m high, and represented an investment of about €2.5 million. Its nominal capacity was set at 28 t/d. Operational and cold tests were carried out in January 2001, and training of the operators completed.

The necessary operational risk evaluation was carried out as well as a worker risk evaluation. The required documentation file was submitted to the respective safety authorities in order to get the startup permit. The conventional and nuclear safety inspection before startup was carried out in the second week of June 2001. As a result, operations were started at the end of June 2001.

At the end of December 2002, after 17 months of operation, 1500 t of concrete had been monitored. All of this material could be unconditionally released and removed from the site after analyses and agreement by the in-house health physics department and the authorities. The material was used in conventional road construction. The cost for the crushing, sampling, milling and analysis activities was in the order of €0.4/kg.

FIG. I–1.4. General view of the concrete crushing and sampling facility.

I–1.6. CONCLUSIONS

The industrial decommissioning of the main process building of the former Eurochemic reprocessing plant was started in 1990, and involved the removal and decontamination of equipment from each cell; the decontamination of cell walls, ceilings and floors; the dismantling of the ventilation system; the complete monitoring for unconditional release; and the demolition of the remaining structures, the removal of the remaining concrete debris and restoration of green field conditions.

About 1500 Mg of metal and 12 500 m^3 of concrete with 55 000 m^2 of concrete surfaces had to be removed and/or decontaminated.

Two fundamental principles were considered for the management of materials, equipment and/or components resulting from the decommissioning of the nuclear installations, which are mainly based on the guidelines of the IAEA's Fundamental Safety Principles with respect to radioactive waste management: (1) the generation of radioactive waste shall be kept to the minimum practicable; and (2) radioactive waste shall be managed in such a way that it will not impose undue burdens on future generations. In a broader context, recycling of materials was considered to be a first order ecological priority in order to limit the quantities of radioactive wastes to be disposed of, to reduce the technical and economic problems involved with the management

of radioactive wastes, and to make economic use of primary material and conserve natural resources of basic material for future generations.

Based on these fundamental principles, Belgoprocess made a straightforward choice for a strategy minimizing the amount of material to be managed as radioactive waste. The objective was achieved using aggressive decontamination techniques and unconditional release of decontaminated materials. Unconditionally released materials were recycled (i.e. metal material was sent to conventional melting facilities) or removed to conventional industrial disposal sites if they had no remaining value. These activities were carried out on an industrial scale with special emphasis on cost minimization, commitment to results within an overall planning and use of state of the art technology on an industrial scale.

Following a pilot project on two storage buildings, the final demolition of the main process building of the former Eurochemic reprocessing plant was undertaken. Based on the results of the pilot project, this required an alternative unconditional release methodology to be developed. A research and development programme was used to deliver a design that was successfully licensed.

Installation of all systems was finalized in the middle of December 2000 and represented an investment of about €2.5 million. Operational and cold tests were carried out in January 2001. Operations were started at the end of June 2001.

At the end of December 2002, after 17 months of operation, 1500 t of concrete had been monitored. All this material could be unconditionally released and removed from the site after analyses and agreement by the in-house health physics department and the authorities. The material was further used in conventional road construction. The cost for the crushing, sampling, milling and analysis activities was €0.4/kg.

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I–2. DISPOSITION OF METAL COMPONENTS FROM THE DECOMMISSIONING OF THE FORMER EUROCHEMIC REPROCESSING PLANT, BELGIUM

I–2.1. INTRODUCTION

As indicated in Annex I–1, Belgoprocess started the industrial decommissioning of the main process building of the former Eurochemic reprocessing plant in 1990, after completion of a pilot project. Two small storage buildings for final products from reprocessing were dismantled to verify the assumptions made in a previous paper study on decommissioning to demonstrate and develop dismantling techniques, and to train personnel. Both buildings were emptied and decontaminated to background levels. They were demolished and the remaining concrete debris was disposed of as industrial waste and green field conditions were restored [I–2.1]. The main conclusions of this pilot decommissioning project noted that emphasis had to be put on the automation of concrete decontamination and the decontamination of metal components.

When decommissioning nuclear installations, large quantities of metal components are produced as well as significant amounts of other radioactive materials. The former mostly show low surface contamination. Having been used or having been brought for a while into a controlled area marks them as 'suspect material'. In view of the very high costs for radioactive waste processing and disposal, alternatives have been considered, and much effort has gone into metal recycling through decontamination, melting and unconditional release.

In a broader context, recycling of materials can be considered to be a first order ecological priority in order to limit the quantities of radioactive wastes for final disposal, and to reduce the technical and economic problems involved with the management of radioactive wastes. It will also help to make economic use of primary material and to conserve natural resources of basic material for future generations.

In a demonstration programme, Belgoprocess showed that it is economically attractive to decontaminate metal components to unconditional release levels using dry abrasive blasting techniques, the unit cost for decontamination being only 30% of the global cost for radioactive waste treatment, conditioning, storage and disposal. As a result, an industrial dry abrasive blasting unit was installed in the Belgoprocess central decontamination infrastructure.

By the end of May 2001, a total of 523 t of contaminated metal had been treated after six years of operation. 182 t of this material was unconditionally released, having been monitored twice by the in-house health physics department. About 303 t of the metal with surfaces that could not be measured due to their shape, was melted for unconditional release in a radiologically controlled melting facility. The suitability of the abrasive blasting system was verified, and it was proven that there was no intrusion of contamination into the material surface.

I–2.2. DECONTAMINATION OF METAL COMPONENTS

From 1990 to 1991, various evaluations and laboratory tests were carried out to identify decontamination techniques that would enable the removal of surface contamination from metal components such that unconditional release levels were met so that the material could be reused without radiological restrictions. In many cases, this required the removal of a thin layer of structural material, which meant that much more aggressive methods had to be used than during normal maintenance operations. Based on the evaluations, it was concluded that appropriate decontamination techniques should be selected based on specific criteria:

— Safety: The application should not result in increased radiation hazards due to external contamination of workers or through inhalation of radioactive dust or aerosols formed during the decontamination activities.
— Efficiency: The surface contamination should be removed to a level that enables the recycling or reuse of the material.
— Waste minimization: The production of large quantities of secondary waste that require excessive work power and costs for treatment and disposal, and that result in additional exposure, should be avoided.
— Cost effectiveness: The decontamination costs should not exceed the costs for waste treatment and disposal of the material.
— Feasibility of industrialization: The quantities of contaminated materials that are produced during decommissioning activities and that may be available for decontamination, in general favour methods or techniques that are not labour intensive, or are difficult to handle or automate.

As a result, a comparative, semi-industrial demonstration programme was developed for the decontamination of metallic components using dry and wet abrasive blasting techniques. It was shown that it is economically attractive

to decontaminate such components to unconditional release levels, when all costs for conditioning and disposal of the resulting wastes are considered. Using adequate dry blasting, 32 t of contaminated profiles and plates were decontaminated to clearance levels, while avoiding the intrusion of contamination into the material. In a wet abrasive blasting system, another 3 t of metal components were decontaminated and measured to be below clearance levels. The results of these tests are indicated in Table I–2.1. They show that the wet abrasive blasting technique presents higher costs, less efficiency, much higher secondary waste production and much greater difficulties in measurements to clearance levels.

Following the demonstration programme, the installation of an automated industrial abrasive blasting infrastructure was evaluated. Estimates indicated that the decommissioning programme for the Eurochemic reprocessing plant would produce more than 309 t of slightly contaminated metal in a short time. This would include flat sheets of metal, profiles, casings of tanks and neutron shields that could be decontaminated to unconditional

TABLE I–2.1. RESULTS OF THE COMPARATIVE DEMONSTRATION PROGRAMME ON DRY AND WET ABRASIVE BLASTING DECONTAMINATION TECHNIQUES

Material treated	Dry system	Wet system
	32 t	3 t
Efficiency	Very high	Lower
Grid consumption	55 g/kg metal	109 g/kg metal
Secondary waste production		
• Intervention clothing	5.3%	8.2%
• Grit waste	5.5%	10.9%
• Water consumption	—	6.9 l/kg metal
Decontamination rate		
• Plates	57.4 kg/h 2.8 m²/h	48.0 kg/h 2.3 m²/h
• Profiles	127.7 kg/h 1.8 m²/h	106.8 kg/h 1.3 m²/h
Grit cost	€0.50/kg	€2.35/kg
Decontamination cost for the demonstration programme	€20.08/kg	€35.70/kg

release levels. In addition, more than 1500 t of other metal structures would be available for decontamination by abrasive blasting.

Based on the results of the demonstration programme, the costs for the decontamination by abrasive blasting of the 309 t of metal components were compared to the costs for normal waste treatment and disposal of the same material [I–2.2]. Three alternative treatment methods were considered. Cutting and decontamination of the 309 t of metal material to unconditional release levels in an industrial wet or dry abrasive blasting system, and treatment and disposal of the resulting secondary wastes were compared to cutting, supercompaction, cementation and surface disposal of the material as LLW. An overview of the costs related to the three alternatives in 1992 is given in Table I–2.2. Figure I–2.1 gives an overview of the total costs for the installation, waste treatment, and intermediate and final disposal for the considered alternatives, as a function of the available quantity of material.

Considering the decontamination factor, the efficiency, the consumption of abrasives, the cost of the used abrasives, the secondary waste production (abrasives and protective clothing), the financial analysis relating to the operation of the installation, the resulting unit cost for decontamination (30% of the cost for radioactive waste treatment, conditioning and disposal) and the

TABLE I–2.2. COSTS FOR THE DECONTAMINATION OF 309 Mg OF METAL IN A DRY OR A WET ABRASIVE BLASTING SYSTEM COMPARED WITH THE COSTS FOR WASTE TREATMENT (IN THOUSANDS OF EUROS IN 1992)

Task	Decontamination by dry abrasive blasting	Decontamination by wet abrasive blasting	Supercompaction and disposal
Capital costs	71.07	991.57	—
Installation costs	254.12	264.20	—
Reducing and packaging	795.89	795.89	1591.77
Decontamination	263.49	549.56	—
Measurements	160.24	160.24	—
Waste treatment	344.20	259.47	3367.63
Intermediate disposal	15.27	11.11	147.15
Final disposal	87.26	63.46	840.85
Total costs	1991.54	3095.50	5947.40

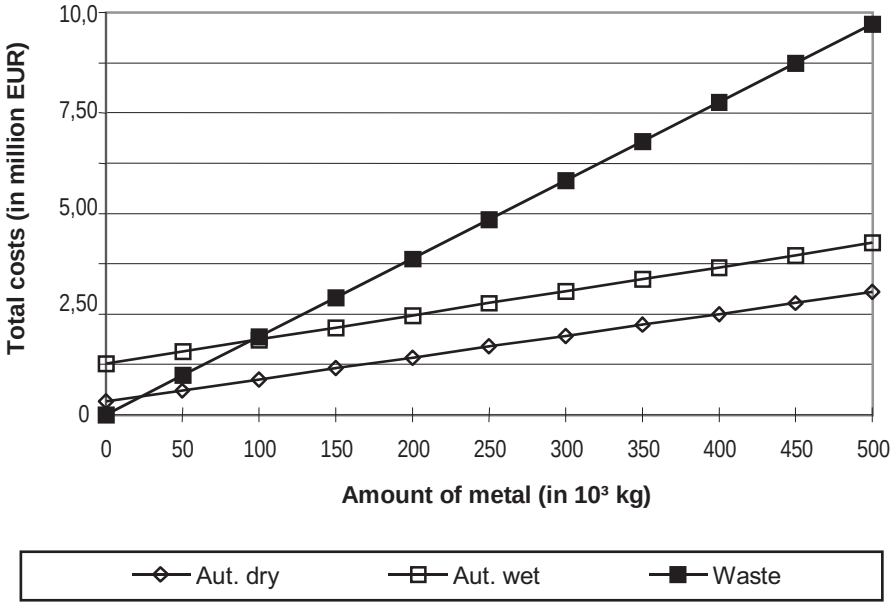

FIG. I–2.1. Total costs for waste treatment, and intermediate and final disposal compared with the costs for decontamination by abrasive blasting as a function of the amount of material.

availability of automated systems on the market, it was decided to install an industrial dry abrasive blasting unit in the Belgoprocess central decontamination infrastructure.

The equipment was ordered in 1995 and was installed during the first trimester of 1996. Functional and cold tests were carried out successfully at the beginning of May 1996 and operational activities started on 9 May 1996. A picture of the abrasive blasting installation is shown in Fig. I–2.2. A schematic diagram of the facility is given in Fig. I–2.3.

The working efficiency of the installation proved to be better than expected. A critical evaluation of the first operating period showed that the overall industrial performance of the activities could still be increased, however. The limiting factor proved to be the time required for making adequate, unconditional release measurements with acceptable accuracy.

The unconditional release of decontaminated material was based on existing procedures. This meant that all equipment, material and areas with contamination levels above background were considered to be radioactive. Materials monitored and found to be under the background levels of the portable contamination monitors used can be disposed of without any

FIG. I–2.2. Operational view of the abrasive blasting installation.

restrictions. With the existing equipment, demonstrating that there was no alpha contamination on decontaminated metal surfaces required a minimum monitoring time of about 6–10 s/50 cm². Surface areas had to be completely monitored twice, and surfaces or areas that could not be monitored were considered to be radioactive.

As the impact of the abrasives could conceivably introduce contamination into the surface layer, the suitability of the abrasive blasting system was verified by means of two independent control actions on samples taken from the material. First of all, contamination levels were monitored by non-destructive gamma measurements on samples taken before and after decontamination. In addition, a control monitoring was carried out on samples taken after decontamination by removing some surface material from the sample through chemical dissolution. A radiological characterization of this solution proved that there was no intrusion of contamination into the material surface.

I–2.3. IMPROVING THE PERFORMANCE OF THE OVERALL DECONTAMINATION PROCESS

To improve the efficiency and the economic productivity of the overall decontamination process, the technical and financial feasibility of alternative

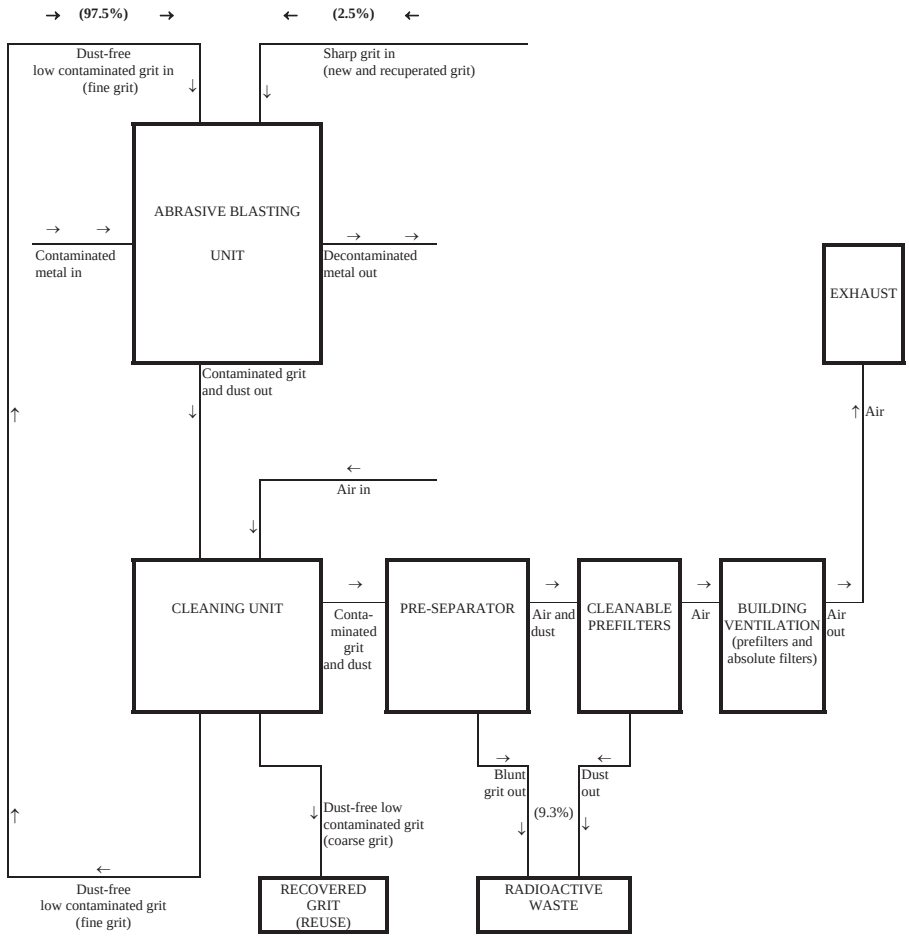

FIG. I–2.3. Schematic diagram of the dry abrasive blasting installation for the decontamination of metal components.

options were investigated in order to minimize the time for monitoring without reducing the required quality of the unconditional release procedures. It was proposed to limit the practical unconditional release measurements of the decontaminated metal to specific dose rate measurements, random sampling and sample analyses, and send the material to a controlled melting facility in order to use the melting process as a monitoring tool for unconditional release. The proposed alternative resulted in a lower overall cost for decontamination and unconditional release of material as:

— Cutting metal components into pieces with geometries and surfaces that could be measured was no longer required;
— Special tooling for handling material that had to be measured was no longer required;
— Unit costs for melting decreased, as larger quantities of material that showed little or no contamination could be melted and monitored for unconditional release;
— The capacity of the abrasive blasting installation could be increased by a factor of 2.5, as waiting times and/or storage times no longer had to be considered, and the material could be immediately removed for melting, reducing the unit cost for decontamination as well.

Furthermore, operators and radiation protection officers, normally involved in time consuming handling and measurement activities, could be integrated in the decommissioning team itself, contributing to increased progress of the actual decommissioning work.

From the technical point of view, unconditional release of metal components via melting offers:

— Adequate opportunities for representative, homogeneous sampling and adapted laboratory analyses;
— Opportunities for the unconditional release of metal components that could be decontaminated, but that cannot be measured due to their shape;
— Increased confidence in discharging the responsibilities of unconditional release of materials at currently applied release limits;
— Reduced operator dose uptake as smaller storage facilities, and limited storage times are required.

In addition, the lower quantities of material to be handled and to be reduced in size, would result in:

— Lower risk of heat stress to the operators cutting in ventilated suits;
— Lower physical load on operators having to handle the material.

Finally, a combination of the proposed alternatives resulted in the best technical and economic compromise, improving the performance of the overall decontamination process as envisaged, without reducing the required quality of current unconditional release practices.

I–2.4. RESULTS AFTER FIVE YEARS OF EXPERIENCE IN ABRASIVE BLASTING OF METAL

At the end of December 2002, after more than six years of operation, 602 t of contaminated metal had been treated and 182 t (30%) of this material had been unconditionally released, having been monitored twice by the in-house health physics department. About 420 t (70%) of the metal, having surfaces that could not be measured due to their shape, were melted for unconditional release in a controlled melting facility. About 16 t of metal grit was required to obtain the result as indicated. The total amount of secondary waste produced was 24 t, including 16 t of grit material and material that was removed from the metal surfaces as a result of the decontamination process, and an additional 8 t of protective clothing, bringing the total proportion of secondary waste production to 4%. The unit cost for abrasive decontamination proved to be about 30% of the global cost for radioactive waste treatment, conditioning and disposal of the same material, as expected.

I–2.5. DECONTAMINATION OF CONCRETE BLOCKS IN THE ABRASIVE BLASTING INSTALLATION

In December 1999, in a specific experiment, some 14.4 t of heavy concrete blocks were decontaminated in the same abrasive blasting installation and 12.2 t of this material could be released after two specific measurements carried out by the in-house health physics department. Only 2.2 t of dust material was recovered as secondary waste.

Since then, some 237 t of concrete and heavy concrete blocks have been decontaminated in the abrasive blasting installation and 209 t (88%) of this material unconditionally released, having been monitored twice by the in-house health physics department. The total amount of secondary waste produced was 28 t, including grit material and material removed from the concrete surfaces as a result of the decontamination process. During the operations, an additional 710 kg of protective clothing was produced as secondary waste, bringing the total amount of secondary waste production to about 13%. The unit cost for abrasive decontamination proved to be about 45% of the global cost for radioactive waste treatment, conditioning and disposal of the same material.

The suitability of the abrasive blasting system was also verified for concrete decontamination. In order to check the impact of the abrasives into the material surface, a small layer of the decontaminated concrete surface was removed by shaving. A gamma spectrometric characterization of the produced

dust proved that there was no intrusion of contamination into the material surface.

I–2.6. CONCLUSIONS

The management of wastes and materials from decommissioning represents a specific type of operation for the nuclear industry. The amount of materials and the very low level of activity present on a large part of the material arising demands new ways of disposition.

The management of this significant flow of material requires specific tools and instruments in order to ensure that the complete system is well under control and that the return of the materials to the 'non-controlled world' is done safely and in an environmentally friendly and economic way.

In all its decommissioning projects, Belgoprocess considers that minimization of the production of radioactive wastes and optimization of the recycling of materials is a first order ecological priority. The examples presented here indicate that such a strategy can be developed in a cost effective way as well, taking into consideration the specific boundary conditions.

REFERENCES TO ANNEX I–2

[I–2.1] TEUNCKENS, L., CLAES, J., GEENS, L.P., "Decommissioning of final product storage buildings at the former Eurochemic reprocessing plant", Large-scale Operations for the Dismantling of Nuclear Installations in the European Community (Proc. Sem. 1990), La Hague/Cherbourg, France (1990).

[I–2.2] TEUNCKENS, L., LEWANDOWSKI, P., WALTHÉRY, R, MILLEN, D., "Decontamination and melting of metallic material from the Eurochemic reprocessing plant", Melting and Recycling of Metallic Materials from Decommissioning (Proc. 3rd EC–OECD/NEA–Studsvik RadWaste European Sem. Nyköping, 1997) (1997).

I–3. THE BR-3 DECOMMISSIONING PROJECT, BELGIUM

I–3.1. INTRODUCTION

BR-3 was a small 10 MW(e) PWR which was shut down in 1987 after 25 years of operation. It was selected as an EU pilot project for the research and development programme on decommissioning of nuclear installations. The decommissioning project started in 1989. The optimization of the management of waste material generated by decommissioning activities has always been an intensive task and the minimization of the radioactive waste a priority. Over the past 16 years, the factors influencing the management of waste have been constantly evolving in Belgium, steered mainly by the following changes in technologies, regulations and economic conditions:

— The publication of the Royal Decree of 20 July 2001, establishing a legal frame on decommissioning and including a set of clearance levels;
— The improvement of the instrumentation used for characterization;
— The increase in the performance of decontamination techniques;
— The cost increase of the waste disposal paths;
— The implementation of international recommendations in areas such as environmental impact, waste categorization, human aspects, ethics, etc.;
— The strengthening of the legislation related to industrial safety and environmental release;
— The diminution of the background radiation level at the decommissioning site itself.

The first part of this annex gives a description of relevant influencing factors in order to define the context in which the dismantling activities took place. The second part puts in perspective the strategy chosen for the management of the waste, recognizing the influencing factors. As mentioned in the scope of this report, the focus is LLW. High and intermediate level wastes for which disposal in dedicated repositories is assumed are outside the scope of this report. They are therefore not examined in detail here.

I–3.2. DESCRIPTION OF INFLUENCING FACTORS

I–3.2.1. Quantities of materials

BR-3 was a small 10 MW(e) PWR research reactor. The relatively small size of the reactor, and the fact that its decommissioning was part of an R&D programme, meant that a fully automated monitoring and material identification system was not established. Nevertheless, the management of the wastes has always been supported by a strong traceability system.

I–3.2.2. Technical feasibility and availability of technology

I–3.2.2.1. Infrastructure

As BR-3 was selected as a pilot project to test dismantling techniques, the range of instruments and techniques available for BR-3 was therefore rather broad.

I–3.2.2.2. Transportation

The waste disposal site is less than 2 km away, which limits the cost of transportation. Most of the monitoring of the material takes place in situ, except for the gamma spectrometry measurements of the 200 L drums which takes place in a separate building leading to extra handling and time consumption.

I–3.2.2.3. Characteristics of radiological contamination

The radiological characterization is facilitated by the dominant presence of the easy to measure isotopes ^{60}Co and ^{137}Cs. The scaling factor for alpha contamination is maximally less than 1%. As the clearance level for alpha emitters is of the order of ten smaller than that of beta emitters, compliance to beta clearance level indicates that the alpha clearance level has also been met. Nevertheless, the absence of alpha contamination is controlled for.

I–3.2.2.4. Disposition pathways

The disposition options are radioactive waste, recycling following melting and clearance. Clearance methodologies are described in detail elsewhere in this publication.

Metals are sent to Duratek in the USA or Studvik in Sweden, if they satisfy the composition and radioactivity concentrations criteria defined by those installations. Metals melted at Duratek are then recycled in the nuclear

industry and none of the secondary wastes are returned back. At Studvik, the metal ingots are released directly or after a decay period and the secondary wastes (dust and slag) are returned.

A melting campaign of lead material contained inside a metallic structure was also organized in the BR-3 controlled area at SCK–CEN. This was done by placing a heating mattress around the metallic structure and applying heat. Once the lead is melted, a tap welded at the end of the structure is opened to run the molten lead into the ingots. As the melting point of the metal structure is higher than that of lead, the structure does not melt during this process.

The categorization of radioactive wastes followed was established by NIRAS/ONDRAF and mostly follows the IAEA waste classification scheme. Measurement methodologies for the characterization of waste had to be approved by NIRAS/ONDRAF.

I–3.2.2.5. Chemical toxicity of radioactive material

When materials are released in Belgium, chemical toxicity is considered in a similar way as industrial wastes. The legislation for industrial wastes is a regional competence whereas clearance is federal. This legislation on industrial wastes has become increasingly severe over the past 16 years of dismantling activities. The categorization of radioactive wastes presenting chemical toxicity has to be defined with NIRAS/ONDRAF. The definition of new categorizations due to particular chemical risks is costly.

I–3.2.3. Economic considerations

Decontamination to clearance levels or to de-categorization and monitoring can be expensive. However, the cost of radioactive waste disposal has also been increasing. It has been demonstrated that minimizing radioactive waste disposal, by using the means described herein, is still economically favourable as it has been over the past 16 years.

I–3.2.4. National policy — legal aspects

Before the implementation of European Council Directive 96/29, clearance criteria and methodologies where agreed with the competent authority on a case by case basis. This was mostly based on the criteria defined in IAEA TECDOC 855.[1] The implementation of the clearance concept (which

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Clearance Levels for Radionuclides in Solid Materials: Application of Exemption Principles, Interim Report for Comment, IAEA–TECDOC-855, IAEA, Vienna (1996).

was not compulsory) was implemented in the Royal Decree of 20 July 2001. The set of mass specific clearance levels is practically the same as the one given in RP-122 part-I. This is a general set of clearance levels and there is no set of clearance levels for conditional release; neither is there any specific set of clearance levels for each type of material.

Although it was stated that surface contamination should be checked for materials that could be manipulated after release, there is no definition of surface contamination clearance levels in the Decree. The IAEA's regulations for the safe transport of radioactive material [I–3.1] was still used for surface contamination clearance levels. A work group from the competent authority and the health physics staff responsible for the installation was established to define common methodologies and surface contamination clearance levels.

Each new measurement methodology had to be approved by the competent authority. The Decree clearly states that "…deliberate dilution with non radioactive material in order to reach the clearance level is forbidden." Conditional release is allowed only if an impact study is performed and approved by the competent authority.

A particularity of the Decree is that the traceability of the material has to be guaranteed up to its first destination.

I–3.2.5. QA and documentation

QA has been considered to be crucial since the beginning of the project as the traceability system of the materials and of the measurements needs to be supported by a good documentation system.

I–3.3. WASTE MANAGEMENT METHODOLOGIES

The approach that applies to metal and other materials from the installation is completely different to the one used for the concrete building itself. The basis of that difference is linked to the difference in mobility of the two types of material.

On the one hand, the dismantled materials of an installation are, after dismantling, reassembled into batches. From there on, the identification applies directly to the batches and no longer to each individual piece of material. The materials inside the batches are re-routable in order to optimize the management of the waste.

On the other hand, the concrete wall of the building remains in place up to the final clearance measurements. As the option of measuring after demolition is not advisable and the final use of the building had not been

defined from the beginning, the option of in situ measurements before eventual demolition or reuse was selected. The identification of the material is made on each wall, floor and ceiling of each room.

Different methodologies were developed for these two types of materials and had to be approved by the competent authority. When clearance measurements were carried out as described in the methodologies, a request for clearance, with the measurement results, was introduced and approved by the health physics department. In all cases, the clearance measurements are always two separate measurements made by two different operators and two different instruments. The released wastes are then considered as industrial wastes and have to comply with environmental legislation. As mentioned, traceability of the materials has to be guaranteed up to its first destination.

I–3.3.1. Metal and other materials

The segregation process occurs in two steps, the first step being the prediction of the disposition routes and the second, the re-routing of the material in order to optimize the management of the waste. The final survey is the measurement to confirm compliance with the clearance levels. It is stressed that flexibility in re-routing material was the key to minimization of the radioactive material.

I–3.3.1.1. First step: Prediction

The first segregation of material is made during the planning of each dismantling activity. At this stage, the person responsible for the dismantling activity has to predict the segregation of the generated material based on the primary characterization information (historical data, incidents, etc.). The segregation is mainly based on the radioactivity concentrations, the geometry, and the physical and chemical form of the materials to comply with the disposition pathways and decontamination criteria. Candidate materials for clearance have to comply with the scope of the selected clearance methodology. They are then reassembled into batches which are identified and traceable. The identification no longer refers to each individual piece of material.

I–3.3.1.2. Second step: Re-routing of the material and final survey

Some materials are directly categorized. This can be as radioactive wastes, as candidates for clearance or as materials that should be sent to a melting facility. Other materials first have to undergo decontamination in order

to be de-categorized or to become candidates for clearance. Candidate materials for clearance then enter into a re-routing system in order to minimize the radioactive waste.

The re-routing measurements and the final survey measurements are described in various measurement methodologies. All materials always undergo these two measurements. Two main methodologies apply to most of the materials. The scope of these methodologies are:

— Complex geometry material: rubble type;
— Clean material (no rust or grease) and flat or within a defined range of diameter.

As the identification applies to the batches and no longer to each individual piece of material, it was not possible to develop a multiple isotopic ratio for different parts of the facilities. The conservatism applied in isotopic ratio is counter-balanced by limiting the measurement errors. This is done by measuring in a geometry that is as close as possible to the calibration configuration.

Methodology for complex geometry material — rubble type

The re-routing and optimization system for the complex geometry material is as follows: each 20 kg of material that is a candidate for clearance is measured by a gross gamma counting device with a 4π geometry (ESM-CCM monitor). This measure is brief, about 20 to 40 s and inexpensive. The alarm level is set to two times the clearance level. Materials with radioactivity concentrations under that level are reassembled in a 200 L drum for the final clearance measurement with a Q^2 gamma spectrometry measurement device. Due to its higher accuracy, this measurement lasts longer, about 20 min, and is more expensive.

The gross gamma counting device has a dual purpose

— It separates materials that can undergo the final clearance measurement from other materials that are re-routed for decontamination, melting or considered as radioactive waste. Decontaminated material can again become a candidate for clearance. As a result, almost all the materials that undergo the final clearance measurement can be released.
— It is a ‘hot spot’ control, as it guarantees that each tenth of the material from the 200 L drum is two times below the clearance levels.

This device is validated to always overestimate the actual activity, by applying conservatism in the isotopic ratio and in the absorption factor of the material itself. This overestimation is kept lower than 30% by measuring material in a geometry that is as close as possible to the calibration configuration. It should also be noted that the poor efficiency of ^{137}Cs compared to ^{60}Co is partly compensated for by a higher clearance level. This limits the overestimation effect. The physicochemical composition of the materials does not have to be defined as the energy of the isotope to be measured is in a region where the attenuation by the material is only due to the apparent density and not to the chemical form of the material.

Methodology for clean material (no rust or grease) that is flat or within a defined range of diameter

The second group of material is the material that complies with the above definition. The conservatism applied to the isotopic ratio is compensated for by choosing a pure beta probe with an efficiency for ^{60}Co close to the one for ^{137}Cs. As the material has to be clean and flat or within a defined range of diameter, no extra conservatism is introduced in the attenuation factors. These materials undergo two measurements made by two different operators and two different instruments. The re-routing principle also takes place after each measurement. After any decontamination, the materials have to be re-measured twice.

The segregation criteria have been set as simple as possible to help the operator easily select the right methodology. For instance, for surface contamination, various methodologies could have been developed for various levels of rust or grease on the material, based on the origins and for various geometries such as the diameters of the tube. Multiplying the number of methodologies in this way could have resulted in operator confusion when segregating the material.

I–3.3.2. Building

Concerning the clearance of the building, the option of an in situ measurement before eventual demolition or reuse was selected. The concrete wall of the building remains up to the final clearance measurement with the identification of the material made on each wall, floor and ceiling of each room. A database has been developed to register any historical data, characterization measurements and decontaminations for each identified surface. Four categories have been defined according to the risk of contamination or activation. Methodologies were then developed for each category.

Decontaminations and characterizations were applied until the surface could be considered as a candidate for clearance.

Clearance measurements can be optimized room by room according to the isotopic ratio allocated to the surface to be measured unlike for metal materials where a multiple isotopic ratio for each surface could be applied to minimize conservatism. However, due to the following elements, the surface contamination measurement of concrete introduces more difficulties compared to the measurement of metal:

— The natural radioactivity of the building material has to be taken into account. This level varies with the origin of the building material. The energy of the beta emitted by ^{40}K is high and contributes significantly to the signal. The level of the signal due to the natural radioactivity in the building material is defined by direct measurement on each type of clean material.
— The decontamination technique can leave quite a rough surface.
— The material obviously cannot be transported to an area with a lower background radiation level. The variation inside a room can be much above the clearance level. Creative methods have to be developed to subtract that effect.

The overall process is summarized in the following flow charts (Fig. I–3.1).

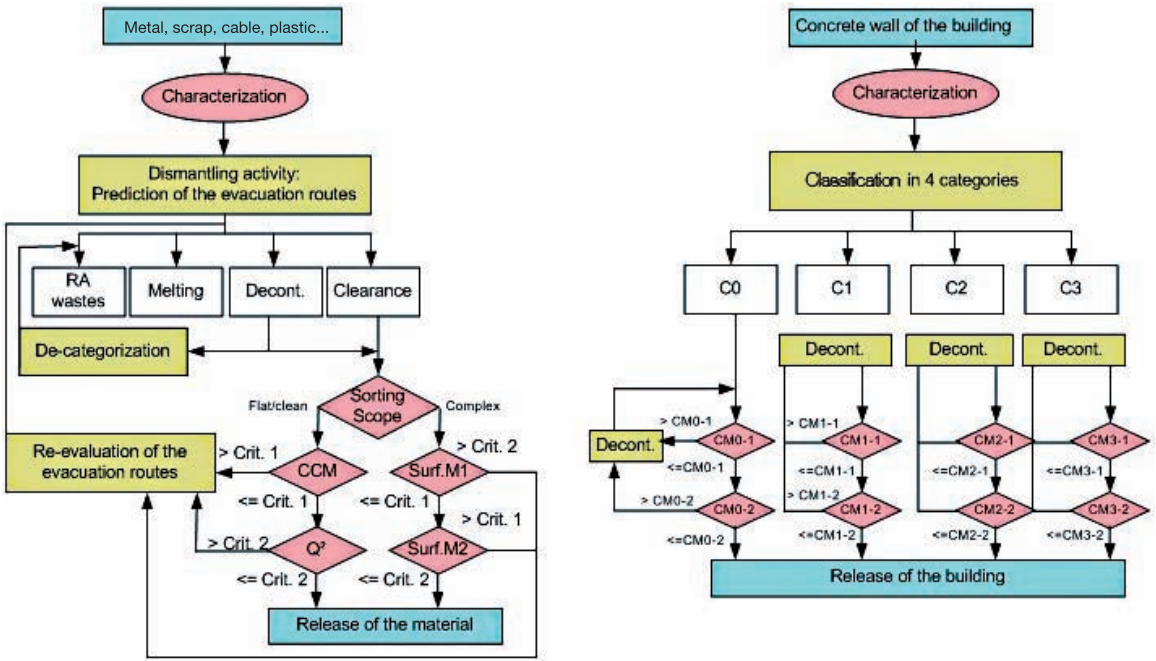

FIG. I–3.1. Flow charts for disposition of material of various types.

I–4. LONG TERM MANAGEMENT OF THE WASTE RESULTING FROM DISMANTLING OPERATIONS. DISPOSAL OF VLLW AT MORVILLIERS, FRANCE

I–4.1. INTRODUCTION

In the coming years, the increasing amount of dismantling work in France will raise the question of the availability of processes and routes for the long term management of the resulting material and waste. Most of the waste will arise after the progressive withdrawal from service of the 58 PWRs now in operation, i.e. not before 2015. However, France is currently faced with the dismantling of nine power reactors (of which six are gas cooled), the first reprocessing plant at Marcoule, obsolete reprocessing facilities at La Hague and the old research reactors and laboratories of the French Atomic Energy Commission (CEA).

The processes and routes used for management of the waste resulting from dismantling are not different from those used for that resulting from operations. For long lived medium and HLW, discussions have begun on the basis of the results of the research carried out in the areas identified in the Law of 30 December 1991. For dismantling, these types of waste contribute relatively small amounts of total waste (some 2000 t for the nine power reactors withdrawn from service) and it will be placed in interim storage until a final decision is made. A dedicated repository should be set up by 2013 for graphite waste (some 23 000 t), which contains significant amounts of long lived radioelements although its overall activity is low.

The bulk of the arisings will be short lived, low or medium activity waste or VLLW. The first type will be sent to the Aube Repository (CSA), which has a capacity of 1 million m^3 of packages. The total volume disposed of at the end of 2004 amounted to 166 500 m^3 with an annual delivery rate of 12 000–15 000 m^3 for a design flow of 30 000 m^3/a. This facility should be capable of absorbing the increased flow associated with the dismantling of the installations (some 50 000 t from dismantling of the nine power reactors). The Aube Repository can also be adapted for the disposal of large pieces of waste such as reactor vessel closure heads (Fig. I–4.1).

Under French regulations, waste produced in any part of a nuclear facility where it is liable to have been contaminated or activated must be disposed of in a traceable manner irrespective of its activity level. Much of it (140 000 t for the dismantling waste corresponding to the nine power reactors) does not require special containment arrangements as its activity is very low; indeed, in many cases, its radioactivity is purely hypothetical. As it would not be appropriate to

FIG. I–4.1. Disposal of large pieces of waste at the Aube Repository.

send it to the Aube Repository, the decision was made to create a dedicated repository for VLLW.

I–4.2. SELECTION OF A SITE FOR THE VLLW REPOSITORY

The environmental protection goals set for the VLLW repository resulted in specific technical requirements for the host site and the associated geological formations. These requirements essentially relate to the containment capability of the host rock, the ability to make disposal cavities in it possessing long term stability, and the monitoring and surveillance possibilities. These criteria have resulted in seeking a geological structure consisting of a thick layer of clay of a nature to isolate waste from deep aquifers and sufficiently large to contain all the disposal installations. The site, selected after two geological characterization campaigns carried out in 1999 and 2000 (Fig. I–4.2), is located in a wooded area in the district of Morvilliers, near the Aube Repository for the disposal of short lived low and medium level waste. The layer of clay (lower Aptian clays) that constitutes the disposal host formation is 15–25 m thick there, and is remarkably impervious and homogeneous.

The hydrogeological context features a captive aquifer underlying the formation in material of low permeability. The repository has been laid out in such a way that the disposal cavities are all entirely above the piezometric

FIG. I–4.2. Prospecting for a site.

surface of the aquifer (i.e. the pressure or head corresponding to the height to which water would rise in an observation well penetrating an aquifer).

In this geological and hydrogeological context, the site selected (which has an area of about 45 ha) met all the project requirements and easily conformed with the environmental protection criteria applying to industrial Class 1 technical burial centres dedicated to the storage of dangerous waste.

The capacity offered by the site is 650 000 m^3 of waste.

I–4.3. REGULATORY PROCESS

The administrative formalities relating to the creation of the repository consisted of two main stages, which took place between 2001 and 2003:

— An application for a declaration of public utility involving a public inquiry, between 28 May 2001 and 5 July 2001, associated with an application for a site deforestation permit. On approval by the inquiry commissioner, the project was declared of public utility in a county by-law on 10 October 2001, and the deforestation permit was granted in February 2002.
— A building permit application, associated with an operating permit for an installation classified for protection of the environment (ICPE). These

two files were the subject of a public inquiry between 3 June 2002 and 3 July 2002. After approval by the inquiry commissioner and granting of the building permit, which was made official in August 2002, the by-law authorizing operation was signed by the Governor of the County of Aube on 26 June 2003.

During this process, stakeholders and especially the sections of the public affected were kept fully informed about the setting up of the centre. The public inquiries proceeded smoothly and were interspersed with meetings to provide information in a manner of a nature to foster dialogue between the French national radioactive waste disposal agency (ANDRA) and the general public. The few questions and comments registered in the inquiry records and the favourable opinions of the municipal councils involved bear witness to the broad acceptance of the project by the local inhabitants. In this context, the exemplary management of the low and medium activity waste at the Aube Repository over more than ten years greatly contributed to the success of the project.

I–4.4. START OF THE CONSTRUCTION WORK

The work began with the deforestation of the site in August 2002. This operation, which was carried out by the National Forestry Office, was completed in December 2002 with the complete clearance of an initial area of 37 ha (Figs I–4.3 and I–4.4).

FIG. I–4.3. Overall view of the site (with the Aube Repository in the background).

FIG. I–4.4. Earthworks at the construction site.

At the same time, an 800 m access road was built between the future centre and the existing road network.

In January 2003, earthworks started. This phase consisted of stripping away the upper layers of the soil (top soil and weathered material), grading and forming the first platform to receive the disposal cells in the host clay, constructing the first two cells and building the ancillary installations (roads and utility networks, ponds). This work was completed by the construction of the first buildings (cell covering building and maintenance building) from April to July 2003.

After an inspection by the Champagne-Ardennes region industry, research and environment bureau (DRIRE), the centre was declared open on 14 August 2003 and received the first packages of VLLW on 1 October 2003 (Fig. I–4.5).

While operation was beginning, construction of the logistics building and processing building continued during the second half of 2003 and early 2004. The VLLW repository thus received its full complement of installations in the first half of 2004.

I–4.5. VLLW ACCEPTANCE CRITERIA

The VLLW mainly originates from the operation, closure and dismantling of the facilities associated with the industrial and military use of nuclear material and the preparation of radioelements. It also comes from research

Fig. I–4.5. Construction of the cavities.

laboratories and a range of industrial facilities where radioactive substances are used.

The requirements that apply to the waste accepted at the VLLW repository are the subject of a number of requirements of a general and technical nature drawn up by ANDRA in a manner consistent with the provisions of the county by-law authorizing operation. These detail the process of prior acceptance by ANDRA of batches of waste at the very low level repository, the radiological and chemical properties, and the required modes of packaging.

As concerns radioactivity, the acceptance criteria for VLLW involve the setting of levels of activity featuring:

— Allowance for the regulatory provisions applicable to facilities classified for protection of the environment. Specifically, Section No. 1711 of the list of ICPEs sets the maximum level of activity for disposal at the very low level repository (total activity below 37 TBq equivalent for Group 1 according to the calculation rules in Section No. 1700).
— Limitation of the impact of the very low level repository on the general public and the operating personnel. As concerns specific activity levels, the specifications imposed on the producers of waste are intended to limit the radioactivity of waste to a level at which their handling at the very low level repository can be conducted without any major constraints for the

operating personnel while guaranteeing that the worker dose limits will be complied with.

The limits on the specific activity of the VLLW were determined to limit the risk of exposure of workers through inhalation or external exposure. These also keep the impact on the general public minimal, even with hypothetical scenarios involving human intrusion into the repository in the distant future. The limit on specific activity is expressed in terms of what is referred to as a radiological index of acceptance for disposal (IRAS), which is defined in the Appendix to this Annex.

Finally, a number of exposure scenarios featuring long term waterborne transfer were also analysed. Study of the migration of radioelements through the clay formation of the repository, their transfer by surface water and the resulting impact on the general public accordingly resulted in limiting the total acceptable activity in the repository for some 20 radioelements, in addition to the criteria laid down in Section No. 1711 of the list of ICPEs.

As concerns the chemical criteria, the VLLW acceptance criteria were based on the body of rules applicable to non-radioactive waste. A distinction was thus drawn between dangerous and non-dangerous waste, as defined in Decree No. 2002-540 of 18 April 2002. For inert non-dangerous waste, consisting of metal or plastic and making up 95% of the VLLW inventory, acceptance at the repository depends on the results of prior chemical characterization of the waste in each batch.

As regards dangerous waste, the requirements relate to:

— Chemical characterization of the waste in each batch;
— Compliance with the criteria concerning the chemically leachable fraction of the waste;
— The contingent need of stabilization treatment to reduce the leachable fraction.

As concerns conditioning, a certain number of packages have been designed for the delivery of waste to the VLLW repository. These packages are intended to enable the transport and handling of waste between the production sites and the repository, to contain the radioactivity and to protect against the risk of dispersal of contamination for the operators, to avoid substances in powder form from becoming airborne and to make sure that any internal voids can be filled to meet repository stability requirements. The packages do not have any long term containment role; this role is played by the clay of the geological formation.

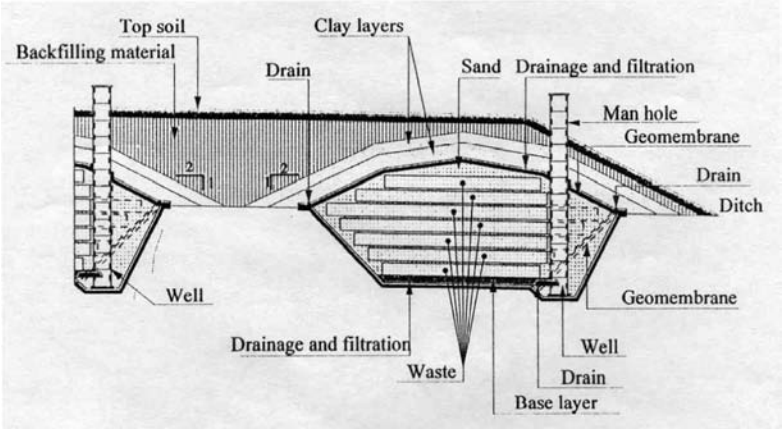

FIG. I–4.6. Cross-section of a disposal cavity.

I–4.6. CONTAINMENT OF WASTE

The VLLW is disposed of in cavities excavated in the clay, the bottoms of which are aged to receive any water that infiltrates in throughout the duration of disposal. They are thus isolated from the environment by an arrangement (Fig. I–4.6) consisting of:

— A synthetic membrane surrounding the waste associated with a monitoring system;
— A thick layer of clay below and on the sides of the disposal cavity;
— A layer of the same clay, placed on top of the waste.

Throughout the operation of the centre, placement of the waste takes place under mobile roofs to provide shelter against rain.

In the long term, the containment of the long lived radioactive elements and chemicals will be guaranteed by the retention properties of the clay formation. As regards the eventuality of total abandonment of the centre, ANDRA has assessed the consequences of long term migration of the substances present in the waste through the underlying clay beneath the repository into the nearest stream.

In considering this conservative scenario, ANDRA calculated the maximum impact on a group of individuals living close to the repository. The critical group would be using the water from the Noues d'Armance stream for its domestic and agricultural requirements. In this severe situation, the

radioactive dose liable to be received by the group would be a maximum of 0.1 μSv/a, well below the regulatory limit of 1000 μSv/a. As regards chemicals, calculation shows that there would be no effect on health under the same conditions.

I–4.7. CONCLUSIONS

The creation of the VLLW repository in the district of Morvilliers, 2 km from the Aube Repository was the subject of two public inquiries, conducted in 2001 and 2002. The positive findings of these inquiries made possible the immediate commencement of the construction phase, the ultimate stage of the project before the repository was declared open in the summer of 2003 for delivery of the first package of waste on 1 October 2003.

The Morvilliers Repository now constitutes an important new tool for the management of the greater part of the waste that will be produced during the forthcoming nuclear facility dismantling work.

After an operating period estimated at 30 years, ANDRA will continue its surveillance of the site and the surrounding environment for decades to guarantee that the repository is harmless to humans and the environment. Thereafter, once the behaviour of the site has been verified, the residual constraints will be simplified.

Appendix to Annex I–4

RADIOLOGICAL CRITERIA FOR ACCEPTANCE AT THE MORVILLIERS REPOSITORY

The limitation of the specific activity of VLLW is based on the division of radioelements into four classes according to their intrinsic toxicity:

- Class 0: radionuclides whose maximum specific activity is 1 Bq/g on average per batch of waste or 10 Bq/g per package of waste, in the case where the waste contains only one of any such radionuclides;
- Class 1: radionuclides whose maximum specific activity is 10 Bq/g on average per batch of waste or 100 Bq/g per package of waste, in the case where the waste contains only one of any such radionuclides;
- Class 2: radionuclides whose maximum specific activity is 100 Bq/g on average per batch of waste or 1000 Bq/g per package of waste, in the case where the waste contains only one of any such radionuclides;
- Class 3: radionuclides whose maximum specific activity is 1000 Bq/g on average per batch of waste or 10 000 Bq/g per package of waste, in the case where the waste contains only one of any such radionuclides.

To determine the acceptability of a batch of waste at the VLLW repository, an IRAS is applied that is defined as follows:

$$\text{IRAS} = \sum_i \frac{Am_i}{10^{Class\,i}}$$

where Am_i is the specific activity of radionuclide i (in Bq/g) in the batch of waste involved (see below) and *Class i* is the number of the VLLW class (0, 1, 2, 3) to which radionuclide i belongs.

For waste to be accepted at the VLLW repository, each batch must have an IRAS not exceeding one. A package may have an IRAS of up to ten provided that the average index of the package to which it belongs is not above one.

The following table gives the classes of the main radioelements:

^{3}H	^{14}C	^{60}Co	^{63}Ni	^{90}Sr	^{137}Cs	^{232}U to ^{238}U	^{236}Pu to ^{240}Pu, ^{241}Am, ^{242}Pu, ^{244}Pu
3	3	1	3	3	1	2	1

It can be seen that reference activity is 10 Bq/g for ^{60}Co and ^{137}Cs, 100 Bq/g for isotopes of uranium and 1000 Bq/g for tritium and ^{14}C.

I–5. RELEASE OF SOLID MATERIALS, VANDELLOS-1 DECOMMISSIONING PROJECT, ENRESA, SPAIN

I–5.1. INTRODUCTION

The declassification of materials with very low activity content was one of the relevant activities in the management of waste materials generated during the Vandellos-1 decommissioning project. The application of the specific radiological criteria on which declassification was based and the methodologies, equipment and measuring systems have had to be adapted to an industrial process.

Declassification may be defined as an administrative activity that allows materials previously subject to controlled practices to be freed from regulatory control and, therefore, managed as conventional wastes, guaranteeing compliance with the authorized levels of declassification.

The declassification of materials as a management route carries with it various advantages and, for this reason, ENRESA has developed and implemented declassification methodologies in order to be able to manage these materials as conventional wastes, such that they do not pose a risk for people or the environment.

Waste materials from radiologically controlled areas in facilities undergoing dismantling are subject to the same requirements as the wastes generated at operating installations. In each case, the removal of all waste materials from the facility requires the relevant prior authorization. Those materials that are to be reused or recycled are also required to meet the acceptance criteria established in the authorization for the facility at which they are to be processed, treated or reused in the future.

Authorizations for the declassification of waste materials may be generic or specific to the facility being dismantled. These authorizations may determine unconditional levels of declassification or levels linked to a given conventional management route foreseen for the materials (disposal at municipal tips, at tips for industrial toxic wastes, recycling at foundries or simple conventional reuse).

I–5.2. MATERIAL RELEASE PROCESS DESCRIPTION

The scope of materials declassification includes all the systems, functions and components located in radiological zones. The different origins of these materials and their applications during operation led to their being classified fundamentally as: (1) metallic equipment and scrap (piping, valves, motors,

support structures, tanks, etc.); (2) concrete rubble; (3) soils; (4) insulation; (5) secondary materials (plastics, protective clothing, etc.); and (6) others. Before being managed as conventional materials, those coming from radiological zones are subjected to a decision making process. A fundamental and critical issue in the management of materials for declassification is the justification of compliance with the radiological criteria established by the authorities. This required measurement, estimation, calculation and comparison of the values of activity with the derived declassification values.

This process required the assignment of 'specific radiological spectra' to the materials to be managed, allowing values to be derived through the use of scaling factors for non-measurable radionuclides on the basis of those detected by means of industrial declassification equipment. Overall, the process includes the following stages:

I–5.2.1. Analysis of radiological history

This phase included the gathering of the largest amount of historical information available on operation and dismantling. This provided an as accurate as possible insight into the potential distribution of residual activity in the different components and systems to be declassified, made it possible to determine whether this activity was fixed and/or detachable, and identified the isotopes to be expected in each case. The information could be obtained through: (1) initial radiometric studies; (2) radiological surveillance of the work performed during plant operation; (3) the location of the functions or systems that were present in the different buildings, areas and installations; and (4) previous and present periodic radiological surveillance. In essence, it was necessary to classify all the functions and systems, along with their components.

This analysis allowed a 'type spectrum' to be assigned to these systems depending on the associated source term, which in most cases is influenced by the fluids that they carried. This also indicated whether or not it was necessary to measure alpha emitting isotopes. The preliminary source term defined in this way made it possible to identify the different zones of the facility and the corresponding radiological spectra.

I–5.2.2. In situ characterization

The main objective of the characterization of systems and components was to gain insight into and assign, qualitatively and quantitatively, the destination of materials to be managed depending on the values recorded.

I–5.2.2.1. Prerequisites

Before beginning work, a check was made to ensure that: (1) the systems, equipment and components were correctly identified; (2) they were definitively tagged out technically and administratively; (3) there was a list of radiological functions with the associated spectrum; (4) the associated documentation was available; and (5) packaging was available for the transfer of the materials to be generated. In order to carry out this characterization, measurements are performed of: (1) the irradiation dose rate and (2) fixed and/or detachable beta–gamma and/or alpha surface contamination.

I–5.2.2.2. Measuring instrumentation

The equipment used for the measurements was the portable equipment usually used in radiological protection, such as radiation counters and contamination probes (proportional gas counter). It is recommended that background measures be performed periodically and that checks be made to ensure that the equipment is not contaminated. If alpha measurements were to be performed, the presence of radon could give a high natural background so the area was ventilated.

Characterization is performed by directly measuring the surfaces of the components, attempting to ensure that this was carried out on accessible areas in most cases. Indirect measures were performed if high background levels were encountered or in the case of inaccessibility of the measuring equipment.

I–5.2.2.3. Physical criteria

Consideration was given to the following: (1) the physical and functional homogeneity of the materials, such that homogeneous batches are produced on the basis of operating history and geographical area; (2) differentiation by material streams having homogeneous densities, the mixing of materials of different types not being allowed; (3) confirmation of the suitability of the measuring equipment and optimization of the number of containers used to hold the materials generated; and (4) weight limitation and homogeneity as regards types of geometries, with no mixing of different geometries.

I–5.2.2.4. Radiological criteria

These criteria are numerical and correspond to: (1) levels of alpha and beta–gamma surface contamination; (2) contact dose rate (for movable elements) or the rate at 1 m (for fixed elements) (Table I–5.1 shows the values

used for the pre-classification of the materials in the zones of origin); (3) function and associated spectrum, considering the radiological and non-radiological function; and (4) determination of streams depending on action levels (Table I–5.2 shows the ranges for these values).

TABLE I–5.1. ACTION LEVELS FOR RADIOLOGICAL CLASSIFICATION

Determination	Level	Value (unit)
Contact dose rate (removable elements)	NR0	10 μSv/h
Dose rate at 1 m (walls)	NR1	1 μSv/h
Overall beta–gamma surface contamination	NB2	100 Bq/cm^2
Overall alpha surface contamination	NA2	0.5 Bq/cm^2
Overall beta–gamma surface contamination	NB1	10 Bq/cm^2
Overall alpha surface contamination	NA1	0.1 Bq/cm^2

TABLE I–5.2. RADIOLOGICAL SEGREGATION CRITERIA — MATERIAL STREAMS

Determination	Level	Stream
Contact dose rate (removable elements)	>NR0	Radioactive waste
Dose rate at 1 m (walls)	>NR1	
Overall beta–gamma surface contamination	>NB2	
Overall alpha surface contamination	>NA2	
Overall beta–gamma surface contamination	<NB1	Potentially declassifiable
Overall alpha surface contamination	<NA1	
Overall beta–gamma surface contamination	>NB1 and <NB2	
Overall alpha surface contamination	>NA1 and <NA2	Decontaminatable

I–5.2.2.5. Performance of measurements

This allowed the destination of the material to be determined a priori with sufficient confidence to eliminate later rejection following release measurements.

I–5.2.3. Homogeneity and radiological representativeness

As has been pointed out above, in order to guarantee the homogeneity of the materials, it is necessary that these are similar in nature and have the same typology, geometry and associated radiological spectrum. This provides optimum conditions for the performance of measurements, since the modelled templates will be homogeneous. Consideration should be given to homogeneity in calculating the mass to surface transfer factors, with a view to estimating the surface activity of the materials.

I–5.2.4. Preparation of authorized handling units (AHUs)

The verification and checking of the previous steps enabled the production of AHUs, such that there would be an optimum quantity of material in individually identified containers as declassification was based on measurements at an AHU level (Fig. I–5.1). These AHUs may be: (1) primary, generated directly as a result of dismantling operations or (2) secondary, produced as a result of the segregation of contaminated parts or the regrouping of units. Administratively, the AHU is regulated by its docket, which reflects all the identifying data: origin, destination, typology, spectrum, degree of filling, type of material, original functions and initial radiological characterization.

FIG. I–5.1. Preparation of AHUs.

I–5.2.5. Checking of prerequisites

Before undertaking declassification measurements, it was necessary to carry out checks ensuring, with a high degree of probability, that all the steps established in the methodology had been adhered to, and that the measurement process had been shown to be able to meet the required standards. For this purpose, the following production requirements were verified: (1) that the origins of the material were in accordance with the original function or system; (2) that the type of material and its nature were specified; (3) that the packaging was correct; (4) that the weight had been recorded; (5) that the isotopic type of origin was correct; (6) that the radiological characterization data were available; and (7) that the destination was assigned.

I–5.2.6. Measurement and calculation

I–5.2.6.1. Measuring systems

In view of the fact that the equipment used for preliminary characterization had no spectrometric capacity, the equipment used for declassification measurements had this capacity and was capable of detecting measurable gamma isotopes. The specific equipment used was the Box-counter manufactured by Canberra. This equipment is fitted (Fig. I–5.2) with a control computer running specific software, allowing movement of the platform and the management of data on the modular measurement of containers and/or drums. It receives, analyses and registers the 12 segmented spectra and the value measured, calculating activities on the basis of the measurement parameters and data. The equipment was verified daily and periodically in order to check for correct

FIG. I–5.2. Box-counter equipment — measuring process.

operation, with functional, and efficiency and energy calibration tests being carried out.

I–5.2.6.2. Release measurement

The system consisted of a platform that determines the measurement geometry to be applied. The containers were placed on this platform for the spectrometric measurement to be performed, in three phases, by means of four Ge detectors configured two by two. The containers are positioned perpendicularly with respect to the detectors. The duration of the measurement varied depending on the spectrum and the type of geometry (15 min to 4 h).

I–5.2.6.3. Activity calculation

The predetermined isotope vectors allowed the estimation of measurable and non-measurable isotopes. The latter were calculated from the former, taking into account the scaling factors and corresponding key isotopes (which are the most significant gamma and beta–gamma emitters) and taking into account the associated uncertainties. In this way, the concentration of mass activity was obtained. Surface activity was calculated using this value and by means of the geometric conversion (surface–mass) factor.

I–5.2.6.4. Comparison with declassification levels

The measured activities were compared to the declassification levels and a check made to ensure that the sum of all the fractions (of activity/declassification level) for all the radionuclides was less than or equal to one, this being the acceptance criterion for the AHU to be declared as declassified as defined by the authorities. A check was also made to ensure that the mass and surface activity criteria were met.

I–5.2.6.5. Verification of measurement quality

This phase included verification of: (1) the equipment and the electronics associated with it by means of internal verifications and calibrations; and (2) the specific software based calculation process, which is performed using the measurement results.

I–5.2.7. Declassification

The documentation generated goes to make up the 'release dossier', composed of: (1) the characterization dossier; (2) the declassification records; (3) the prerequisite verification records; and (4) the release certificate. On completion of this step, the material is declared to be conventional and its subsequent management begins.

I–5.3. OVERALL LICENSING OF THE PROCESS

The application of the methodology, including the industrial measuring equipment incorporated, had to pass official tests witnessed by inspectors from the authorities. The tests performed consisted of: (1) equipment verification and calibration; (2) measurement of actual activity; (3) the calculation and comparison of activities; and (4) the documentation and records generated. Following successful performance of these tests, the regulatory body required the performance of an additional checking plan, covering the entire process of declassification for each of the authorized routes. The scope of the additional testing plan was as follows:

— Verification of all the steps of the methodologies on at least 5% of each releasable material stored at the time of initiation (concrete, ferrous scrap, non-ferrous scrap, cables, plastic materials and miscellaneous materials).
— Testing of at least two AHUs of each material representative of all the isotopic spectra existing at the facility and generating declassifiable materials. The selection of AHUs was made on a random basis.

In addition to verification of the quality of the declassification process, an additional control was performed through estimation of the activity of the existing radionuclides, by means of radiochemical analysis at an exterior laboratory.

As part of the conditions issued by the regulatory body in its favourable report, ENRESA was required to perform two additional control programmes:

(a) Six-monthly certification by an independent organization of the quality of the measurement performed by the measuring equipment used in the release process;
(b) Data acquisition, radiochemical analysis and contrasting of the results, for the following:

(i) Analysis of at least 1% of the AHU processed for each combination representative of the different types of spectra of the release material;
(ii) Storage at the facility of the AHUs used as a sample, until the corresponding analytical results verify that the activities assigned during declassification have been conservative;
(iii) The planned actions and investigations, if the analytical results of the AHUs analysed exceeded the levels of activity authorized for declassification.

I–5.4. CONCLUSIONS

The following are specific and generally applicable conclusions of ENRESA's experience from Vandellos-1:

— Control and certification of the radiological content of materials that are candidates to be released are essential in the different phases of materials management, since they underpin the clarity and correctness of the decision making process. They begin with the operating history of the facility and are ratified through the application of accepted methodologies confirming the reliability of the results obtained.
— Any release methodology should be made up of a generic set of steps or stages that must be adapted on an application specific basis.
— As part of these methodologies, the measuring equipment to be used in each case must be able to ensure that compliance with the applicable criteria can be justified and be compatible with the needs of the production process.
— The process of licensing before the regulatory body has required a far reaching plan of justifications and validations, covering all the different stages and activities of the methodologies used. This generated a significant workload and resources, and required the generation of substantial documentation of different types and challenges. This aspect should possibly be reconsidered in designing future campaigns.

Appendix to Annex I–5

MAIN INFLUENCES ON THE MATERIALS MANAGEMENT APPLIED BY ENRESA IN THE DISMANTLING OF THE VANDELLOS-1 NUCLEAR POWER PLANT

There are numerous factors that influence the design and practical implementation of the materials management methodology. The main influences at Vandellos-1 were as described below.

A–5.1. QUANTITIES OF MATERIALS

As may be appreciated in Fig. I–5.3, important volumes of materials have been managed during the dismantling of the Vandellos-1 nuclear power plant, of which only a minor part has been considered as constituting radioactive waste.

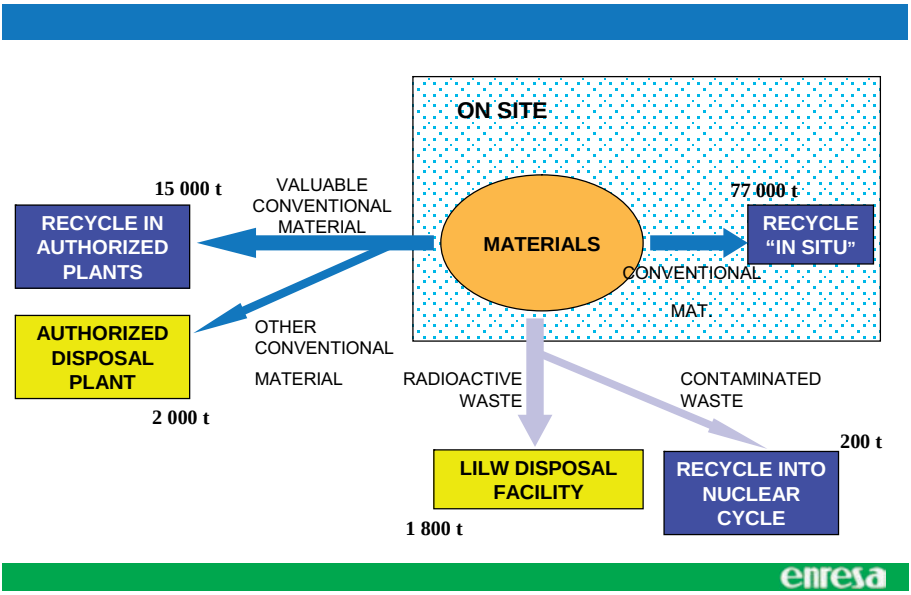

FIG. I–5.3. Vandellos-1 material recycling diagram.

In the presence of such large quantities of materials, many of them coming from active zones, there is a high risk of more materials being declared as radioactive wastes than actually warrant this consideration.

The objective of the methodology applied by ENRESA was to keep the generation of radioactive material as low as reasonably possible and allowed by the techniques and equipment available on the market.

In the short term, this decision implies a longer duration of the dismantling process and a higher cost, this being more than compensated for in the medium and long term by the cost saving from reduced disposals of radioactive waste.

A–5.2. TECHNICAL FEASIBILITY AND AVAILABILITY OF TECHNOLOGY

A–5.2.1. Infrastructure

Spain possesses infrastructure for the disposal and/or recycling of most of the wastes generated during dismantling, both radioactive and conventional:

— El Cabril low and intermediate level radioactive waste disposal facility;
— Recycling plants for conventional materials having an intrinsic value (metals, paper, electronic and organic products, etc.);
— Controlled tips for toxic and/or hazardous conventional materials having no intrinsic value (paints, plastics, asbestos, etc.).

At present, there is no centralized facility for the management of HLW, although individual solutions have been adopted in certain cases (Trillo nuclear power plant), with management based on dual purpose (storage and transport) casks. A facility allowing for the overall management of such wastes is expected in the future. However, this has not conditioned the materials management strategy applied to any large extent, since the spent fuel was sent to France prior to the start of dismantling.

No decision has yet been taken regarding the large quantities of graphite present at the site, which was either part of the reactor pile (moderator) or the fuel sleeves themselves. The interim option adopted has been to keep the graphite from the fuel assembly sleeves at the site during the safe enclosure period, to be jointly managed along with the rest of the graphite present in the reactor pile when Level 3 dismantling is undertaken.

A–5.2.2. Equipment/technology

The technology available for the performance of characterization, decontamination, cutting activities, etc. has a radical influence on the design and practical application of the materials management methodology.

In the case of Vandellos-1, the main agent influencing materials management has been the monitoring tool used for the declassification measurements, which required characteristics such as dimensions, weights, grouping criteria, etc. to be identified for each material type. This has significantly affected the overall management of the materials requiring a large number of cutting operations, packaging of materials in boxes, etc.

The decontamination tools available (decontamination workshop based on sand blasting) have also had an influence since, in view of their low efficiency, they have made it necessary for most of the decontamination operations to be carried out in situ prior to disassembly.

A–5.2.3. Dilution with non-active materials

The deliberate mixing of contaminated and non-contaminated materials in order to achieve declassification levels has been expressly prohibited during the dismantling of Vandellos-1 nuclear power plant.

In this respect, the surface declassification methodology expressly required the need to completely decontaminate both faces of all walls, floors and ceilings before undertaking their demolition.

In practice, this decision has meant high monitoring and decontamination costs, due to the difficulty in accessing many of the zones.

A–5.3. ECONOMIC CONSIDERATIONS

In general, the costs of managing materials during the dismantling of Vandellos-1 nuclear power plant can be subdivided as follows:

— PC: Production costs (cutting and packaging);
— TC: Treatment costs (decontamination, declassification, conditioning, etc.);
— DC: Disposal costs (once the materials have left the site).

The types of materials generated may in turn be classified as follows:

— LILW: Low and intermediate level radioactive wastes;

— VLLW: Very low level radioactive wastes;
— DESC: Materials declassifiable following a process of decontamination and monitoring;
— CONV: Conventional materials.

The influence of the economic factors in the selection of one or another materials management alternative has been based on comparative cost studies incorporating the parameters indicated in Section 4.3 (manpower, equipment, etc.).

In general, the results of these studies showed that the additional costs of decontamination and monitoring of a declassifiable material are lower than the excess cost that would arise if a conventional material were disposed of as LILW or VLLW. This is shown in the following table and equation:

	LILW	VLLW	DESC	CONV
PC	M	M	M	L
TC	L	L	H	L
DC	VH	H	L	L

Note: Costs are shown schematically as L: low, M: medium, H: high and VH: very high.

$$(PC + TC + DC)_{LILW} > (PC + TC + DC)_{VLLW} > (PC + TC + DC)_{DESC} >$$

$$(PC + TC + DC)_{CONV}$$

However, depending on the nature of the material, and on the type and distribution of its radiological contamination, TC may increase such that declassification would cease to be economically viable. In such cases, these materials should be classified directly on the basis of expert judgment and at the place of origin as LILW or VLLW, instead of as declassifiable materials.

A–5.4. FINAL STATUS OF THE DECOMMISSIONING ACTIVITIES

The objectives of the project established the need for most of the land making up the site to be released for any type of future use, leaving just the reactor and its surrounding areas inside a new protected perimeter.

This obviously meant that all buildings and other materials had to be removed from the area to be released. Consequently, the dismantling strategy

does not include the possibility of storing contaminated material arising from dismantling in the zone to be released. However, the existence of a temporary store for graphite within the protected safe enclosure perimeter is allowed as this will be removed when Level 3 dismantling is undertaken.

A–5.5. QA AND DOCUMENTATION

The materials management process was designed such that the activities performed on each batch of materials were easily traceable, especially those relating to the origin and composition of the different material batches (containers) and their monitoring.

A–5.6. ETHICAL ASPECTS

As a public company, ENRESA provides a service to society, through which it accepts the commitment to:

— Minimize the wastes produced to the extent to which this is technically and economically feasible;
— Leave the plant free from contamination to the extent to which this is technically and economically feasible.

As a result, the methodology applied emphasized segregation, decontamination, cleaning and declassification routes rather than those corresponding to radioactive waste generation and disposal.

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I–6. IMPLEMENTATION OF STAGE 3 DECOMMISSIONING AND OPTIMIZATION OF RADIOACTIVE WASTE GENERATION, TRITON FACILITY, FRANCE

I–6.1. INTRODUCTION

The CEA centre of Fontenay-aux-Roses was created in 1946, when the French nuclear energy programme started. Two generations of facilities have been built and operated. The first generation remained operational for 15 years and was dismantled in the late 1950s. It was replaced by a new generation of facilities, as part of the French electronuclear programme, and these included the Triton and Nereïde research reactors (hereafter called the Triton facility).

In accordance with the CEA strategy and taking into account its urban location, in 1998 the CEA Fontenay-aux-Roses centre decided to launch an extensive cleanup programme to be implemented from 2010 onwards. This included the Stage 3 decommissioning of the Triton facility.

In the frame of this decommissioning project, a decommissioning strategy was developed making it possible to optimize the volume of radioactive waste generated.

I–6.2. TRITON AND NEREÏDE DESIGN AND HISTORY

The Triton facility was built in the late 1950s. Triton was first operated in June 1959. It was a 6 MW research reactor working with high enriched uranium as fuel, and light water as moderator and coolant. Nereïde was first operated in 1960. It was a 600 kW pile also using enriched uranium as fuel and light water as moderator and coolant.

Triton was used mainly for radioelement production, radiological protection studies, neutron diffraction experiments (Fig. I–6.1) and fundamental physics experiments. Five neutron beam ports were built in the concrete radiation shielding of the Triton pool to make neutron experiments possible (Fig. I–6.2).

Nereïde was a mobile system, which when in remote position, made it possible to conduct experiments in a dry pool called Naïade. In contact position, the Nereïde pile was used to irradiate experimental devices. The Nereïde pile and Naïade pool were mainly used for radiation protection experiments and for studies regarding graphite gas subcritical assemblies

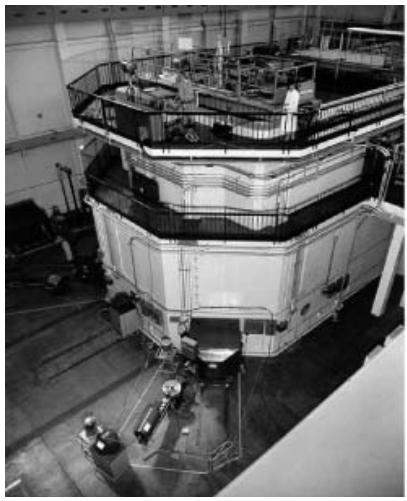

FIG. I–6.1. Physics experiment at the Triton facility.

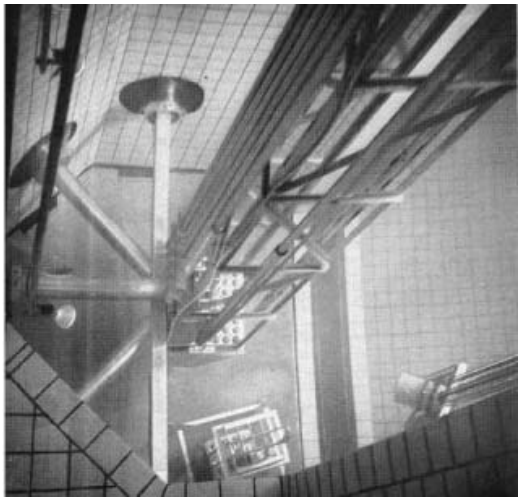

FIG. I–6.2. Triton pile with the five beam ports.

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A hot cell equipped with remote handling systems was built against the Triton pool. It was used to transfer spent fuel, and activated materials and equipment from the pool, for conditioning and packaging, before 'evacuation'.

Triton and Nereïde were shut down in May 1982 and December 1981, respectively. From 1983 to 1986, Stage 2 decommissioning operations of the Triton facility were performed.

The Triton facility, which had been registered in the records as basic nuclear installation (BNI) No. 10, was removed from the BNI list in 1987. It was reclassified on the basis of an installation having environmental protection concerns.

In 1993, it was decided to perform a complementary decontamination of the hot cell and to remove the spent radioactive sources present in the facility, in order to send them to an interim storage facility.

From 1994, the Triton facility was definitively closed and from that date, only simplified radiation monitoring of the facility has been performed.

In March 2001, the chief executive of the CEA Fontenay-aux-Roses centre decided to launch the Stage 3 decommissioning of Triton.

I–6.3. TRITON STAGE 3 DECOMMISSIONING PROJECT

I–6.3.1. Decommissioning plan

Following international recommendations, a decommissioning plan, i.e. a conceptual study including the technical and economic analysis of the decommissioning, was drawn up in 2001. It was called the preliminary project study. At that time, the plan was to dismantle the entire facility by cutting it into blocks to be sent to an interim storage facility as VLLW. As a result, the Stage 3 decommissioning of the Triton facility would have generated about 1700 m^3 (3500 t) of VLLW, mostly concrete. No low or intermediate level radioactive waste was expected.

Based on the facility history, its operating conditions and its radiological inventory, it was decided in 2002 to change the decommissioning strategy. Thus, new studies were carried out, based on a comprehensive radiological characterization of the facility and on the implementation of zoning called specific zoning for dismantling. This zoning would aim at defining work areas generating conventional waste and others generating radioactive waste.

Specification of the zoning for dismantling was composed of three parts:

— A first part, describing the risk points, their history and radiological status;
— A second more analytical part, precisely identifying the past events having consequences on the radiological status: at this point, a preliminary specific zoning can be established;
— A third part making it possible to finalize the specific zoning by means of radiochemical analysis of samples, use of calculation codes, in situ measurements by gamma spectrometry, etc.

I–6.3.2. History of the facility

National and local archives and documentation were collected, including historical documents, safety documents, photographs, slides and video records, and documents (records, plans, etc.) related to all the life phases of the plant (design, construction, operation, shutdown, Stage 2 decommissioning and modifications of the installation that have occurred since the end of Stage 2 decommissioning).

The collection of evidence from workers was also organized. Evidence was obtained from retired employees as well. The evidence collected allowed the project team to bring consistency to the data collected.

I–6.3.3. Radiological characterization of the Triton facility

The artificial radioactivity present in the facility came from two distinct sources:

— The first origin was surface contamination due to contact of the inner surfaces of the Triton pool with contaminated water, or contact of the inner walls of the hot cell with contaminated dust. The contamination was composed of fission products (^{137}Cs, ^{90}Sr) and activation products (^{60}Co, ^{63}Ni).
— The second type was neutron activation inside the concrete radiological shielding surrounding the Triton pile and particularly around and along the five neutron beam ports, due mainly to the neutron streaming effect. The main radionuclides were ^{60}Co, ^{63}Ni, ^{55}Fe, ^{152}Eu and ^{3}H.

The radiological status of the facility was based on:

— A set of measurements carried out at the end of Stage 2 decommissioning in 1987;
— A second set of measurements performed in 1998, which brought more consistency and credibility to the 1987 measurements.

Indeed, following complementary cleanup work in the hot cell, the dose rate in the cell dropped significantly, from 450 μGy/h in 1987 to values lower than 0.1 μGy/h in 1998.

To evaluate the activity still present in the facility, complementary measurements were performed in addition to the 1998 inventory. Hundreds of dose rate measurements, smear tests and direct surface contamination

FIG. I–6.3. Lamas, mobile laboratory unit for site characterization.

measurements were carried out. More than 500 samples and core drillings were taken and analysed.

Moreover, an in situ gamma spectrometry method was developed that made it possible to measure 10 m² of wall surface at a time. It was successfully used to identify the 'major' gamma emitting nuclides present in the surface contamination and to confirm results obtained from direct surface contamination and sampling methods. Dozens of such gamma spectrometry measurements were carried out. Figure I–6.3 shows the Lamas, a mobile laboratory unit ensuring in situ measurements outside or inside facilities. Figure I–6.4 shows a gamma spectrometer, connected to the Lamas, measuring surface and shallow deep gamma activity of internal walls of the Triton pool, with models developed by the CEA.

To evaluate the neutron activation in the radiation shielding around the Triton core, both calculations and measurements were performed.

Calculations were used to obtain a profile of neutron activation distribution inside the radiation shielding but not the absolute values of the specific activities. The calculations were made by experts of the CEA/DEN/SERMA (Reactor Studies and Applied Mathematics Division of CEA), using a model of the Triton pile concrete shielding and iron structure combined with a core model. The TRIPOLI-4 code was used to compute neutron transport and the DARWIN-PEPIN-2 code made it possible to calculate the activities and decay of the activation products.

FIG. I–6.4. Gamma spectrometry of the inner Triton wall.

Dozens of core drillings and samples were taken. The activities of beta and beta–gamma emitters were determined by laboratory methods. Both in the calculation and in the measurements, the predominant activation nuclides were found to be ^{52}Eu, ^{154}Eu, ^{60}Co, ^{63}Ni, ^{55}Fe and ^{3}H.

The calculated relative profile of the activity distribution within the concrete shielding was in good agreement with experimental results. A factor of ten was found between calculation results and measurement results, due to conservatisms in the assumptions used for the calculations (thermal power, time of functioning, concentration of impurities in materials, etc.).

Finally, the radiological characterization was completed. Areas and work pieces, where significant radioactivity remained, were called 'risk points'. These risk points were studied in detail, to prepare the dismantling operations, using the ALARA (as low as reasonably achievable) approach, and in order to define 'specific zonings for dismantling'.

I–6.3.4. Specific zoning for dismantling

The radiological characterization described above made it possible to define the reference zoning of the Triton facility, with a categorization of 'non contaminating area' with 19 risk points, including the Triton pool, the dry pool Naïade, the hot cell, the liquid waste tanks, etc.

In accordance with the reference zoning, all the waste generated by the cleanup and decommissioning of the Triton pool and the hot cell would have been considered as radioactive waste. However, processing of measurement

and historical data showed that the reference zoning could be reviewed in order to reduce radioactive waste streams. Consequently, a 'specific zoning for dismantling' of each part of the facility which contains one or several risk points was defined.

As an example, the specific zoning of the Triton pool revealed that:

(a) Water was responsible for contamination transport inside the Triton pool. As a result, the contaminated areas were located:
    (i) On the internal walls of the pool, through the epoxy resin coating and to a depth of 2 cm inside the concrete;
    (ii) On the floor of the pool, including tiles, seals and cement coating.
(b) The radioactivity induced by neutron activation was located:
    (i) Within a 1 m radius around the five neutron beam ports set in the radiological shielding of the Triton pool;
    (ii) Around the irradiation window, between the Nereïde pile and the Naïade dry pool.

Finally, according to the specific zonings of dismantling, waste generated by the dismantling work was expected to be:

— 1200 m^3 of conventional waste, mainly concrete and scrap metal. This waste was disposed of in municipal and industrial repositories, accepted by the CEA Fontenay-aux-Roses centre.
— 500 m^3 of VLLW, mainly concrete and technological waste. This waste was disposed of at ANDRA's Morvilliers disposal facility for VLLW.

Low level radioactive waste production was not expected and no waste of this category was generated during the decommissioning work

I–6.4. DECOMMISSIONING SCENARIO

The decommissioning scenario was broken down into phases of conventional work and radioactive work.

In the first phase, all the floors, rooms and footbridges around the Triton pool and the hot cell were demolished in order to gain full access around them before demolition (Fig. I–6.5). This phase generated conventional waste (concrete and scrap metal).

In the second phase, the hot cell was cleaned up and partially dismantled. The concrete walls had already been cleaned up in 2002 (see next section).

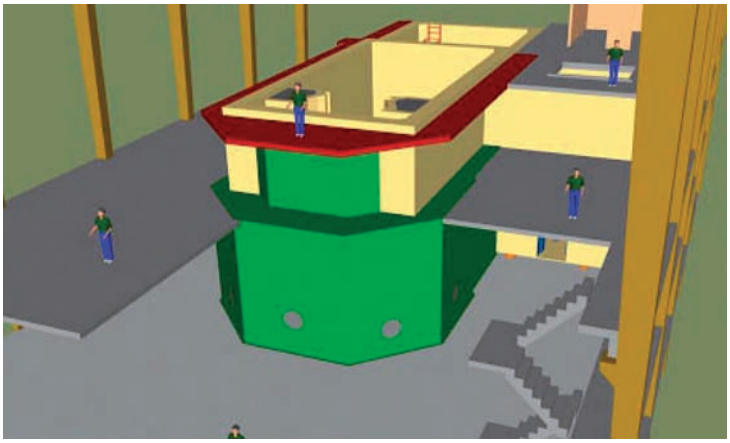

FIG. I–6.5. Floors, rooms and footbridges around the Triton pool and the hot cell.

Finally, the metal inserts and lead glass were removed by cutting the concrete blocks around them (wiresawing). Waste (concrete rubble, metal dust and concrete blocks) was conditioned as VLLW and sent to ANDRA's Morvilliers disposal facility for VLLW.

The next phase consisted of the cleanup of the inner walls of the Triton pool by scarifying and scraping them with hand held tools. The bottom floor of the pool was cleaned up by means of a demolition hammer to remove the tile floor and 5 cm of concrete below. Thereafter, the radiation shielding was cut into 124 blocks by a wiresawing technique, which were conditioned and sent to ANDRA's Morvilliers disposal facility for VLLW.

The fourth phase consisted of the final cleanup of three liquid waste tanks, which had already been cleaned up in 2001 (removal of internal polymer coating), but needed to be scraped (hand held chipping hammer) to remove the residual activity on a 2 cm thickness of concrete.

At the end of the cleanup and removal of all risk points (Fig. I–6.6), the demolition of the remaining concrete structures was then performed by conventional civil engineering techniques (mechanical shovel, pneumatic hammer, etc.).

During all phases related to the cleanup and demolition of risk points, vacuum and shrouding attachments were included in the process. During all the phases related to the cleanup or demolition of radioactive parts of the facility (risk points), the work areas were contained and equipped with absolute filtering systems, in order to avoid cross-contamination of the building.

An important requirement of the project team, imposed on the subcontractor, was that under no circumstances would the demolition of a

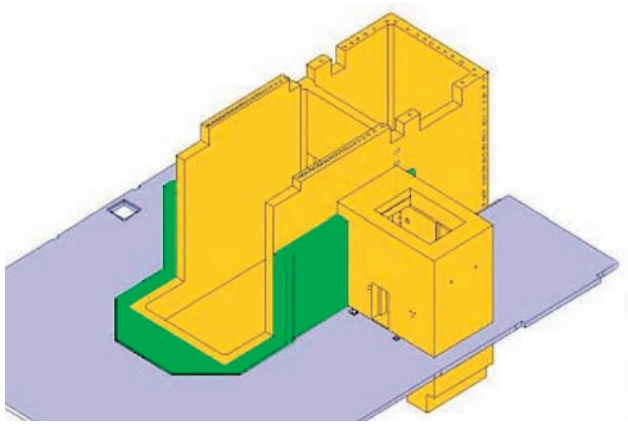

FIG. I–6.6. Remaining civil engineering structures after cleanup and dismantling of all risk points.

conventional area be carried out at the same time as cleanup of a radioactive risk point. All radioactive waste had to be removed from the facility before generation of conventional radioactive waste or vice versa.

Moreover, the facility had to be kept as clean as possible, even during demolition of conventional civil engineering, even when heavy machines and tools were used. To comply with this, the subcontractor planned many cleaning operations every week.

It was decided by the project team to avoid, when possible, techniques generating contaminated liquids that would need to be collected and processed. However, the wiresawing technique required water as a lubricant. Special precautions were taken by the subcontractor to collect effluents during the sawing operations and to transfer them to a processing unit including a decantation unit and a filter press, in state of the art containment conditions.

Special attention was given to performing a risk analysis related to the civil engineering work to be carried out during Stage 3 decommissioning of the Triton facility, considering particularly:

— Lifting and handling of heavy loads (equipment, waste containers, etc.) by means of a top running crane. This required a precise circulation scheme in the working area and continuous surveillance.
— Using civil engineering demolition machines (e.g. 12 t mechanical shovel) requiring surveillance around the machine by an operator to prevent circulation of workers.

— Use of hand held tools, for cleanup operations requiring the use of individual protective equipment (special gloves, dust or filtration masks, safety glasses, earplugs, etc.).

A goal fixed for the project team by the facility's owner was that decommissioning operations should be carried out under QA and traceability from one end of the 'processing' line to the other.

I–6.5. MONITORING OF CONVENTIONAL CONCRETE RUBBLE

As described in the decommissioning scenario, conventional concrete rubble was generated before heavy cleanup operations of the hot cell and the Triton pool, and after cleanup, when demolishing the civil engineering structures.

The following examples describe how monitoring operations were performed on the final phase of the decommissioning of the Triton facility. Concrete rubble was produced by means of a mechanical shovel equipped with a pneumatic hammer or crushers, and concrete rubble was conditioned in a 2 m^3 container. The content of three 2 m^3 containers was gathered in a 7 m^3 open top container to be shipped to the landfill.

Monitoring of concrete rubble was performed in two steps:

— A first step, called first level of control, was carried out by the subcontractor on the civil engineering structures, by means of in situ gamma spectrometry. A 30% relative efficiency HPGe detector was used associated with collimators making it possible to measure average areas of 1 and 9 m^2. Samples were taken every ten surfaces by scrabbling and the samples measured by gamma spectrometry in a low background laboratory, to ensure that the in situ gamma spectrometry measurements were valid.
— A second step, called second level of control, was performed by the radiation protection division of the CEA Fontenay-aux-Roses centre, based on samples and measurements of samples by gamma spectrometry. The sampling methodology is described hereafter.

The content of a 7 m^3 container was considered as one batch, i.e. about 6 m^3 of concrete rubble and total weight of about 8 t.

A sample of about 30 kg was taken (grabbed) from each individual 2 m^3 container. These three individual samples were added, crushed and mixed to constitute the 'main sample' of about 90 kg. A 'measurement sample',

conditioned in three normalized flasks type SG 3000, was taken from the 'main sample' by the 'coning and quartering' method. The total weight of the measurement sample was about 12 kg. The three SG 3000 flasks were measured individually by gamma spectrometry and the average of the three results was compared to the activity levels set by the CEA.

Thus, from a batch of about 8 t of concrete rubble, a main sample of 90 kg (3 × 30 kg) was taken, i.e. 1.1% wt/wt of the batch, and a 'measurement sample' of 12 kg (3 × 4 kg) was performed, i.e. 13% w/w of the main sample and 0.15% wt/wt of the batch.

It is worth mentioning that a third level of control was performed on the 7 m^3 containers by means of high sensitivity scintillation portal monitors, controlling the shipment before it left the CEA Fontenay-aux-Roses centre.

I–6.6. CONCLUSIONS

Triton Stage 2 decommissioning was completed in 1986 and additional decontamination work was performed in 1993. The Stage 3 decommissioning project started in March 2001 and work was completed in November 2004. Comprehensive studies were carried out as well as preliminary dismantling work prior to decommissioning, particularly in order to optimize the volume of VLLW.

Given the low radiological hazards and experience feedback gathered from similar decommissioning projects, due consideration was given to the industrial hazards associated with the use of the techniques selected, and especially to the lifting and handling of heavy loads.

The most important parameters for decision making in the process of choosing a subcontractor was waste minimization, safety arrangements, technical efficiency and cost effectiveness.

The implementation of the specific zoning resulted in a considerable reduction (about 60%) of the amount of VLLW radioactive waste generated.

I–7. POLICY AND REGULATORY FRAMEWORK GOVERNING THE DISPOSITION OF WASTE ARISING FROM NUCLEAR SITE DECOMMISSIONING IN THE UNITED KINGDOM

I–7.1. INTRODUCTION

The United Kingdom policy and regulatory framework governing the approach that can be taken to the disposition of decommissioning wastes has evolved as the issues have evolved. With regard to radioactive waste management, the United Kingdom invokes a regime that requires pre-authorization by the regulators for certain practices. Constraints and limitations to what may be done by a site operator are provided through conditions and limitations attached to site authorizations and registrations. Existing Government policy and regulations were set very much with nuclear operations in mind, and the Government is now considering revising the existing framework to address the new focus on decommissioning as it has proven complex to implement the current policy in that context.

The purpose of this annex is to set out how the regulatory framework applies to decommissioning and waste management practices on a nuclear site, and how this is an important factor in the adoption of certain disposition options for disposal and material reuse or recycling.

I–7.2. THE NUCLEAR DECOMMISSIONING AUTHORITY

The United Kingdom has a large and diverse nuclear industry comprising nuclear power plants of different designs (Magnox, advanced gas-cooled reactor (AGR) and PWR), fuel fabrication and reprocessing facilities, nuclear research and development facilities including research reactors, and defence related facilities.

Several of the early facilities dating from the 1940s, 1950s and 1960s are now closed, and some have begun to be decommissioned. In 2004, the United Kingdom Government established a new organization, the Nuclear Decommissioning Authority (NDA), to take charge of cleaning up the United Kingdom's civil public sector nuclear legacy by means that are safe, secure, cost effective and that safeguard the environment for this and future generations.

On 1 April 2005, the NDA took responsibility for 20 sites around the United Kingdom that represent this nuclear legacy. The locations of these sites are shown in Fig. I–7.1.

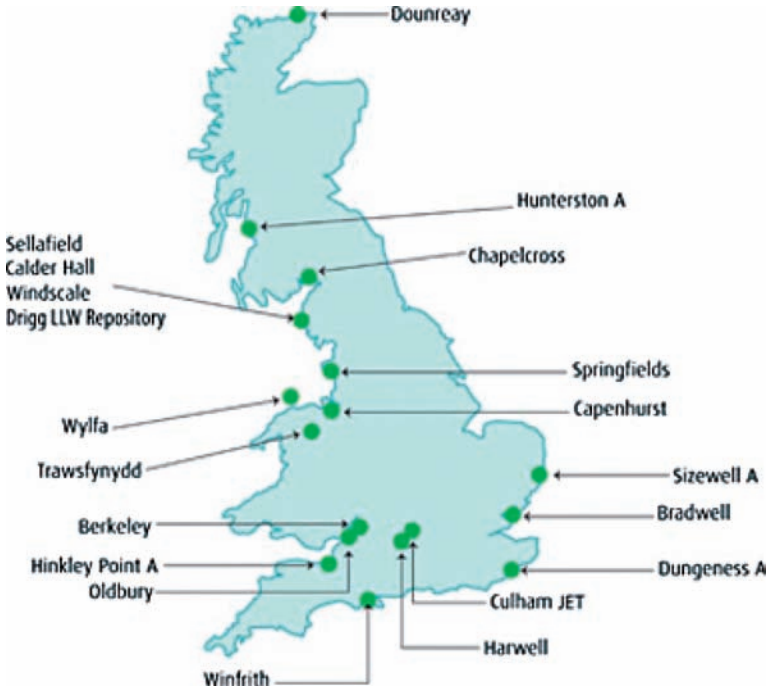

FIG. I–7.1. The locations of the nuclear sites in the United Kingdom under the control of the NDA.

These sites include:

— Those nuclear sites and facilities now operated by the United Kingdom Atomic Energy Authority (UKAEA) and British Nuclear Fuels plc (BNFL), which were developed since the 1940s to support the Government's research programmes into nuclear power, together with the wastes, materials and spent fuels produced by these programmes;
— The fleet of Magnox nuclear power plants designed and built in the 1960s and 1970s that are now operated on the Government's behalf by BNFL, together with facilities at the Sellafield site used for the reprocessing of spent Magnox fuel and all associated wastes and materials.

The NDA does not have responsibility for the 14 AGRs and the single PWR operated by British Energy, nor the nuclear sites and materials managed by the Ministry of Defence (MOD) and other site licensees.

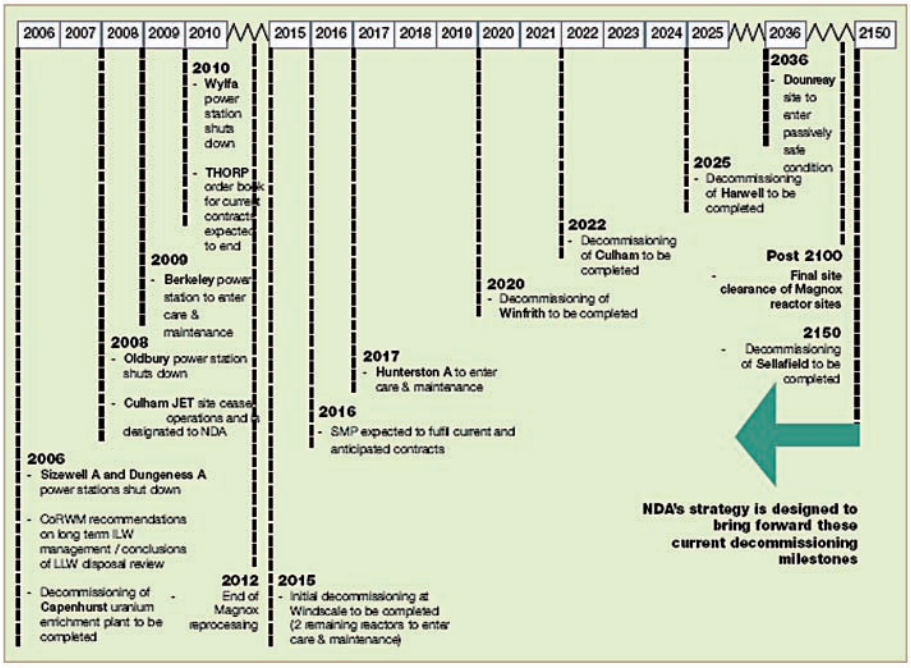

FIG. I–7.2. The programme and milestones proposed by the NDA for decommissioning work in the United Kingdom.

I–7.3. PROGRAMME FOR DECOMMISSIONING IN THE UNITED KINGDOM

During its first year of operation, the NDA reviewed and prioritized the decommissioning and remediation work needed on its sites and, in 2005, published a draft strategy and work plan [I–7.1]. After a period of public consultation, the final strategy was ratified by the United Kingdom Government in March 2006.

The strategy sets out a programme of work stretching for 75 years into the future (Fig. I–7.2).

Key milestones for decommissioning and remediation proposed by the NDA include within five years to have:

— Begun to reduce the high hazards in legacy facilities, especially at Sellafield;
— Determined an approach to ILW interim storage and LLW disposal;
— Defined end-states and agreed timescales for all sites.

And within 25 years to have:

— All 11 Magnox reactor sites cleared and available for alternative uses, subject to long term management arrangements being available;
— Achieved final site clearance at Harwell, Winfrith, Culham and Capenhurst, and possibly Springfields, depending on future commercial opportunities;
— Achieved final site clearance at Dounreay, subject to long term management arrangements for HLW and ILW.

Within 75 years, all plants and facilities at Sellafield would be decommissioned, and all wastes placed in the long term management facilities. Once decommissioning of Sellafield is completed, it is anticipated that the site would remain under institutional control indefinitely.

It is unlikely that a common end state would be achieved for every nuclear site due to the different activities that have taken place. For example, while there is the potential to remove all radioactive waste and contamination from many of the reactor sites, this would be more difficult to achieve for sites such as Sellafield, which produces and stores large quantities of HLW. Largely as a result of historic leakages from legacy facilities at Sellafield (and to a lesser extent some other sites), there is a significant volume of contaminated land to be managed. The full extent of this contamination is not well understood, and the contaminated land issue will be an important factor in determining the final end state and the appropriate remediation strategy.

I–7.4. DECOMMISSIONING OF THE MAGNOX REACTORS

Currently, 7 of the 11 Magnox sites are at various stages of defuelling or decommissioning. During defuelling, the fuel is removed and sent to Sellafield for reprocessing, which removes 99% of the radiological hazard from the site.

Apart from work related to dealing with operational ILW and the arrangements for the disposal of LLW, most of the remaining decommissioning work is concerned with decontamination and dismantling of buildings and other structures. There are at least three approaches to decommissioning Magnox reactor sites after defuelling that may be followed:

— The current approach which is being pursued by BNFL. This assumes 20–25 years of initial decommissioning work, followed by a further 100 years or so in care and maintenance, before final reactor dismantling (Fig. I–7.3). During care and maintenance, the reactor would be managed in a passive

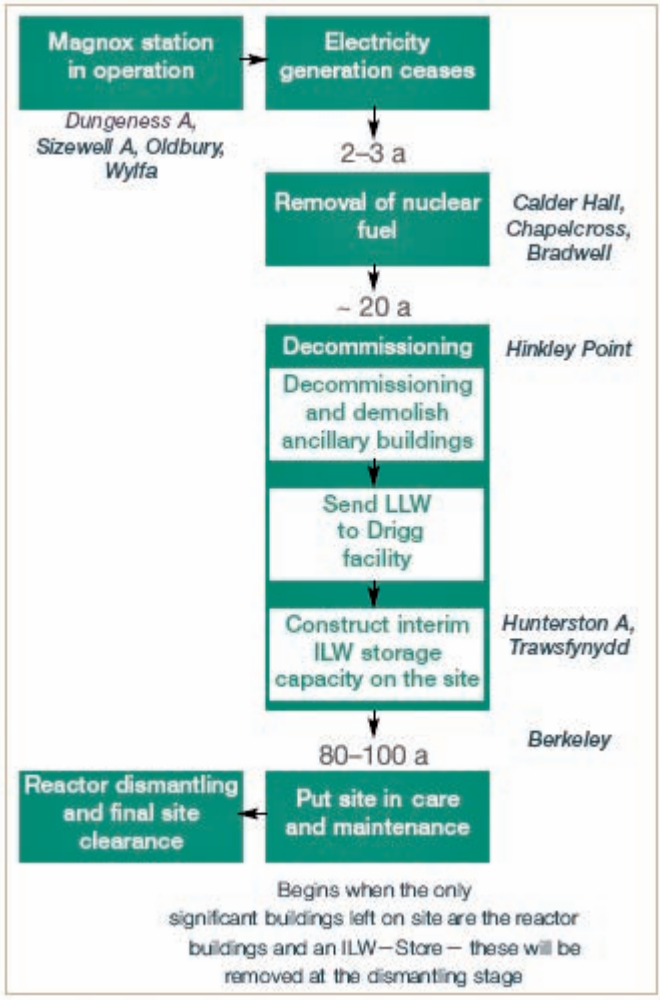

FIG. I–7.3. The current Magnox decommissioning approach.

state, allowing for radioactive decay so that when final dismantling is undertaken activity levels are much reduced and certain components may be declassified.

— BNFL's proposed 'Magnox Innovation' approach which is similar to the current approach but entails a shortened period of initial decommissioning to reach care and maintenance in as little as 5 years, followed by 100 years or so of care and maintenance, before final reactor dismantling.

— The NDA's proposed approach which would entail a radically accelerated decommissioning programme without a period of care and maintenance.

This prompt decommissioning approach may allow for release of the site for alternative use in 25 years or less. This approach means that activity levels would be higher during final reactor dismantling and more active waste would need to be managed but these disadvantages are offset by the following anticipated benefits:

- Better use of the existing knowledgeable workforce and associated socioeconomic benefits for the local area;
- Earlier availability of the site for other uses;
- Fewer ILW interim stores needed around the United Kingdom with consequential cost savings;
- Visible signs of decommissioning and cleanup, including reductions in visual impact;
- A reduced threat of coastal erosion and climate change at a number of coastal sites.

Which decommissioning option will be followed will depend on the Government. Either of these approaches could also be adopted at non-Magnox reactor sites, such as the AGR reactors and the Sizewell PWR reactor operated by British Energy.

Research and prototype reactors on the UKAEA sites are planned to be decommissioned without a period of care and maintenance, in an approach similar to that proposed by the NDA for the power reactors.

I–7.5. WASTE MANAGEMENT TO SUPPORT DECOMMISSIONING

A decommissioning programme is dependent on the availability of suitable stores or disposal facilities to take the waste arisings. In the United Kingdom, the only operating repository is at Drigg, Cumbria which accepts certain conditioned and packaged LLW.

The NDA does not have responsibility for building or operating waste management facilities in the United Kingdom, with the exception of the existing LLW repository at Drigg. Current Government policy on radioactive waste management in the United Kingdom is for the continued temporary storage of HLW and ILW, and continued disposal of LLW to the Drigg repository.

The Drigg repository is operated by BNFL (under contract from the NDA) and accepts the majority of the LLW arising in the United Kingdom. Disposal to Drigg is expensive (disposal costs are around £1800/m^3) and disposal volume is running out. The United Kingdom Radioactive Waste Inventory indicates that about 2 million m^3 of LLW will arise from the NDA's

strategy to decommission the nuclear sites. Contaminated land from Sellafield could add a further 18 million m^3 if it was ever proposed to extract and move this for disposal elsewhere. These volumes compare to the remaining capacity of the Drigg repository of 0.8 million m^3.

In light of this volume constraint, in February 2006, the United Kingdom Government launched a consultation [I–7.2] on the future management of LLW that considers whether a new LLW facility should be built, whether greater consideration should be given to increased use of the other options available for the long term management of LLW (such as reuse and recycling), and whether a new designation of VLLW should be introduced that would allow certain high volume, very low activity decommissioning wastes to be disposed on site.

As part of the United Kingdom Government's ongoing radioactive waste management consultation programme, in 2003 it tasked the Committee for Radioactive Waste Management (CoRWM) with making recommendations for how to manage in the long term HLW and ILW, and certain LLW that cannot be disposed of at the repository at Drigg. CoRWM is due to report on its recommendations by July 2006 and, thereafter, the United Kingdom Government is expected to make a decision on the future policy on management of these wastes, taking into account their recommendations. While the outcome of CoRWM's deliberations are not yet known, the alternatives that CoRWM have been considering are and these include options for long term storage and final disposal in facilities that are either centralized in one or a few places or in many facilities located at or close to the locations where wastes arise.

In summary, the status of radioactive waste management policy in the United Kingdom is in a state of change, and firm decisions are not expected to be made until late 2007. Until such time as a new policy is implemented to build and operate new waste management facilities (stores or repositories), plans cannot be finalized for the decommissioning and remediation of the nuclear sites or for the disposition of decommissioning wastes.

I–7.6. NUCLEAR SITE DECOMMISSIONING AND WASTE MANAGEMENT PLANS

It is a requirement of current Government policy and the terms of the NDA that the operators of nuclear sites establish strategies for the decommissioning of their sites, and plans for the management of decommissioning wastes in the absence of a final disposal route for most radioactive wastes.

The decommissioning strategies are called life cycle base line (LCBL) plans and near term work plans (NTWPs). The LCBLs set out the work required for the decommissioning of the sites in the longer term, whereas the NTWP identifies the detailed work to be done in the next few years. It is the NDA's responsibility to consolidate the overall LCBLs into an overall national plan that will outline the work needed to be undertaken to achieve decommissioning and remediation using best practice and value for money for the tax payer.

The waste management plans are called integrated waste strategies (IWSs) and these are intended to show how a site plans to manage in a safe and coherent manner all wastes they generate, including the large volumes of wastes resulting from decommissioning. An IWS is a plan to ensure that waste management approaches are both optimized and applied consistently across a site to all actual and potential sources of waste, both radioactive and non-radioactive, as well as materials that may become waste in the future. IWSs address what wastes are disposed of, what wastes are required to be stored, as well as waste minimization and treatment issues. Specifically, when formulating an IWS, a site needs to develop policies and strategies, including principles that explain how they will manage their wastes so that:

— Wastes are stored and treated in ways that are consistent with the ALARP principle (i.e. any radiological doses to the public that arise are 'as low as reasonably practicable');
— Wastes are treated and disposed of using methods that represent the best practicable environmental option (BPEO) with associated best practicable means (BPM) abatement and monitoring arrangements;
— Decommissioning plans are prioritized with respect to safety, health and the environment;
— The operator can demonstrate compliance with all relevant regulatory requirements and the waste strategy has been subject to stakeholder consultation.

The current LCBLs and IWSs have been reviewed by the NDA and by the regulatory authorities. Common themes running through many of these reviews were the obvious constraint on site operators to manage the large volumes of radiologically clean and slightly radioactive decommissioning wastes in the face of a potential lack of disposal space at the Drigg repository, and the lack of stores to contain ILW and HLW pending a decision on the long term management of these wastes. It was this observation that is driving the Government's consultation process on radioactive waste management.

I–7.7. UNITED KINGDOM REGULATIONS GOVERNING DECOMMISSIONING WASTE MANAGEMENT

As mentioned earlier, United Kingdom policy for radioactive waste management is in a state of change. The following sections set out the current system of regulation but these may be modified by Government when they complete their review of policy.

The nature of decommissioning works obviously will vary from site to site but, in most cases, will involve either the extensive clean out, refurbishment or demolition of buildings and other facilities, and remediation of the land. Any wastes and materials generated through decommissioning (and ongoing nuclear operations) that are contaminated or activated with radioactivity must be managed in accordance with the requirements of the Nuclear Installations Act 1965 (NIA'65) and disposed of in accordance with the requirements of the Radioactive Substances Act 1993 (RSA'93).

In the United Kingdom, those wastes of relevance to this report are currently categorized as follows:

— LLW: This waste type has a radioactivity content below the lower limit for ILW (4 GBq/t alpha or 12 GBq/t of beta–gamma activity). It arises mainly from contamination of equipment, clothing and cleaning materials during routine operations and maintenance of nuclear facilities, and during decommissioning. The waste can be chemically and materially heterogeneous, and includes a wide range of materials such as metal, soils, building rubble and miscellaneous scrap. There is no formal lower threshold for LLW, but in practice many would regard it to be 0.4 Bq/g which is the level laid down in the substances of low activity (SoLA) exemption order issued under RSA'93 (see below).
— VLLW: This waste is a subset of LLW and is uniquely defined in terms of activity and volume because it is intended to cover small volumes of low activity wastes that may be disposed of with ordinary refuse. It is defined as each 0.1 m^3 containing less than 400 kBq of beta–gamma activity or single items containing less than 40 kBq of beta–gamma activity. The key constraint here is that VLLW is designated for only very small volumes of material that would be disposed of along with large volumes of non-radioactive domestic wastes. As currently defined, the VLLW category cannot be applied to large volumes of decommissioning wastes. The ongoing Government consultation on LLW is addressing the future designation and treatment of high volume, very low activity wastes and is proposing to withdraw the volume constraint on VLLW.

The management of all radioactive waste arising on a nuclear site will be subject to the requirements of NIA'65 and any disposal of radioactive waste from nuclear licensed sites requires prior authorization under RSA'93 unless it can be demonstrated to the satisfaction of the environmental regulators that the wastes are radiologically clean or excluded. Determining whether a waste may be released from further controls under RSA'93 is on the basis of whether the activity is below exclusion or exemption levels. This concept is widely referred to as free release but a better term is controlled release because it reflects the strict regulatory controls governing the process. There are a number of terms that are used in relation to controlled release which are important in the context of the potential reuse and recycling of waste materials. Their meanings in the United Kingdom regulatory context are defined as:

— Clean: An article or substance which has had no reasonable potential to have become contaminated or activated, or upon or within which no radioactivity other than normal background is detectable when suitable comprehensive measurement (monitoring and sampling) is practicable and has been undertaken.
— Clearance: The process to confirm that an article or substance is clean (free from radioactivity), or excluded or exempt from further control under all relevant legislation on the basis of its radioactivity.
— Excluded: An article or substance that is not radioactive under RSA'93 and not subject to any control under the Act because it does not contain levels of any of the specified radioelements above the limits in Schedule 1 of RSA'93 (see Table I–7.1) or any non-specified radioelements at levels above normal backgrounds.
— RSA exempt: An article or substance that is radioactive or contaminated under RSA'93 because it contains levels of specified radioelements above RSA'93 Schedule 1 exclusion limits or because it contains other radioelements wholly or partly attributable to either an artificial process or as a result of the disposal of radioactive waste, but in both cases at levels below relevant limits in exemption orders under the Act. An RSA'93 exempt article or substance may be subject to control as radioactive under other legislation. The two main exemption orders that are likely to apply to decommissioning wastes are:
  • The substances of low activity exemption order 1986 (the SoLA Exemption Order) which exempts waste that has an activity that does not exceed 0.4 Bq/g and is substantially insoluble in water;
  • The Radioactive Substances (Phosphatic and Rare Earths, etc.) Exemption Order 1962 (the PSRE Exemption Order) which exempts material that is radioactive solely because of the presence of one or

TABLE I–7.1. SCHEDULE 1 FROM RSA'93. THE SPECIFIED ELEMENTS AND THEIR LIMITING SPECIFIC ACTIVITIES

Element	Activity (Bq/g)		
	Solid	Liquid	Gas or vapour
Actinium	0.37	7.40×10^{-2}	2.59×10^{-6}
Lead	0.74	3.70×10^{-3}	1.11×10^{-4}
Polonium	0.37	2.59×10^{-2}	2.22×10^{-4}
Protoactinium	0.37	3.33×10^{-2}	1.11×10^{-6}
Radium	0.37	3.70×10^{-4}	3.70×10^{-5}
Radon	—	—	3.70×10^{-2}
Thorium	2.59	3.70×10^{-2}	2.22×10^{-2}
Uranium	11.1	0.74	7.40×10^{-5}

more of the Schedule 1 elements and is substantially insoluble in water provided that the specific activity of each of the Schedule 1 elements present does not exceed 14.8 Bq/g (expressed in the Exemption Order as 4×10^{-4} mCi/g).

Both exemption orders allow wastes meeting these criteria to be disposed of without prior authorization from the relevant environmental regulator — the Environment Agency in England and Wales; and the Scottish Environmental Protection Agency (SEPA) in Scotland. Disposal can also be taken to mean reuse and recycling in the non-nuclear sector. Such potential reuse, recycling or disposal will, however, be subject to regulation as a conventional waste which may restrict some options for reuse or disposal on the basis of their material or chemical properties (see below).

Before any decommissioning waste could be reused or recycled for use either on or off a nuclear site, appropriate demonstrations need to be made to the regulators that it is either radiologically clean or that its levels of radioactivity are appropriate for it to be classed as RSA exempt or excluded. An industry code of practice on clearance and exemption [I–7.3] has been promulgated that is likely to be adequate when making demonstrations to regulators in support of waste management proposals.

Exemption does not mean that wastes are no longer designated as radioactive, and this causes some confusion and practical problems for waste managers. For example, exempt wastes cannot strictly be disposed of to a conventional waste landfill if that landfill has waste acceptance criteria that

forbid the acceptance of radioactive wastes. Similarly, there are differences between the Environment Agency and SEPA over the application of exemption orders to decay storage. SEPA does allow the use of decay storage to exempt waste from the requirement for authorization. This approach is based on a legal interpretation of RSA'93 and views of the Scottish Executive. Put simply, the interpretation of radioactive waste in Scotland is that once a substance has been declared as a waste it must always be treated as a waste, which is not the case in England and Wales.

I–7.8. DISPOSAL, REUSE AND RECYCLING OF CLEAN AND EXEMPT WASTE

Radiologically clean and RSA'93 exempt wastes may be disposed of, reused or recycled in the same manner as conventional municipal and industrial wastes without further control under RSA'93 but they may remain subject to control under the Environmental Protection Act 1990 and associated regulations depending, in part, on their physical and chemical properties, their potential for causing harm to the environment and their proposed method of disposition. The relevant regulations in terms of the reuse and recycling of decommissioning wastes are the Waste Management Licensing Regulations, Pollution Prevention Control Regulations and Landfill Regulations.

An important issue is that material is defined as a waste under the Waste Framework Directive (75/442/EEC as amended by 91/156/EEC) as any substance or object that the holder discards, intends to discard or is required to discard and this will include exempt decommissioning wastes. As a result of European and national case law over the last few years, the circumstances under which a substance or object may be said to have been discarded have broadened considerably. Furthermore, it is considered that once a substance or object has become waste, it will remain waste until it has been fully recovered and it no longer poses a potential threat to the environment or human health.

If the proposed disposition option is disposal as a non-radioactive waste, then charges will apply in accordance with the landfill tax, which for chemically inert materials is at a rate of £2/t and for reactive and hazardous materials is £13/t but subject to annual increases. There is an ongoing debate concerning the use of decommissioning materials for site landscaping at final site clearance. Certain authorities consider this use to be de facto landfilling and, therefore, subject to landfill tax whilst other authorities do not. This is an issue that requires clarification.

If the proposed disposition option is uncontrolled reuse and recycling, then the waste manager or recycler needs to demonstrate that their product has

been fully recovered and is no longer a waste. For the most commonly reused decommissioning material (concrete), the waste and resources action programme (WRAP) has developed a quality protocol [I–7.4] for the production of aggregates from inert waste that addresses some of the difficulties in the interpretation and application of the Waste Framework Directive. The purpose of the quality protocol is to provide a uniform control process for producers from which they can reasonably state and demonstrate that their product has been fully recovered and is no longer a waste.

In general, the clean and exempt decommissioning wastes arising from a nuclear site are similar in material characteristics to those wastes that arise from any other construction or demolition project and, therefore, the potential applications to which these wastes may be put are essentially the same.

Various services are available to waste producers and recyclers to ensure their products are processed to achieve appropriate quality standards. AggRegain is a free web based sustainable aggregates information service provided by the WRAP aggregates programme [I–7.4]. It is designed to assist with the specification of recycled and secondary aggregates, and is an important input to the development of plans for the sustainable reuse and recycling of decommissioning wastes from nuclear sites.

I–7.9. SUMMARY

The United Kingdom policy and regulatory framework governing the approach that can be taken to the disposition of decommissioning wastes has evolved as the issues have evolved. Existing Government policy and regulations were set very much with nuclear operations in mind, and Government is now considering revising the existing framework to address the new focus on decommissioning as it has proven complex to implement the current policy in that context.

Uncertainty over future waste management policy is a potential hindrance to waste managers on decommissioning sites, and is a major factor that needs to be considered when making decisions on waste disposition. Waste management plans currently being developed by the sites have to be sufficiently flexible to allow decommissioning work to continue but, at the same time, keep options open for whatever future waste management policy the Government decides to implement.

Under the current framework, the majority of radioactive waste arising from decommissioning is designated as LLW and can only be disposed of to the operating LLW repository at Drigg. An extension to Drigg or an entirely new facility (or facilities) will be necessary to cope with the volumes of waste that

are anticipated to arise in future. It is not possible to dispose of large volumes of VLLW in the United Kingdom under the current regulations but Government proposals for future policy may allow VLLW to be managed differently from the remainder of LLW, subject to controls commensurate with the radiological risk the waste presents. CoRWM and the NDA are both assessing options for the on site disposal of LLW (and VLLW) as an alternative to centralized disposal at Drigg or its replacement.

There is a permissive regime for the exemption of certain radioactive wastes from regulatory control that allows them to be disposed of, reused or recycled without further intervention under the Radioactive Substances Act. Demonstrating that a waste material meets the exemption criteria remains difficult, and this process is made more complicated by differing interpretations of the exemption orders by the Environment Agency and SEPA. The industry code of practice on clearance and exemption is intended to help waste managers demonstrate that exemption criteria have been met to the satisfaction of the regulators.

Exempt materials are, however, subject to control under the strict regulations that apply to conventional industrial wastes, and these may place further restrictions on disposition options. Exempt solid wastes that cannot be reused or recycled will need to be disposed of to conventional or hazardous waste landfills (subject to the relevant waste acceptance criteria) and will be subject to landfill tax.

To allow exempt solid wastes to be unconditionally reused or recycled, it will be necessary to demonstrate that the proposed product has been fully recovered and is no longer a waste as defined in law. For the reuse and recycling of concrete, there is a quality protocol that simplifies this process.

The appropriate application of the industry code of practice on clearance and exemption, and the quality protocol will enable decommissioning waste managers to reuse and recycling greater proportions of inert solid wastes than have been achieved to date. Anticipated changes to the policy and regulatory framework should enable a greater range of disposition options to become available to waste managers in the future.

REFERENCES TO ANNEX I–7

[I–7.1] NUCLEAR DECOMMISSIONING AUTHORITY, Strategy (Draft for Consultation and Approved Versions) (2005), www.nda.gov.uk

[I–7.2] DEPARTMENT FOR ENVIRONMENT, FOOD AND RURAL AFFAIRS, A public consultation on policy for the long term management of solid low level radioactive waste in the United Kingdom, 28 February (2006).

[I–7.3] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Clearance and Exemption Principles, Processes and Practices for Use by the Nuclear Industry. A Nuclear Industry Code of Practice (2005), www.ukaea.org.uk/reports/generalpdf

[I–7.4] WASTE & RESOURCES ACTION PROGRAMME (WRAP), The Quality Protocol for the Production of Aggregates from Inert Waste, revised edition (2005), http://www.wrap.org.uk/downloads/0083_Quality_Protocol_A4.8b89bbd0.pdf

I–8. DISPOSAL OF VLLW AT THE GRAND VIEW, IDAHO, HAZARDOUS WASTE SITE, USA

I–8.1. INTRODUCTION

Historically, almost all LLW in the USA was disposed of at facilities licensed under Nuclear Regulatory Commission (NRC) regulations or at US Department of Energy disposal facilities developed and operated exclusively for nuclear weapons-related waste. Three NRC regulated facilities currently serve the nation. These are located in Barnwell, South Carolina; Clive, Utah; and Richland, Washington. Access to these facilities is limited either by special fees, geographic service area restrictions or radionuclide concentration limits. These restrictions reduce the options practically available to waste producers. Under these conditions, disposal prices have increased significantly and unused capacity has dwindled.

In 2005, the NRC indicated that work on a proposed clearance rule for unrestricted release was indefinitely delayed due to higher priorities. A prior Below Regulatory Concern proposal was withdrawn in 1993 in response to negative public comment including concerns of the metal recycling industry. As a result, a clearance standard is not expected to represent a significant near term alternative in the USA.

The USA is presently served by 18 hazardous waste disposal facilities permitted under Environmental Protection Agency (EPA) regulations implementing Subtitle C of the Resource Conservation and Recovery Act (RCRA). Wastes regulated under these regulations are typically produced by chemical and other manufacturing industries, steel mills, petroleum refineries, electric utilities, electroplaters and pharmaceutical producers as well as waste from past practice remediation projects.

In contrast with the low level radioactive waste situation, hazardous waste disposal capacity exceeds demand, prices have generally been stable and substantial unused disposal capacity is readily available nationally. Like NRC regulated facilities, however, hazardous waste sites are also subject to site selection and design requirements, and are extensively regulated once in operation.

In response to market demand, hazardous waste facilities in the western states of Colorado, Idaho and Texas have recently modified their permits to allow co-disposal of very low activity radioactive waste along with hazardous

waste. Acceptance criteria for this very low activity waste are specified in each facility's specific operating permit, rather than by national regulations.[2]

These specifically permitted hazardous waste sites offer a cost effective disposal option for large volumes of contaminated soil, concrete rubble, building debris and rare earth processing residue as well as certain consumer products and accelerator produced wastes. Such wastes may or may not be NRC regulated at the point of generation. Diffuse, naturally occurring and accelerator produced radioactive material (NARM) is not subject to NRC regulation.[3]

If a waste stream is regulated by the NRC, it must first be exempted from requirements to utilize an NRC regulated disposal site before being shipped to a hazardous waste site. General NRC exemptions apply to various consumer products (e.g. smoke detectors, electron tubes, watches, compasses, etc.), magnesium thorium aircraft parts and other items. In addition, NRC can specifically exempt an individual generator waste stream following a case-specific safety evaluation. This is done through an alternate disposal authorization and related exemption decision.[4]

This annex provides a case study of the Grand View, Idaho, hazardous waste site's experience obtaining permit approvals and disposing of very low activity radioactive waste. To date, the Idaho facility has accepted more than 1.3 million t of low activity material. While rare earth processors and other industry facilities have utilized the Grand View site for low activity waste, most waste has been shipped from federal government remediation projects involving large volumes of contaminated soil and debris.

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[2] The Environmental Protection Agency issued an Advance Notice of Proposed Rulemaking on Approaches to an Integrated Framework for Management and Disposal of Low-Activity Radioactive Waste (Federal Register Vol. 68, No. 222, November 18, 2003) seeking comment on standards for disposal of such waste at hazardous waste sites and is evaluating comments received.

[3] Discrete NARM (e.g. radium sources) is now regulated as by-product material by the NRC in accordance with the Energy Policy Act of 2005 (Public Law 109-58).

[4] Title 10 Code of Federal Regulations Part 20.2002 regarding alternate disposal authorizations and Parts 30.11. 40.14 and 70.17 regarding source, by-product and special nuclear material exemptions.

FIG. I–8.1. General view of the Grand View facility.

I–8.2. GRAND VIEW, IDAHO, HAZARDOUS WASTE FACILITY EXPERIENCE

I–8.2.1. Location and physical setting

The Idaho hazardous waste facility is located on a remote 5.3 km^2 site in the Owyhee Desert of southwestern Idaho, 18 km west of the small town of Grand View (Fig. I–8.1). The property is owned by the facility operator, US Ecology Idaho, Inc., a subsidiary of American Ecology Corporation which also operates the Richland, Washington low level radioactive waste facility discussed above.

Average annual total precipitation in the area is 180 mm. Mean pan evaporation is approximately 1.35 m. A deep geothermal aquifer (77°C) lies 900 m beneath the sedimentary alluvial soils present at the site. Groundwater is also present within fine-grained sand beds and interbedded silts 43–107 m below ground level. Wells in this upper saturated zone yield less than 1.9 L/min. A thick layer of natural clay occurs at a depth below the upper saturated zone.

I–8.2.2. Hazardous waste disposal unit design

Hazardous waste disposal facilities in the USA must meet minimum design standards set by the EPA.[5] These standards require a triple liner system to form an impermeable barrier beneath the buried wastes as well as a leachate collection system to remove accumulated liquids.

The Grand View facility liner system is comprised of two synthetic liners constructed of high density polyethylene underlain by an approximately 1 m thick natural clay liner. The clay liner is constructed of materials excavated at the site. Two synthetic geonet layers allow free drainage to a sump at the low end of the disposal unit where leachate is pumped to collection tanks. The leachate is routed through carbon canisters to remove organic constituents and is then piped to lined surface impoundments for evaporation. The site's arid setting and a prohibition on liquid waste disposal result in very low leachate production.

The EPA also prescribes the required cover system. This generally consists of a synthetic liner and a compacted clay layer covered with about 1 m of soil to promote re-vegetation. The EPA can approve alternate cover systems, however, and the Grand View facility is completing an evapotranspiration cover system pilot project to take advantage of the site's favourable soil characteristics and semi-arid environment. The facility permit also requires that buried radioactive materials be placed approximately 4 m beneath the top of the cover to control radon emanation.

The combination of waste form restrictions (no liquids), cap design and low rainfall climate conditions protect against accumulation of liquids and the so-called 'bath tub effect' which had produced problems at humid disposal sites in the eastern USA (Fig. I–8.2).

I–8.2.3. Regulatory framework for low activity radioactive waste disposal

The EPA has delegated to the State of Idaho the authority to regulate RCRA wastes. The assigned regulatory agency within the state is the Idaho Department of Environmental Quality (IDEQ). The Grand View site was historically permitted by IDEQ to accept small quantities of naturally occurring radioactive material. In 2000, IDEQ authorized the facility to accept wastes containing source material in concentrations less than 0.05% by weight. This material is subject to general exemption from NRC regulation.

[5] Title 40 Code of Federal Regulations Part 264, Subpart N — Landfills.

FIG. I–8.2. Newly constructed lined disposal trench.

With this authorization, the facility began receiving significant amounts of low activity waste from remediation of former Manhattan Project uranium and thorium ore processing sites under a contract with the federal government. In response to this increased disposal activity, the State of Idaho enacted a law in 2001 requiring IDEQ to develop specific regulations for disposal of radioactive materials at hazardous waste sites such as the Grand View facility. These regulations were issued by the IDEQ in late 2001 following a public hearing and comment period on a draft.[6]

The IDEQ regulations provide radiation protection standards and other requirements for acceptance of non-NRC regulated radioactive waste at permitted hazardous waste facilities, but prohibit disposal of such waste at municipal solid waste landfills.[7] General radiation protection standards include the following:

[6] Idaho Administrative Code 58.01.10 Rules Regulating the Disposal of Radioactive Materials.

[7] Other states, including Michigan and Tennessee, permit disposal of specified low activity waste at industrial waste landfills regulated under Subtitle D of RCRA.

— Total effective dose equivalent (TEDE) during operations cannot exceed 1 mSv/a to any member of the public;
— For unrestricted post-closure use, TEDE to the reasonably maximally exposed individual may not exceed 0.15 mSv/a excluding natural background.

The rule also requires each facility to obtain the following site specific permit approvals:

— Waste acceptance criteria defining the specific wastes that may be received;
— A closure plan providing reasonable assurance that the radon emanation rate from the closed disposal unit will not exceed 20 pCi (0.74 Bq)·m^{-2}·s^{-1} averaged across the area of the disposal unit;
— An environmental monitoring programme for air, ground water, surface water and soil sufficient to demonstrate that no member of the public is likely to receive a dose in excess of 1 mSv/a.

Shortly after the new regulation was issued, the facility operator submitted an application to IDEQ to modify its hazardous waste permit along with a safety assessment documenting conformance with the above requirements. This modification request was approved by IDEQ in December 2001 following a public hearing and comment period.

In 2005, the permit was again modified after a public hearing and comment period to allow the disposal of fission and activation products in very low concentrations that have been specifically exempted by the NRC if also approved by IDEQ. This review process requires the facility operator to submit documentation to IDEQ that the NRC has exempted a particular waste stream along with the operator's safety assessment demonstrating that the required radiation protection standards will not be exceeded. IDEQ then performs a concurrence review of the NRC exemption determination and the operator's findings.

I–8.2.4. Safety assessment, monitoring and waste acceptance criteria

The RESRAD programme, a public domain code accepted by the EPA and NRC, was employed by the facility operator for the initial 2001 safety assessment. For the 2005 permit modification, the RESRAD default parameters were changed to use site specific geology and hydrology information that more accurately reflect local conditions.

Low activity wastes are received in two distinct physical configurations. Most of the waste is diffuse contaminated soils, rubble and debris. These wastes are identified in Tables A.8.1 and A.8.2 of the Waste Acceptance Criteria included in the appendix. The second category is accelerator produced material and NRC exempt waste, identified in Tables A.8.3 and A.8.4 of the appendix. These are usually discrete items or devices packaged in non-bulk (metal drums) or other containers.

The concentrations of wastes that may be received are driven by dose constraints. Potential external dose is most important during operations, while potential internal dose drives the criteria for the post-closure period. The occupational monitoring programme includes monitoring air for radionuclides attached to dust particles and radon. Calculations based on maximum concentrations allowed by the waste acceptance criteria and actual measurement of airborne radionuclide concentrations demonstrate that inhalation of airborne radionuclides is not a significant contributor to dose during operations.

I–8.2.4.1. Operational dose

Limits for external dose resulting from transportation and handling during disposal are dictated by the NRC's exemption policy. In general, the NRC requires waste generators seeking an exemption to demonstrate that doses to individual workers will not exceed a few tens of microsieverts per year with a maximum of 50 μSv/a. Dose estimates demonstrating how these limits will be achieved must be submitted and approved on a case by case basis prior to authorization being given.

At the Grand View site, external doses are maintained at a low level by an overriding concentration limit of 70 Bq/g for the sum of the concentrations of all diffuse radionuclides present in a single container of waste.

Doses from NRC exempt items and accelerator produced materials are controlled by a limit on contact dose rate of 0.01 mSv/h for containers. As these wastes are predominantly disposed of directly in the landfill and are received infrequently, contribution to external dose is insignificant. These materials are additionally required to be packaged and are disposed of in their packages to avoid potential inhalation.

Air sampling is conducted in office areas at the site to assess potential inhalation doses to non-occupationally exposed individuals. The air filters are analysed for the primary radionuclides received: ^{232}Th, ^{234,235,238}U and ^{226}Ra. Recently, with the approval of the 2005 permit modification, ^{241}Am and ^{137}Cs were added. Analyses of the filters consistently demonstrate airborne

radionuclide concentrations from slightly greater than one to three orders of magnitude below allowable concentrations for the general public.

Potential dose to a member of the general public is also calculated annually using the Capp88pc model. An annual emission rate is calculated based on the mass of diffuse waste received each year and the average radionuclide concentration in that total waste mass. The model predicts a potential dose at a hypothetical receptor located 1 m from the facility in the predominant downwind direction. These potential doses have also consistently been several orders of magnitude below the EPA's limit.

Ambient gamma radiation levels are also measured quarterly along the periphery of the facility just inside the security fence. The highest reading noted each quarter is approximately one-fifth the maximum allowable dose of 1 mSv/a.

I–8.2.4.2. Post-closure dose

As noted above, IDEQ rules require that the operator demonstrate that the maximum annual dose to a potential resident on site does not exceed 0.15 mSv within 1000 years of closure. With the exception of the 1 m natural clay liner beneath the disposal units, none of the artificial barriers used in the disposal unit design are credited in the post-closure dose evaluation. Dose contributors are radon and water dependent pathways. Major factors in assessing the magnitude of the post-closure dose are the physicochemical nature of the waste and the site's hydrogeological setting including annual precipitation and pan evaporation rate.

Compliance with the dose limit must be demonstrated at the first contacted ground water, the saturated interbedded silt and clay zone. Based on the upgraded 2005 RESRAD analysis, the time of transit for precipitation to this saturated zone is approximately 220 years. This transit time provides for significant decay of radionuclides with short or intermediate half-lives. The major contributor to post-closure dose is from the water dependent pathways, predominantly from ^{14}C, ^{99}Tc and ^{129}I.

Doses from discrete accelerator produced radionuclides and NRC exempt waste are evaluated separately from the diffuse, bulk waste materials. Many accelerator produced radionuclides are not represented in safety assessment models because they do not have sufficiently long half-lives to warrant concern for long term dose and/or because the models were constructed with an emphasis on reactor-produced nuclides. Also, numerous NRC exempt items (Table I–8.4) can contain higher concentrations of the same radionuclides, principally uranium and thorium, in diffuse waste streams (Table I–8.1). Consequently, a separate analysis using the approved RESRAD

model is made for all accelerator produced materials and exempt items received annually. The results of theses analyses demonstrate a negligible contribution from these materials to post-closure dose.

I–8.2.5. Socioeconomic factors

The Grand View hazardous waste facility is the largest property tax payer in rural Owyhee County. In addition to job production and local spending, the facility also paid the State of Idaho waste tipping fees that totalled approximately $3 million in 2005. Of this amount, 5% is paid to Owyhee County. The facility operator also provides discretionary local community grants of approximately $25 000 to schools, libraries, senior citizen centres and civic organizations.

In addition to the Grand View operation, the disposal facility's parent company American Ecology is based in the state capitol of Boise, Idaho. An independent economist issued a report in 2006 estimating the company's total direct and indirect annual economic impact to the state at $51 million.[8]

The facility operator regularly provides information on its activities to State and Owyhee County elected officials, and seeks input on potential local community concerns. To provide an adequate buffer zone surrounding the facility, the property of several former neighbours were acquired and a land trade was completed with the Federal Government. Expansion of the buffer zone also produces long term protection benefits by eliminating potential doses to the nearest members of the general public and water supply well.

As a result of this proactive approach, no negative public comments were received at a well attended 2005 public hearing on the operator's application to expand the Grand View facility waste acceptance criteria to include exempt fission and activation products. Conversely, a large number of citizens provided written comments and spoke at the hearing in support of the application, which was subsequently approved by IDEQ.

Finally, it is the facility operator's experience that achieving and maintaining public acceptance for expanded waste acceptance criteria at a long existing facility is more likely to be successful than establishing a new facility.

[8] READING, D., An Analysis of the Economic and Fiscal Impacts of American Ecology Corporation's Idaho Operations, Ben Johnson Associates, Inc. (February 2006).

I–8.2.6. Conclusions

The use of hazardous waste disposal facilities in the USA to dispose of very low activity radioactive material is providing a cost effective alternative to use of more expensive NRC regulated disposal facilities. This stringently regulated alternative also provides a more substantial level of protection and public transparency than clearance approaches based on unrestricted release. In addition, the remaining disposal capacity at the nation's three existing NRC regulated disposal sites in South Carolina, Utah and Washington is preserved for higher concentration wastes that require the heightened level of protection provided by these facilities.

The NRC has recognized these advantages and recently clarified its policy for case by case alternate disposal authorizations and exemptions to allow specified wastes to take advantage of the hazardous waste facility option. The EPA has also recognized these advantages and is evaluating potential rulemaking to establish standardized regulations.

To date, the only hazardous waste sites accepting low activity waste are located in western states with low rainfall and high evaporation rates. The successful Grand View, Idaho facility experience offers a useful case study on actions taken by regulatory agencies and the private sector facility operator to develop and expand this alternative.

Appendix to Annex I–8

GRAND VIEW, IDAHO LOW ACTIVITY WASTE ACCEPTANCE CRITERIA

TABLE I–8.1. UNIMPORTANT QUANTITIES OF SOURCE MATERIAL UNIFORMLY DISPERSED[a] IN SOIL OR OTHER MEDIA[b]

Status of equilibrium	Maximum concentration of source material	Sum of concentrations parent(s) and all progeny present[c]
Natural uranium in equilibrium with progeny	422 ppm / 141 pCi/g	≤2000 pCi/g
Refined natural uranium (^{238}U,^{235}U,^{234}U; ^{234}Th, ^{234m}Pa)	500 ppm / 333 pCi/g	
Depleted uranium (^{234}Th, ^{234m} Pa)	500 ppm / 169 pCi/g	
Natural thorium (^{232}Th- + ^{228}Th)	500 ppm / 110 pCi/g	
^{230}Th in equilibrium with progeny	0.01 ppm / 200 pCi/g	≤2000 pCi/g
^{230} Th (with no progeny)	0.1 ppm / ≤2000 pCi/g	
Any mixture of thorium and uranium	Sum of ratios ≤1[d]	≤2000 pCi/g

Note: 1 pCi = 0.037 Bq.

[a] Average over conveyance or container. The use of the phrase 'over the conveyance or container' is meant to reflect the variability on the generator side. The concentration limit is the primary acceptance criteria.

[b] Other media do not include radioactively contaminated liquid (except for incidental liquids in materials).

[c] Diffuse waste with a total concentration (sum of concentrations of all radionuclides present) which is 2000 pCi/g or less may be accepted at the site (i.e. the controlling limit is 2000 pCi/g).

[d] $$\frac{\text{Conc. of U in sample}}{\text{Allowable conc. of U}} + \frac{\text{Conc. of Th in sample}}{\text{Allowable conc. of Th}} \leq 1$$

TABLE I–8.2. NATURALLY OCCURRING RADIOACTIVE MATERIAL OTHER THAN URANIUM AND THORIUM UNIFORMLY DISPERSED[a] IN SOIL OR OTHER MEDIA[b]

Status of equilibrium	Maximum concentration of parent nuclide	Sum of concentrations of parent and all progeny present[c]
^{226}Ra or ^{228}Ra with progeny	222 pCi/g	≤2000 pCi/g
^{210}Pb with progeny (Bi and ^{210}Po)	666 pCi/g	≤2000 pCi/g
Any other NORM		≤2000 pCi/g

Note: 1 pCi = 0.037 Bq.

[a] Average over conveyance or container. The use of the phrase 'over the conveyance or container' is meant to reflect the variability on the generator side. The concentration limit is the primary acceptance criteria.

[b] Other media do not include radioactively contaminated liquid (except for incidental liquids in materials).

[c] Diffuse waste with a total concentration (sum of concentrations of all radionuclides present) which is 2000 pCi/g or less may be accepted at the site (i.e. the controlling limit is 2000 pCi/g).

TABLE I–8.3. ACCELERATOR PRODUCED RADIOACTIVE MATERIAL

Acceptable material	Activity or concentration
Any accelerator produced radionuclide, the half-life of which is ≤3 years. Longer half-life materials may only be accepted based on IDEQ review and approval of a specific proposal.	All materials shall be packaged in accordance with USDOT packaging requirements. Any packages containing iodine isotopes or volatile radionuclides will have lids or covers sealed to the container with gaskets. Contamination levels on the surface of the packages shall not exceed those allowed at point of receipt by USDOT rules. Gamma or X ray radiation levels may not exceed 10 mrem/h (0.1 mSv/h) anywhere on the surface of the package. All packages received shall be directly disposed of in the active cell. All containers shall be certified to be 90% full.

TABLE I–8.4. NRC EXEMPTED PRODUCTS, DEVICES OR ITEMS

Exemption 10 CFR Part	Product, device or item	Isotope, activity or concentration
30.15	Timepieces, lock illuminators, balances, auto shift quadrants, marine compasses, thermostat dials and pointers, internal and external calibration sources for radiation measurement devices, spark gap irradiators	Various isotopes and activities as set forth in 30.15
30.16	Resins containing ^{46}Sc for sand consolidation in oil wells	Activity by manufacturing license. Surface radiation level must not exceed 10 mrem/h (0.1 mSv/h)
30.19	Self-luminous products containing tritium, ^{85}Kr, ^{3}H or ^{147}Pm	Activity by manufacturing license
30.20	Gas and aerosol detectors for protection of life and property from fire	Isotope and activity by manufacturing license
30.21	Capsules containing ^{14}C urea for in vivo diagnosis of humans	^{14}C, 1 μCi per capsule
40.13(a)	Unimportant quantity of source material: see table above	≤0.05% by weight source material
40.13(b)	Unrefined and unprocessed ore containing source material	As set forth in rule
40.13(c)(1)	Source material in incandescent gas mantles, vacuum tubes, welding rods, electric lamps for illumination	Thorium and uranium, various amounts or concentrations, see rules
40.13(c)(2)	(i) Source material in glazed ceramic tableware	≤20% by weight
	(ii) Piezoelectric ceramic	≤2% by weight
	(iii) Glassware not including glass brick, pane glass, ceramic tile, or other glass or ceramic used in construction	≤10% by weight
40.13(c)(3)	Photographic film, negatives or prints	Uranium or thorium
40.13(c)(4)	Finished product or part fabricated of or containing tungsten or magnesium–thorium alloys. Cannot treat or process chemically, metallurgically or physically	≤4% by weight thorium content
40.13(c)(5)	Uranium contained in counterweights installed in aircraft, rockets, projectiles and missiles or stored or handled in connection with installation or removal of such counterweights	Per stated conditions in rule

TABLE I–8.4. NRC EXEMPTED PRODUCTS, DEVICES OR ITEMS (cont.)

Exemption 10 CFR Part	Product, device or item	Isotope, activity or concentration
40.13(c)(6)	Uranium used as shielding in shipping containers if conspicuously and legibly impressed with legend "CAUTION RADIOACTIVE SHIELDING – URANIUM" and uranium encased in at least 1/8 in (3.175 mm) thick steel or fire resistant metal	Depleted uranium
40.13(c)(7)	Thorium contained in finished optical lenses	≤30% by weight thorium, per conditions in rule
40.13(c)(8)	Thorium contained in any finished aircraft engine part containing nickel–thoria alloy	≤4% by weight thorium, per conditions in rule
30.11	Diffuse material such as contaminated soil, rubble, pavement, etc. 1. Fission and activation products – 25 pCi/g 2. Transuranics – 0.1 pCi/g 3. ^{3}H – 1000 pCi/g 4. ^{129}I – 0.01 p/Ci/g 5. ^{99}Tc – 1.0 pCi/g 6. ^{14}C –10 pCi/g 7. ^{40}K – 818 pCi/g	As determined by specific NRC or Agreement State exemption, IDEQ authorization and related safety determination. The sum of the concentrations of all radionuclides present shall not exceed 2000 pCi/g

Note: 1 pCi = 0.037 Bq.

Annex II

LESSONS LEARNED

The following examples present lessons learned. Some brief technical details are provided for each decommissioning project featured with a description of the problems encountered in the disposition of decommissioning wastes. The situations described are typical of the difficulties that can arise when planning or implementing decommissioning where the planned approach for the management of the resulting decommissioning wastes proves inadequate. The information presented is not intended to be exhaustive. The reader is encouraged to evaluate the applicability of the lessons learned to his or her specific decommissioning project.

CASE 1. CHANGES IN RELEASE CRITERIA, GERMANY [II–1]

Problem encountered

Due to technical problems, the Niederaichbach Power Plant (KKN) in Germany was shut down on 31 July 1974. The licence for safe enclosure, the first step of decommissioning, was granted in 1975. Implementation of safe enclosure took until 1981, costing a total amount of €18 million, including €1.1 million for regulation and technical supervision. By then, all operating fluids including the heavy water and the fuel elements had been disposed of and the controlled area had been restricted to the containment accommodating the entire radioactive inventory of about 74 TBq (1982).

Upon completion of these activities, the ambient dose rate was less than 0.01 mSv/h in the accessible areas. The annual cost of safe enclosure (checks by the technical control board (TÜV), radiation protection and monitoring, conservation, repairs and safeguarding) amounted to about €0.3 million, increasing to about €0.6 million by 1987.

During the implementation of safe enclosure, a research project was sought to demonstrate total dismantling of a power reactor to ‘green field’ status. Due to its short operating time and the comparatively small radioactivity inventory, the KKN was selected as a demonstration dismantling project. The decision in favour of complete dismantling of KKN was taken in 1979. A licence application was made in accordance with Article 7.3 of the German Atomic Law. This licence was granted on 6 June 1986 and became effective with the order of immediate execution of 1 July 1987.

In 1979, a general contractor was hired to prepare planning and licensing documents based on free (unrestricted) release levels of 3.7 Bq/cm^2 for ^{60}Co averaged over an area of 100 cm^2. The contract for complete dismantling of KKN was also awarded at a value of €35 million. Subsequent changes in release criteria resulted in the licence granted in 1986 specifying release levels of 0.37 Bq/cm^2 for ^{60}Co (10% of the original value applied for).

Following difficult negotiations with the contractor, the reduction of the release level resulted in a very substantial increase in costs of approximately €15 million.

Lessons learned

Ideally, all relevant licence requirements and criteria should be fixed before any tendering process is undertaken. In practice, regulations are subject to change and the potential strategic or contractual impact of such changes needs to be recognized in a risk assessment conducted before contractual or other commitments are made and contingent risk management options should be identified. In this case, with the tender already placed, the tenfold reduction in release levels resulted in a massive increase in work content with the consequent cost escalation demanded by the contractor.

CASE 2. HOT LAUNDRY, USA [II–2]

Problem encountered

Containers of various chemicals were left at the Hot Laundry when it was shut down in 1981. The labels on some of these containers were missing and some containers were in a poor physical condition. Since the chemicals were not identified and documented during the pre-decommissioning characterization, the disposal of these chemicals required considerable time and effort, which resulted in unanticipated costs and project delays.

Lessons learned

Pre-decommissioning records and characterization should include the identification and documentation of stored chemicals in order to properly plan for their disposal. Failure to label material essentially violates the waste minimization principle as the material will have to be treated, at least initially, as having the most demanding credible radiological and chemical properties.

CASE 3. AUXILIARY REACTOR AREA II FACILITY, USA [II–2]

Problem encountered

A radiological characterization of known or suspected radiologically contaminated areas at the auxiliary reactor area II facility was performed in 1983 to provide data for hazard evaluation and waste disposal. Radiation surveys of the interiors of buildings and structures were performed and smear samples for detecting removable contamination were collected. Surface and subsurface soil samples were collected and analysed. Samples were also taken of building materials such as insulation, lumber, metal sidings and sheet rock to identify the extent of the hazards and to establish possible waste streams for the demolition process.

Radiation surveys from the 1983 characterization indicated that most of the buildings contained no smearable (transferable) contamination. However, as the decommissioning progressed, it was discovered that most of the metal building had been painted over with a heavy metallic paint after the SL-1 reactor (Stationary Low Power Reactor No. 1) accident to fix contamination in place. During decommissioning, it was also discovered that concrete caps had been poured over the top of the original floors to cover and fix the contamination in place. This is why the smears from the original survey were negative — the smears were collected from on top of the clean covers.

The samples of insulation taken from the buildings also showed measurable contamination on them. Additional surveys performed during the interior dismantlement process confirmed that contamination had concentrated behind the sheet rock walls and in the attic space of the buildings. All the building components in these spaces (lumber, insulation, sheet rock, ceiling tiles, electrical wiring and conduits) were contaminated above allowable release limits. All this material therefore had to be manually disassembled, size reduced, placed in waste boxes and disposed of at the Idaho National Engineering and Environment Laboratory low level radioactive waste burial grounds. Contamination was also found under the concrete floor caps and under the heavy metallic paint, which required that unexpected amounts of the metal structures and concrete from the buildings be sent to the Radioactive Waste Management Complex instead of being released or sent to a landfill site for disposal.

Lessons learned

The characterization surveys performed before the project, both physical and radiological, are not always a good indication of the levels of

contamination that will be found on the site or of the actual physical characteristics of the site. Those who do the decommissioning should be prepared to deal with these unknowns in the process. Incident occurrence and associated remediation reports are extremely valuable for future decommissioning. Characterization needs to utilize all sources of information on the radiological status of the facility.

CASE 4. KOREA RESEARCH REACTORS (KRR) [II–2]

Problem encountered

It is required that the radioactive solid waste arising from decommissioning activities has to be stored until the start of operation of the low and intermediate level radioactive waste repository. Thus, the option of changing the reactor hall of KRR 2 into a temporary storage site for radioactive waste was selected. Unfortunately, it was very difficult to read the construction drawings, as KRR 2 was built more than 30 years ago.

An investigation of the structural analysis of the reactor hall of KRR 2 was required to satisfy the requirements for the temporary storage of radioactive waste. Some information was obtained from the drawings. Since some detailed records were lost, additional investigations such as non-destructive examination and electric resistivity prospecting could be necessary to acquire the required information on the reactor hall of KRR 2. This may cause an increase in the decommissioning cost.

Lessons learned

Maintaining accurate records of design and construction is necessary to ensure information is available for eventual decommissioning. This applies not only to the means of dismantling a structure but also to its prospective reuse as this example shows. Operators should ensure that such information is available in the planning stages of decommissioning rather than waiting until operation has ceased.

CASE 5. WASTE CLASSIFICATION OF CONCRETE BIOSHIELDS BY COMPUTER CODE CALCULATIONS (PROTOTYPE AGR) [II–3]

Problem encountered

Neutron activation calculations using standard codes can overpredict the activity of concrete bioshields by factors of five to ten, leading to waste overclassification. To avoid this, ensure that iron rod enforcements (composition, size and spacings) are taken into account during calculations. Furthermore, the different water (hydrogen) concentrations in the concrete can also affect the neutron flux estimations.

Lessons learned

Off-the-shelf codes developed for reactor physics/shielding calculations need to be used with care in predicting neutron activation. A comprehensive input data set is necessary, reflecting local conditions accurately.

CASE 6. WASTE CLASSIFICATION OF CONCRETE BIOSHIELDS — INCLUSION OF TRITIUM (^{3}H) INVENTORY (PROTOTYPE AGR) [II–3]

Problem encountered

Tritium production from ^{6}Li (n,g) is the major source of activity in concrete bioshields, resulting from the activation of, typically, about 20 ppm Li parent. Furthermore, after production, tritium can migrate towards the outside of the bioshield, resulting in underprediction of waste quantities.

Lessons learned

Behaviour of mobile species is important during waste classification and can lead to waste cost underpredictions if unaccounted for. Therefore, take into account diffusion characteristics during modelling of mobile species and validate results by sampling and analysis.

CASE 7. SELECTION OF RADIONUCLIDES FOR WASTE CHARACTERIZATION DURING DECOMMISSIONING PLANNING PHASES (PROTOTYPE AGR) [II–3]

Problem encountered

The selection of radionuclides for estimation required optimization on a cost–benefit basis since all forms of estimation added costs to the waste disposal process and these increases could be significant. There was a need for early contact with regulatory and disposal authorities to select appropriate radionuclides and to determine the levels below which there was no concern. An appropriate waste quality plan was then drafted so that waste processing, including assay and packaging, could be optimized. This needed to recognize that radionuclides present in minor quantities, in terms of activity, could become significant for ultimate disposal, e.g. ^{36}Cl in graphite for disposal in a deep repository

Lessons learned

Uncertainties in waste characterization can be reduced by measurements and sampling but it will be at a cost. If plans are not agreed in advance with the relevant authorities, then measurements may need to be repeated at extra cost. Minor radionuclides may not affect clearance measurement outcomes but may be important for waste repositories and need to be considered.

CASE 8. DECLASSIFICATION OF LARGE SURFACES: THE STATISTICAL APPROACH FOR THE VANDELLOS-1 REACTOR BUILDING, SPAIN

Problem encountered

The reactor building of the Vandellos-1 nuclear power plant measured 42 m in width by 54 m in length and 86 m in height. This building had been considered a controlled zone during dismantling, so it had to be declassified prior to its disassembly/demolition. The surface declassification methodology in force required characterization of 100% of the surface of the building, which in view of its dimensions meant a serious problem from the point of view of logistics and cost.

Taking into consideration the operating history of the building, which indicated that it was subject to radiological effects, an alternative

declassification methodology based on statistical principles (MARSSIM) was developed. This methodology, which was approved by the regulatory authority, significantly reduced characterization efforts without reducing the levels of confidence required.

Lessons learned

Methodologies based on statistical methods may be a particularly suitable alternative for the declassification not only of areas of land but also of buildings having difficult to access surfaces, since they allow costs and time to be reduced without decreasing levels of confidence.

CASE 9. LAUNDERABLE VERSUS DISPOSABLE PERSONNEL PROTECTIVE CLOTHING, LOS ALAMOS NATIONAL LABORATORY, USA [II–4]

Problem encountered

The Los Alamos National Laboratory's Hazardous Waste Operations Group spent approximately $100 000 to dispose of solid waste, LLW, and mixed LLW generated during work with stored waste streams. Additionally, the group spent approximately $30 000 annually for disposable type coveralls. An alternative product had superior features of operator comfort, lower static attraction of radioactive particles and was able to be laundered. The laboratory replaced disposable personal protection clothing with launderable/reusable, chemical-resistant anti-contamination cloth coveralls and accessories to enhance worker comfort, reduce the amount of waste generated and make cost savings through waste avoidance.

Switching to launderable clothing for this group alone realized cost savings of $100 000 by avoidance of approximately 360 ft^3 (10.2 m^3) per year of waste. The laboratory sent its launderable clothing off-site for cleaning (transportation costs covered by supplier); however, some users surveyed used coveralls and wash contamination-free clothing on-site for additional savings.

Lessons learned

The authorities require that the management of waste, including radioactive, hazardous and mixed waste, be accomplished in a manner that minimizes the generation of such wastes; moreover, that ongoing and future waste management activities implement pollution prevention and waste

minimization programmes. Launderable protective clothing should be evaluated to determine if it is an appropriate replacement for disposable clothing and if it offers a cost effective means of minimizing waste in accordance with authority requirements.

CASE 10. APPROVED RELEASE OF MATERIAL FROM SITE LATER SUSPECTED AS CONTAMINATED, USA [II–5]

Problem encountered

Yankee Nuclear Power Station (YNPS) is a deactivated PWR situated in northwestern Massachusetts, USA which began dismantling and decommissioning activities in 1993. Concrete shield blocks from within the reactor support structure (RSS) were removed, sand blasted, surveyed and released from licensee radiological controls in 1999. At the time of the shield block release, analyses of the radionuclide content of concrete within the reactor support structure indicated values less than the minimum detectable activity. Based on these results and surface contamination surveys, the shield blocks were determined to be free of detectable licensed radioactive material.

Forty of the shield blocks from the steam generator cubicles were approved to be removed from the site and used to construct a retaining wall at a private residence. In 2004, as part of preparing for demolition and with plans to retain RSS concrete on-site, the licensee performed further volumetric sampling and analysis of radionuclides. A lower limit of detection of 10 pCi/g (0.37 Bq/g) was established for tritium for the additional volumetric sampling, based upon the concrete derived concentration guideline limits and the requirements of the license termination plan (LTP). This analysis identified the presence of tritium in essentially all concrete within the RSS. Levels of tritium from samples taken in the proximity of the former location of the steam generator shield blocks averaged approximately 200 pCi/g (7.4 Bq/g). The licensee subsequently had samples from the released shield blocks in Vermont analysed for the suite of radionuclides listed in the LTP, using detection limits consistent with the requirements of the LTP. The results indicated detectable levels of tritium and ^{14}C. Subsequent to the discovery of radioactive contamination in the concrete blocks, it was considered that there was inadequate legal authority for the removal of the shield blocks from YNPS. Therefore, the licensee submitted the subject request for disposal pursuant to 10 CFR 20.2002.

The NRC concluded that there were no significant radiological environmental impacts associated with allowing the shield blocks to remain in

place on the private property. As an alternative to the proposed action, the NRC considered denial of the proposed action. The implications would have been that the blocks currently being used as a retaining wall would have to be removed and disposed of at an appropriate disposal facility. This alternative would require a significant industrial activity with an associated risk of injury. Although the contamination level is low, this alternative would also result in an increase in occupational exposure as a result of the removal and relocation process. Additionally, the transportation of the blocks from their present location to a disposal facility would add an air quality and transportation risk impact. Finally, the property owner had indicated his desire to retain the blocks for the retaining wall. The removal of the blocks would necessitate a change to property usage or construction of an alternative wall, either of which would pose a significant financial impact on the property owner. The NRC determined that the impacts of the alternative were greater than that of the proposed action.

Lessons learned

When material is released for uncontrolled use, there should be robust procedures in place such that a later re-evaluation is most unlikely to reverse the disposition choice. In addition, when assessing disposition options, a whole risk approach should be taken that recognizes the non-radiological hazards as well as the radiological ones.

CASE 11. STAKEHOLDER INFLUENCE ON UNRESTRICTED RELEASE CRITERIA, USA [II–6]

Problem encountered

Maine Yankee Atomic Power Plant originally developed decommissioning plans and cost estimates based on the NRC unrestricted release criteria of 0.25 mSv/a to the maximum exposed individual. The intent was to allow rubblized building concrete meeting acceptable residual radioactive material limits to be reused as subgrade fill. Subsequent interaction with key stakeholders, including the State of Maine, resulted in a revised decommissioning plan and cost estimates in support of unrestricted release criteria of 0.1 mSv/a from all pathways and 0.04 mSv/a from the groundwater pathway with the removal of all rubblized concrete. This change resulted in more than a significant increase in projected radioactive waste volume.

Lessons learned

Early agreement with key stakeholders can assist in reducing risk to project schedule and costs arising from release criteria.

REFERENCES TO ANNEX II

[II–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Planning, Managing and Organizing the Decommissioning of Nuclear Facilities: Lessons Learned, IAEA-TECDOC-1394, IAEA, Vienna (2004).

[II–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Record Keeping for the Decommissioning of Nuclear Facilities: Guidelines and Experience, Technical Reports Series No. 411, IAEA, Vienna (2002).

[II–3] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes, Technical Reports Series No. 389, IAEA, Vienna (1998).

[II–4] US DEPARTMENT OF ENERGY, Launderable Versus Disposable Personnel Protective Clothing, Lesson ID: 2000-LA-LANL-ESH7-0005 website: www.eh.doe.gov/DOEll/index.asp (available upon subscription)

[II–5] NUCLEAR REGULATORY COMMISSION, Environmental Assessment and Finding of no Significant Impact for Proposed Disposal Procedures for the Yankee Atomic Electric Company in Accordance With 10 CFR 20.2002, License DPR-003, Rowe, MA (2006).

[II–6] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, The Decommissioning Handbook, page 5-7, ASME, New York (2004).

GLOSSARY[1]

clearance. Removal of radioactive materials or radioactive objects within authorized practices from any further regulatory control by the regulatory body.

clearance level. A value, established by a regulatory body and expressed in terms of activity concentration and/or total activity, at or below which a source of radiation may be released from regulatory control.

conditional release.* Applies to material that has met clearance criteria which are specific to an identified first usage and for which a specific set of relevant exposure scenarios have been used to calculate doses to potentially affected persons. (See also **restricted use**.)

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are 'closed' and not 'decommissioned').

decommissioning materials/waste.* Materials or waste generated in the course of decommissioning of a nuclear facility.

decontamination. The complete or partial removal of contamination by a deliberate physical, chemical or biological process.

disposition.* Consignment of or arrangements for the consignment of, radioactive material and waste for some specified (interim or final) destination, for example, for the purpose of processing, recycling, reuse, storage or disposal.

[1] Except where noted by an asterisk, all definitions are taken from the IAEA Safety Glossary: Terminology used in Nuclear Safety and Radiation Protection, 2007 Edition, IAEA, Vienna (2007).

monitoring. The measurement of dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of the results.

nuclide vectors.* (See **scaling factors**.)

regulatory body. An authority or system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

regulatory control. Any form of control or regulation applied to facilities or activities by a regulatory body for reasons related to radiation protection or to the safety or security of radioactive sources.

restricted use. The use of an area or of materials, subject to restrictions imposed for reasons of radiation protection and safety.

scaling factors.* A mechanism for the evaluation of activity of difficult to measure radionuclides based on the activity of key radionuclides where the relative proportions of each are well understood.

unconditional release.* Applies to material that having met clearance criteria is subject to no further radiologically based restrictions on its further usage or disposal. (See also **unrestricted use**.)

unrestricted use. The use of an area or of materials without any radiologically based restrictions.

waste, low and intermediate level (LILW).* Radioactive waste with radiological characteristics between those of exempt waste and high level waste (HLW). Often divided into intermediate level waste (ILW) and low level waste (LLW).

waste, minimization. The process of reducing the amount and activity of radioactive waste to a level as low as reasonably achievable, at all stages from the design of a facility or activity to decommissioning, by reducing waste generation and by means such as recycling and reuse, and treatment, with due consideration for secondary as well as primary waste.

waste, radioactive. For legal and regulatory purposes, waste that contains, or is contaminated with, radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body. It should be recognized that this definition is purely for regulatory purposes, and that material with activity concentrations equal to or less than clearance levels is radioactive from a physical viewpoint, although the associated radiological hazards are considered negligible.

waste, very low level (VLLW).* Radioactive waste at activity concentrations above clearance levels but within the lower end of the activity concentration range defined for LLW. The waste may be disposed of either to dedicated facilities that meet less demanding design criteria than those for LLW or, subject to specified conditions, with ordinary waste in facilities not specifically designed for radioactive waste disposal.

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Vandellos, Spain: 25–29 April 2005; Vienna, Austria: 27 February–3 March 2006

This report presents options for the management of decommissioning materials to inform the production of a materials disposition strategy consistent with current IAEA guidance on clearance. It includes a review of the relevant safety, regulatory, technological, economic, social and administrative factors influencing these options. The subject is examined in the context of the value, practicality and viability of the various disposition options, and the availability of suitable tools, techniques and instrumentation to monitor compliance with release criteria. Each of the range of disposition options discussed is feasible in principle, and successful applications in Member States are described.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–109907–5
ISSN 0074–1914