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ON DECOMMISSIONING

Material Management and Characterisation Techniques

Final Report
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1. Introduction

For nuclear facilities, decommissioning is the final phase in the life-cycle after siting, design, construction, commissioning and operation. It is a process which involves operations such as decontamination, dismantling of plant equipment and facilities, demolition of buildings and structures, and management of the resulting materials. All of these activities take into account the health and safety requirements of the operating personnel and the general public, and also any implications these activities have for the environment.

As a result of these decontamination and decommissioning operations, a wide range of materials arise. Some of them will be radioactive; some will continue to have an economic value and/or are in a form which can be recycled or reused. Others will have little or no economic value, and these are the wastes that have to be disposed of, or which must be stored if no accepted method of disposal exists, in either case at major economic cost to the industry and, ultimately, to the community at large.

It should be reminded that the ultimate goal, i.e. the end 'product' of the decommissioning and decontamination operations, is the unconditional release or reuse of sites, facilities, installations, or materials for other purposes. Intermediate products of decommissioning and decontamination operations can be facilities or installations that have been dismantled, or materials that have been decontaminated to permit their release or reuse for other nuclear applications. Materials that cannot be conditionally or unconditionally released or reused, and which have to be treated as radioactive wastes, can be considered as by-products of the decommissioning and decontamination process. As such, waste minimisation can be considered as a strategy for avoiding, as much as possible, the production of these undesirable by-products. Where by-products are unavoidable, steps are required to minimise their volumes.

2. Objectives

The objective of this report is to provide an overview of current experience in characterisation and management of materials and wastes, monitoring systems and procedures, and unconditional release and reuse of materials from decommissioning operations.

This objective has been achieved by reviewing available experiences in:

- Material management (sources/quantities/characteristics of materials, material management routes/options, waste minimisation, conditioning and packaging, cost evaluation of options).
- Clearance process (methodology, procedures, monitoring systems, detection devices, monitoring campaigns, scaling factors, statistical analyses, documentation, compliance with clearance levels).
- Organisation and information systems (on site organisational systems for material management, waste/material management information systems, material traceability systems).
- Waste characterisation (techniques, instrumentation, monitoring of alpha-contamination/isotopes difficult to measure, characterisation of historic wastes and waste for final disposal).
- Management of special wastes, alpha wastes, graphite, Na, Be, solvents, resins, mixed wastes, etc. (waste quantities/characteristics, packaging, conditioning, treatment/storage, disposal).

3. Preparation of the report

This report has been prepared under the Work Package 3 (WP3) of the Co-ordination Network on Decommissioning of Nuclear Installations (CND) under the contract no. 0508855 (FI60) from the European Commission's Research and Technological Development (RTD) Division to a consortium of European nuclear organisations. This work was carried out by DECOM (Slovak Republic) as the acting work package manager supported by AF-Colenco AG of Switzerland.

The activities in the framework of the working package were managed as follows:

- Facilitating the discussions among experts by sharing knowledge, practical experience, priorities, points of view, etc. and promulgating the practical experience relating to technologies and quality assurance.
- Collecting information within projects and associated sources.
- Evaluating good and best practice from participants' presentations of the experience.
- Analysing, interpreting and correlating the collected information.
- Concluding the work with this final report

The document represents an overview of the state-of-the-art in the area based on different documents to which participants of the Co-ordination Network on Decommissioning of Nuclear Installations (CND) have contributed in a substantial way.

4. Management of materials arising during the decommissioning process

4.1 General characteristics of materials from decommissioning

Potentially radioactive material and radioactive wastes may arise throughout the lifetime of any type of nuclear facility. They 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 arising 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 the final decommissioning period. These volumes of materials will require careful management to ensure their disposition for reuse, recycling, storage or disposal in both environmentally and economically appropriate conditions [1].

4.1.1 Range of decommissioning materials

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. For such decommissioning materials proper planning and selection of appropriate material management and eventually waste management options is of particular importance from the organisational, safety and economic point of view.

On the other hand, materials or wastes similar to those produced during the operation period (process fluids from the facility, secondary waste like decontamination solutions, spent ion exchangers or air-conditioning filters) 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 during decommissioning.

From the radiological point of view, a majority of the material arising from the nuclear facility decommissioning process will be non-radioactive. In addition, there are also relatively large quantities of materials in which the radionuclide concentrations are close to those for release from regulatory control or clearance, with or without conditions being placed upon their further use. This means that a high percentage of decommissioning materials should be able to be released into the environment and for only a limited proportion of materials (2 - 6 %), disposal as radioactive waste may be the only option. This is normally the case with activated and contaminated material, where decontamination is not a viable alternative such as internal reactor core parts or other large volume items of the primary circuit like heat exchangers.

Most of the non-radioactive materials will be concrete and brick rubble from building demolition (concrete, masonry, roof construction, prefabricates, etc.), materials from the secondary circuits and other auxiliary facilities (steel, colour metals, insulations), and site clean-up material (soil, roads). Thus, a major effort in material or waste management is associated with segregation, measurement for verification of release limits and decontamination to reach those same limits. It means, if the segregation, decontamination and clearance verification process will be done properly, the amount of radioactive materials arising in the decommissioning process requiring treatment and disposal in radioactive waste repositories can be reduced significantly.

Materials from decommissioning may also include so-called special wastes containing toxic or hazardous materials, such as sodium, beryllium, lead, asbestos, etc. These materials have to be treated in a special way considering their toxic and hazardous characteristic (see Section 5) [1, 2].

The radioactive waste management strategy is one of the key considerations in the planning and timing of the nuclear facilities decommissioning process. At the early stage of decommissioning planning, it is required to make an estimation of the amount of decommissioning materials from different types in relation to the methods and processes for their further management. An accurate estimate of the mass or the volume of the decommissioning materials should comprise the following activities [2]:

- Classification of systems and structures of the nuclear facility with respect to activity (activated, contaminated, alpha bearing versus non-alpha bearing, etc.). This characterises the material type that will be generated, as well as further treatment, handling, packaging and disposal requirements.
- Development of a detailed mass or volume inventory of the systems and structures of the facility.
- Determination of the quantities and volumes of materials that can be decontaminated and/or measured in view of conditional or unconditional release or recycle and reuse, including items generated during decommissioning.
- Determination of the quantities and volumes of compactable and burnable contaminated solid materials, including items generated during decommissioning.
- Determination of the quantities and volumes of contaminated solid materials which cannot be compacted or incinerated. As this category of materials has a large impact on the technical equipment required for handling and conditioning, an accurate determination is required.
- Determination of the volume and characteristics of contaminated liquids. The volume of liquids generated during decontamination and flushing operations will largely depend on the type of facility and its representative contaminants, the number of decontamination steps and their efficiency.
- Determination of the quantities of secondary waste that is likely to be produced in the decommissioning process.
- Determination of gaseous effluents and aerosols. Aerosols containing finely dispersed radioactive materials result from cutting and abrasive surface cleaning methods. Some cutting and cleaning methods produce large volumes of toxic smoke and fumes. Contamination control coupled with filters in the ventilation streams should be adequate to collect and retain the particulate material.

4.1.2 Material types arising in the decommissioning of specific nuclear facilities

This section describes the nature of the materials arising from the decommissioning of facilities from different parts of the nuclear fuel 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. The management of materials from facilities housing irradiated nuclear fuel is covered in terms of reactor nuclear facilities and spent fuel reprocessing plants. Finally, some coverage is also provided for smaller, more specialised facilities [1, 3, 4, 5].

4.1.2.1 Refining and conversion

Refining is the processing of uranium ore concentrates 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 different sites. A general sequence of different processes resulting in UO_3 and UO_2 production briefly considers:

- *Purification*: uranium ore concentrate is dissolved in nitric acid and then purified from a broad spectrum of impurities to obtain the very high purity levels required for nuclear fuel. The purity of the final product is > 99.95 % pure uranyl nitrate.
- *UO_3 and UO_2 production*: there are three basic processes usually used to produce UO_3 and UO_2 from the purified uranyl nitrate: the thermal denitration process, the ammonium diuranate process, and the ammonium uranyl carbonate process.

A major part of the waste generated during the refining process is associated with the purification stage. The uranium nitrate after the purification stage is quite pure, and a very low amount of waste is linked with the end of the refining process.

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

- *Reduction*: the UO_3 is reduced to UO_2 by reaction with hydrogen or cracked ammonia in reactors of moving bed, fluidised bed or rotary kiln types.
- *Hydro-fluorination*: the UO_2 is converted to UF_4 using wet process technologies (calcium fluoride CaF_2 is arising) and dry process technologies (no significant waste is generated).
- *Fluorination*: UF_4 reacts with fluorine to form UF_6 either in a flame reactor or in a fluidised bed reactor. The only waste arising in the fluidised bed process is the calcium fluoride that is stored to allow the decay of the short-lived, daughter products of ^{238}U (^{234}Th and ^{234m}Pa). The spent calcium fluoride, after drying, can either be consigned to a non-nuclear waste repository due to its low uranium content, or reused.

The uranium refining and conversion facilities are similar to many chemical plants. The described processes are generally carried out in closed systems within vessels and pipes. Also the handling of wet solids, solutions, solvents and gaseous products is involved. As such the plant components become contaminated by process fluids on internal surfaces and externally as a result of leakages and spills. Normally only uranium isotopes and their daughters are present. Hence, the radiation hazard is quite low. But on the other side the heavy metal nature of the uranium and the properties of non-radioactive materials (acids, organic solvents, fluorine, hydrogen fluoride, etc.) used in the process, can evoke health risks for personnel. Apart from the physical size of conversion facilities (occupy the area of several hectares) and the presence of conventional hazards, the decommissioning of these facilities is usually more straightforward compared with the other facilities included in the nuclear fuel cycle. It is very analogous to the decommissioning 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 [1, 3].

4.1.2.2 Enrichment

Enrichment is understood as increasing the proportion of ^{235}U, from the natural level of 0.712 % to an average level of 3 to 5%, in UF_6, according to requirements from fuel fabrication plants. This is done predominantly by two different industrial methods:

- *Gaseous diffusion* enrichment is based on the different diffusion rates of gaseous $^{235}UF_6$ and $^{238}UF_6$ through semi-permeable membranes. The lighter $^{235}UF_6$ diffuses slightly quicker than the $^{238}UF_6$. Repetition of the operation in cascade diffusion columns leads to increasing of the enrichment to the required level.
- *Centrifugation* is the process in which enrichment is achieved by differential centrifugation. UF_6 is injected into a high speed centrifuge and the lighter ^{235}U is separated from the heavier ^{238}U. The cascade arrangement of centrifuges leads to progressively enriched fractions. Compared with diffusion, centrifugation is more efficient. Thus the plant is smaller for the same output and the energy consumption is significantly lower for a given capacity.

Centrifugation and gaseous diffusion processes produce only very minor quantities of waste. The reason is that the plant handles a single process medium (UF_6), which is completely contained in a high integrity system throughout the operation. Since the processes are not chemical, there are no auxiliary inflows of material and no rejects of intermediate or waste products in the accepted sense. But enrichment of 1000 Te of uranium in the form of UF_6 leads to the generation of around 850 Te of depleted uranium with a ^{235}U content of approximately 0.2 % - 0.3 %. This material may be classified as a by-product or as a waste.

As for conversion plants also the decommissioning of enrichment facilities is parallel to many chemical plants. However, the radiological risks are greater than in uranium conversion due to the enhanced levels of ^{235}U. What is more, given the adaptability of enrichment processes to the development of uranium-based nuclear weapons, the decommissioning of such plant introduces special security issues. Steps may be taken to ensure that sensitive components are destroyed.

The final decommissioning of enrichment facilities are largely repetitive and involves the dismantling of a very large number of almost identical units that are installed inside very large buildings [1, 3].

4.1.2.3 Fuel fabrication

Fuel fabrication means the manufacture of the fuel assemblies that should be loaded into the nuclear reactor. For fuel fabrication, two products are used as a starting material:

- *Uranium dioxide (UO_2)*: for UO_2 fuel manufacture, natural or enriched uranium can be used. In the first step, UO_2 in powder form is produced. The blended powder is pre-compacted and granulated. The granulated powder is compacted into a cylindrical form. The pellets are then sintered in a high temperature furnace under a hydrogen atmosphere. After sintering, they are loaded into zirconium alloy tubes that are filled with helium and then welded. The last step involves inserting the fuel pellets into fuel assemblies. The above described order for fuel fabrication is typical for most common light water reactors where the uranium enrichment and the cladding materials may vary but the manufacturing process from UO_2 powder to the finished fuel is basically the same.
- *Metallic uranium*: only natural uranium is used for the production of metallic uranium fuel. This type of fuel, for which the starting material is natural UF_4, is typical for

older type of reactors, e.g., Magnox (magnesium non-oxidising) reactors in the United Kingdom.

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 for the production of $(Pu\text{-}U)O_2$ fuel pellets and incorporation of these pellets into fuel assemblies. The plant may use a process involving mechanical blending of Pu and U solutions, followed by co-precipitation and calcination to form MOX fuel.

The fuel fabrication part has the potential to produce a significant amount of material scrap. Most portion of this scrap is not considered as waste, and what is more, it has a significant value and is recycled and reused.

Radioactive waste from fuel fabrication that should be classified as low level radioactive waste (LLW) and intermediate level radioactive waste (ILW) is represented by filter media from wash water clean-up, waste oils, spent acids and bases, spent analytical solutions, decontamination and cleaning solutions and discarded scrap metals and equipment. Any of these wastes may also be contaminated with hazardous chemicals and uranium. Plutonium contamination is present from facilities manufacturing mixed oxide fuel.

The fuel fabrication technology may also include support processes or secondary waste treatment technologies such as solvent extraction, ion exchange or precipitation to recovery the effluents and liquid waste evaporation system, followed by solidification (cementation, bituminisation) of resulting concentrates.

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. In the decommissioning process of mixed oxide fuel fabrication plants, the measurement of the residual inventory to avoid criticality hazards should be done because PuO_2 and $(Pu\text{-}U)O_2$ powders may remain in some parts of the process. Another feature for Pu plants is the possibility of significant radiation exposure of personnel from inhalation or external irradiation from gamma or neutron emitters, wherever residues exist in the plant.

The decommissioning of typical fuel fabrication plants includes operations in manufacturing, maintenance, decontamination and storage areas situated in the main building. To release the whole site that occupies an area of several hectares, other buildings like laboratories, waste treatment facilities, waste recycle plant and other auxiliary facilities, such as tanks and pumps, warehouse and storage areas also have to be decommissioned in a proper way [1, 3].

4.1.2.4 Nuclear power plants

Around the world, there are a large number of nuclear power plants which ended operations and that have to be decommissioned. What is more, this number is still increasing due to shutdown of the older types of nuclear power plants or due to political decisions not to develop a nuclear energy programme.

From the radiological point of view, the main sources of radioactive materials from dismantling may be divided into two separate groups:

- The reactor itself with the pressure vessel, internal structures and biological shielding. The materials of these components are primarily activated and account for more than 90 % of the total activity in the nuclear installation.
- The complete primary coolant circuits and secondary radioactive technological equipment which are primarily contaminated.

Taking the above into the account, consideration may be given to preserve and defer the dismantling of the reactor parts for a longer period of time. But the dismantling of the primary circuits and auxiliary parts may be done immediately just after decontamination, thus reducing the nuclear power plant surveillance and maintenance annual costs.

On the other hand, because of the radioactive decay process, the quantity of radioactivity decreases with time after plant shutdown, particularly for the reactor and primary circuit components where ^{60}Co is the dominant radionuclide. As a result, selecting a strategy of deferred dismantling for the nuclear power plant technological equipment, due to the radiological decay of the radio-nuclides, should involve a subsequent decrease of the amount and the activity level of the radioactive waste produced during the decommissioning activities. It means that more materials should be able to be recycled and reused outside the nuclear power plant area, which may result in a reduction of the necessary radioactive waste repository capacity, while also a non-negligible amount of money could be saved.

An example of approximated masses and activities of dismantled steel at various time periods after final shutdown for a 1'000 MW_e pressurised water reactor (PWR) is shown in Table 4.1. The main benefits from the radioactive decay are usually from reductions in the gamma radiation levels. The table shows the decreasing proportion of beta-gamma emitters in low-level radioactive steels, as time progresses [1, 4].

Table 4.1 Effect of radioactive decay on amounts and activities of steel from a 1000 MWe PWR

<table>
<tr><td colspan="2" rowspan="2"></td><td colspan="6">Time after reactor shutdown</td></tr>
<tr><td colspan="2">5 years of decay</td><td colspan="2">25 years of decay</td><td colspan="2">100 years of decay</td></tr>
<tr><td>Surface activity ($Bq.cm^{-2}$)</td><td>Activity concentration ($Bq.g^{-1}$)</td><td>Mass (t)</td><td>Total activity (Bq)</td><td>Mass (t)</td><td>Total activity (Bq)</td><td>Mass (t)</td><td>Total activity (Bq)</td></tr>
<tr><td>37 – 370</td><td>10</td><td>800</td><td>8.0×10^{9}</td><td>440</td><td>4.4×10^{9}</td><td>240</td><td>2.4×10^{9}</td></tr>
<tr><td>3.7 – 37</td><td>1</td><td>1600</td><td>1.6×10^{9}</td><td>880</td><td>8.8×10^{8}</td><td>480</td><td>4.8×10^{8}</td></tr>
<tr><td>0.37 - 3.7</td><td>0.1</td><td>3200</td><td>3.2×10^{8}</td><td>1760</td><td>1.8×10^{8}</td><td>960</td><td>9.6×10^{7}</td></tr>
<tr><td colspan="2"></td><td colspan="2">99.9 % beta-gamma, 0.1 % alpha</td><td colspan="2">99 % beta-gamma, 1 % alpha</td><td colspan="2">95 % beta-gamma, 5 % alpha</td></tr>
</table>

Table 4.2 Estimated quantities of typical material types generated during the decommissioning of two nuclear power plants with different reactor types

Radioactive material generation	250 MWe GCR (t)	900-1300 MWe PWR (t)
Irradiated carbon steel	3'000	-
Activated steel	-	650
Graphite	2'500	-
Activated concrete	600	300
Contaminated ferritic steel	6'000	2'400
Steel likely to be contaminated	-	1'100
Contaminated concrete	150	600
Contaminated lagging	150	150
Contaminated technological wastes	-	1'000

From the above mentioned facts is clear, that the inventory of radioactive materials and radioactive wastes arising from the decommissioning of a nuclear power plant depends on the chosen decommissioning strategy.

Another important factor determining the decommissioning inventory is the type of reactor and also the characteristics of the operational period (operation duration, number and seriousness of accidents). As an example Table 4.2 shows the estimated amounts of different material types for a gas cooled reactor (GCR) and a pressurised water reactor [1, 4].

The radioactivity level of most of these materials is usually low. That is the reason why they should be available for unconditional release to the environment after cleaning and/or adequate decontamination to the required release levels. The quantities in Table 4.2 are dominated by steel and in the case of a gas cooled reactor also by graphite, being discussed further in Section 5.

Another important task in the management of decommissioning material is the large quantities of concrete, not shown in this table, because reinforced concrete around the reactor core may be activated. In addition, there is a possibility particularly along joints or through leak paths that the concrete structure will also be contaminated [1, 4].

4.1.2.5 Spent fuel reprocessing

In a spent fuel reprocessing plant, the components of spent nuclear fuel (SNF) such as reprocessed uranium, plutonium, minor actinides, fission products, activation products or cladding materials are separated. The first stage of reprocessing usually involves the separation of the uranium and plutonium via the solvent phase from minor actinides, fission and activation products which remain in the aqueous phase and form the high level liquid waste. This operational high level waste (HLW) from reprocessing is stored to await conditioning, e.g., by vitrification. In most countries the disposal strategy for high level waste is still under development, however. All liquid wastes from reprocessing, all fuel cladding and fuel assembly debris, all maintenance wastes and discarded equipment, support laboratory and analytical equipment and solutions are considered to be low or intermediate level waste (LLW/ILW).

A number of reprocessing plants have been decommissioned, and hence, data on the waste arising from these operations are available. The quantities and the disposition of materials from the Karslruhe WAK reprocessing plant decommissioning (annual throughput ~30 t of uranium) are given in the Table 4.3. The high level waste in liquid form is operational waste, which had been stored to await conditioning upon final plant decommissioning. The table illustrates the domination of the decommissioning wastes by structural materials, and about 98 % were cleared for recycling/reuse. Also a major proportion of the plant components and other wastes from decommissioning were cleared.

Table 4.3 Summary of waste arising form the decommissioning of the WAK reprocessing plant

Material	Quantity (m^3)	Nature	Disposition
High level liquid waste	52 (vitrified)	Radioactive waste 130 of 400 litres casks	Storage
Contaminated plant, decontamination, wastes, etc.	1’681	Radioactive waste conditioned (3’360 m^3)	Disposal
	2’840	Cleared material	Recycle/reuse
Building rubble, structural materials, etc.	2’279	Radioactive waste conditioned (4’560 m^3)	Disposal
	110’000	Cleared material	Recycle/reuse

Spent fuel reprocessing plants are usually contaminated by alpha emitters and fission products. Even after several decades, the resulting radioactive decay makes no significant benefits for personnel protection, radioactive material management or potential minimisation of decommissioning waste. The annual costs of maintenance and surveillance in the reprocessing plant after shutdown are substantial. The combination of these facts makes the immediate dismantling of reprocessing plant desirable. What is more in a plant with a plutonium inventory, the radiation level rises due to ^{241}Am ingrowth. Therefore, early clean-out of such facilities is recommended [1, 4].

4.1.2.6 Research, institutional and industrial facilities

Research, institutional and industrial facilities vary widely in the nature of the work undertaken and therefore also their operational wastes should vary too. Generally, in this kind of nuclear facilities only low and intermediate level wastes are produced.

Materials arising from the decommissioning of research reactor facilities are quite similar to those from nuclear power plants. Dominated material types are concrete, steel (or aluminium) and often graphite, although there may be special considerations depending on the construction, the experimental use and the operational history of the reactor.

The decommissioning phase may also include the management of stored operational wastes. They are sometimes generated in small amounts from various experiments, processes and operations, which scope changed over the time very often. Therefore, they usually have unique characteristics and it is very difficult to find and define general waste management requirements and final disposition solutions for such kinds of wastes. Hence, a case-by-case approach is very often needed and also finally applied. The structural materials generated during the decommissioning process of these installations will be quite similar to those that are typical for other types of nuclear facilities, but they are likely to be relatively low in volume and except for research reactors without activation products.

4.1.2.7 Comparison of waste amounts for different parts of the nuclear fuel cycle

In the following Table 4.4, a comparison of operational and decommissioning wastes from the whole fuel cycle is given on the basis of m^3/GWe/year. It considers a pressurised water reactor with open fuel cycle (enrichment 3.5 %) and a pressurised water reactor with mixed-oxide fuel cycle in which pressurised water reactor fuel is reprocessed, and recovered plutonium is used for producing mixed oxide fuel (MOX) (5% of plutonium content) and recovered uranium is recycled in a conversion plant. Table 4.4 only comprises details of the final radioactive waste arising. Materials that were either not radioactive or decontaminated to clearance levels were not included [5].

4.2 Strategies and options for the management of decommissioning materials

4.2.1 General considerations

The decommissioning process is the final phase in the life cycle of a nuclear facility during which a wide range of materials is produced. Most of these materials are non-radioactive and from the materials that are classified as radioactive, most of these show only low levels of surface contamination. Only a small part represented by some reactor components will be activated.

The management of the materials resulting from the decommissioning activities remains one of the most critical aspects of the decommissioning process. Managing tens of thousand tons (and more) of decommissioning materials is not trivial and requires a dedicated organisation. In addition, the costs involved in the radioactive waste management process are a significant element of the overall decommissioning costs. These facts indicate the need for an accurate mass and radiological characterisation of the inventory of the nuclear facility and for maximising opportunities for materials recycle and reuse in order to minimise the amount of materials requiring treatment, conditioning, storage and final disposal as radioactive waste.

Table 4.4 Radioactive waste volumes arising from single parts of the nuclear fuel cycle (m^3/GWe/year)

	Facility type		Tailings	LLW	ILW	HLW	SNF
PWR uranium open fuel cycle	*Mining/Milling*		52’578				
	Conversion	Operations		26-92			
		Decommissioning		76			
	Enrichment	Operations		60			
		Decommissioning		4.8			
	Fabrication	Operations		2-9			
		Decommissioning		3-9			
	Reactor Operations	Operations		86-130	1.1-33		
		Decommissioning		175-230	9		
	Interim Storage	Operations		0.2	1.9		
		Decommissioning					
	Disposal	Operations		0.2	4.9		37
		Decommissioning					
PWR MOX closed fuel cycle	*Mining/Milling*		36’576				
	Conversion	Operations		21-74			
		Decommissioning		61			
	Enrichment	Operations		48			
		Decommissioning		3.9			
	MOX Fabrication	Operations		5	12		
		Decommissioning		0.2	0.4		
	Reactor Operations	Operations		86-130	1.1-33		
		Decommissioning		175-230	9		
	Interim Storage MOX	Operations		0.03	0.3		
		Decommissioning					
	Reprocessing	Operations		56-77	16-36	1.7-3.1	
		Decommissioning		4	0.6		
	Disposal MOX	Operations		0.03	0.7	2.4	5.4
		Decommissioning					

In the principles of radioactive waste management of the International Atomic Energy Agency it is stipulated that the generation of radioactive wastes during the whole life-cycle of a facility shall be kept to the minimum practicable, in terms of both activity and volume, by appropriate design measures, facility operation and decommissioning practices [6]. This includes the selection of appropriate technology, the selection and control of construction and operational materials, the recycle and reuse of materials, and the implementation of appropriate procedures. Emphasis should be placed on the segregation of different types of materials in order to reduce the volume of radioactive waste and facilitate its management.

In addition, it is also important to minimise the spread of radioactive contamination with a view to reducing to the strict minimum the need for decontamination, and hence also minimise the creation of secondary radioactive 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 [1].As such, waste minimisation (see more in Section 1.3) can be considered to be a strategy for avoiding, as much as possible, the production of undesirable by-products like non-releasable materials that should be disposed in a radioactive waste repository. Where the production of these by-products is unavoidable, steps are required to minimise their volumes [2].

From the above mentioned, it is possible to say that there are strong incentives to minimise the generation of radioactive decommissioning waste and 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. Ecological and sustainable development considerations demand maximum reutilization of non-renewable resources. Furthermore, the intrinsic value of the materials for recycle, in the case of metals, or for use in construction in the case of concrete, is considerable. Also the disposal of conventional (non-radioactive) waste is generally much cheaper than the disposal of radioactive wastes. Nevertheless, other factors such as the likelihood of regulatory approval and stakeholder acceptance also need to be taken into account. Consequently, some level of optimisation 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 [1].

4.2.2 Possible strategies for the management of decommissioning materials and wastes

In general, a vast technical area deserving attention is related to the decision making in the decommissioning material/waste management. An important element of the decision making is the methodology to evaluate and characterise the materials in view of their safe and cost-effective management. Materials origin, radiological characterisation, chemical and physical properties and available disposition paths should be taken into the consideration.

The subject requires that factors be investigated relevant to management strategies for large amounts of decommissioning materials including unconditional or conditional release or storage/disposal as radioactive or conventional waste. In addition, evaluation and characterisation methodologies should be defined for the planning and implementation of such strategies, and practical guidance provided on the flexibility and options offered by current management systems, including methodologies to ensure traceability of materials, activities and related information required in the various decommissioning material management options.

In order to enable such investigation, an overview of the possible management routes for components or materials arising from decommissioning of nuclear facilities should be given. This should be examined in the context of the need, practicality and viability of the various management options and the availability of suitable technologies, equipment and tools, criteria and instrumentation to monitor compliance with release criteria. Adequate flexibility should enable the operator to select the appropriate option meeting the boundary conditions of the specific situation of the nuclear facility.

An overview of the main options for the segregation and routing of materials arising from the decommissioning activities is illustrated in Figure 4.1. It should be obvious that alternative options or combinations of options may be applicable considering specific cases in individual decommissioning projects [1].

In general, part of the buildings or infrastructure of a nuclear facility may have been limited to conventional non-nuclear use. The materials resulting from the decommissioning of these parts may comprise technological equipments, office furniture, tools or structural components that should not be radioactive. After verification (if necessary), these materials may be:

- Reused in industry without any further transformation, e.g., tools, tanks, motors, pumps and also large components such as diesel generators.
- Recycled and then further used.

- Disposed as conventional waste on a municipal or an industrial disposal site. For these materials no economic use is available or they are requiring expensive recycling techniques. Depending on their chemical/toxic content, the latter may be subject to special disposal provisions.

The materials resulting from the decommissioning of the nuclear part of the facility may require adequate segregation and characterisation. Some of these materials may be presumed to be non-radioactive. Having been used or having been brought for a while in a controlled area marks them as 'suspected material' and they can only be withdrawn from the radioactive waste management system by a thorough and intensive demonstration that a possible residual contamination is well below specific levels, taking into consideration the historical use of the materials. Components or materials that after dismantling and simple cleaning activities can be measured to meet the actual unconditional or conditional clearance levels can be released for recycling or reuse, or for disposal as conventional waste on a municipal disposal site.

The decommissioning 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 to be radioactive should be adequately segregated and characterised in order to define further options. The first group is represented with the materials or components that can easily be cleaned or decontaminated and after meeting the clearance levels, they are released into the environment.

For the remaining materials further options for releasing them are available:

- Implementation of aggressive decontamination techniques and monitoring of the materials in order to meet the unconditional release levels.
- Storage of radioactive materials to allow for radioactive decay and then meet the unconditional release levels. Time decontamination storage should be applicable only for materials contaminated with short lived radioisotopes.
- Disposal of the materials as very low level waste without treatment, or just using less difficult and expensive technologies, e.g., low-pressure compaction.
- A combination of decontamination and storage to achieve the unconditional release levels or to reclassify the radioactive waste.

Finally, materials which are radioactive and for which the unconditional release levels cannot be achieved because decontamination or decay storage is not a cost-effective way, will have to be treated and finally disposed as radioactive waste. However, depending on local factors and costs for waste conditioning, packaging, transport, storage and disposal, adequate decontamination (e.g., melting) for re-categorisation may still be realised, reducing the overall costs for radioactive waste handling.

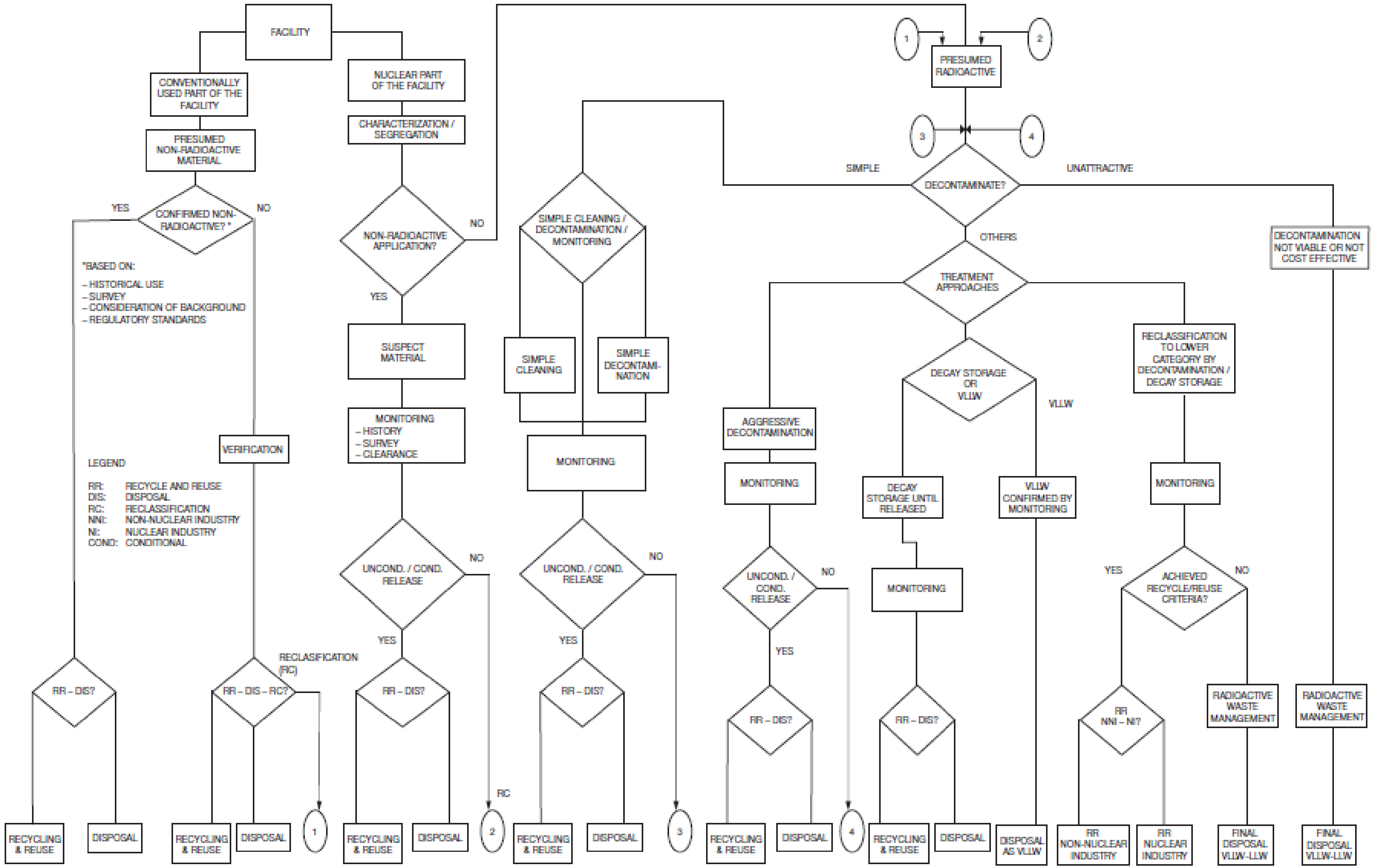

Figure 4.1 Flow sheet from generation to disposition of decommissioning waste/materials

4.2.3 Potential approaches for the final disposition of decommissioning materials

In the previous sectiob the detailed scheme of material management strategies was discussed. In Figure 4.2, a reduced diagram with available options for decommissioning materials or final waste disposition is shown. According to Figure 4.2, following disposition options may be identified [1]:

- Unconditionally released materials for free recycling and reuse.
- Unconditionally released materials for final disposal.
- Conditionally released materials for recycling and reuse within the non-nuclear industry.
- Conditionally released materials for recycling and reuse within the nuclear industry.
- Very low level waste (VLLW) disposed to an appropriate repository or storage facility.
- Low and intermediate level waste (LLW/ILW) disposed to an appropriate repository or storage facility.

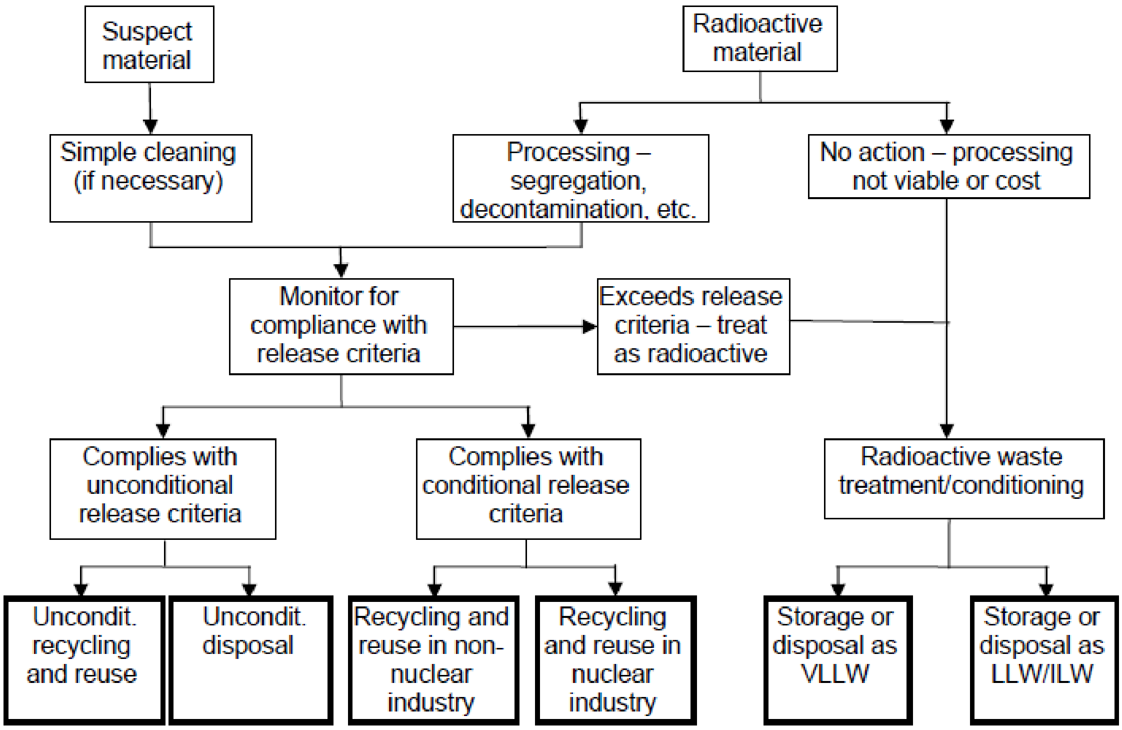

Figure 4.2. Disposition options for decommissioning materials

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 presented in [7].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 without specifically considering their radioactive properties. Also, reusable materials below clearance levels may be reused or recycled in an unrestricted manner.

The current situation in the field of clearance levels 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.

In the IAEA classification of radioactive wastes, the category of waste after cleared material is low level waste. There are two sub-categories: short-lived wastes and long-lived wastes. Depending on their specific characteristics, these wastes can be disposed into near surface, sub-surface or deep geological disposal facilities. Some countries have in their radioactive waste classification system also the category very low level waste (VLLW) and have established the disposal facilities for that kind of waste. The design criteria to be met and the related cost need not be as demanding as those for general low level waste, due to the lower radioactivity levels present [1].

The following sections explore in more detail the above defined decommissioning materials disposition options.

4.2.3.1 Unconditionally released materials for recycling and reuse

Unconditional release (it means recycling, reuse or disposal) is a designation by the regulatory body in a country, enabling the release of equipment, materials or buildings without radiological restriction. The full and complete unconditional release of a material requires that all reasonably possible exposure routes be examined and taken into account in the derivation of the unconditional release levels, irrespective of how that material is used and to where it may be directed.

During the decommissioning of nuclear facilities, large quantities of valuable metals and equipment may become available for unconditional recycling or reuse in any other industrial area. The IAEA work has been performed to define the basis for establishing general suitable criteria for unconditional release, and for applying these criteria to actual waste management and decommissioning cases.

On the other hand, it must be emphasised that national laws define their own processes and criteria for unconditional release of materials. Some countries have already implemented unconditional release of materials on a case-by-case basis.

To conclude this section it can be said that suitable criteria, measurement methodologies and instrumentation must be available to facilitate any release practice. This is particularly important for materials for unconditional release, which usually represents a large proportion of all decommissioning materials. Release criteria should be established in a manner that safety requirements are achieved but also that waste minimisation through recycling and reuse will be encouraged. Achieving international consistency in criteria definition can improve the prospects for a wider reuse of cleared material [1].

4.2.3.2 Unconditionally released materials for final disposal

If it is neither practical nor economical reasonable to reuse components or materials from decommissioning process that are considered to be radiologically clean or have been decontaminated to bring their activity below clearance levels, they may be sent for disposal as conventional industrial waste (e.g., in a municipal landfill site). This represents 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 and characteristics. Such provisions should not differ from the provisions defined for the disposal of other similar industrial or municipal waste materials. It means that these wastes can be safely disposed, applying conventional techniques and systems, without special measures related to their radioactive status, probably again at lower cost than if treated as radioactive waste [1].

4.2.3.3 Conditionally released materials for recycling and 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 radioactivity. Alternative approaches for material release that meet health protection objectives 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.

There are numerous examples of national regulations and international recommendations allowing the use of conditional or restricted release approaches. For example the radiological criteria based on specific planned destination of the conditionally released, e.g., scrap metal melting, building for demolition are prescribed. Such levels can be replaced by a site specific radiological analysis on the basis of a 10 μSv/a dose criterion.

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. 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 [1].

4.2.3.4 Conditionally released materials for recycling and reuse within the nuclear industry

The recycling and reuse of decommissioned components and scrap materials from a nuclear installation within the nuclear sector has the potential to significantly reduce either the amounts of waste requiring disposal as radioactive waste or large scale unrestricted release campaigns, thereby offering significant savings in management and final disposal costs. The application of materials that are conditionally released within the nuclear industry includes:

- Fabrication of containers and overpacks for radioactive wastes, e.g., steel ISOs.
- Cementitious grout and backfills to infill intermediate level waste and low level waste packages.
- Incorporation into the reinforced concrete structures of radioactive waste repositories and storage facilities.
- Construction of waste processing equipment such as super-compactors.
- Backfilling materials for waste repositories.

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.

If the management of steels is considered, it is possible to say that work within the nuclear sector has been driven by efforts to allow unrestricted release of this material into the conventional marketplace and 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.

The prospect of utilising 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 wastes, rather than for the release of material to conventional recycling markets [1].

4.2.3.5 Low/intermediate level waste disposed in an appropriate repository or storage facility

The radioactive waste produced during operational and decommissioning activities in different nuclear facilities varies considerably by activity level, half-life, volume, physical and chemical nature. The IAEA waste classification system distinguishes between, on the one hand, short-lived and long-lived waste and, on the other hand, between low, intermediate and high level waste [7].

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. 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 and type of the radioactive waste package and the difficulty in handling the package during disposal.

The principal methods for disposal of low/intermediate level waste are surface or sub-surface disposal in facilities with appropriate engineered barriers. Disposal in deep geological repositories is generally envisaged only for intermediate level waste with significant quantities of long-lived radio-nuclides and for heat producing high level waste. The choice of the final disposal method is dependent on conditions specific for every country and on many other factors connected with the disposal system to be developed. Disposal facilities for low/intermediate level waste 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 on the other hand, on-site disposal needs to be consistent with the planned end state of the site [1].

4.2.3.6 Very low level waste disposed in an appropriate repository or storage facility

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

The benefit of designating very low level waste separately from low/intermediate level waste 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 low/intermediate level waste. The design and safety barriers of the very low level waste repositories are often similar to disposal facilities for hazardous or toxic waste. Moreover, the conditioning and packaging requirements of very low level waste are likely to be simpler and more inexpensive compared with low/intermediate level waste.

Disposal facilities for very low level waste can be located on the nuclear site at which it was generated or elsewhere. The advantages of on-site disposal are as for onsite disposal of low/intermediate level waste.

One of the more challenging problems in site clean-up is how to deal with large volumes of soils, or similar materials, containing low concentrations of radioactive materials which could potentially be designated as very low level waste. The traditional clean-up approach is to excavate the contaminated soil and dispose of it in a licensed radioactive waste disposal facility. However, many operating organisations 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. One of the possible solution is to construct on-site disposal cells with safety barriers corresponding to very low level waste repositories, in which large volumes of contaminated or potentially contaminated soils or other materials can be disposed [1].

4.2.4 Factors and constraints influencing the management of decommissioning material

There are many factors that may influence decisions in the area of materials and waste management. The factors affecting the decision what management strategy for the various material types with different characteristics should be finally adopted include:

- Quantities of materials.
- Technical feasibility and the availability of needed technology and infrastructure.
- Final status of the decommissioning activities.
- Radiological factors, application of the clearance principle.
- National policy, regulatory frameworks, public acceptance and legal issues.
- Economic considerations.
- Hazards and risks to people and the environment.
- Quality assurance (QA) and documentation.
- Environmental and ethical issues.

The factors mentioned above are in detail discussed in the following sections [1, 2]. Not all factors will apply in every case but several of them are likely to be relevant. Some may be deemed to be more important than others and may have a larger influence on decisions connected with decommissioning materials management. Consequently, some level of optimisation is likely to be done to determine which segregation, release, recycling, reuse and disposal practices will be applied within particular decommissioning projects.

When considering what material management 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. 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) and this complexity may increase costs compared to a simpler generic approach.

4.2.4.1 Quantities of materials

As mentioned in the previous sections, large quantities of materials will be generated during the decommissioning of nuclear facilities. This fact significantly influenced the demands on the necessary technologies, storage places, disposal facilities needed in the material management process. Also due to the large quantities of materials, there are strong requirements on the quantity and good qualification of personnel. A well developed system of material management flow is required too.

A significant proportion of the decommissioning 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.

4.2.4.2 Technical feasibility and availability of technologies and 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 (recycling, reuse, release, disposal) of materials from the decommissioning of nuclear facilities is essential to any waste management strategy. Technically feasible methods of decontamination, dismantling and waste processing should not give rise to large quantities of secondary waste generation, the further processing and disposal of which would involve substantial additional cost, negative impact on workers, the public and the 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. There are also numerous other technologies to support recycling and reuse options that are at laboratory or at pilot scale but these will require additional time, resources and efforts for further development to prove viability on an industrial scale. Also regulatory approval will be required to use these technologies in the nuclear sector.

The decision on whether to proceed with any material management option will be influenced 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 combustion).

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

- Type and composition of the characterised material, 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 detection limit.
- Radioactivity distribution on or within the material.
- Types of radionuclide to be measured and the presence and significance of difficult to measure radio-nuclides.
- Required confidence level.
- Cost and performance levels of available detection devices.

Special attention has to be given during the management of chemically toxic and other hazardous materials that could also be activated or contaminated. Therefore, their treatment, conditioning and disposal should consider both radiological and non-radiological hazards associated with these materials and wastes. Normal, dual purpose technologies for radioactive waste treatment combined with special methods and processes for destruction or stabilisation of chemically toxic materials provide a promising and demonstrated technical basis for the processing of toxic constituents in wastes. Few individual technologies can provide a combination of organic destruction, radionuclide immobilisation and other toxic materials immobilisation after transferring them into a less toxic form. During the final disposal, the chemically toxic components have to be taken into consideration. There is also a need to ensure that the substances in the wastes do not compromise the performance of the repository engineered and natural barriers.

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 these 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 or treatment.

4.2.4.3 Radiological factors applied to clearance levels and release practices

Clearance levels and release criteria may vary widely between countries. In some cases, criteria were based on prescriptive national regulations, while in others they were based on a case-by-case evaluation. Some countries (Finland, Belgium) have specified separate limits for alpha and beta-gamma emitters, while others (USA, United Kingdom, France) maintain nuclide specific limits. Some of the regulations specifically indicate that decontamination prior to clearance is considered acceptable (Belgium, Germany and the USA).

The variability in criteria applied in different decommissioning 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 and reuse principle. This inconsistency of values is due to the absence of an international agreement on rules concerning clearance values. 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. Clearance criteria must be based on reasonable assumptions with respect to dose and other hazards and associated risks and considered in the context of global optimisation, thereby helping to conserve the world’s non-renewable resources. If clearance criteria were excessively restrictive, large quantities of material would require disposal as radioactive waste, resulting in potentially greatly enhanced costs and changed environmental impacts.

What is more, many derived clearance levels are close to, or below, current limits of detection for practicable field instrumentation. Consequently, instrumentation and/or operational procedures which are expensive in both time and cost are required. Where this is not feasible, materials must be deemed to be above the clearance level and treated accordingly, again with significant cost and changed 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).

Aspect of non-nuclear hazards such as chemical toxicity on specific releases to controlled disposal facilities are, in most cases, limited by national environmental protection acts (Sweden, Germany etc.). Additional limitations have sometimes been imposed on the basis of the type of material (Canada), the disposal destination (France) or the conditions specified by inspectors (United Kingdom) [1, 2].

4.2.4.4 Radiological factors applied to clearance levels and release practices

4.2.4.4.1 Radiological factors applied to clearance levels and release practices

The expected end state of the decommissioning site has a significant impact on the material management strategy. Leaving buildings on-site for further use may prevent the generation of waste materials. If the site is intended for further nuclear operation, it may be possible to store radioactive materials 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 waste storage or disposal facility, if this option is acceptable by the authorities and the public. Indeed, the decommissioning strategy could be the entombment of the nuclear facility with the waste retained on-site, perhaps in situ.

4.2.4.4.2 Availability of radioactive waste disposal facilities

The availability or the access to fully developed treatment and disposal routes for large volumes of radioactive waste from decommissioning on a national or international basis may encourage the use of these disposal routes rather than pursuing recycling and reuse principles. 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 and reuse on environmental grounds. The unavailability of disposal facilities will promote the development of recycling and reuse technologies, particularly if large costs for construction and operation of storage facilities can be avoided.

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

4.2.4.4.3 Availability of radioactive waste storage facilities

Storage can be an interim solution for decommissioning materials and waste management, where it fits into the overall decommissioning strategy. This may be pending the availability of radioactive waste treatment technologies or final repository sites, or may be to allow later release for disposal or recycling as conventional material/waste as a result of radioactive decay. Storage may prove costly in the long run, unless disposal or other costs are thereby avoided.

Instead of building an interim on-site radioactive waste storage facility, existing buildings may be used provided that such an approach is consistent with the overall site decommissioning plan and time schedule [1, 2].

4.2.4.5 National policy, public acceptance and legal liability

4.2.4.5.1 National policy

National policy will provide a constraint over which material management strategies can be followed in any particular circumstance or may provide an indication as to which strategy may be preferred, all else being equal. The policy practices must be supported by a coherent dialogue among legislators, competent authorities, decommissioning operators and the public in order to gain acceptance for release practices and to promote options for the recycle or reuse of decommissioning materials, rather than for their disposal as radioactive waste.

Waste managers have no direct control over national policy. In spite of that, 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 radioactive waste management strategy and potentially the decommissioning plan as a whole.

4.2.4.5.2 Public acceptance

Depending on the national policy, it may be necessary to gain local or national public support for the preferred material management 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 waste management 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 whole decommissioning programme.

Recycling/reuse outside the nuclear industry and disposal/replacement each presents different public acceptance issues. Gaining public acceptance for the practice of recycling materials containing traces of radio-nuclides may be challenging because of the stigma associated with the nuclear industry in many countries.

However, products containing low levels of added or naturally occurring radioactivity are widely used and substantial quantities of radioactive scrap metal have been successfully recycled in a number of countries. Public perceptions of risk related to products containing radioactive materials (e.g., smoke detectors) are influenced by product familiarity, benefit and the extent to which the radioactive aspects of the product are publicised.

Notwithstanding, the large quantities of naturally occurring radio-nuclides released in the course of mining/refining of metals, petroleum, phosphate and coal, the public generally does not attach a nuclear stigma to these industries.

Repositories for very low level waste and low/intermediate level waste are also subject to similar public scrutiny and heightened sensitivity. Replacement/disposal options will present requirements for increased disposal capacity in excess of the capacity of currently operating facilities. Moreover, the siting and licensing of radioactive waste facilities has been the subject of intense political debate.

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

- Simple unified clearance system for deciding whether material is subject to regulatory requirements.
- Clear separation of political judgements from technical assessments. If political inputs are necessary they should be explicit.
- Maximum use of recycling within the nuclear industry.

Ultimately, public perceptions regarding the acceptability of material management 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.

4.2.4.5.3 Legal liability

When analysing and evaluating decommissioning material management strategies, waste managers 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 melting 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 material management strategy than another.

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 the regulations may attract a financial penalty but may also damage stakeholder confidence in the decommissioning process [1, 2].

4.2.4.6 Economic considerations

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

- The cost of recycling or reuse versus the cost of storage and disposal.
- The cost of processing materials, including removal, characterisation, decontamination, monitoring, size reduction, treatment, melting, transport, licensing etc.
- Price and marketability of the material, as determined by the availability of new resources and the alternative costs of new (basic) material.
- The availability of adequate funds to pursue the preferred option.
- Contingency funding required to offset the financial risk due to unforeseen events from causes such as legislative aspects, technical issues and public acceptance requirements.
- 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 characterisation and monitoring programmes largely depend on the chosen strategy. The more handling and the lower the target activity levels, the higher these costs will be. In general, the costs of clearance increase with decreasing permissible residual activity levels and the costs can be very high for a survey near state-of-the-art detection limits at high confidence level.

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 radioactive wastes) and the scrap value of the item or its component materials can be realised. 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 costs 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 costs of reclaiming the scrap can also be substantial and may include the labour, the material costs and the radiation dose:

- to decontaminate the materials.
- to treat and dispose of any secondary radioactive waste arising.
- to undertake the extra monitoring to select the items for recycling and to ensure that they are below the release limits.
- to decontaminate the treatment process equipment [1, 2].

4.2.4.7 Hazards and risks to people and the environment

Decommissioning on a nuclear site and material management activities may involve both radiological and other more conventional hazards for workers, the public and the environment. The radiological health risks from either of the material management options are relatively low, whereas risks from other hazards may be potentially high. Some options may carry health risks from workplace and transportation accidents, as well as exposure of workers and the public to chemicals that are carcinogenic or toxic. The overall hazards and risks need to be considered when selecting a material management strategy, not simply focusing on radiological issues.

Many aspects of disposal and further needed materials replacement processes are connected within the environment that are less stringently regulated than the environment in which recycle/reuse alternatives would operate. Replacement necessarily involves coal mining, iron ore mining and coke production, occupations that have relatively high accident rates. Moreover, because of the multiple stages involved in replacement/disposal practices, transportation requirements usually exceed those associated with recycle/reuse practices. Replacement must consider not only shipment of wastes, but also transportation of the coal and ores necessary for steel production. Consequently, the risks to workers from disposal and replacement alternatives exceed those from recycling alternatives.

Similarly, the potential for adverse environmental impacts is also much higher for disposal and replacement alternatives. Although recycle and reuse alternatives will impact the environment by utilising relatively small amounts of radioactive waste disposal capacity, replacement/disposal presents adverse impacts of greater severity to the environment from land use, disruption and the damage that results from ore mining and related processes [1, 2].

4.2.4.8 Quality assurance and documentation

A well planned quality assurance (QA) programme in accordance with the applicable codes, standards and jurisdictional requirements to cover the material management activities should be available. Such quality assurance programme should ensure the required transparency in sampling, analysis, monitoring, documentation, interpretation and use of data generated for a selected material management option in order to demonstrate compliance with the prescribed regulations, to enable a regulatory authority to verify the results of the material management activities and to enhance public acceptance.

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.

Transparency and traceability, two key components of a quality assurance programme, are crucial for clearance as quality assurance 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. In cases where the radioactive materials remain under regulatory control, the need for accurate records is important for any future decision.

A constraint on the material management strategy may be the availability of sufficient numbers of suitably qualified, trained and experienced personnel, particularly at times of peak demand. The minimum personnel qualification and training requirements should be clearly specified in the quality assurance 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 facility decommissioning phase [1].

4.2.4.9 Environmental and ethical issues

4.2.4.9.1 Ethical aspects and sustainability

Policies for the long term protection of the environment stem from ethical considerations that the current generation should protect the environment for future generations. This is the basic concept behind sustainable development and combines environmental issues and socio-economic priorities. The present generation may benefit financially and achieve higher standards of living from nuclear practices 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.

Minimisation of radioactive and clean wastes is a regulatory requirement in many countries. 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 wastes are responsible for managing these wastes safely and responsibly. These responsibilities are not limited to bearing the costs of managing and disposing of the wastes by themselves, but also research and development costs undertaken by themselves and by the regulatory bodies.

Consequently, all of the final disposition options considered in the foregoing sections need to be assessed for compliance with sustainability goals. The 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.

4.2.4.9.2 Global optimisation

In general, a material management option 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 optimised.

The overall objective of any material management option should be to reduce the environmental impact of the materials, as well as the total costs involved. When considering global optimisation, it is important to consider the costs of the individual contributions to obtain a net benefit status for the selected material management option. In practice, it is usually a trade-off between the benefits accruing from the programme and the costs to achieve these benefits. A full analysis should not only include radiological impacts, but also the risks and the environmental impacts associated with the selected option. To improve the effectiveness of choice between various material management and waste management alternatives, in addition to radiation protection, a global optimisation should be considered, including a broad range of issues, 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.

Overall, ecological and sustainable development considerations promote the reutilisation of non-renewable resources by way of direct reuse of equipment or buildings and by recycling of useful materials [1].

4.2.5 Methodology for decision making in view of selecting a material management option based on specific influencing factors

As mentioned in the previous sections, various factors can influence the decision as to which a material management strategy should be adopted and some level of optimisation is required as, on a case-by-case basis, the relevance and relative importance of these factors may differ.

When evaluating the various influencing factors for a specific material management option, a linear decision tree approach could be adopted as indicated in Section 2.

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 material management options 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 each option. An example of a decision matrix is given in Table 4.5 [1, 2].

Table 4.5 Example of a decision matrix with available final disposition options

Option	Costs		Technical feasibility		Hazards and risks		Availability of disposal		Environmental impact		FINAL SCORE
	Weight	Score	Weight	Score	Weight	Score	Weight	Score	Weight	Score	
Unconditional release – free 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 %
Unconditional release – disposal as conventional waste	V_2 %	C_2	W_2 %	F_2	X_2 %	R_2	Y_2 %	D_2	Z_2 %	I_2	Σ_2 %
Conditional release – 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 %
Conditional release – 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 and disposal as LLW/ILW	V_5 %	C_5	W_5 %	F_5	X_5 %	R_5	Y_5 %	D_5	Z_5 %	I_5	Σ_5 %
Storage and 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 %

4.3 Radioactive waste minimisation

4.3.1 Definition of the radioactive waste minimisation principle

Radioactive waste minimisation is defined as the minimisation of the generation and the spread of radioactivity and the minimisation of the volume of radioactive wastes arising from the management of materials from decommissioning operations, to levels ‘as low as reasonably achievable’” (ALARA) both safety and economic factors being taken into account. This involves minimisation of their impact on the environment and reduction in overall decommissioning costs.

It should be considered that the ultimate goal of the decommissioning operations is the unconditional release or reuse of sites, facilities, installations and materials for other purposes. Intermediate products of decommissioning process can be facilities or installations that have been dismantled, or materials that have been decontaminated to permit their release or reuse for other nuclear applications. Materials that cannot be conditionally or unconditionally released or reused, and which have to be treated and conditioned as radioactive wastes, can be considered as by-products of the decommissioning activities. The radioactive waste minimisation process can be considered as a strategy for avoiding, as much as possible, the production of these undesirable by-products. Where mentioned by-products are unavoidable, steps are required to minimise their amounts and volumes [2].

4.3.2 Impact of decommissioning strategies on the waste minimisation process

When selecting a decommissioning strategy, the following technical, regulatory, economic and social considerations have to be taken into account. Some of which form part of the main elements of a radioactive waste minimisation strategy:

- The material condition of the nuclear facility after final shutdown, which involves an evaluation of the state of the technological equipment and building structures, making allowance for how this will change in the long term. The material condition defines the maintenance, surveillance and inspection requirements necessary to keep it in a safe shutdown state for the required period, avoiding degradation of equipment, structures, minimising the spread of contamination and preventing decommissioning operations from becoming more difficult in a later phase.
- The radiological condition of the facility, which involves assessing the potential hazards, either while work is going on or during waiting periods. This will provide guidance on waste management and the waste minimisation options to be adopted.
- Constraints due to nuclear safety and radiation protection, industrial safety and the related risk analysis studies make it possible to evaluate the best means of protection, to assess how the radiological aspects can be optimised and to determine the requirements of maintenance, inspection, monitoring and surveillance. The possible deterioration of equipment, structures and containment should also be considered, as well as minimising the spread of contamination and preventing decommissioning operations from becoming more difficult in a later phase.
- The availability of a radioactive waste management infrastructure, which includes treatment and conditioning equipment, storage and disposal facilities. Evaluation of the different amounts of radioactive materials which will be produced by the dismantling operations is also included (i.e., their characteristics, quantities, production rates, etc.).
- The services concerned with operation, maintenance, instrumentation and surveillance to guarantee safety and to keep the equipment remaining in service (handling equipment, electrical supplies, ventilation, radiological surveillance instruments, fire monitoring, etc.) running and properly maintained, with particular attention paid to those parts of the plant which may deteriorate over the long term.
- The possibility of reusing the site and buildings and of recovering plant, equipment and materials for nuclear or other industrial purposes (without neglecting the social and political aspects), which presents important incentives for considering decontamination practices and significantly reducing the potential amount of radioactive wastes remaining.
- The existence of technical resources, qualified personnel and local support for dismantling, decontamination and contaminated material handling, which includes considering the available means of waste minimisation and evaluating how existing facilities on-site can be modified to meet the needs with minimum expenditure.

- The costs and financing, as knowledge of the cost of each possible approach is needed, including the cost of labour, materials and supplies, as well as financing costs and cost savings involved when applying waste minimisation principles and techniques.
- Social considerations, which include public perception of radioactive waste treatment versus recycle and reuse options, that is usually taken into account in the procedure whereby proposals are submitted for approval by the safety authorities.

These factors, when considered, should facilitate the choice of decontamination and dismantling tactics and should result in the most appropriate way of dealing with the contaminated materials produced.

Tactical decisions take account of the regulatory constraints and the specific features of the installation to be decommissioned. Within a given strategy, it is necessary to determine the tasks that need to be carried out in order to determine the technical approaches for their implementation, and to manage these tasks in order to optimise the balance of costs, time schedule, waste minimisation and worker doses. In this stage, the main technical approaches are considered such as:

- To decontaminate or fix contamination.
- To use equipment in-situ or in a site workshop.
- To handle radioactive materials directly on-site or in centralised facilities.
- To cut materials into large pieces, adopting large piece disposal (e.g., steam generators, reactor pressure vessels) or additional size reduction in specialised areas or cut radioactive components directly in-situ taking into account the transfer and disposal requirements.
- To choose means and modes of access to working areas and decide the routing of contaminated materials.
- To identify suitable manipulation/handling equipment (robot manipulators, carriers, etc.).
- To determine methods to be used for protection, safety and security.

Taking technical decisions involves choosing the most appropriate technical facilities with which to carry out the operations as determined by the tactical decisions taken, including the choice of cutting tools and remotely controlled systems, processes for decontamination and for management of radioactive materials and effluents, and methods of radiation protection and industrial safety.

On the basis of the foregoing, it should be clear that waste minimisation is an inherent and important part of any decommissioning strategy.

4.3.3 Fundamental principles of waste minimisation

The objectives of waste minimisation are to limit the generation and spread of radioactive contamination and to reduce the volume of radioactive waste for storage and disposal in the radioactive waste repositories, thereby limiting any consequent environmental impact, as well as the total costs associated with contaminated material management. The main elements, described in the following four sub-sections, of a waste minimisation strategy can be grouped into four areas that are [2]:

- Radioactive waste sources reduction.
- Prevention of contamination spread.
- Recycle and reuse of materials.
- Waste management optimisation

The above four elements important for any waste minimisation process define four fundamental principles which should be considered when planning and implementing the waste minimisation programme. These fundamental principles can be summarised as follows:

- Keep the generation of radioactive waste to the minimum possible or practicable.
- Minimise the spread of radioactivity leading to the creation of radioactive waste as much as possible by containing it to the greatest extent possible.
- Optimise possibilities for recycle and reuse of valuable material components from existing and potential waste streams.
- Minimise the amount of radioactive waste that has been created by applying adequate treatment technology.

4.3.3.1 Environmental and ethical issues

The generation of radioactive waste during the decommissioning of nuclear facilities (including secondary waste) needs to be kept to the minimum practicable, in terms of both its activity and its volume. This requires the selection of appropriate decommissioning strategies, timely and overall work preparation, application of efficient decommissioning practices, tools and procedures, and by appropriate training and qualification of personnel. Emphasis needs to be placed on the effective segregation of different types of materials at their place of production in order to reduce the volumes of radioactive waste and facilitate their management.

4.3.3.2 Environmental and ethical issues

A key aspect of minimising the generation of radioactive materials is to minimise the spread of radioactive contamination whilst undertaking decommissioning activities. This will help to reduce the production of contaminated materials and minimise the need for decontamination and hence the creation of secondary radioactive wastes. All practicable means of preventing contamination should be used, provided that they are economically justified and do not lead to unacceptable additional risks and complications in other decommissioning activities.

4.3.3.3 Environmental and ethical issues

Consideration of the amounts of material arising from the decommissioning of a nuclear facility highlights the importance of recycle and reuse within a waste minimisation strategy. In addition, considering the ultimate goal, it means unconditional release or reuse of sites, facilities or materials for other purposes, opportunities for release or recycle/reuse of materials should be maximised. Implementation of the recycle and reuse principle requires the existence of suitable clearance criteria and the availability of measurement methodologies and instrumentation. Initiatives to support waste minimisation in the decommissioning process should be set up in order to promote options for recycling or reusing materials, rather than to restrict this practice.

4.3.3.4 Environmental and ethical issues

In addition to reducing the amount of radioactive waste generated, it is also important after generation to minimise their volumes by appropriate treatment. The volume of radioactive waste resulting from decommissioning operations may be reduced by increasing the use of volume reduction processes that include compaction, incineration, filtration, and evaporation. Waste volume minimisation will extend the operating life of current disposal sites, limit the need for interim storage, if disposal is not available, and reduce the number of waste shipments [2].

4.3.4 Practical implementation of the fundamental principles

Practical implementation of the mentioned fundamental principles can be achieved using administrative or organisational arrangements and technical approaches as given below. It must be re-emphasized that the first step of any waste minimisation strategy is to keep the generation of radioactive wastes to a minimum. Application of adequate waste management technologies should be considered as a final step, when the creation of radioactive waste is unavoidable [2].

4.3.4.1 Operation culture

The minimisation of radioactive waste can be most easily achieved by minimising opportunities for the creation and spread of radioactivity. The establishment of an appropriate policy and culture to achieve this is the primary responsibility of the management of the nuclear facility, during both operation and decommissioning. The management must create this culture through leading by example and by creating a consultative team environment. This involves not just applying management procedures, but also education of the workforce to instil in all appropriate understanding of the problems, attitudes, behaviour and personal responsibility.

Preparing efficient work plans with adequate work organisation and selection of appropriate tools, instruments and materials will limit potential cross contamination, provide optimised working times and minimise the generation of secondary waste from intervention work.

Each worker needs to minimise wastes generated in the tasks assigned to him. This involves training workers in waste minimisation and contamination control techniques and empowering them to identify and apply new ways of minimising all wastes. In decommissioning operations, major reductions in waste generation can be achieved through training and administrative controls, contamination control tenting and confinement, and decontamination of selected material and components to releasable or reusable levels, etc.

Contamination control tenting needs to be used wherever there is potential for airborne contamination. Tents need to be fitted with cleanable pre-filters and high efficiency particulate air (HEPA) filters with adequate air flow away from the worker in order to prevent inhalation. Plastic sheeting covered with absorbent pads (reusable if available), can be used to minimise the spread of activity under all pipe cuts, where a potential exists for liquid spills.

Components and tools used in the work may be decontaminated to clearance levels by one or more techniques. Tools decontaminated and cleaned in this manner may be reused repeatedly. In each case, the cost-benefit needs to be evaluated [2].

4.3.4.2 Administrative controls and management initiatives

Administrative controls and management initiatives in nuclear facilities can contribute significantly to an adequate waste minimisation strategy. Steps, which can be taken, include:

- Collection and update of all information related to the design of the plant (drawings, material specifications, various modifications and implementations) and also information related to the operating performance during operation of the nuclear facility. The development of computerised database techniques can solve the problem of preserving and updating the information needed for performing decommissioning.
- Establishment of an organisational structure which ensures that the responsibilities for all aspects of contaminated material management are clearly defined and assigned.
- Establishment of an accounting and tracing system to quantify the sources, types, amounts, activities and final dispositions of contaminated materials.

- Identification of all points in the working areas and all stages in the process where it is possible to prevent materials from becoming radioactive or radioactively contaminated, e.g., by excluding packaging, by using recycled materials in the process or by making equipment changes.
- Improvement of operational practices and management techniques and exchange of information and experience on sorting and segregating wastes at their sources in order to prevent mixing of different waste categories.
- Provision for the comprehensive education of operators. Through introductory courses and regular courses, the attempt must be made to foster operator awareness of the need to keep the generation of contaminated materials to a minimum.

In order to assist the operators of nuclear facilities or decommissioners in minimising contaminated material consistent with safe operating practices, the relevant regulations must be in place and enforced.

The implementation of any waste minimisation strategy is always an optimisation exercise taking into account factors such as worker doses, costs of recovering materials generated, disposal routes available for specific types of contaminated material, quantities of material generated in each category and duration and costs of interim storage of wastes compared with the estimated ultimate disposal costs. Ideally, a waste minimisation strategy should be considered at the planning stage of any process development [2].

4.3.4.3 Technical factors to avoid the production of radioactive materials

The significant technical factors to consider in avoiding or minimising the production of radioactive materials are:

- Design of facilities and technological equipment.
- Choice of construction materials (especially for the components in contact with primary coolant).
- Maintenance of installations and systems during the operational period.
- Cleanliness and decontamination (electro-polishing of metals, coating for porous materials).

These factors have a function of preventive actions during the planning and operation phase used to avoid or to minimise the generation of radioactive materials during decommissioning. But, the current decommissioning activities generally involve facilities that were designed in the past without adequate consideration of these factors [2].

4.3.4.4 Characterisation of materials during decommissioning

Proper characterisation and segregation of materials arising during decommissioning activities are very important factors in the waste minimisation process. Characterisation helps develop a complete understanding of the physical, chemical and radiological characteristics of these materials in order to enable them to be segregated. Segregation favours the maximisation of unconditional release, allows consideration of conditional release, reuse or recycling of materials permits reduction in the radioactive wastes that should only be disposed in the repositories.

4.3.4.5 Measurement techniques and infrastructure

Measurement of radioactivity in materials during decommissioning is very important, both for characterisation/segregation of materials and for providing proper control before release of materials (conditional or unconditional) after dismantling or decontamination.

Techniques for the measurement of radioactivity can be organised into three general groups:

- *Direct measurements* are taken using a radiation detector positioned near contact with the surface or object to be measured. The choice of the detector depends on a number of factors, including the type of radiation to be measured, the size and shape of the object or surface to be measured and the anticipated level of radiation. Direct measurements may be used in waste minimisation for the final release of buildings, objects and materials, provided that there are no circumstances which reduce the confidence of the direct measurement to unacceptable levels (inaccessible surfaces, radioisotopes that cannot be efficiently measured, high radiation backgrounds, etc.).
- *Indirect measurements* are taken using a paper smear to swipe a known area of a surface or object in order to assess whether loose contamination is present. The smear is evaluated for contamination by performing a near contact measurement of it using suitable radiation detectors. Some of the technical challenges associated with smears involve the estimation of pick-up efficiency (as not all of the loose contamination on a material will be removed), and the fact that the measurement, once taken, is not reproducible.
- *Measurements by sampling* are characterised by the fact that materials are analysed by laboratory processes (i.e., chemical separation, and alpha, beta and gamma spectrometry) in order to identify the radio-nuclides present in the material and to determine their concentrations. Samples are often taken before direct and indirect measurements, in order to determine which radio-nuclides, and their relative proportions, are to be expected in the material. Sampling methods may be very useful for measuring the radioactivity of objects having inaccessible surfaces. Uncertainty is introduced because of the challenges associated with obtaining representative samples from surfaces, objects and materials. Samples are also used to develop scaling factors for radio-nuclides in materials in order that an easily executed direct measurement (i.e., gamma dose rate due to ^{137}Cs) may be used to infer the concentrations of other radio-nuclides that were generated in a similar fashion (i.e., other fission products) [2].

4.3.4.6 Release of materials

A further requirement of waste minimisation during the decommissioning process is the maximisation of the amount of material which can be released either for unconditional or conditional recycle or reuse within and outside the nuclear industry.

During the decommissioning of nuclear facilities, quantities of valuable metals and equipment may become available for recycle or reuse, provided that the radioactivity on or in them can be reduced to acceptable levels. Current work is aimed at defining the basis for establishing suitable criteria for unconditional release of materials and equipment and for applying these criteria to actual waste management and decommissioning cases. In some examples the case-by-case principle is implemented in materials releasing.

Melting of scrap metals is a promising method of metals conditioning with the purpose of achieving volume reduction, immobilisation of radioactivity in homogenous products and possible recycling of the steel. During the melting, the reduction in activity of the final product results from the decontamination processes employed before melting and by the partition of the radio-nuclides within the melt, slag and dust generated during the melting operation.

In order to facilitate the release practices, suitable criteria, measurement methodology and instrumentation must be available. This is particularly important for material destined for unconditional release, which usually represents a very large proportion of the material produced. The establishment of unconditional release criteria is a critical step towards satisfying the need for a consistent, internationally accepted standard.

Another important factor which must be considered in the unconditional reuse or recycle of material is the economic impact. The monetary benefits of recycling or reusing materials are relatively obvious. There are no disposal costs for the materials and the scrap value of the item or its component materials may be realised.

On the other hand, the monetary costs of reclaiming the scrap can be significant and include:

- Material and labour costs of decontaminating the materials.
- Costs of treating and disposing of the secondary radioactive waste arising, e.g., from decontamination or melting.
- Cost (labour and radiation dose) of undertaking the extra monitoring needed to select the items for recycle and of ensuring that they are below the release limits. The cost of this activity will increase as the level of the acceptable limits for release decreases and the main problems with very low level waste are the measurability of the inherent activity and verification of the measuring techniques used [2].

4.3.4.7 Processing of radioactive waste for volume reduction

Contaminated materials that are not subject of the waste minimisation techniques discussed previously have to be considered as a radioactive waste.

Radioactive waste processing includes handling, treatment, conditioning, packaging, storage, transport and disposal. The methods of processing the waste arising from decommissioning will, in general, be similar to those used in other parts of the nuclear industry. However, owing to the specific nature of certain decommissioning wastes, special consideration may be necessary in certain areas. The main purpose of radioactive waste processing is, in the first step, waste volume reduction using the technologies of compaction or incineration and then immobilisation of the waste into suitable form that should be disposed to the radioactive waste repository site.

The major requirement of any radioactive waste management strategy is to guarantee the safety of the waste operations. Detailed consideration needs to be given to the types of wastes and to the packaging, transportation and disposal requirements. The waste forms and packaging have to comply with national transport regulations and with the acceptance criteria at radioactive waste disposal sites [2].

A more detailed discussion about radioactive waste processing (including decontamination techniques) is available in Section 4.4.

4.4 Processing of radioactive decommissioning waste

By appropriate management of primary radioactive waste, a considerable reduction of waste volumes can be achieved. In order to realise this option, it is important to ensure the support and commitment of the top management of the operating facility. Some of the steps that can be taken are the following:

- Characterisation by proper methods (the source of the waste, type, category and physical and chemical properties), accounting and tracking systems (the date of generation, amounts, location) of the waste generated.
- Selection of suitable technologies for radioactive waste pre-treatment, treatment and conditioning. Providing an appropriate storage and disposal facilities.
- Determination of the life cycle costs of various waste management process stages to identify the true costs for the management of material arising.
- Encouraging the exchange of technical information on waste management with emphasis on waste minimisation from other companies and/or institutions to share best practices.

In addition, non-releasable radioactive wastes need to be treated according to the type of waste, concentrations of radio-nuclides and requirements for waste storage and/or final disposal.

The choice of a waste management processes and options will depend on a variety of parameters including:

- The physical, chemical and radiological properties of the radioactive waste.
- The type of waste management technologies and processes available.
- The location of the requisite processing equipment, e.g., whether the treatment facilities are on-site or at a nearby location.
- The transportation, storage and disposal alternatives available.
- Economic considerations.

In the following sections, the individual technological steps, as indicated in Figure 4.3, of the radioactive waste management process are described in more detail.

The characteristics and typical types of radioactive wastes arising from the decommissioning of different nuclear facilities has been given in Section 4.1.

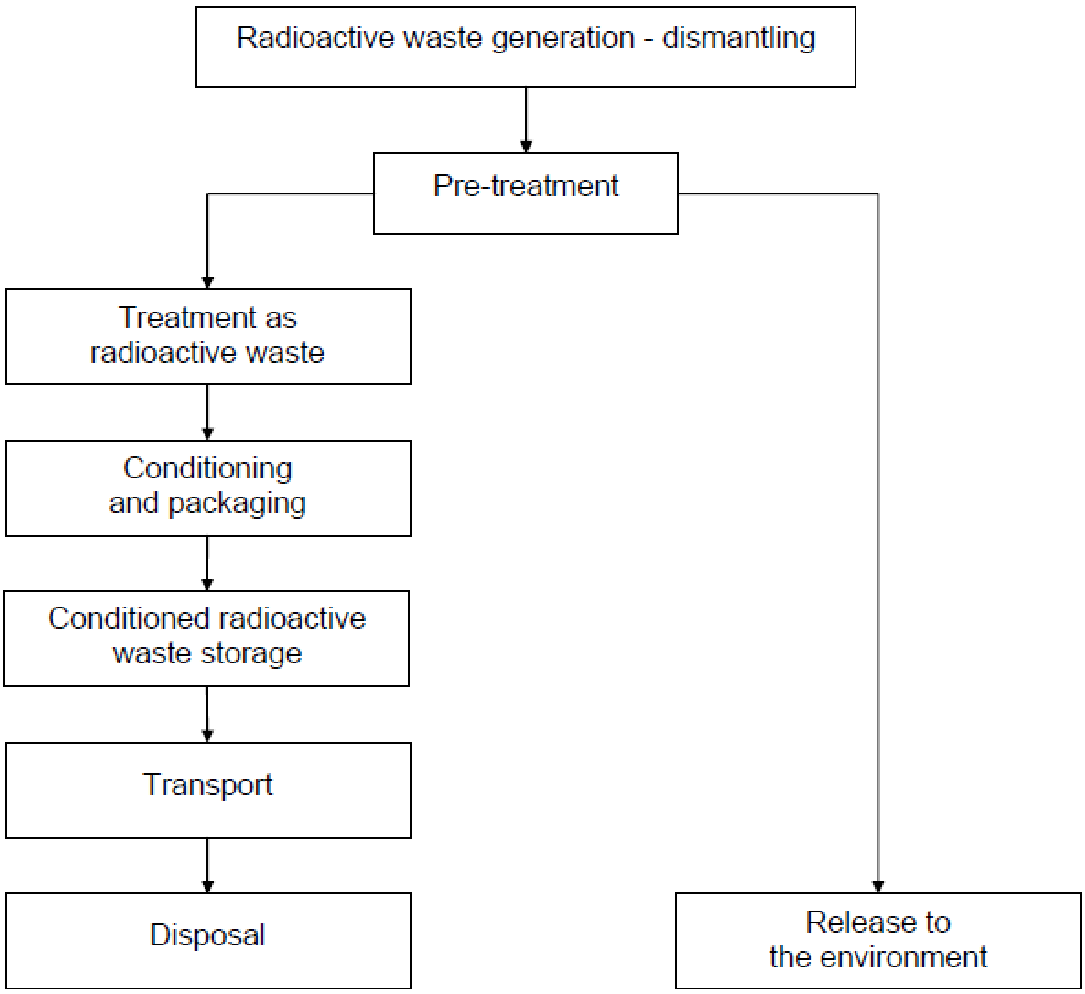

Figure 4.3 Technological steps in the radioactive waste management

4.4.1 Pre-treatment of radioactive waste

Where possible, the radioactive waste is pre-treated to provide appropriate preparation and to facilitate subsequent waste treatment steps by reducing the risk arising from physical, chemical and radiological characteristics of the radioactive materials. The main goal of the pre-treatment operations is to minimise the amount of the further processed radioactive waste and to maximise the volume of released materials (see also Section 4.3). Pre-treatment steps could include:

- Administrative steps, including documentation of the details of the waste for accountability and operational purposes.
- Collection, segregation, sorting and characterisation of the waste in order to classify it for further suitable treatment.
- Basic chemical treatment to reduce the chemical toxicity of the waste if necessary.
- Size reduction and fragmentation of the primary radioactive waste if needed.
- Packaging in containers (bags, drums) that are suitable for transport to the treatment area.
- Decontamination, in order to facilitate further handling and treatment or to release the materials.
- Intermediate decay storage, which allows decay of short lived isotopes and thereby facilitates subsequent steps.

4.4.1.1 Collection, segregation, sorting, characterisation and packaging

Collection practices for waste normally consist of distributing suitable containers or cans (tanks) for liquid radioactive waste, throughout the working area to receive discarded radioactive materials. The containers should be marked with brightly coloured paint (normally yellow) and the radiation symbol to distinguish them from bins meant to receive inactive waste. Multiple containers, one for each category of waste produced at a facility, are often used to encourage segregation at source.

Segregation more specifically refers to an activity where waste or materials (radioactive or cleared) are separated or are kept separate according to radiological, chemical and physical properties which facilitate waste handling and/or processing including disposal (conventionally or as radioactive waste). It may be possible to segregate radioactive from cleared material at the point where it is generated and thus reduce the radioactive waste volume. The sorting and segregation of both solid and liquid wastes with different characteristics is widely practised by generators. It provides a number of advantages:

- Simplifying documentation requirements for shipment and handling by minimising the volume of low and intermediate level radioactive wastes with toxic constituents.
- Simplifying treatment by limiting the input to any process to those items it is specifically designed to treat the radioactive waste.
- Simplifying disposal of wastes for both paths conventional or radioactive by segregating them according to disposal waste categories and disposal acceptance criteria.

Wastes that were generated and simply stored without regard for their treatment requirements may require re-sorting in order to be compatible with new treatment technologies or with disposal site acceptance criteria. However, the sorting of stored wastes may be disadvantageous as it exposes staff to radioactive and physical hazards during the sorting process.

Experience shows that even at sites where waste treatment facilities are not yet available, it is highly recommended to start to segregate the waste from the beginning. Non-segregated bulk storage makes final waste treatment inevitably complicated.

Proper sorting requires staff training for segregation at the various points including generation and treatment. Sufficient additional space and containers in the waste collection and storage areas are required.

Characterisation derives from the knowledge of the radiological state of an installation both before and during decommissioning. In practice, this involves:

- drawing up maps of contamination and of other radiation sources in order to determine the tactics and resources to be used for the decommissioning of the installation.
- continuously measuring dose rates and air contamination levels in the working area with a view to make the necessary arrangements for protecting workers and the environment.
- checking the outcome of decontamination operations.
- segregating and packaging the materials resulting from dismantling and decontamination.
- monitoring the waste packages to ensure that they conform with the requirements for further treatment and/or with the regulations governing transport, storage and disposal.
- measuring and documenting the radioactivity of materials, buildings and land, when these are to be returned to the public domain.

A thorough knowledge of the operating history of the nuclear facility will also aid in the selection of measurement methods and equipment and in the development of measurement and characterisation procedures.

The following information is also required as part of an appropriate waste characterisation:

- Type of emitter (alpha, beta, gamma, X ray).
- Source of emission (loose or fixed contamination, induced radioactivity).
- Level of radioactivity.
- Potential for interference from several sources of radiation.
- Physical state (solid, liquid or gas).
- Geometry, surface area and volume of components to be measured.
- Chemical composition and characteristics [2].

To characterise radioactive waste properly, it is necessary to have suitable instruments and well developed techniques and infrastructure for the measurement of the waste characteristics.

Whenever possible, these operations (collection, segregation, sorting, characterisation) should be carried out at the source of the waste production and they should be put (after size reduction, if necessary) with the packaging into appropriate containers (usually plastic bags or 200 litres metal drums) bearing adequate indication (labelling and colour coding) for the subsequent management steps and treatment.

4.4.1.2 Size reduction

Some wastes may require reduction in size or shape changes to be acceptable for feeding to technological equipment for treatment. Size reduction should normally be carried out in the treatment facility. However, it can be advantageous to start at the source, whenever this is possible.

Shredding, chopping, shearing, grinding, or torch cutting can be used for reducing waste volumes for disposal and also for sizing of wastes for feeding to the treatment processes. Larger pieces of rubble type waste, such as spent process equipment, may be cut with remote saws or other techniques. In some cases such size reduction techniques may enhance the process effectiveness by providing increased surface areas for reaction. Furthermore, size reduction can also reduce waste handling, processing, storage and shipment costs. However, grinding dust and saw chips may be hazardous and may require the introduction of secondary waste control. There may also be increased hazards to workers due to the use of cutting tools and the physical hazards of handling the sharp and heavy materials [8].

4.4.1.3 Decontamination

Decontamination is defined as the removal of contamination from the surfaces of facilities or equipment by washing, heating, chemical or electrochemical action, mechanical cleaning, or other techniques. During decommissioning there are two main reasons for considering the use of decontamination techniques:

- Reduction of the radiation exposure (dose levels) before dismantling (pre-dismantling decontamination).
- Reduction of the contamination of technological components (post-dismantling decontamination) or building structures (building surfaces decontamination) to such levels that wastes can be managed in a lower, and therefore more economical, waste treatment and disposal category or disposed (released) as waste (materials) exempt from regulatory concern.

A decontamination programme may also require a facility capable for treating secondary wastes from decontamination, e.g., processing chemical solutions, sludge, aerosols, debris. The concentrated wastes, representing a more significant radiation source, must be solidified and shipped for disposal in licensed radioactive waste disposal facilities unless properly treated within the waste reduction/recycling/reclamation processing alternative.

The requirements for post-dismantling decontamination in the decommissioning period are quite different to those required for maintenance decontamination in the operational period. During decontamination for maintenance, components and systems must not be damaged and the use of very aggressive decontamination methods is not appropriate. However, in decontamination during decommissioning, it is mainly the use of somewhat destructive techniques that present the possibilities of meeting the objectives to release the materials at clearance levels. Another aspect is the need for industrialisation. The large amount and various types of contaminated materials produced during decommissioning procedures and available for decontamination, generally do not favour methods or techniques that are labour intensive or difficult to handle or that present difficulties when automation is envisaged.

4.4.1.3.1 Selection of decontamination techniques

Very early in the process of selecting decontamination technologies for decommissioning, it is important that a cost-benefit analysis will be done to see, if it is worth decontaminating the component or facility, or to determine whether a mild decontamination at low cost is more advantageous than a severe decontamination at a higher cost. This analysis is usually accompanied by extensive experimental work on selected samples from the facility in view of characterisation before the final choice of a decontamination technique is made.

To achieve a good decontamination factor (DF), a decontamination process must be designed for site specific application taking into account a large variety of parameters:

- Type of nuclear facility (nuclear power plant, reprocessing plant, etc.).
- Operating history of the facility (duration of operation, number and seriousness of accidents).

- Type of material (steel, color metals. Zircaloy, concrete, etc).
- Type of decontaminated surface (rough, porous, coated, etc.).
- Type of contaminant (oxide, crud, sludge, loose, etc.).
- Distribution of contamination (surface, cracks, homogeneous distribution, etc.).
- Nuclide composition and specification of the contaminant (activation products, fission products, actinides).
- External or internal surface to be cleaned.
- Required decontamination factor.
- Quantity and type of the secondary radioactive waste.
- Destination of the components being decontaminated (disposal as radioactive waste, disposal as industrial waste, recycling, reuse).
- Time required for application.
- Proven efficiency of the process for the contamination in the facility.
- Type of decontaminated component (pipe, tank, etc.).

Other factors which are important in selecting the method but which do not affect the required decontamination factor are:

- Availability, cost and complexity of the decontamination equipment.
- Need to condition the secondary waste generated.
- Personnel and public doses resulting from the decontamination process.
- Availability of trained personnel.
- Extent to which the plant needs to be decontaminated to achieve acceptable conditions for decommissioning.
- Salvage value of materials which would otherwise be disposed.
- Extent to which the facility must be modified to do the decontamination (making decontamination loops, system isolation, enclosed and ventilated spaces, etc.).
- Other safety, environmental and social issues.

In addition, the choice of a process or of a combination of several processes will finally depend on several other factors such as the:

- Specific nature of the application, the complexity of the decontamination system.
- Feasibility of industrialisation.
- Cost-benefit analysis taking into account all aspects of the decontamination operation.

4.4.1.3.2 Characterisation of decontamination techniques

In the following section, the characterisation, advantages and constraints of some techniques used for the decontamination of technological equipment that is used during radioactive waste pre-treatment in order to release the materials to the environment, or to facilitate the parameters for further radioactive waste treatment [2, 8].

- *Chemical decontamination*:

  Chemical decontamination is a relatively simple well known practice in many nuclear facilities. It comprises the use of concentrated or dilute solutions of different reagents in contact with the contaminated item to dissolve either the metal substrate or the contamination layer covering it. The required decontamination levels can be obtained by continuing the process as long as necessary, care being taken to ensure that tank walls or piping are not penetrated by corrosion. Chemical decontamination is usually carried out by circulating the appropriate reagents in the circulating loops or systems. However, segmented parts can be decontaminated by immersing them in a tank containing the reagent, which must then be agitated.

Chemical techniques are generally suitable for use on complex geometries, as well as for the uniform treatment of the inner and outer surfaces of equipment (e.g., pipes, tanks), where good contact between the chemical and the surface is provided. It can also remove radioactivity from hidden surfaces. However, in this case its effectiveness may be low and measurement at release levels will be a problem. Also, the chemical decontamination is usually not effective on porous surfaces.

With a proper selection of chemicals, it can remove all the radio-nuclides from the decontaminated surface. Problems of recontamination can be reduced by continuously rinsing the surface with water.

Mild chemical decontamination techniques have generally been used for items where the main purpose is to remove contamination without attacking the base material. Their advantages are low corrosion rates and low chemical concentrations which ease the problem of treating the spent decontamination solutions. Mild decontamination techniques have low decontamination factors and require long contact times, however.

Aggressive chemical decontamination techniques can involve one or more stages using different chemical solutions with intermediate rinses. Process advantages include short time application and high decontamination factor (usually 10 - 100) and the clearance level should be easily reached, but the reuse of material is often impossible. Process limitations include high chemical concentrations and the creation of potential problems for effluent treatment systems.

The main disadvantage of chemical decontamination is the generation of relatively high volumes of liquid secondary radioactive waste compared with other processes such as electrochemical decontamination. Spent decontamination solutions have to be treated and disposed as a radioactive waste. Also in some cases, corrosive and toxic reagents may need to be handled. The regeneration of chemicals has become a fundamental step in all chemical decontamination processes. Several conventional chemical processes can be used for regenerating the spent solutions, either on their own or in combination, including ion exchange, evaporation/distillation and electro-dialysis.

- *Electrochemical decontamination:*

Electrochemical decontamination (electro-polishing) can be considered in principle to be a chemical decontamination assisted by an electrical field. Electro-polishing is a process widely used in non-nuclear industrial applications to produce a smooth polished surface on metals and alloys. Major equipments are relatively inexpensive and the technological procedures are quite simple.

Electrochemical decontamination uses direct electric current, which results in the anodic dissolution and removal of metal and oxide layers from the component. The dissolution can be achieved by soaking the items to be decontaminated in an electrolyte bath fitted with anodes. This method is useful for decontaminating items whose surfaces are not easily accessible. Electric current can also be delivered to a component by moving a pad over the surface to be decontaminated, and this is an efficient method to use on regular surfaces. The volume of electrolytes for electrochemical decontamination is relatively lower than the volume of liquids for chemical decontamination.

Electrochemical decontamination processes can only be applied when removing radionuclide contamination from conducting surfaces, such as iron based alloys (including stainless steel), copper, aluminium, lead and molybdenum. They are highly effective, removing practically all radio-nuclides and give a high decontamination factor (about 100).

Phosphoric acid is normally used as the electrolyte in electro-polishing because it is stable, non-aggressive and applicable to a variety of alloy systems. Moreover, the non-drying nature of phosphoric acid helps minimise airborne contamination and the good complex characteristics of phosphoric acid with metal ions is a significant factor in minimising recontamination by the electrolyte.

The use of electrochemical decontamination is limited by the size of the bath and by the necessity to remove equipment before realising decontamination (when soaking is used), and by the geometry of the surfaces and the available clearance around the part being treated (when the pad is used). The effectiveness of the decontamination can be limited by the presence of materials adhering to the surface of the items to be decontaminated. Materials such as oil, grease, oxides (rust) and paint or other coatings should be removed before decontamination. Despite the fact, that the volume of effluents is minimised, handling the parts to be soaked or to be treated with the pad can lead to additional exposure to workers.

- *Mechanical decontamination:*

Mechanical decontamination methods can be classified as either surface cleaning (e.g., washing, wiping, scrubbing, using foaming agents) or surface removal (e.g., blasting). Mechanical decontamination can be used as an alternative, employed simultaneously or sequentially with chemical decontamination. In general, mechanical decontamination methods can be used on any surface, with very good results being achieved. When these methods are used in conjunction with chemical methods, an even better result is realised. Moreover, when dealing with porous surfaces, mechanical methods may be the only option.

Surface cleaning techniques are used when contamination is limited to near surface material. These techniques generate contaminated liquids that need to be collected and treated. Many surface cleaning techniques can be used for decontaminating both equipment and buildings.

Abrasive blasting system is a typical surface removal decontamination technique. Generally, abrasive blasting techniques have proven themselves to be effective. In many cases, the equipment is well developed and commercially available. Several methods are able to remove strongly adhering material, including corrosion layers. On the other hand, in some cases it is very difficult to control and regulate the amount of metal substrate removed. Care must also be taken not to introduce contamination into the material surface (the so-called ‘hammering’ effect).

A wet abrasive blasting system in a closed loop is a liquid abrasive decontamination technique. The system uses a combination of water, abrasive media and compressed air and is normally applied in a self-contained, leak proof, stainless steel enclosure. Wet abrasive cleaning is being used in many nuclear facilities to remove fixed contamination from metal surfaces such as structural steel, scaffolding, components, hand tools and machine parts. During wet blasting application, large volumes of waste are produced including waste water, abrasives and removed debris. Solid waste is mechanically separated from the cleaning media, e.g., by cyclone/centrifuge separation, sieving. Water can be filtered and recycled and no soluble or hazardous chemicals are being required.

The dry abrasive blasting technique (sand blasting, abrasive jetting) uses abrasive materials suspended in a medium that is propelled onto the surface being treated, results in the uniform removal of surface contamination. Removed surface material and abrasives are collected and placed in appropriate containers for treatment and/or disposal. Dry abrasive blasting is applicable to most surface materials except those that might be shattered by the abrasive, such as glass or plexiglas. Its use on aluminium or magnesium should also be avoided owing to the risk of dust explosions. It is most effective on flat surfaces and because the abrasive is sprayed, it can also be used on ‘hard-to-reach’ areas. Nonetheless, materials such as oil and grease, or obstructions close to or bolted to components, must be removed before application and precautions should be taken to stabilise, neutralise, or remove combustible contaminants because certain abrasives can cause some materials to detonate or can cause dust explosions. Using dry blasting, dust control measures may be needed to control dusts and airborne contamination. This problem can be reduced by using filtered vacuum systems in the working area [9].

- *Decontamination by melting:*

  Specifically during the decommissioning of nuclear power plants, large amounts of slightly contaminated metallic scrap are generated. Much of these wastes consists of bulky equipment (e.g., heat exchangers, moisture separators, steam generators) that, if disposed in appropriate repositories, would occupy considerable volumes of the available space. Moreover, in many cases, this equipment contains valuable material that can be recycled. By melting slightly contaminated scrap, it is possible to recover much of these valuable metals while simultaneously conserving valuable space at final disposal facilities. The pieces of equipment considered frequently also have complex geometries, making the determination of the exact location and level of radioactivity on the internal surfaces extremely difficult, time consuming and expensive tasks. Using melting the problem of inaccessible surfaces is eliminated. The remaining radioactivity is homogenised throughout the total mass of the ingot and can be precisely determined and measured. Moreover, an ingot can be released for restricted or unrestricted reuse, stored for decay until the limits are fulfilled.

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

  Melting presents the particular advantage of redistributing a number of radio-nuclides among the ingots, slag and filter dust resulting from the melting process, thereby decontaminating the primary material. A particularly advantageous consequence of melting is its decontaminating 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; the dominant nuclide remaining in the ingots (for most reactor scrap) being ^{60}Co. This element has a half life of only 5.3 years. Consequently, ingots with reasonably low activity concentrations can be stored for release in the foreseeable future. Moreover, radiation exposure of foundry workers during the subsequent re-melting of ingots is drastically reduced as a result of the removal of ^{137}Cs. The secondary waste consists of the slag from segmenting and melting, as well as dust from the ventilation filters. This secondary waste only comprises 1 - 4 % of the melted scrap weight.

- *Other decontamination techniques*

  In special cases, other decontamination techniques, e.g., ultrasonics, high pressure water jetting or steam spraying, thermal erosion, pastes, gels, foams have also been used in decommissioning. Some of these, however, require more or less complex application procedures or still need more development to allow industrial applications.

4.4.1.4 Decay storage

Carefully organising radioactive waste interim storage allowing the decay of short lived radioisotopes can be also an effective pre-treatment method, considerably facilitating subsequent treatment steps or permitting the clearance level acquirement. Decay storage is of particular importance for low level and short-lived radioactive waste representing the majority of decommissioning waste. A properly administered decay storage system can result in substantial cost savings and a high degree of safety when managing the decommissioning waste.

The constraints of the decay storage are mainly related with the non-radioactive risks. Volatile or combustible waste presents a hazard through fire. Also biological instability and toxicity are the main risk factors to be taken into consideration.

The practical implementation of decay storage requires the operation of a protected, selective storage capacity with matching identification and other administrative procedures. This ensures that packages are stored for the correct length of time and the correct package is retrieved for disposal. Usual radiological protection requirements should apply regarding the handling of active or potentially active materials, even in the simplest facility [8].

4.4.2 Treatment of radioactive waste

Radioactive waste treatment is presented as a complex of technological operations and procedures intended to the creation of such radioactive waste form that allows further effective conditioning and disposal process. The main goal of radioactive waste treatment is to increase the safety and to improve technical and economical parameters of further radioactive waste management phases by:

- Radioactive waste volume reduction.
- Removal or concentration reduction of radio-nuclides from the waste.
- Changing radioactive waste characteristics.
- Changing radioactive waste composition.

In the next sections, the basic conditions, characteristics and technologies used for treatment of liquid and solid radioactive wastes are discussed.

4.4.2.1 Treatment of aqueous liquid waste

During decommissioning, significant volumes of liquid radioactive wastes are arising. The process for the decontamination of technological equipment (pre-dismantling and post-dismantling) mainly produces aqueous liquid waste decontamination solutions. The waters from the sanitary locks should also be considered as liquid radioactive waste but they are expected to have such low radioactivity levels that allow them to be directly discharged into the environment.

The most obvious problem during the treatment of liquid radioactive wastes is the potential for the uncontrolled release of liquids with the creation of extensive contamination, and therefore every effort must be made to avoid leaks, spills or other unplanned releases. The management of corrosive aqueous wastes requires that the collection, storage and treatment facilities can withstand prolonged exposure to this type of wastes. Otherwise, they must be conditioned directly without any prior treatment.

The selection of a liquid waste treatment system involves making decisions on a number of factors. These can be grouped into five main categories:

- Characterisation and segregation of liquid waste.
- Discharge requirements.
- Available technologies and their costs.
- Conditioning of the concentrates resulting from the treatment.
- Storage and disposal of the conditioned concentrates.

To a large extent, the selection of a primary treatment process for liquid waste depends upon its radiological and physicochemical properties and the quantity of arising. It is therefore important to know these properties, not only those predicted by the facility design characteristics, but also those that result from the actual operating conditions in a nuclear facility.

Liquids containing suspended matter must be treated to remove the particulate. This can occur either before primary treatment (such as prior to ion exchange) or after (such as to remove the precipitate/sludge produced in a chemical precipitation process). Various possibilities for treatment include sedimentation and decantation, filtration or centrifugation.

A number of treatment processes are available for low and intermediate level aqueous liquid wastes arising from the decommissioning of nuclear installations. Following, the main techniques such as precipitation, ion exchange and evaporation are shortly described.

The objective of a chemical precipitation process is to use an insoluble, finely divided solid material to remove radio-nuclides from a liquid waste. The majority of precipitation methods use metal hydroxide flocs under neutral or alkaline conditions to remove the radio-nuclides. In these processes, a number of radio-nuclides are extensively hydrolysed and are likely to be either co-precipitated or sorbed on a floc. Precipitation normally involves four main stages:

- The addition of reagents and/or adjustment of pH to form for precipitation.
- Flocculation.
- Sedimentation.
- Solid/liquid separation.

The process of ion exchange involves the exchange of ionic species between a liquid solution and a solid matrix containing ionisable polar groups. When exchangers become fully loaded they are removed from service and treated as radioactive waste. Alternatively, many organic ion exchange materials may be regenerated by strong acids or bases, yielding radioactive liquid waste with a high salt and activity content. Ion exchange media are available in many combinations of natural or synthetic, organic or inorganic materials and in cation or anion exchange forms. Many media are also available in a variety of physical (e.g., bead or powdered) and chemical forms (e.g., H^+ or Na^+ counter ions).

The evaporation involves distilling the solvent from a waste effluent, leaving a smaller volume of residue containing both the radio-nuclides and the inactive salts. The condensate resulting from evaporation is an almost salt free solution of very low activity that may be subsequently polished by ion exchange before it is discharged or recycled. The concentrate containing the radio-nuclides can either be dried to produce a salt cake or be incorporated into a suitable matrix (e.g., cement, bitumen) for final disposal as radioactive waste.

In Table 4.6, a brief description of the main characteristics of specific liquid radioactive waste treatment technologies is given [8].

Table 4.6 Main characteristics of basic liquid waste treatment processes

	Chemical precipitation	Ion exchange/sorption	Evaporation
Liquid radioactive waste characteristics	- Not sensitive to highly salt laden solutions. - Possible negative effects when oils, detergents and complexing agents are presented.	Suitable for: - Low suspended solids content. - Low salt content. - Absence of non-ionic active species.	- Low detergent content required (owing to foaming problems). - Not suitable for volatile radio-nuclides (tritium).
Decontamination factor (DF)	10 - 100 (β/γ); 1'000 (α); exceptionally > 1000 (α).	10 to 10'000; average 100 to 1'000.	10'000 - 100'000.
Volume reduction factor	10 - 100 (wet sludge); 200 – 10'000 (dried solids).	500 to 10'000.	Depends on the salt content in the solution.
Specificity	Can be tailored to a variety of species.	Can be tailored to a variety of species.	Not species specific, but can be tailored for various chemical solvents.
Conventional combination with other processes	Possible with evaporation and ultra-filtration.	Possible with evaporation.	Condensate can be subsequently treated by ion exchange.
Process drawbacks	- Volume of flocs may be important - Dewatering system needed for sludges. - Chemical hazards associated with reagents.	- Limited radiation stability. - Limited heat resistance.	Sensitive to scaling, foaming, salt, precipitation and corrosion.
Application types	Concentration of active species.	Demineralisation/ decontamination of effluent when salt content <1 g/l.	Concentration of the solution (active and non-active species).
Scope of application	- Utility liquid waste from nuclear power plants. - Low and intermediate level streams in reprocessing operations. - Nuclear research centres liquid waste.	- Maintenance of pond water quality. - Water conditioning in reactor circuits. - Various treatments in reprocessing operations. - Post-treatment for all other operations.	- Primary coolant clean-up. - Utility liquid waste from nuclear power plants. - Various uses in reprocessing operations.
Maintenance	Possible blockage of feed lines and corrosion.	Possible blockage of ion exchange beds.	Possible foaming, scaling and salt precipitation problems, corrosion.
Cost	Relatively low cost.	Relatively expensive, mainly for synthetic ion exchangers.	Expensive (high energy consumption).

4.4.2.2 Treatment of organic liquid wastes

Liquid organic radioactive waste includes oils, scintillation fluids and solvents. In most cases the volume of these wastes is smaller by comparison with aqueous wastes. While most aqueous wastes may be discharged to the environment following treatment or decay storage, organic liquids may require more elaborate treatments to remove or destroy chemically or biochemically hazardous components. The factors of hazardous components removal must be considered when planning and implementing a radioactive organic waste management system.

In the pre-treatment period, organic liquid waste should be collected and stored until a sufficient quantity has accumulated to justify its transport to a central radioactive waste management facility. The different waste types should be segregated during waste collection. The most important fact is that organic liquids should not be mixed with aqueous liquids, since this complicates further treatment for both. Because of the chemically active and flammable nature of organic liquid waste, its collection and storage area should be isolated from other activities and there should be adequate fire protection and ventilation.

A number of treatment processes and technologies are available to treat organic radioactive liquids. Many of these processes treat a variety of wastes and an individual waste stream may be treatable by a variety of methods. The waste management strategy may require treating the waste by processes such as distillation to concentrate the radioactivity and/or separating the components by evaporating those that are non-active, or by incineration to destroy the organic material. The other used treatment methods are wet oxidation, acid digestion or electrochemical oxidation. Often, substantial advantages can be accrued by selecting a combination of two or more processes, rather than a single process. The detailed description of available organic liquid treatment technologies should be found in [8]. The treated waste may then need to be immobilised and conditioned to prevent radioactivity from escaping into the environment [8].

4.4.2.3 Treatment of solid waste

Solid waste is produced by virtually all applications and uses of radioactive materials, including normal operations, maintenance and mainly decommissioning activities. In many facilities, dealing with solid radioactive waste may be the largest part of the radioactive waste management programme. A detailed evaluation is required to determine whether treatment is to be used and if so, which processes should be selected. In cases where only small volumes of waste are produced, it may not be economical to install the radioactive waste treatment equipment; in these cases direct conditioning prior to storage or disposal may be less costly. In cases where large volumes of waste are produced, the cost of treatment to reduce the volume may be less than the cost of more storage and/or disposal space for the added waste. In cases where only short lived radioisotopes are involved, storage for decay and release is the preferred option, since it eliminates the costs associated with radioactive waste processing.

In general, the solid radioactive waste non-releasable to the environment should be in accordance with their further processing divided into [8]:

- *Compactable solid radioactive waste:*

  Compaction is a widely used method to volume reduction of dry radioactive solid waste through the application of a mechanical force. The volume reduction factors obtained depend largely on the waste material involved and on the pressure applied, but in general are between 3 and 10. Optimum operation is achieved if the appropriate sorting and pre-treatment of the waste has been accomplished. Both low and high pressure hydraulic and pneumatic presses are in use.

  In-drum low pressure compaction with forces up to about 100 t and volume reduction factors between 3 - 5 is used to reduce the volume of solid compactable waste primarily to facilitate packaging for transport to another waste treatment facility, where further compaction or other treatments will be carried out or to an interim storage facility, where it will await disposal.

  The drum compaction (super-compaction) process is characterised by the fact that drums containing compactable waste are compressed into stable compacts. The degree of compaction depends on the force of the compaction press and the physical properties of the waste material. Drums containing either compactable waste collected at the place of production or waste already treated by in-drum compaction can be fed to the drum compaction equipment. In the latter case, high pressure compactors using forces up to 1'500 t are used to achieve an additional volume reduction. However, drum compaction equipment is much more expensive than simple in-drum compactors, and is not recommended for low volume producers.

The main constraints should be summarised:

* Large, non-compactable components could damage the equipment and should be eliminated.
* Chemical reactivity of the material compacted might be enhanced and the press should be provided with appropriate fire fighting facilities.
* Absorbed or incidentally contained liquids can be released during compaction and should be collected.
* Air enclosed in the primary waste packages will be released during compaction. This can lead to airborne contamination, so there are requirements for an appropriate air filtration system.
* Materials like plastics and rubber show a tendency to expand or spring back after release of the compaction pressure. Compaction of waste in steel drums can minimise the spring-back effect.

- *Combustible solid radioactive waste:*

Incineration is the process used for treatment of solid combustible waste with decontamination factors up to 10^7. The principal objective of incineration is to achieve the complete combustion of the organic components of the waste into inorganic products. The process also significantly reduces the volume and the mass of the radioactive waste and converts the waste into a form suitable for subsequent immobilisation and final disposal. The products from complete incineration are carbon dioxide and water and the oxides of other constituent components, e.g., phosphorus, sulphur, metals.

A radioactive waste incineration system must provide containment of the radioactive species throughout the process. The incinerator itself must provide physical containment of the volatile organic and radioactive wastes so as to avoid the escape of gases and vapours. The incineration system should also include the following components:

* Waste receipt and storage area with capabilities to sort and blend materials.
* Primary combustion system with a design specifically selected to address the waste form, waste stream quantity and contaminants of concern.
* Secondary combustion chamber to complete destruction of the organic compounds in the off-gases from the primary chamber.
* Ash removal system to safely cool and remove non-combustible material from the combustion chamber.
* System for immobilisation of the ash.
* Air pollution control system to remove particulate matter and other pollutants of concern from the combustion gases.

From the above mentioned facts, it should be stated that the use of incineration for combustible radioactive waste involves high investment and operation costs. Also in some cases, the incinerator may not be employed to its full capacity.

There are also a number of different incineration technologies available, e.g., air incineration, pyrolising, fluidised bed incineration, slagging or rotary kiln incineration, agitated or multiple heart incineration and cyclone incineration.

- *Non-compactable and non-combustible solid radioactive waste:*

  Non-compactable or non-combustible solid radioactive waste often requires special treatment, depending on its particular characteristics. Usually, there are directly put into the waste package (steel drum) and immobilised using an appropriate grouting (concrete) mixture. This form should be transported to conditioning equipment, to a storage facility or directly to the radioactive waste repository.

4.4.3 Immobilising and conditioning of radioactive waste

Waste immobilisation has become an important step in the field of waste management and in the philosophy of environmental containment. This step converts radioactive waste, usually a liquid or semi-liquid, into a solid form that can be handled, stored and disposed more safely and conveniently.

There are a variety of matrix materials and techniques available for immobilising and conditioning waste. For long term storage and disposal, the method of solidification is not a reversible process that allows the solids to return into a liquid form. Estimation of the rate of leaching from a solidified matrix during disposal is one of the important considerations in the assessment of a solidification method. Low matrix solubility improves the safety of waste management through isolation, which further reduces the likelihood of radionuclide release.

It should also be pointed out that problems have been encountered with the solidification of some waste streams, and waste generators frequently use high integrity containers to provide the required stability. These containers are generally very expensive, but this disadvantage may be offset by the simplified processing of the waste.

In general, the following parameters may be used to categorise radioactive waste immobilisation and conditioning processes:

- *The matrix material*: the selection of any matrix material will be governed not only by the waste form criteria stipulated by the licensing and regulatory bodies, but also by the composition of the waste and the extent and type of treatment prior to the conditioning step.
- *The size and type of container*: some systems are built to use a single type of container, while other systems can use many different types.
- *The type of mixing process*.

Several different processes and techniques, discussed in the following parts, are available for radioactive waste conditioning [8]:

- *Cementation:*

  The cementation process has been extensively used throughout the world for immobilising and conditioning of different waste types such as aqueous waste, sludge, ion exchange resins, biological waste and also solid materials. It is practised on different scales, depending on the specific local situation with respect to the types and volumes of waste.

  Cements and grouts represent an inorganic matrix. Ordinary Portland cement is used in waste stabilisation by the blending and casting of small items and shredded or chopped solid wastes into drums or boxes producing a packaged monolithic form for disposal. Finely divided wastes, such as rubble, contaminated soil, ion exchange resin, etc., are mixed with Portland cement similar to the sand and gravel used normally in making concrete. Special grout formulations have been developed to incorporate high salt content wastes that could not be retained with Portland cement alone. These formulations use other grout type materials including foundry slag and coal fired power plant fly ash.

The most commonly used cementation process is in-drum mixing that involves mixing the waste and cement inside a prepared container. The components are blended until a homogeneous mixture is obtained. After mixing, the cement waste mixture is allowed to set, the container is capped with fresh cement to minimise void spaces and to avoid surface contamination and a lid is fitted. The filled container (drum) represents a suitable form for final disposal.

- *Bituminisation:*

Bitumisation is a proven immobilisation process for a wide variety of radioactive wastes mainly for aqueous wastes or sludges. But it is suitable for stabilisation of both organic and inorganic wastes. The use of bitumen to solidify low and intermediate level radioactive waste has been successfully applied on an industrial scale for many years in different countries.

Bitumen (asphalt) is a thermoplastic material and can behave mechanically as either a viscous liquid or a solid, depending on its temperature. As an organic material, bitumen has an advantage in its potential for retention of mobile organic compounds in the waste. It has two major components: asphaltene compounds, which give bitumen colloidal properties and malthene compounds, which impart viscous properties.

In the bituminisation process two types of matrix materials are usually used – straight run distillation bitumen and oxidised (air blown) bitumen. Both batch and continuous processes have been used, but the continuous process is generally preferred because of the higher throughput. The process is also energy intensive owing to the need to heat equipment, bitumen storage tanks and feed lines.

- *Polymerisation:*

Polymer processes have only been used to a limited extent for immobilising radioactive waste. Generally, they have been applied to ion exchange resins owing to the difficulties encountered when using cement or bitumen. Polymers may also be used for macro-encapsulation of solids such as lead bricks or whole entire pieces of equipment. Both thermoplastic and thermosetting polymers have been used. In most cases the waste must be pre-dried when polymers are used as the immobilisation matrix; an exception to this is, if a water extendible vinyl ester or polyester resin forms the matrix.

- *Vitrification:*

Vitrification was developed for high level radioactive waste applications, especially for spent fuel reprocessing plants, but has been suggested as a technology for solid and liquid low and intermediate level radioactive waste immobilisation. Vitrification uses heating of glass matrices to melt inorganic materials such as ash. The product, the final waste form, is a high-density glass. The advantages of vitrification include:

* Leach resistance.
* Low volume of final waste form.
* A high volume reduction for combustible waste.
* Organic and nitrate destruction efficiency equivalent to incineration.

- * For waste streams treatable by direct vitrification, overall waste processing is simplified compared to a separate organic destruction process followed by blending and solidification.

  However, there are certain limitations associated with vitrification. Care must be taken to avoid unacceptable emissions of volatile metals, such as mercury, dioxins and furans. The associated gas cleaning system generates a secondary radioactive waste. Furthermore, if there are variations in the waste feed, it is necessary to control the melt composition and therefore assure the melted waste form properties are satisfactory.

- *Sorption:*

  Liquid solutions may also be stabilised by sorption onto particulate materials and through reactive formation of stable insoluble chemical compounds. There are many variations of physical sorption used to produce stable, disposable waste forms. Acidic or basic solutions may react and affect effect the cement grouts adversely when combined directly with Portland cement. Alternatively, acids, bases, or aqueous/organic solutions may be adsorbed onto vermiculite or similar solids. These particulate type materials with the sorbed solution may be satisfactory for disposal or may be further stabilised by incorporation into grout or cement formulations. The sorption of the liquids onto the inert particulate materials may reduce the potential for the liquid to interfere with the solidification reactions of the grouts or cement. Specially modified clays can be added to organic liquids and sludge to form a solid waste. These final form solids are reported to have desirable physical properties and good leach resistant characteristics.

4.4.4 Packaging of radioactive waste

Typically, carbon steel drums of at least 200 litres are the waste containers used for immobilising liquid and wet solid wastes because they are easy to handle and are available for a low cost. Also concrete drums are used, the disadvantage of these drums being their added weight and thickness requirements that may affect their volumetric efficiency.

The primary purpose of a container is to provide integrity for a waste package during handling, interim storage, transportation and disposal. In most cases it also serves as the vessel used during solidification. Depending on the materials being shipped (the types of radio-nuclides and their gross activities), transport regulations may require that waste packages have either protective overpacks or inner linings to provide additional mechanical integrity and/or radiation shielding. The following general guidelines should be considered in selecting a suitable transport container:

- All packaging should provide at least two independent barriers designed to contain radioactive materials throughout handling, transport and storage/disposal operations.
- The container should be small enough in volume and light enough in weight to be easily handled and be of a uniform shape.
- The container should be watertight and capable of resisting the storage/disposal environment.
- The container should have sufficient structural strength to withstand stacking, dropping, penetration tests.
- The container should meet the shipping criteria for transportation without the requirement for the waste to be repackaged [8].

4.4.5 Storage of conditioned radioactive waste

Storage of conditioned radioactive waste may have to be included in the waste management process for the following reasons:

- To achieve the needed mechanical or chemical characteristics of the waste package or fixing matrix before disposal (short term storage).
- The repositories for final radioactive waste disposal are not available (should be long term storage up to 50 years).
- The existence of technical, economical or legislative problems with radioactive waste transport or disposal.

In the case of conditioned waste storage, it is necessary to guarantee the integrity of the radioactive waste package during the whole storage period. This storage period should not evoke such changes of the radioactive waste characteristics that complicate or make further transport and disposal impossible.

Two hundred or four hundred litre steel drums are the generally recommended containers for storing conditioned waste. Depending on the level of activity, other containers are also commonly used. Ad hoc arrangements could be made for larger shielded containers, if required. The storage facility should be designed to accept all required immobilising matrix forms (cement, bitumen, polymer, etc.) and to handle the volume of waste to be produced during the planned period of operation.

If a conditioned waste storage building is also used to store unconditioned waste, the two waste types should be separated, especially when storing unconditioned flammable liquids such as scintillation fluids.

Other general considerations for a storage facility are security and fire protection. The careful selection of non-flammable construction materials when building a storage facility will greatly reduce this hazard. A storage facility should not be used to house any highly flammable or highly reactive materials. A radioactive waste storage facility should be also well protected against unauthorised human intrusion. It should be constructed, operated and maintained in such a way that unauthorised removal of radioactive waste is prevented [8].

4.4.6 Final disposal of radioactive waste

Final disposal of radioactive waste is understood as an emplacement in an appropriate facility without the intention of retrieval. A radioactive waste repository is a nuclear installation that ensures the long-term isolation of radio-nuclides from the environment using the cooperating system of engineering and geological barriers (multi-barriers principle). In general, the following principal methods of radioactive waste disposal are available:

- Near surface repositories are suitable for short-lived low/intermediate level waste disposal. The radioactive wastes are disposed in trenches or vaults and the environmental protection is provided mainly by engineered barriers (sorption layers, concrete boxes, leakage monitoring equipment, etc.).
- Sub-surface repositories, built in a rock cavity or abandoned mines in depths of about 30 to 150 meters, should be suitable for disposal of all kinds of solid low/intermediate level wastes.
- Deep geological repositories are considered to be established in depths of about 500 to 1000 meters in stable geological formations (granite, clay). They are envisaged mainly for high level waste and spent fuel disposal.

The choice of a disposal method is dependent on the conditions prevailing in the country and on many other factors specific to the disposal system to be developed. Generally, near surface disposal and rock cavity concepts appear to be the most viable for the disposal of decommissioning waste.

5. The clearance process

5.1 General considerations

As an introduction to the further considerations, it will be useful to repeat the definition and the approaches to ‘exemption’ and ‘clearance’. Within decommissioning practices, both of these terms are frequently used very freely, often without appreciation of their real content.

‘*Exemption*’ determines *a priori* which practices and sources within practices may be excluded from the requirements for practices on the basis of their meeting certain criteria. In essence, exemption may be considered a generic authorisation granted by the regulatory body which, once issued, releases the practice or source from the requirements that would otherwise apply and, in particular, the requirements relating to notification and authorisation [10]. Exemption may be granted if the regulatory body is satisfied that the justified practices or sources within practices meet the exemption principles and criteria specified in Schedule I of the Basic Safety Standards [11], or other exemption levels as specified by the regulatory body on the basis of the exemption criteria specified in Schedule I of the Basic Safety Standards.

The activity concentrations and total quantities of radio-nuclides specified in Schedule I of the Basic Safety Standards were derived by establishing a set of representative exposure scenarios and determining the activity concentrations and total activities that would give rise to doses to the appropriate critical groups that correspond to the dose criteria for the exemption of practices set out in Schedule I of the Basic Safety Standards, modified to take account of low probability exposure events, as described in [12]. These derived radionuclide specific values were based on calculations in which only moderate quantities of material were assumed to be present. A footnote to Schedule I of the Basic Safety Standards indicates that ‘exemption for bulk amounts of materials with activity concentrations lower than the guidance exemption levels may nevertheless require further consideration by the [regulatory body]’. Thus the quantitative guidance given in the Basic Safety Standards for exemption levels is limited to ‘moderate quantities’ of material; that is, amounts ‘at most of the order of a tone’ [12]. There are situations for which the exemption of considerably greater amounts than one tone of material may be appropriate, and the quantitative guidance provided in the Basic Safety Standards may not be suitable for these situations.

While exemption is used as a part of a process to determine the nature and extent of the application of the system of regulatory control, ‘*clearance*’ is intended to establish which material under regulatory control that can be removed from this control. ‘*Clearance*’ is defined as the removal of radioactive materials or radioactive objects within authorised practices from any further regulatory control by the regulatory body. Furthermore, the Basic Safety Standards state that clearance levels ‘shall take account of the exemption criteria specified in Schedule I and shall not be higher than the exemption levels specified in Schedule I or defined by the regulatory body’. A footnote indicates that ‘Clearance of bulk amounts of materials with activity concentrations lower than the guidance exemption levels of Schedule I may require further consideration by the regulatory body’.

As a conclusion, there are two principal differences when approaching ‘*exemption*’ and ‘*clearance*’ in practice:

- the amount of the bulk material that should be exempted or cleared;
- the time when the exemption/clearance approaches are applied: while ‘*exemption*’ is applied a priori, i.e., before the given practice, ‘*clearance*’ is applied a posteriori, i.e., after the practices have been regulated.

Clearance applied with good understanding of these aspects as well as the aspects discussed in the following sub-sections should represent an effective tool to rationalise the system of management of solid radioactive wastes arising from various nuclear activities, particularly from the decommissioning of nuclear facilities.

5.2 Methodology and procedures

The clearance process should be considered as a part of the overall decommissioning waste management system. As is shown in Figure 4.2 of this document, four from six waste disposal routes represent various options for the clearance:

- Unconditionally released materials for recycling or reuse;
- Unconditionally released materials for disposal;
- Radioactive material for conditional recycling or reuse within the non-nuclear industry;
- Radioactive material for conditional recycling or reuse within the nuclear industry.

The fifth route (disposal of solid very low-level radioactive waste) can be considered to be a borderline between the clearance routes and disposal as radioactive waste. Disposal of cleared materials is very near to the disposal of very low level waste. In some countries, there is not an unambiguous separation of disposal of cleared materials from the disposal of very low level waste: neither legislatively, nor by siting of the disposal facility, nor technically or from the long- and short-term safety point of view.

Issuing from the analysis of the material streams arising from decommissioning activities, there are three basic solid material or waste streams (see Figure 4.1):

- Materials demonstrably considered as non-radioactive, e.g., for trivial reasons, as materials arising from activities outside of the radiological controlled areas of the facilities; the management of these is realised according to the regulations on non-radioactive waste management, according to their physicochemical characteristics.
- Potentially radioactive and/or clearable materials.
- Waste materials which should be primarily considered as radioactive, but containing also potentially clearable components.

Within both of the last two basic waste streams, there is a need for decision whether the given waste streams can be considered to be potentially clearable. There are many factors that may influence decisions in the area of clearance of materials. Factors affecting the decision which disposition strategy should be adopted include [13]:

- 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 the public and the environment;
- quality assurance (QA) and documentation issues;
- environmental and ethical issues.

The decision process is briefly illustrated in Figure 5.1. Not all these factors will apply in every case but several of these are likely to be relevant. The interrelationship between the 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 optimisation is likely to form an inherent part of the determination of which segregation, release, recycle, reuse and disposal practices will be applied within particular decommissioning projects.

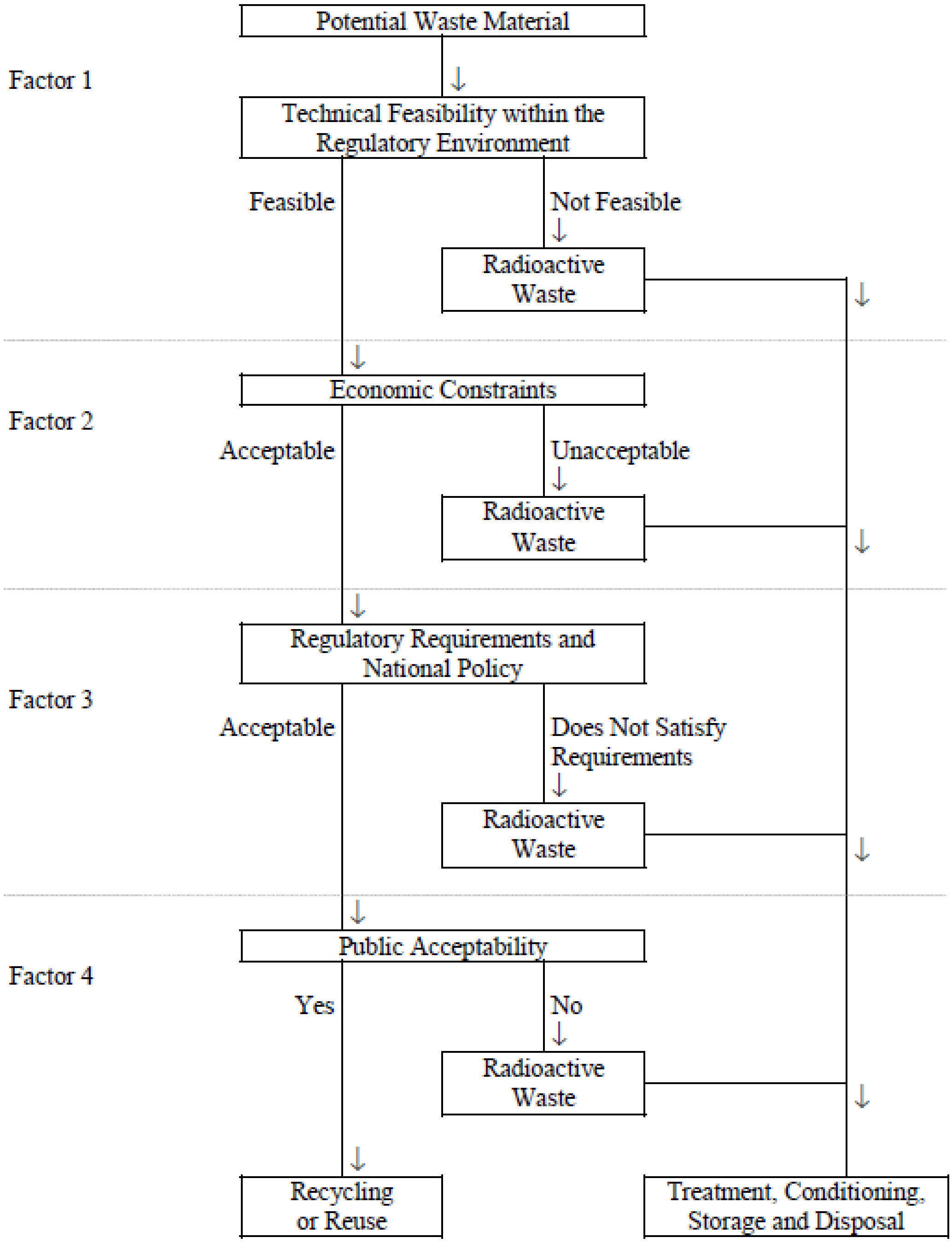

Figure 5.1 Linear decision-tree approach for a recycling and reuse application

Sometimes, 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 'rubblelisation', whereby above-ground structures are partially decontaminated, demolished and disposed of in the below-ground portions of the structures.

In fact, there are no principal differences between recycle and reuse of low activity materials within the non-nuclear and the nuclear industry. Both of the disposal routes could differ from each other in limitations of safety significant radionuclide concentrations derived by corresponding safety assessments in the framework of given scenarios and exposure pathways.

When a specific disposition option has already been identified as the preferred option, a linear decision tree approach could be adopted to test and confirm this option's acceptability for a particular application. This is illustrated in Figure 5.1 for testing a preference to recycling and reuse but with concerns about the 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. 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 each option. This alternative has been shown in Table 4.5 of Section 4.2.5.

As addressed in Section 4, procedures relating to the management of radioactive and suspect (from the content of radioactivity point of view) materials arising during decommissioning activities, performed with the intention to release the materials from regulatory control, consist of two principal steps:

- processing, i.e., decontamination (if possible, needed and advantageous), segregation and primary characterisation for preliminary determination of compliance with the release criteria. Regarding clearance, by comparison with established clearance limits or with clearance limits approved by the regulator, this step results into a definition of what material/waste stream could be considered as potentially clearable and what should be managed as radioactive waste.
- final checking of the conformity of the waste characteristics with the criteria for the release of the materials by a given clearance route, i.e., for changing the status of the materials from 'potentially' clearable to 'really' clearable.

As described above, characterisation of waste with the potential for being released is always one of the principal steps within the clearance methodology and procedures. Nevertheless, it is necessary to consider that the clearance process is not, in the first place, a way to improve the safety of the decommissioning activities. It is a way to economise these. Even though the best methodologies and procedures are applied, including characterisation practices, they do not need to eliminate fully all doubts on the content of potentially released materials from both the side of the regulator and/or the decommissioner. In such cases, for instance when materials are, due to complex geometries, difficult or expensive to monitor, safety and economic considerations may lead to their disposal as radioactive waste.

5.3 Detection devices and monitoring systems

For the release of materials from regulatory control, in principle two distinct types of measurements can be considered [14]. One is the monitoring of surface contamination, i.e., the measurement of radioactive materials which may be attached to the surface of another object. These radioactive materials may well be subject to relatively easy removal, leading to potential ingestion or inhalation by persons involved in for example handling, transportation and reprocessing of scrap metal. They may also directly irradiate workers by the emission of X-, γ- or energetic β-radiation. As a result of the recycling, the radioactive material may be concentrated in the recycled material or, as is more likely, discharged to the atmosphere or concentrated in the waste from recycling. Assessment of surface contamination is therefore an important part of the sentencing of decommissioned materials.

The other category is the measurement of activity in bulk material, caused either by direct neutron activation of the material and its inherent impurities, or by the diffusion of radionuclides through the surface of the material into a significant depth. These contaminants are inherently less accessible, which has the advantage that they are less likely to be ingested or inhaled during recycling. They may, however, continue to be a source of irradiation for the workers and may also be concentrated in the recycled materials, discharged to the atmosphere or concentrated in waste from recycling. They are also more difficult to measure as the geometry is less defined and as self shielding is inevitable.

The two types of measurements also differ in the sense that surface contamination monitoring can normally be performed on a relatively well defined area whereas the assessment of contamination in bulk usually implies a significant averaging volume.

The extra costs of monitoring to a level sufficient to justify free release have to be balanced against the costs of on-site storage and/or off-site disposal. As a first step, it is important to assemble as much information as is easily available. For a plant still in operation and some period of time away from decommissioning there should be the opportunity to develop monitoring and recording programs that will help in the decommissioning phase. The usable information could come from:

1. *Existing radiation protection monitoring data*: monitoring information should be available throughout the lifetime of the plant. Such monitoring will normally have been biased mainly towards the designation of working areas in terms of dose rate, air and surface activity levels. Measurements in areas where maintenance was performed regularly should be well documented. Inaccessible areas may not be documented at all. The information may well be quite basic, such as count rate from a particular type of beta counter, gross alpha count rate on wipes, gamma ambient dose equivalent rate or neutron dose equivalent rate.

2. *Authorized discharge route monitoring*: for some areas useful information may well be available, especially where there are authorised discharge routes. Stack discharges on power stations for the heating, ventilation and air conditioning systems are normally monitored using high quality γ-spectrometers generating activity levels per m^3 for a wide range of nuclides. Liquid discharge routes are also monitored in a similar manner.

3. *Plant integrity monitoring*: monitoring may have been performed also to check plant integrity. A good example is the secondary cooling circuit on pressurised water reactors where regular monitoring is undertaken to identify heat exchanger leaks, or the burst can detector system on gas cooled reactors which is designed to identify fuel failure.

4. *Activation calculations*: knowledge of the likely neutron exposure and composition of structural materials will enable the calculation, if sometimes only very approximately, of the levels of activation nuclides such as ^{60}Co and ^{54}Mn.

5. *Process throughput*: in areas where fuel fabrication, reprocessing or other chemical operations have been undertaken, the total throughput of radioactive material may be well known. Care has to be taken where the deposition varies significantly between

the various chemical species. A good example is the tendency of ^{99}Tc from reprocessed uranium to plate out on the surface of a fuel fabrication plant.

Impediments to this process of gathering relevant data are many. In an old plant, monitoring results may have been lost or many may be intrinsically inadequate. Accidents may have happened leading to a much larger release of activity into the system than was anticipated at the construction stage. Complicated parts of the plant, such as pipe bends, may have much higher contamination levels than would be expected from measurements in other areas where deposition is less likely. The plant may have been deliberately run in an unusual condition or may have been modified and components replaced at some unspecified time. Alloys of a different composition to those specified may have been used. The construction of the plant may differ from the available plans. The plant may have been operated under some form of security condition which may mean that full monitoring data are not available.

The older the plant, and the more unusual the plant, the poorer the information will be that is likely to be available in written records. In these circumstances, it is important to trace as many of the workers as possible, including those who have left the plant or retired, in order to get a picture of the real history of the plant as complete as possible.

Regarding surface contamination, the process for the characterisation of materials can be formulated generally within the clearance procedure:

a) Identify likely contaminants from the operating data.

b) Weight the contaminants using the free release levels.

c) Identify those contaminants which are likely to be important.

d) Look at the decay scheme.

e) Identify nuclides which are likely to be present and emit reasonably penetrating radiation, e.g., beta emitters with an E_{max} in excess of 0.6 MeV.

f) Choose a suitable instrument, i.e., one which responds to the likely dominating contaminants, has a suitable area and is sufficiently robust.

g) Identify contaminated areas. Evaluate whether there are any problems with gamma background.

h) Sample contamination and analyse by γ-spectrometry and radiochemical analysis of alpha and beta emitters with insufficient γ-radiation.

i) Compare (h) with (a) and evaluate whether the results are in reasonable agreement with the predictions. Evaluate whether the radionuclide mix is reasonably consistent, at least in terms of the nuclides which will influence the free release of the material. Evaluate whether there are problem nuclides, in the sense of nuclides which are important in terms of the release criteria but which are difficult to detect on the material under consideration. A good example would be steel which is rusty and has significant α-contamination.

j) Decide on whether direct monitoring is possible for the material and the condition in which it exists. Evaluate whether cleaning would be possible if direct monitoring is not feasible. If this is the case, evaluate whether the initial instrument is appropriate. Evaluate whether there would be advantages in negotiating with any supervisory authority for an increase in any initial defined averaging area using the initial results for justification.

k) Decide on the most appropriate instrument, i.e., one which provides the most appropriate balance of characteristics, i.e.:

   - has good sensitivity over background.
   - is not dependent on changes in levels of unimportant nuclides.

- is as robust as possible having taken account of the demands above.
- is simple to operate by the staff selected.
- can be tested easily.
- is easy to repair.
- has an appropriate averaging area balanced against cost, complexity of the shape of the material to be monitored and the permitted averaging area.

l) Decide on the number of instruments required. Negotiate a repair service or identify a member of the team who can maintain the equipment. Order sufficient spares, having estimated the likely damage rate and considered the delivery time for supply of components from the manufacturer or component supplier.

m) Organize training for the workforce on both the instrument chosen and the monitoring technique.

n) Write monitoring and maintenance procedures.

o) Set up an auditing process, whereby results are traceable to national standards (metrological ones, for instance) and also where a random sample of the result is checked by another competent and independent person.

p) Decide on the frequency of sampling for γ-spectrometry and radiochemical analysis in order to support calculated release levels and to obtain acceptable representativeness of results.

q) Start monitoring for release.

r) Take an early piece of the material which is contaminated at or around the release level. Ask the workforce to monitor it, concealing, as far as possible, that others will have monitored it. Compare the results and evaluate whether the spread is acceptable. Would significant errors have occurred in the sense that either:

   (i) a piece which was definitely over the acceptable level was marked for free release, or

   (ii) a piece which was definitely within the limit was marked as excessively contaminated,

   find out why the error occurred and take steps to prevent it, such as further training or a modification to a written procedure.

s) Continue the process bearing in mind the need to continue to be confident that the instrument indication corresponding to free release is unchanging. The process should be reviewed to ensure that samples are being sorted correctly.

An similar action list should be established for the determination of bulk contaminated and/or neutron activated materials:

a) Identify likely contaminants from operating data.

b) Identify a hand held instrument which will respond to those contaminants.

c) If possible, remove samples for analysis, using the hand held instrument to aid selection.

d) Analyse samples by γ-spectrometry and by radiochemical analysis for alpha and beta emitters with insufficient γ-emission.

e) Weight the measured activities using the free release levels.

f) Identify those nuclides which are likely to be important. Look at the decay scheme for each nuclide.

g) Evaluate whether the samples are reasonably consistent.

h) Choose, if possible, a useful emission, such as an energetic γ-line which is present on a level proportional to the weighted activity levels in the samples. In the absence of one useful energy evaluate whether there is a limited combination of gamma emissions which could be used to assess the waste.

i) In the absence of useful γ-emissions evaluate whether there is some other possible means of monitoring, such as the measurement of gross alpha or gross beta emissions from prepared samples.

j) Evaluate whether the monitoring be performed in situ. Evaluate whether the monitoring can be performed after the materials have been removed, but are still intact, such as steel beams. Evaluate whether monitoring will have to be performed on what is essentially debris, such as concrete rubble.

k) Identify a suitable monitoring technique or techniques. It may well be advantageous to split the materials into 3 groups: the obviously clean, based on in-situ measurements, which can go for free release; the obviously excessively active, again based on in-situ measurements; and the borderline group. Materials in this group may require more sophisticated monitoring using installed equipment.

l) Identify suitable equipment to operate the monitoring techniques chosen.

m) Decide on the number of instruments required. Organize maintenance. Order sufficient spares, having estimated the likely damage rate and considered the delivery time for the supply of components from the manufacturer or component supplier.

n) Organise training for the workforce on the instruments and techniques chosen.

o) Write monitoring and maintenance procedures.

p) For installed monitoring, identify building needs and power requirements. As examples a sodium iodide scintillator conveyor monitoring system for crushed rubble will require a large but fairly basic building, whereas a drum monitoring system using large intrinsic germanium detectors will require a building which offers a good environment and the provision of a liquid nitrogen supply.

q) Set up an auditing process, whereby results are traceable to national standards and also where a random sample of the results is checked by another competent and independent person.

r) Decide whether there is a requirement for more detailed analysis of samples, by, for example, germanium detector spectrometry and/or radiochemical analysis. If so, evaluate what should be the frequency.

s) Start monitoring for release.

t) In the case of in-situ monitoring ask all the work force to monitor at defined positions concealing, as far as possible, that others will have monitored at the same position. Compare the results and evaluate whether the spread is acceptable. Would significant errors have occurred in the sense that either:

    (i) a piece which was definitely over the acceptable level was marked for free release, or

    (ii) a piece which was definitely within the limit was marked as excessively contaminated,

    evaluate whether written procedures can be improved or whether extra training is required.

u) Continue the process bearing in mind that the operator has to be confident that significant changes in radionuclide composition are identified and that the release criteria are adjusted accordingly.

Basic approaches to measurement of both surface contamination and specific activity have been normalised in the corresponding ISO standard [15]. A brief general overview of the measurement methods for various types of emissions from contamination nuclides and various measurement situations is described in Table 5.1 [16].

Table 5.1 General approaches used to measure activity of releasable materials for various surfaces or bulk materials

Type of contamination	Surfaces or bulk material		
	Smooth, impervious, clean surface	Rough, porous, dirty or painted surface	Bulk activity/contamination
Alpha radiation	Direct measurement is possible using, for example, alpha-sensitive scintillation detectors or proportional counters. Swipes or swabs can also be used for measurement of removable contamination. Long-range alpha detection techniques might be useful.	Direct measurement is not possible, although consider whether the contaminant also emits low-energy X-rays. Depending on the surface, it could be possible to use swipes or swabs for measurement of removable contamination.	Direct measurement is very difficult or impossible. Radiochemical analysis is likely to be the appropriate approach.
Beta radiation	Direct measurement is possible using, for example, scintillation detectors, proportional counters or thin-walled or mica end window Geiger-Müller detectors. Swipes or swabs can also be used for measurement of removable contamination.	Direct measurement is possible using, for example, scintillation detectors, proportional counters or thin walled or mica end window Geiger-Müller detectors. Depending on the surface, it may be possible to use swipes or swabs for measurement of removable contamination.	Direct measurement may be possible using, for example, large area scintillation detectors or proportional counters filled with a low atomic number gas. Radiochemical analysis of representative samples may prove necessary.
Gamma radiation	Large surfaces can be monitored with in situ gamma spectrometers with collimator.	Large surfaces can be monitored with in situ gamma spectrometers with collimator.	Direct measurement likely possible using scintillation or solid state detectors, bulk monitors, etc.; also in-situ gamma spectrometers with collimators can be used if volumes are not too thick.
Low-energy gamma or X-radiation	Direct measurement is possible using, for example, NaI(Tl) scintillation detectors, or sealed, xenon filled, titanium windowed proportional counters. Swipes and swabs may also be used for measurement of removable contamination.	Direct measurement is possible using, for example, NaI(Tl) scintillation detectors, or sealed, xenon filled, titanium windowed proportional counters. Depending on the surface, it may be possible to use swipes or swabs for measurement of removable contamination.	Direct measurement may be possible using scintillation or solid state detectors.

A basic comparison of the various detection systems is discussed in [14]. For surface contamination, some considerations are summarised in Table 5.2 illustrating the ability of a typical instrument in each category to demonstrate compliance with clearance levels for each nuclide, for direct reuse of metal items as indicated in [17].

Table 5.2 Illustration of the advisability of various detection systems - surface contamination - for direct detection of nuclides at the clearance level

Radio-nuclide	Clearance level Bq.cm^{-2}	Detectable at the clearance level					
		100 cm^2 anthracene scintillator	100 cm^2 xenon proportional counter	100 cm^2 butan proportional counter	20 cm^2 Geiger-Muller detector	100 cm^2 ZnS scintillator	100 cm^2 thin NaI(Tl) scintillator
^{3}H	1 x 10^4	-	-	-	-	-	-
^{14}C	1 x 10^3	✓	✓	✓	✓	-	-
^{54}Mn	10	-	-	-	-	-	
^{55}Fe	1 x 10^3	-	✓	✓	✓	-	✓
^{60}Co	1	✓	✓	✓	-	-	-
^{59}Ni	1 x 10^4	-	✓	-	✓	-	✓
^{63}Ni	1 x 10^3	✓	-	✓	✓	-	-
^{90}Sr	10	✓	✓	✓	✓	-	✓
^{94}Nb	1	✓	✓	✓	✓	-	-
^{99}Tc	1 x 10^3	✓	✓	✓	✓	-	-
^{106}Ru	10	✓	✓	✓	✓	-	-
^{106m}Ag	1	-	-	-	-	-	✓
^{110m}Ag	1	-	-	-	-	-	-
^{125}Sb	10	✓	✓	✓	✓	-	✓
^{134}Cs	1	✓	✓	✓	✓	-	-
^{137}Cs	10	✓	✓	✓	✓	-	-
^{147}Pm	1 x 10^3	✓	✓	✓	✓	-	-
^{151}Sm	1 x 10^3	✓	-	✓	✓	-	-
^{152}Eu	1	-	-	✓	-	-	✓
^{154}Eu	1	✓	✓	✓	✓	✓	-
^{238}U	1	-	-	✓	✓	✓	-
^{237}Np	0.1	-	-	✓	-	✓	-
^{238}Pu	0.1	-	-	✓	-	✓	-
^{239}Pu	0.1	-	-	✓	-	✓	-
^{240}Pu	0.1	-	-	✓	-	✓	-
^{241}Pu	10	-	-	-	-	-	-
^{241}Am	0.1	-	-	✓	-	✓	-
^{244}Cm	0.1	-	-	✓	-	✓	-

Problem nuclides are confined mainly to very low energy β-emitters such as tritium or ^{241}Pu. No instrument is suitable for monitoring every nuclide. The butane filled proportional counter is probably the most versatile but is the only instrument considered which requires regular maintenance, in the shape of refilling with counting gas. In most situations, the choice of the monitor will be determined by initial finger printing of the contamination. The actual monitoring limit will generally have to be calculated in counts per second taking into account the nuclide mix and the monitor characteristics, and, in some cases, may well be dominated by a relatively low fraction of a particular radionuclide which has very high detection efficiency. A typical example would be ^{90}Sr and its daughter, ^{90}Y, both of which are very easy to detect but are acceptable, for recycling, at a relatively high level.

Regarding the measurement of bulk/specific activity [14, 16], there is no such as a universal technique, given the wide range of nuclides, emission types, emission energies and types and shapes of materials. It seems likely that for any practical, large scale decommissioning of a plant, which is in any way complicated, all methods may have to be employed.

It is very difficult to detect directly the specific activity of nuclides emitting α- and β-radiation in bulk samples, in spite of the fact that the proposed clearance levels for β-emitters are generally higher than for γ-emitters.

There are many methods for the direct measurement of specific activity of γ-emitting radio-nuclides by hand held instruments. Circumstances may arise where it is possible to use conventional radiation protection instruments to estimate the gamma activity per unit mass of samples of decommissioning waste. This technique can be particularly useful for the clearance of large volumes of wastes where contamination is unlikely and where the potential contaminant or contaminants are known and are energetic γ-emitters. The technique is not appropriate for samples of complicated shape, wide and varying nuclide mix and for nuclides which emit non-penetrating radiations. Typical suitable situations include soil which might possibly have been contaminated by ^{137}Cs from fuel pond leaks, concrete which might contain ^{60}Co activated reinforcing steel, or potentially activated steel beams or piping which can be assumed to be uncontaminated. Interpretation of measurement results will differ for two sample approximations. One is where the mass of the material is so large that the sample is close to a semi-infinite source, and the other is where the sample has a relatively simple shape and where self absorption is very small.

Some hand held instruments have also spectrometric capability. These vary from completely self-contained instruments which will acquire spectra and identify contributions from up to 10 nuclides, to instruments which acquire and store spectra, and require connection to a computer for interpretation. Such instruments are inevitably more expensive and will take longer to make each measurement, because of the processing involved. However, they are extremely useful in checking that the expected mix of gamma radio-nuclides has not changed significantly and that the calculated maximum acceptable count rates used for the simpler equipment are still valid.

A second option for direct, in-situ measurement of specific activity is an installed device equipped by an appropriate detector system (generally based on NaI(Tl) detectors or on plastic scintillators or on both) and on the moving part (conveyors, rotating device) providing the movement of wastes to the detection system and appropriate scanning. Comparing with portable devices, the advantages of such systems are that the detectors can be larger, a wider range of detector types can be employed, shielding and collimation is much easier and the available electronic processing power is much greater. The disadvantages are that the equipment is much more expensive and that the objects to be monitored have to be transported to the monitoring station.

The method of choice for situations where a wide range of nuclides may be present is the intrinsic germanium semiconductor spectrometer. These devices have the ability to determine the energy of γ-emissions with great accuracy and to separate effectively emissions of very similar energy. In this way they are superior to sodium iodide based equipment, which has a much poorer energy resolution.

Nevertheless, it is not possible to produce single detectors of very great size, i.e., their efficiency, comparing with the NaI (Tl) detectors is generally lower. They are also more expensive, require a more sophisticated pre-amplifier and have to be operated at low temperatures, requiring either an electrical refrigerator or the supply of liquid nitrogen. Inevitably, the decision on which type of spectrometric equipment to employ requires the balancing of the metrological advantage of the intrinsic germanium detector with the lower cost and greater convenience of the scintillation detector.

Basic parameters that characterise semiconductor detectors are:

- *energy resolution*: this parameter is more than one order of magnitude better in comparison with NaI(Tl) detectors.
- *peak to Compton ratio*: is defined as the ratio of the count in the highest photo peak channel to the count in a typical channel of the Compton continuum associated with that peak.
- *escape peaks*: the effect is even more prominent for very high energy interactions, where pair production generates a position which can combine with an electron, generating two 511 keV photons; one or both of these can escape, leading to two peaks 511 keV and 1022 keV below the main peak.

Semiconductor detectors have to be calibrated by energy and efficiency calibration. Effective energy calibration is a tool to control that the system is working properly. There are only a limited number of tools to calibrate the detector efficiency for changing of measurement geometry (shapes of sample). This could lead to measurement uncertainties.

The merits of the germanium and sodium iodide detectors can frequently be combined by finger printing materials. In this technique, samples are taken of the material of interest and assessed using a germanium detector for the γ-emitters, and radiochemical analysis techniques for the α- and β-emitters. This generates a detailed record of the emitters present. The contents can be assessed and a limit chosen for a prominent γ-emitter, in $Bq.g^{-1}$, based on the total radioactive content. Sodium iodide based equipment can be used to assess the effective activity of samples by gating it on that γ-line and setting the rejection level at an appropriate point. Hence the higher precision, higher cost equipment can be used to control a larger number of simpler, operationally more convenient, units. A regular sampling programme is required either to demonstrate that the original nuclide mix is being maintained or to correct the setting of the sodium iodide equipment. Changes in nuclide mix can occur, for example, as a result of the change in neutron spectrum with distance from a reactor giving rise to different nuclide ratios in a steel beam.

Table 5.3 Typical drum scanner detection limits

Nuclide	Lower limit of detection [$Bq.g^{-1}$]	
	Average density 0.1 $g.cm^{-3}$	Average density 1.8 $g.cm^{-3}$
^{137}Cs	7×10^{-3}	1.5×10^{-3}
^{134}Cs	6×10^{-3}	1.1×10^{-3}
^{60}Co	6×10^{-3}	1.1×10^{-3}
^{232}Th	2×10^{-2}	4×10^{-3}
^{235}U	1×10^{-2}	4×10^{-3}
^{239}Pu	400	100

Another typical use for germanium detectors is in a drum scanner. This comprises a detector, provided with a collimator, which is mounted close to a drum, mounted on a turntable. The collimator is designed so that it sees, typically, about 10 % of the height of the drum. The drum is rotated and spectral data generated for the first segment. The detector (or drum) is then lifted, so that the detector is viewing the adjacent disc shaped segment of the drum. This process is repeated until the entire drum has been scanned. The segments are analyzed in turn and a complete inventory of the contents generated. This technique is capable of measuring total activity and identifying any excessively high activity volumes within the drum. Table 5.3 illustrates a typical system performance for a 10 minute counting time and three 20 % germanium detectors [14].

As demonstrated, there are several nuclides which are difficult to monitor either as a surface contaminant or as a bulk contaminant, for example ^{3}H and ^{241}Pu. Such radio-nuclides can often be assessed by radiochemical methods. These are much slower and more expensive than direct methods, because of the much longer preparation time before counting. Therefore the main domain of radiochemical analyses is in determination of correlation/scaling coefficients of difficult-to-measure radio-nuclides. For samples representing the given waste stream intended to be released, it is often provided by external, metrologically authorised laboratories.

5.4 Monitoring campaigns

While the measurements at the places of dismantling and demolition are often performed continuously using portable devices, mostly being part of the segregation of the materials in view of their further treatment, the activities (i.e., the measurements itself and other corresponding activities) leading to confirmation of compliance with the clearance levels are more difficult. Therefore, to improve the reproducibility of the measurements and the overall effectiveness of the determination of the values significant for the release from regulatory control, it is advisable to group potentially cleared materials according to their practical characteristics: origin, type of materials, shape and dimensions, package form, type of contamination, etc. and organise the determination of the needed characteristics by campaigns. The use of mobile sophisticated measurement facilities shared by more operators performing decommissioning activities, can improve the effectiveness of the determination of the required characteristics.

5.5 Scaling factors

As indicated above, there are many radio-nuclides in potentially releasable materials, which are considered as safety significant, i.e., with existing clearance limits (see, for instance, [18]) but difficult-to-measure or even impossible to measure by routinely performed monitoring within the given monitoring campaigns. Acceptable determination of such nuclides is built on correlation and/or scaling factors (coefficients)[1].

It must be recognised that the use of scaling factors is a serious and complex matter in waste characterisation, also for clearance purposes [19]. There are many pitfalls and it is easy to be misled. The following general guidance addresses many of these potential pitfalls:

1. Changes to chemistry or process is likely to change the distribution of nuclides, so scaling factors will need to be re-assessed following any significant change.
2. Scaling factors require a technical basis for the existence of a correlation. If there is no technical basis for the correlation, it may not be valid or defensible in practice.
3. Scaling factors are waste stream and case specific, not reactor type specific. As relative distributions of nuclides in wastes are highly process and chemistry dependent, it is uncommon for generic scaling factors to be developed, although it is recognised that the existence of such generic sets would be highly desirable to a waste manager. Where generic scaling factors are used, they usually are applied to only a few specific pairs of nuclides. It is necessary that all scaling factors are demonstrably

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[1] The term ‘scaling factor’ is frequently used very freely. Correlation factors represent coefficients of dependence of derived values on easy-to-measure values, which is generally not linear in the large extent of values. For routine determinations, supposing relatively narrow repeating extent of values, such dependence should be linearised and in this case we can speak about ‘scaling factors’. Establishing scaling factors requires a demonstration on which scale they are valid for an acceptable level of accuracy. In some documents, the term ‘vector’ is used as synonymous for the term ‘factor’. See also Section 4.

valid and the uncertainty for the specific case has been assessed. Before using any generic set of scaling factors, it would be necessary to demonstrate that the real conditions fall within the bounds of conditions that led to the development of the set. For the set to be truly representative, the population assessed would have to exhibit similar design materials, process history and process chemistry. If there are many variables, then there may not be an underlying technical basis for the correlation. In addition, with large populations, the resultant uncertainty may be very large. This is not to say that a generic set of scaling factors could never be valid, but it would have to be assessed and shown to be valid prior to use.

4. It is common practice when developing scaling factors or fingerprints to pre-survey the stream to identify regions of elevated activity. Samples are taken specifically from this upstream area to ensure good quality statistics of the ratios of activities of the various nuclides. This initial scoping is often most practically performed using γ-spectroscopy. There are a few rare occasions where this approach can miss entirely some difficult-to-measure radio-nuclides.

5. The national (or commercial) policy regarding whether to use mean values or conservative values for scaling factors will have to be identified. While there is a sound scientific basis for using mean values as the overall inventory of the repository will be more accurate, it may not be permitted in some jurisdictions to allow underestimating of scaling factors based on the log-log plots used. A situation might be achieved where the real radioactivity in a repository is much smaller than the declared one, resulting in much higher and superfluous costs.

6. There is a trend in certain organisations to use scaling factors to assess also easy-to-measure nuclides in addition to difficult-to-measure ones. In principle, this is not necessary because these nuclides can be directly measured. However, there may be particular advantages for this approach. It is necessary to fully understand the uncertainty implications of both approaches and it is recommended to use the approach with the smaller uncertainty. In addition, it is advantageous to periodically use the alternative approach and compare results to continuously confirm the validity of the scaling factors.

7. If scaling factors are used to assess difficult-to-measure values in waste or waste forms, it is important to validate these factors through destructive analysis. This is specially so for the radio-nuclides affecting the long-term dose to man during the disposal of the waste.

The basic application procedure for the scaling factor method follows four steps, starting from basic studies, sampling and evaluations, and proceeding to the determination of the actual scaling factor values in accordance with the basic application flow shown in Figure 5.2 [20].

When using analytical data on small samples to calculate scaling factors, it is highly important to upscale to full size waste packages. This is especially true in case of variable waste streams. A sufficiently large set of samples should be taken and analysed according to a sampling strategy. Non-destructive analysis on full size waste package will be helpful to verify the scaling factors.

The following nuclides, for instance, should be considered as difficult-to-measure [13]: ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{99}Tc, ^{129}I, actinides. Their determination for the purposes of monitoring of compliance with clearance levels is based on the known relation between activity of easily measurable nuclides (‘key’ nuclides) and difficult-to-measure nuclides based on radiochemical determination of these in representative samples of given waste or material streams. To establish the appropriate accuracy of measurements/monitoring for the overall time period when the clearance of a given waste stream is performed, periodical validation of the scaling factors is needed. Obviously, the determination of scaling factors and their validation should be performed in metrologically authorised, well equipped radiochemical laboratory, often external. Additional measurement data (from sources such as clearance surveys, air sample data, etc.) should also be evaluated for confirmation of the continued validity of scaling factors and may be utilised for improving the accuracy of the scaling factors.

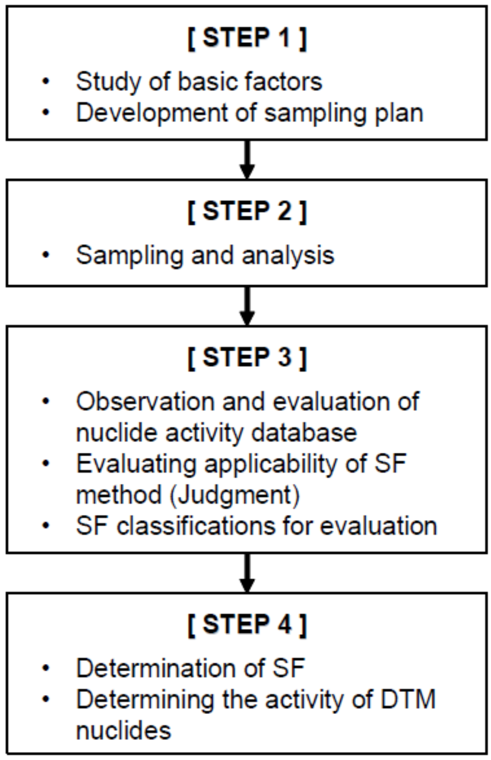


Figure 5.2 Basic flow of application for the scaling factor (SF) method

Typically, characterisation of contamination associated with reactor decommissioning waste uses ^{60}Co for corrosion products, ^{137}Cs for fission products and some α-emitters as key nuclides. ^{241}Am is used for actinides when they are present in appreciable quantities. The key radionuclide is measured by radionuclide specific analysis (e.g., γ- or α-spectrometry). The scaling factor is applied to the results to ensure that all relevant radio-nuclides are accounted for when calculating the total activity. The isotopic ratio is also used to determine reference efficiencies for gross β- or γ-counting instruments.

Corrosion product radio-nuclides show generally 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 α-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 wastes because of differences in solubility and chemical properties. Therefore, it is necessary to determine scaling factors for each waste stream and key radionuclide.

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 radio-nuclides 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 radio-nuclides 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 radio-nuclides 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 radio-nuclides (e.g., ^{90}Sr and ^{137}Cs) than for others and scaling factors may be very different before and after decontamination steps.

The summation formula through all safety significant radio-nuclides is used generally for confirming of the compliance with clearance level. Therefore, some radio-nuclides that are important for waste characterisation may be trivial in clearance measurements. The key determining factors in the former application are the half-life and radio-toxicity 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.

5.6 Statistical analyses

There are two areas within the measurements for clearance purposes, where there is a real need to use statistical methods:

- to study the appropriate sampling which guarantees an acceptable level of representativeness of results, i.e., to assist in the determination of the sampling strategy according to acceptable levels of uncertainty;
- to manage uncertainties on the whole.

Of course, both aspects relate to each other. The first one has a particular importance when the homogeneity of the materials under study can not be guaranteed, i.e., the homogeneity of the material itself and the homogeneity of the activity distribution.

The measurements implemented on the samples are generally used to obtain information about the activity distribution in the material as a whole for a decision on whether the material complies with the clearance criteria or not. Increasing the number of samples gives a better estimation of the median value and the standard deviation of the activity (surface or mass) values in the material. The minimum number of samples needed to make a statistical compliance test depends on the median value and the standard deviation of the activity, the statistical test used and the levels put on the decision errors that are used in the test (confidence level). In some cases the number of samples has to be increased as a result of the measured activity concentrations if the median value obtained from the given sampling is higher than anticipated. The number of samples needed for the clearance decision raises steeply when the median value is close to the clearance value and if the standard deviation on the measured activity is high.

The stratified random sampling approach is used in the ISO standard [15] to determine the number of samples n for statistical acceptability:

$$n \geq 45 \frac{s^2}{\bar{X}^2}$$

where *s* is the standard deviation for the sample and $\bar{X}$ the mean value of the chosen parameter of the *n* samples.

The decision whether to clear a material or not is made by using a statistical test on the measured activity. Different tests might be applied [16]. In the most simplified case of only one radionuclide present in the samples and no background contamination, the null hypothesis states:

> *'The candidate material has a median activity concentration higher than or equal to the clearance activity'*

and is tested versus the alternative hypothesis which states:

‘*The candidate material has a median activity lower than the clearance level*’.

If the distribution function of the activity in the material is unknown, the non-parametric Sign-test can be used. The number of samples needed in a Sign-test is illustrated in Table 5.4 as a function of a normalised median value and normalised standard deviation on the activity. The values are valid for decision errors $\alpha = 5$ % (probability of rejecting the null hypothesis while it is true) and $\beta = 10$ % (probability of accepting the null hypothesis while it is wrong).

Table 5.4 Number of samples needed in a Sign-test as a function of the median activity and the standard deviation of the activity normalised to the clearance level (decision error levels $\alpha = 5$ % and $\beta = 10$ %)

Normalised standard deviation	Normalised median activity						
	0.2	0.3	0.4	0.5	0.6	0.7	0.8
0.2	11	11	11	11	12	15	23
0.3	11	11	12	14	17	23	40
0.4	12	13	15	18	23	37	71
0.5	14	16	18	23	32	52	107
0.6	17	18	23	32	47	71	185
0.7	20	23	30	40	71	107	214

Another testing of hypotheses is recommended in [16] for the less idealistic situation when the contaminant is also present as background. In this case, the null hypothesis can be formulated:

‘*The candidate material has a median activity that exceeds the median background activity by more than the clearance level*’

and, comparing with the formulation of the hypotheses above, the alternative hypothesis can be formulated similarly.

The second area of the application of statistics is the treatment of uncertainty. An ideal monitoring technique could have to be able to distinguish with certainty whether any given measurement is above or below the clearance values. In reality, any measurement of activity will have its associated uncertainty. The fraction of the material which will fall into each of three categories (meets the clearance values, exceeds the clearance values, uncertain) can be estimated using the anticipated activity distribution of the material and the uncertainty associated with the monitoring technique.

Measurements are normally expressed along with an uncertainty at a given confidence level (quoted as either a percentage or a number of standard deviations). 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:

- the distribution of radioactivity in the material;
- the homogeneity of the density inside the material (affecting self absorption of radiation);
- the variation in the radionuclide composition of the contamination;
- the modelling assumptions;
- human error.

Furthermore, for laboratory measurements there will be a 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. 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.

5.7 Quality management issues, documentation

All clearance activities must be performed under a quality assurance program, which is obviously a sub-programme of the quality assurance for decommissioning. This sub-programme will provide a framework within which material disposition must comply. It will need to demonstrate that applicable regulations, codes and standards were met in clearance activities.

Control monitoring of the content of safety significant radio-nuclides in cleared materials, including correct declaration of the content of the difficult-to-measure radio-nuclides is the most significant issue within the clearance process. The purpose of a quality assurance 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 the use of data generated for this purpose.

Transparency and traceability, two key components of a quality assurance programme, are crucial for clearance as quality assurance 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. 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).

Quality management is an integral part of the process for releasing materials from regulatory control. The assurance of the quality of results obtained and used during the release process is critical for ensuring and demonstrating that the established values have been met. Quality management includes, where appropriate, verification that each step of the monitoring process has met the objectives and any necessary corrective action has been implemented. It also needs to provide confidence in the use of data selection of monitoring measurements, sampling, analytical techniques and equipment and interpretation of results. Since monitoring results have important regulatory, public health and societal implications, quality management needs to satisfy the recognised standards established by the regulatory body or internationally.

A quality management programme needs to be designed and implemented by the operator/performer of the clearance process to assure that [16]:

- the relevant requirements and criteria relating to monitoring are clearly defined and met.
- the management arrangements (organisation, roles and responsibilities of managers and other staff members, their competencies, detailed instructions and procedures) are in place documented and have been applied.
- an adequate monitoring strategy has been periodically reviewed, approved, selected, implemented and if necessary modified.
- adequate monitoring techniques have been selected, periodically reviewed, approved, implemented and if necessary modified; this includes selecting appropriate sampling and measurement methods, frequency of measurements, etc.
- procurement control, including sub-contractor services is adequately planned and implemented. As clearance of contaminated material is authorised following the demonstration that residual contamination is less than established limits, the quality management programme needs to particularly focus on equipment or services directly related to measurement. The metrological authorisation for performing given measurements should often be a good confirmation of the sub-contractor capabilities.
- verification and analysis of results has been undertaken. Verification that results of all measurements are accurate, acceptably precise and reliable to enable the appropriate values to be compared with the established clearance limits. Inter-laboratory comparisons (circle analyses) and testing by standard reference materials is a good tool thereof.
- recording and reporting procedures are in place. The quality management programme needs to emphasize documentation of calibration, control and testing procedures, sample management, non-conformity with established values and the proper reporting of results.
- selection, calibration, testing and maintenance of equipment is carried out and covers equipment involved and procedures for their operation.
- appropriate qualification, experience and training of managers and other staff members involved in the monitoring activities is assured. It should be noted, that there are significant differences between radiation measurement within routine operation of facilities and measurements in the context of release of material from regulatory control.
- adequate auditing covering internal and external audits and regulatory inspections is planned and undertaken with a view to ensure that the results are assigned to proper material, location or sample. Such a procedure needs to include inspection of labelling of samples, field-book notations, step-by-step recording and sample tracking.

- measures for identification of non-conformance and adequate corrective actions are provided.

Proper and accurate documentation is required for an operator or its contractors, who frequently participated in the characterisation activities, to demonstrate the acceptability of clearance, release and reuse 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.

Operators responsible for the clearance activities need to retain key records to demonstrate at any moment that clearance has been carried out appropriately. As the release of material will usually not allow further verification of the activity after it has been released, the safe keeping of proper and sufficiently detailed and accurate records is of particular importance. Records (such as sampling forms, measurement protocols, final compliance reports) need to be protected from loss, destruction or falsification by controlled storage in facilities with appropriate protection against fire and other aggression. The storage period should be approved by the regulatory body and usually lasts up to tens of years.

For the purpose of release of materials from decommissioning of nuclear facilities, the results of analyses (surface and/or specific activity) shall document the following information [16]:

- Information enabling unequivocal identification of the material.
- Relevant values and radio-nuclides.
- Reference to the initial characterisation of material.
- Origin and type of material, its history prior to monitoring.
- Clearance values for inscribed ways of release: averaging mass and/or surface, activity concentration/surface contamination of radio-nuclides.
- Contact data of involved organisations and experts.
- Type of contamination: bulk/surface.
- Packaging (if any), weight of package or released material.
- Outputs of the characterisation of released materials with references to relevant procedural documents and standards:
    * the way of declaration of given values (direct measurement, declaration of difficult-to measure radio-nuclides, including the used scaling factors and way of their determination).
    * equipment used and data significant from the metrology point of view.
    * dose rate data, data on the total activity values.
    * sampling (e.g., from the representativeness point of view, data on wipe tests, etc.).
    * results, including statistical treatment of the measurement data.
- Final interpretation of results, i.e., final statement on compliance with the clearance levels, recommended follow-up actions with uniquely determined end-point, respectively.
- Additional remarks (if any); solution of non-conformity, including contact data, date and signature of the manager of the organisational unit which has been notified.

- Administrative information: the protocol number and date; contact data of the company and its responsible organisational unit; contact data, date and signature of the person(s) responsible for performing the measurement; contact data, date and signature of the responsible organisational unit manager.

5.8 Compliance with clearance levels

All measurements, other activities and procedures described in the previous sections lead to the demonstration of compliance with the clearance levels for the given way(s) of release of very low activity materials from regulatory control. On the other hand, the compliance process is a way of establishing the clearance levels themselves.

The primary radiological basis for establishing values of activity concentration for the exemption of bulk amounts of material and for clearance is that the effective doses to individuals should be of the order of 10 μSv or less in a year. To take account of the occurrence of low probability events leading to higher radiation exposures, an additional criterion was used, namely, the effective doses due to such low probability events should not exceed 1 mSv in a year. In this case, consideration was also given to doses to the skin; an equivalent dose criterion of 50 mSv in a year to the skin was used for this purpose. This approach is consistent with that used in establishing the values for exemption provided in Schedule I of the Basic Safety Standards.

The second radiological criterion for exemption set out in Schedule I of the Basic Safety Standards concerns the collective effective doses associated with a practice. It has generally been concluded in many studies that the individual dose criterion will almost always be the limiting factor and that the collective effective dose commitment from one year of the practice will usually be well below 1 man.Sv.

Many studies undertaken at national and international levels have derived radionuclide specific levels for the exemption and clearance of solid material [13, 18, 21, 22, 23, 24, 25]. The calculations are based on the evaluation of a selected set of typical exposure scenarios for all material, encompassing external irradiation, dust inhalation and ingestion (direct and indirect). The values selected were the lowest values obtained from the scenarios. Foodstuff and drinking water pathways of intake were taken into account to consider the radiological consequences.

To be consistent with the Basic Safety Standards approaches to clearance, the derivation of activity concentration values is a principal task for decision whether radioactive material arising from decommissioning activities should be cleared or managed through the radioactive waste management chain. The limiting nuclide concentration values represent at the same time entry for development and application of waste characterisation techniques (see Section 7) supporting such decision and demonstrating the compliance with established limits.

Calculations for deriving the activity concentration values for material containing radio-nuclides of artificial origin proceeds along the following lines [18]:

- Selection of radio-nuclides for which the calculations are carried out.
- Definition of suitable scenarios and parameter values.
- Calculation of annual doses relating to the unit specific activity (i.e., 1 Bq/g) for each radionuclide.
- Identification of the limiting scenario for each set of calculations, i.e., the one that gives the highest dose.
- Derivation of the radionuclide specific activity concentration values by dividing the reference dose level (10 μSv/a, 1 mSv/a or 50 mSv/a, as appropriate) by the annual dose calculated for 1 Bq/g for the limiting scenario for that nuclide.

- Application of rounding procedures to the activity concentration values.

Initially, the radio-nuclides for which activity concentration values should be calculated are those for which exemption levels exist in the Basic Safety Standards. This robust set contains those nuclides that are most relevant to nuclear installations, such as nuclear power plants or fuel cycle facilities, and the application of radio-nuclides, including short lived radio-nuclides, in research, industry and medicine. Definitive establishing of the set of safety significant radio-nuclides depends on the original designation and use of the facility in decommissioning. A number of additional radio-nuclides should also be considered because of their practical relevance in some cases (e.g., ^{41}Ca and ^{79}Se).

Regarding calculations, dose coefficients are used to calculate (annual) doses from a given activity in general. More specifically, dose coefficients are used for the following exposure pathways:

- external exposure;
- inhalation exposure;
- ingestion exposure.

Table 5.5 Scenarios for derivation of the activity concentration values for clearance

Scenario	Description	Exposed individual	Relevant exposure pathway
WL	Worker on landfill or in other facility (other than foundry)	Worker	External exposure on landfill
			Inhalation on landfill
			Direct ingestion of contaminated material
WF	Worker in foundry	Worker	External exposure in foundry from equipment or scrap pile
			Inhalation in foundry
			Direct ingestion of contaminated material
WO	Other worker (e.g. truck driver)	Worker	External exposure from equipment or the load on the truck
RL-C	Resident near landfill or other facility	Child (1–2 a)	Inhalation near landfill or other facility
			Ingestion of contaminated foodstuffs grown on contaminated land
RL-A		Adult (>17 a)	Inhalation near landfill or other facility
			Ingestion of contaminated foodstuffs grown on contaminated land
RF	Resident near foundry	Child (1–2 a)	Inhalation near foundry
RH	Resident in house constructed of contaminated material	Adult (>17 a)	External exposure in house
RP	Resident near public place constructed with contaminated material	Child (1–2 a)	External exposure
			Inhalation of contaminated dust
			Direct ingestion of contaminated material
RW-C	Resident using water from private well or consuming fish from contaminated river	Child (1–2 a)	Ingestion of contaminated drinking water, fish and other foodstuffs
RW-A		Adult (>17 a)	

Reference [18] provides the generic calculation of radionuclide concentration values considering a general and robust set of scenarios (see Table 5.5).

For each scenario, general parameters are defined that characterise the exposure situation:

- Exposure time.
- Decay time allowed before the scenario starts.
- Decay time during the scenario.

Values used for calculations within each type of scenario depend on and reflect the difference in assumptions: realistic versus low probability.

Calculations lead to the derivation of limiting radionuclide concentrations for the clearance route. Considering the complex and robust approach to a scenario followed by the calculation of all possibilities, as illustrated in Table 5.5, and applying the most limiting results (most limiting scenarios results) as values, according to which the clearance is limited and controlled, is called 'unconditional clearance'. On the other hand, the route of disposal of radioactive materials by clearance, i.e., in fact a scenario, given a priori by legislation, decision of regulatory bodies and practice (depending on the regulatory infrastructure, the clearance issue is frequently lying on a border between the competence of the nuclear/radiation safety regulatory body and the regulatory body responsible for environmental protection) is called 'conditional clearance', where the given scenario and corresponding calculation parameters must be invariably unbroken.

When applying the derived activity concentrations, the regulatory body needs to consider methodologies for sampling, averaging, monitoring and detection of radio-nuclides. In doing this, the regulatory body needs to recognise that these activity concentrations were derived for bulk amounts and that the averaging should be done accordingly. These aspects were discussed in previous parts of this section.

It is also necessary to mention that activity in a material is not in all cases fully characterised by the activity concentration. A major portion of the activity may be concentrated on the surface of the material. This is in particular relevant for metals and buildings, but other materials may also exhibit surface contamination depending on their nature and on the origin of the contamination. The difference between contaminants present preferentially on the surface compared with the bulk of a material plays only a minor role for the important pathways of external irradiation and food ingestion, and does not affect exposure estimates significantly. For the inhalation and ingestion of contaminated dust, however, this difference can become very important. A well known example is the massive release of surface-bound radio-nuclides during the thermal cutting of metals, which gives rise to several times the doses that would be expected if the radio-nuclides were evenly distributed throughout the bulk of the material. Therefore, it has to be recognised that for specific situations such as the clearance of metals or the reuse of buildings from nuclear installations, additional criteria relating to surface contamination may have to be applied. This may lead to a decision of the regulatory body not to release some material even if the activity concentration values are not exceeded for the bulk quantity.

6. Material Management Systems

6.1 On-site organisational systems for material management

The first records for a nuclear facility are produced and stored at its siting and conceptual design stage. Subsequent phases in its life cycle (i.e., its detailed design, construction, commissioning, operation and shutdown) will include the production and retention of a large variety of records (Table 6.1). Design records, as built drawings and operational records are essential for the safe and efficient operation of any nuclear facility. This set of records needs to be constantly updated and augmented during the operation of a facility and should include details of any modifications to it, the fuel and waste management records, the radiological conditions, the operational records and details of any unusual events that may lead to the unplanned contamination of systems and structures [26].

Records from all phases of a nuclear facility are important for planning its decommissioning. Although not all of these records need to be included explicitly in the decommissioning plan itself, the process of initial, ongoing and final planning utilizes the pertinent records for, and ultimately achieves, safe and cost effective decommissioning.

As the operating experience of a nuclear facility may be lost when it is shut down, one important element of planning is therefore to identify, secure and store the appropriate operational records needed to support its decommissioning. This process is preferably initiated during the design and construction phase and continues throughout its operation and shutdown. Part of the records inventory from the operation of a facility will become the records for its decommissioning, and it is cost effective to identify these records before it is shut down.

Experience shows that a lack of attention to record keeping may result in an undue waste of time and other resources and may incur additional costs. In addition, the systematic management of records is an essential part of quality assurance (QA) and is often a condition of a facility’s licence. A good comprehensive decommissioning records management system is one specific application of the broader concepts of the protection of future generations and burden on future generations.

6.1.1 Design and operational data required for decommissioning

The current convention is that the requirements for decommissioning are reviewed and incorporated into the design and operational procedures for a new facility. Accordingly, it is important that managing the records generated receives serious and proper consideration at this stage.

During the operation of a facility the information on the original design and modifications to it is normally maintained as a recoverable record. In addition, careful attention is given to the operational records of the facility, for example dose rate surveys, dose commitments, contamination maps, unplanned events and waste management records. It should be noted that these records will form the basis for the records needed for the post-operational phase, including the decommissioning phase.

For existing operating or shutdown facilities without a decommissioning plan, the establishment of a decommissioning plan and strategy is a high priority. This includes the consideration and identification of the records important for decommissioning.

Inattention to the proper identification and management of records from the design, construction and operation of a facility may cause delays during its decommissioning, increase costs and may affect safety and/or the environment; for example, the requirement to reconstruct information could require plant interventions and hence unnecessary exposures of workers to radiation. Some examples of the undesirable consequences of poor record management practices are given in the following:

Table 6.1 Documentation typically collected and archived for decommissioning

Design, construction and modification data	The following design, construction and modification documentation are typically collected and archived: - Site characterisation, geological and background baseline radiological data; - Complete drawings and technical descriptions of the facility as built, including design calculations; - Construction photographs with detailed captions; - Schedules of any construction modifications and their drawings; - Procurement records that identify the types and quantities of the materials used in construction; - Engineering codes; - Equipment and component specifications, including pertinent information (i.e., the supplier, weight, size, materials of construction, etc.); - Facility construction material samples; - Facility design inventories of chemical and radiological material flow sheets; - Quality certifications; - Safety cases for the operation of the facility; - Environmental impact statements; - Pre-operational facility testing and commissioning records; - Licensing documentation and operating requirements; - Preliminary decommissioning plans.
Operating, shutdown and post-shutdown data	The following documentation should be collected and archived during the operation, shutdown and post-shutdown phases of a facility: - The licence and licensing requirements; - Safety analysis reports; - Technical manuals; - Details of environmental releases; - Facility logbooks; - Facility and/or site radiological survey reports; - Operating and maintenance procedures and records; - Abnormal occurrence reports; - Decontamination plans and reports; - Technical specifications (limits and conditions); - Design change reports and updated drawings; - Hazardous material inventories; - Process and service interfaces with other facilities; - Process flow-sheets, including for services; - System, structure and component inspection records; - On-facility waste management records; - Site hydrology and groundwater contamination records; - Records of equipment terminations (e.g., piping and cables) during operation and at shutdown; - Records of staff leaving debriefings; - Quality assurance records; - Fuel geometry, performance (i.e., damage) and accounting records; - Records of neutron fluxes and distributions; - Records of waste management strategies and locations of waste; - Records of radiation sources and their locations; - Samples of irradiated and embrittled materials; - Relevant laboratory test reports.

- *Design, construction and modification data:*

  Site characterisation, geological and background baseline radiological data:

  * No target for the restoration of the site.
  * Site termination surveys more technically difficult.
  * More time, resource and equipment use required.
  * Future litigation, owing to inadequate data.
  * Significant regulator interface on the potential environmental, health and safety issues.
  * Licence termination documentation potentially large and complex.
  * Impact on decommissioning strategy and cost (i.e., significantly increased waste management).
  * Considered to be a significant issue for facilities handling naturally occurring radioactive material.

  Complete drawings of the facility as built and the technical description of the facility, including design calculations:

  * Complicates the knowledge of and access to contaminated areas.
  * Time and money spent on reconstructing the record and on calculations.
  * The safety case may be delayed.
  * Direct effect on the decommissioning strategy and an impact on time scheduling.
  * Much more safety and environmental planning to deal with unknown situations and more contingency required, for example for resources and financing.
  * Considerable increased regulatory interaction to clear the safety case.
  * Cannot move to decommissioning without this data being available or reconstructed.

  Procurement records of materials during construction and through the lifetime of the facility:

  * An adequate theoretical assessment of neutron activation materials (for reactors) of is more difficult and hence waste cost estimates become difficult, which leads to considerably more sampling of the facility (this has implications for workforce safety and decommissioning costs).
  * Can affect the waste management aspect of the decommissioning strategy.
  * Causes difficulty with estimating the potential dose uptake, which leads to conservative decommissioning strategies, which will affect decommissioning work packages.
  * Implications for the selection of decontamination techniques.
  * More regulatory intervention by the regulators.
  * Time, resource and cost implications for the strategy to be used, and a time delay.

- *Operating, shutdown and post-shutdown data:*

  Environmental releases (over the lifetime of the facility):

  * Lack of assurance on off-site and site contamination.
  * Public concern on potential and potential long-term litigation.
  * Will need regulatory intervention regarding previously unrevealed historical events.
  * Will need to reconstruct data via extensive sampling.
  * Potential to be forced to do clean-up operations that are not the responsibility of the facility.
  * Unable to confirm adequately the baseline site characteristics.
  * Potential difficulty in releasing land for other uses.

  Abnormal occurrence reports:

  * The need to deal with unknowns, which can give rise to an unexpected operator risk, and will give the regulator, public and workforce a lack of confidence in the management of the decommissioning.
  * Unexpected waste arising and workforce dose and/or chemical exposure.
  * Will have an impact upon the decommissioning strategy.
  * Will cause delays.
  * Will cause a substantial change in the strategy.
  * Will increase time, costs and resources, which can have an impact upon the ability to release land.

  Records of terminations (disconnections, removals, etc.) of pipes, cables and vessels:

  * Unexpected hazards arise.
  * Lack of records will lead to a lack of confidence by the regulators, the public and the workforce.
  * Potential for cross-contamination.
  * Will interfere with the development of the work programmes, and hence contingency will be required.
  * Extensive surveys will be required.
  * Additional waste generated.

All these issues can affect contract bids. Inadequate planning contingency could lead to increased safety hazards, worker dose implications and a financial shortfall.

6.1.1.1 Decommissioning strategy

The two most common decommissioning strategies are immediate dismantling and deferred dismantling. A combination of these options, known as phased decommissioning, which consists of periods of active dismantling interspersed with safe enclosure phases, is also common. A third strategy, on-site disposal (entombment), which is the permanent disposal of the facility, or parts thereof, within the site on which the facility operated, is generally used only in special cases.

Immediate dismantling is the strategy of active decommissioning being undertaken soon after the facility is shut down. For the purposes of planning and implementation it is important that a complete, up-to-date and validated set of records is available to those who carry out the decommissioning. In the event that the operational records needed for decommissioning planning are incomplete, the knowledge of key operational staff becomes an important component to improve and enhance the operational record. It is useful to have within an organisation's management system a requirement to debrief key staff when the facility is shut down or when they cease working for it.

It is clear that record keeping for a deferred dismantling strategy involves long term record storage and that retrievability concerns are significantly greater than for immediate dismantling. There may only be a few people with a detailed knowledge of the shut down facility at the beginning of its dismantling; debriefing staff at the shutdown of a facility, or when they cease working for it, is therefore particularly important for this strategy. It is important that the debriefing is structured, of good quality and is itself a well maintained record. In case of deferred dismantling, which may happen decades after closure of the facility, the opportunity for debriefing personnel will probably no longer exist when the decommissioning actually begins. Full reliance will therefore have to be given to the records assembled during the design, the construction, the operation and the shutdown of the facility, and to earlier personnel debriefings. These records will have to be stored for future use over a period of several decades.

It is evident that issues such as legibility, preservation and retrievability over such long time spans are important for this strategy.

6.1.1.2 Primary data sources for decommissioning

The main sources of data from which a records management system for decommissioning can be selected and assembled at the end of the lifetime of a facility are:

- Design, construction and modification data;
- Operating, shutdown and post-shutdown data.

Typical questions concerning which design and operational records should be selected to support the decommissioning are:

- Is the record needed to support and authorise the continued safe operation of the facility?
- Is the record needed to comply with a licence condition and/or other statutory requirement?
- Is the record needed to quantify and characterise waste on the site or to be sent for disposal?
- Is the record needed to provide information for future decommissioning activities?
- Is the record needed to support the long term care and maintenance of the facility and site?
- Is the record needed to preserve and/or record the staff dosimetry and health records or for staff welfare?
- Is the record of a type that is neither directly related to operations nor decommissioning but that nevertheless needs to be retained?
- Is the record a new data that has arisen since the last review of the records?
- Is the record likely to be needed to defend against any possible litigation?
- Is the record considered to be non-permanent?

6.1.1.2.1 *Design, construction and modification data*

The data usually generated during the design, construction and modification of a facility are given in Table 6.1.

A baseline radiological, environmental and geotechnical characterisation of the site for the proposed facility will normally be required for the purposes of its licensing.

A quantification of the natural activity in backfill soil and the building materials used in construction is an essential component in demonstrating compliance with future clearance and target clean-up levels. Samples of selected soil and construction materials for future analysis are also typically part of the records archive. This information is important for the future restoration of a site to its baseline condition.

Geotechnical surveys are normally carried out for structural reasons and to identify site hazards. These surveys provide important records, particularly for the purposes of the reuse of the site after the decommissioning process.

Full details of the design specifications and information relevant to the siting, final design and construction of a facility should be retained as part of the information needed to assist in its operation and eventual decommissioning. This information should be maintained, reviewed and updated throughout the operational lifetime of a facility. The maintenance of a records management system used during this process is clearly a responsibility of the operating organisation. As noted in Table 6.1, such information may include as built drawings, models and photographs, the construction sequence, piping schematics, the details of construction, cable penetrations and repairs to or deviations in components and structures. In addition, all relevant information relating to the environmental condition of a site prior to the operation of a facility is essential and relevant for any environmental impact study.

As a means of assuring adequate attention to maintaining up-to-date drawings, strong procedural emphasis on quality assurance during the design and construction period is essential and should be extended throughout the operating phase and into decommissioning (see Sub-section 6.1.4). During the operation of a facility (see Sub-section 6.1.1.2.2), modifications to the buildings and systems will occur, which will lead to modifications to the design and construction records.

During the lifetime of a facility it is important that the documentation be regularly and independently audited to assess specifically its fitness for decommissioning purposes.

6.1.1.2.2 *Operating, shutdown and post-shutdown data*

To facilitate a successful decommissioning, accurate and relevant records should be kept during the operating phase of a facility. Table 6.1 outlines the records produced during the normal operation of a facility. These records include safety and licensing information, operational manuals and logs, maintenance records, radiological surveys, as well as any information pertaining to abnormal occurrences.

If these records have not been or are not being maintained, it is desirable that such record keeping is initiated as soon as possible. The records should be organised so that those most relevant to decommissioning can be identified. It is important that data on modifications to a facility or its processes are recorded and maintained. In addition, the record of the final condition of a facility at shutdown is essential, in particular for identifying any systems that have been terminated or isolated prior to the decommissioning.

The records of maintenance are particularly important as they give information that includes:

- Special repair or maintenance activities and techniques (e.g., temporary shielding arrangements or techniques for the removal of large components);

- Details of the design, material composition and configuration of the facility as built and the location of all temporary experiments and devices.

It is recognised that documenting good practices during the operational lifetime of a facility will also be valuable for decommissioning. Specific operational benefits realised during maintenance or refurbishment from the use of good practices such as minimising radiation doses and from greater working efficiencies will also be relevant.

The experimental irradiation of specimens of selected materials used in the construction of the installation may assist in comparing measured and calculated activation levels for the final radioactivity inventory.

The management of records becomes particularly important at the end of the operation of a facility. If adequate attention has been paid to records management during design, construction and operation then the data for decommissioning will be readily available. If the recommended approach of continuous record keeping has not been properly implemented then immediate corrective measures are necessary in order to identify and flag those records from all the records of the facility. That will enable, if required, the transfer of information to the decommissioning records management system.

6.1.2 Process of selecting decommissioning records

Reaching the decommissioning stage of a facility will have a significant impact on the importance of the surviving records. It is clear that many of the records derived from the operation of a facility are not required for its decommissioning, but it is also clear that additional data may be necessary. Other records may need to be retained for legal reasons even though they may not be directly relevant to decommissioning. Creating the full set of records essential for decommissioning only after the shutdown of a facility is time consuming and difficult. Planning for decommissioning requires the relevant data of all stages of the life cycle of a plant (Table 6.1) and the creation of new records. This normally includes radiological measurements (i.e., dose rates and contamination levels), a cross-check of the drawings of the facility as built and may include a three dimensional computer aided design simulation. To minimise delays and profit from the experience of the operating staff it is preferable that most of these new record keeping activities be completed as early as possible, before the final shutdown of the facility. However, data may change during the remaining lifetime of the facility (and characterisation activities will have to be repeated). Some caution is needed when using historical data, as they might be obsolete or inaccurate.

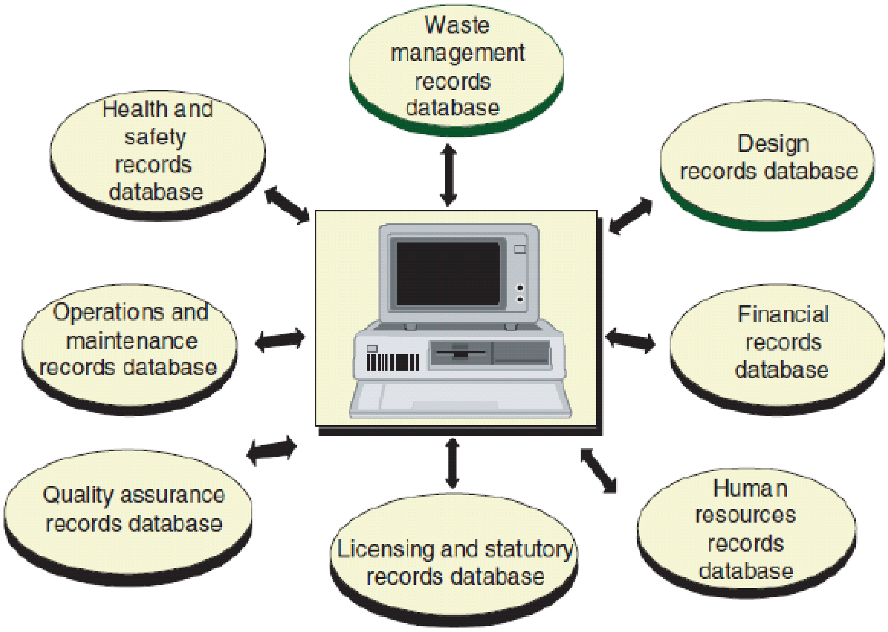
6.1.2.1 Establishing the records management system

There exists an opportunity to build a data collection and record keeping system to operate as part of an integrated facility information system for a nuclear facility at its design and construction stage, as noted in Sub-section 6.1.1. Information is available from different sources (the operation of the facility, maintenance, radiological protection activities and waste management) and in several forms (as figures, images, samples and reports). An integrated facility information system can be designed to provide retrieval and manipulation of the data in a transparent way for its users. Records should be indexed by elements such as the classification of the record system, their type and their location. Further, it may be particularly helpful in any such database to flag data that may be of particular importance to decommissioning.

Scanning the hardcopy records required for decommissioning into an electronic records management system may provide enhanced search and retrieval capabilities, as well as providing a back-up for the hardcopy records.

Figure 6.1 illustrates the typical elements of a computer based integrated facility management information system. A computer system can enable access to all categories of records. If records are in a hardcopy form, their storage location is important for their retrieval, whereas if records are in an electronic format they can be accessed directly and displayed. It is important to audit an information system regularly to determine throughout the lifetime of a facility if the system contains adequate data for its decommissioning (see Sub-section 6.1.2.3).

Figure 6.1 Integrated management information system



6.1.2.2 Selection of decommissioning records

An initial decommissioning plan should be developed during the design and construction of a facility and should be regularly reviewed during its operation. In order to start decommissioning, an overview decommissioning plan is usually required for the licence. Subsequently, detailed work programmes should be produced to make choices on the technology to be used, to develop a waste management strategy and to identify the resources that will be required. In addition, these plans contribute to the selection of the decommissioning records (Table 6.2) for further guidance. A validation of the documents and data to be utilised for decommissioning is essential for the safe and cost effective planning of the work programme. This process will in general entail a considerable reduction in the number of documents from those used for the operation of the facility. The selection criteria therefore are to identify the documents needed for decommissioning and waste management strategies, as well as those required by legislation or for addressing potential future litigation. The selection is based on a review of the existing records and of their relevance:

- The statutory and regulatory requirements.
- Their support for engineering and the safety of immediate and future facility decommissioning activities.
- The operator's legal defence against possible future litigation.

A typical systematic selection process for records may benefit from reviewing the set of questions given in Sub-section 6.1.1.2.

Records are selected for the purposes of decommissioning depending on a number of factors, such as the:

- Decommissioning strategy to be followed (see Sub-section 6.1.1.1).
- Full availability of up-to-date as-built design, construction, modification, operational and shutdown records or the need to reacquire any or all of these.
- Availability of the human, technical and financial resources, in house or through contractors.
- Legislative and regulatory aspects, including requirements for sensitive records, their redundancy, etc.
- Characteristics of the facility, operational records, system specifications, main piping isometrics and supports, layout drawings, etc.
- Radiological characterisation data and records.

Table 6.2 Typical records required before and during decommissioning

Important records to be produced in preparation for decommissioning	- The decommissioning strategy selection document and associated plans. - The design, construction and operational records to be retained or transferred to the operating organisation at the start of decommissioning. - The decommissioning plan and subsequent amendments. - The decommissioning project quality assurance programme. - Decommissioning safety assessments and reports. - The work programme and associated work packages and records. - Manufacturing and construction records as built, including engineering drawings for any installation or construction work done to assist, or as part of, decommissioning. - Initial radiological survey reports. - Environmental assessment reports, including environmental impact assessments. - Project management plans. - Funding and financial documents, including costs and schedules. - Licensing documentation. - The decommissioning organisation.
Important records produced during decommissioning	- Engineering drawings that indicate the state of the facility on the completion of each defined decommissioning phase. - Decommissioning team personnel radiological dose records. - Radioactive and chemical material waste records and disposal records. - Release material records. - Photographs taken of the facility and site during decommissioning. - Details of significant abnormal events during decommissioning and the actions taken. - Project progress and status reports. - Intermediate and final radiological survey reports. - Routine surveillance, maintenance and monitoring records. - The final decommissioning report.

6.1.2.3 Auditing of the records management system

As noted in Sub-section 6.1.2.1, it will be beneficial to audit regularly the design, construction and operational records management system to assess whether it properly manages the records important for decommissioning. The auditing process is intended to ensure that proper records are flagged for inclusion or consideration at the time of decommissioning. The auditing team typically includes decommissioning specialists, information specialists and facility operations staff. In addition, the auditing should emphasize to the operational staff the importance of decommissioning and the need to manage the records management system properly. Well before the facility is shut down, a decommissioning team should be put in place to select the documents from the operational records management system needed to form the decommissioning records management system. Where necessary and ahead of the shutdown any gaps in the records should be reconstructed and any remedial action taken.

6.1.2.4 Documentation prepared for decommissioning

Typical decommissioning records required for and produced during decommissioning are listed in Table 6.2.

Detailed decommissioning plans contain a description of the planned decommissioning activities. These plans include a description of the methods used to ensure the protection of the workers, the public and the environment against radiation and other hazards. In addition, an estimate of the waste expected to be generated during a project is included. A detailed radiological and material inventory is crucial to the planning of any decommissioning project. Information such as the levels and locations of contaminants and quantities of specific radio-nuclides present in the areas of the facility to be decommissioned is needed to develop these reports. One special aspect is the topic of the characterisation of inaccessible areas. Certain areas may not have been routinely accessible during normal operation, but workers may need to work in them during decommissioning operations. Acknowledge of the radiological conditions in these areas, for example around the reactor enclosure or within the reactor bio-shielding, will serve to facilitate decommissioning by minimising occupational radiation exposures.

Another input into the decommissioning plan is the records of spills or other unusual occurrences that took place over the operating lifetime of the facility and that may have resulted in contamination that remains and that suggest potential locations of inaccessible or concealed contamination (e.g., under repainted surfaces or floors). Records of such events indicate conditions in a facility that could adversely affect health and safety and are therefore needed to assist in the progress of the overall decommissioning project. These records could be used to minimise radiation exposures during decommissioning activities. For example, the decommissioning records would contain information on radiation sources that could otherwise be overlooked after the period of operation, such as buried pipes, remote surface locations contaminated with radioactive material or multiple layers of paint that may conceal contaminants. A generation and long term management of such records during the operation of a facility is important to assist in decommissioning. Another important characterisation aspect is that of waste management. It is essential that the location, physical and chemical content and concentrations of both the hazardous and radioactive wastes stored at a facility or on its site be well documented and readily available. Of particular importance is waste and debris placed in temporary cells, pits or vaults. Details of modifications to the plant and maintenance experience include the records cited in Sub-section 6.1.1.

It is also important that material test certificates and coupons be retained to assess the influence of radiation exposures (e.g., neutron activation). Activated or contaminated materials reports generated during the operation of a facility (e.g., during maintenance) are useful because of their applicability to future decommissioning (e.g., to validate activation codes).

Up-to-date information on the systems and components of the facility is essential. Undocumented changes will cause some filed drawings to be inaccurate. If as built drawings are poorly maintained or incomplete, they may need to be reconstructed and verified.

Typical applications of records for reactor decommissioning purposes are given in Table 6.3.

The decommissioning documents and records produced for legal and regulatory purposes play a key role in the process of records management. Typically they are identified in licensing documentation and in the decommissioning plan. Ideally these documents may serve purposes other than complying with regulations, for example for the engineering of decommissioning activities. It is recognised, however, that there will be some documents generated exclusively or primarily for legal and/or regulatory purposes. Depending on national legislation and regulations, decommissioning plans, or relevant parts thereof, should be submitted by the operating organisation to the regulatory body for review and/or approval. If the selected decommissioning option results in a phased decommissioning with significant periods of time between phases, a higher level of detail for the items may be required for the next decommissioning phase being executed. As a result of completing an individual phase of decommissioning, some modification to the planning for subsequent phases may be needed. In such cases, subsequent sections of the decommissioning plan may require updating.

Table 6.3 Sample applications of records for decommissioning a nuclear reactor

Design and construction information	Decommissioning application
Structural details, including concrete pour drawings, rebar placements, penetration locations (as built).	Demolition support: core drilling, blast placement, access considerations.
General arrangement drawings.	Material flow, traffic control and activity sequencing.
Fabrication specifications of reactor vessel and internal packages identifying and disassembly procedures, material planning specifications, construction details and arrangements, including vessel support and recirculation system interface details.	Radionuclide activation analyses, assembly, disassembly and segmentation automated cutter and manipulator designs and mock-ups, reactor cavity modifications.
Nuclear steam supply system component drawings as built, arrangement drawings with supporting structural interfaces.	Removal and disassembly procedures, rigging, transportation and disposal scenarios.
Equipment and system specifications, manufacturers' as built and as installed associated arrangement drawings and piping layouts.	Removal and dismantling sequencing and scheduling, system and equipment turnover and/or conversion for decommissioning operations.
Construction aids: photographs, installation and placement records, scale models and mock-ups.	Decommissioning and dismantling planning support.

6.1.3 Retention of decommissioning records

6.1.3.1 Records produced during the decommissioning phases

Some of the selected records will only be preserved for a limited period of time, owing either to current regulations or because they are related to the service life of equipment. Examples of the types of documents produced during the decommissioning phases are given in Table 6.2. The majority of these types of records are technical documents that relate to systems to be dismantled. They will be processed on a regular basis and are then either classified for further retention (for deferred dismantling) or discarded. Some of the benefits of creating an integrated decommissioning records management system are:

- to assist in substantiating the integrity of a facility at each stage of its decommissioning and dismantling.
- to assist in substantiating the manner and means of the decommissioning of a facility, including interim maintenance provisions.
- to estimate costs and waste quantities.
- to enable the identification, recovery, safe storage and disposal of radioactive material.
- to assist in minimising occupational radiation doses during decommissioning.

- to identify shipments and the disposal locations of waste.

Once the decommissioning records management system has been established, operating protocols, methods and an organisation to maintain it will be required. By training the personnel and building awareness of the work involved, any record produced by, transmitted to or received at the facility will be capable of being integrated into the records management system.

A strict observance of records management procedures to ensure the control and integrity of the original master copies is required to preserve the integrity of the system.

Periodic review and update operations should be performed, in order:

- to sort the temporary archives after each dismantling phase:
- to include in the archives certain important documents, the significance of which was noted in the course of the decommissioning process.

The records management system created prior to decommissioning is part of the project management system for the decommissioning operations. It also serves as a basis for the utilisation of feedback from decommissioning activities.

6.1.3.2 Records produced after termination of the nuclear licence

An issue to resolve is that of which decommissioning records are to be preserved and for what period of time after completion of the decommissioning activities. This matter needs to be viewed in the light of a possible transfer of ownership of the site after the final clearance. Also, once the project has been completed, there must be a long term management of the knowledge base of the facility. Any caretaker responsibility, including keeping relevant records for potential litigation or other purposes, is then likely to be transferred to other institutions, as required by the country laws and regulations. After a facility is decommissioned, the records may be required because of, among other things, new regulatory positions (e.g., on clearance levels) or the development of more advanced, higher resolution detection equipment. Typically, the national regulatory body or another institution would take over keeping the records from the operator of the facility. The duration of records control is usually determined by a country regulation for records for, for example, occupational exposures and potential future liability. Other records may need to be kept for institutional purposes or other ad hoc reasons.

The same considerations are necessary for the long term preservation of decommissioning records, as discussed in Sub-section 6.1.5.

6.1.4 Quality assurance

6.1.4.1 Record keeping as part of the quality assurance programme

The responsible organisation will have a quality assurance programme for a nuclear facility as an integral part of its management system. The quality assurance programme should be modified at various stages (e.g., at the siting, design, construction, commissioning, operation and decommissioning of a facility) at a time consistent with the schedule for accomplishing stage related activities. In order to ensure that all activities are continually carried out under controlled conditions, the decommissioning quality assurance programme is normally developed from the operational quality assurance programme, and may overlap it. For facilities not having a quality assurance programme in place it is important that an ad hoc quality assurance programme for decommissioning will be developed before the decommissioning activities begin. Typical elements of a quality assurance programme are:

- A description of the quality assurance programme;
- Personnel training and qualification;

- Quality improvement;
- Documents and records;
- Work processes;
- Design;
- Procurement;
- Inspection and acceptance testing;
- A management assessment;
- An independent assessment.

A records management system (see Sub-section 6.1.5) is an important part of the overall quality assurance programme for each facility. This system ensures that records are specified, prepared, authenticated and maintained, as required by applicable codes, standards and specifications. The records management system ensures that records are:

- categorised and organised (e.g., Table 6.1);
- registered upon receipt;
- readily retrievable;
- indexed and placed in their proper location;
- stored in a controlled environment;
- corrected or supplemented to reflect the actual status of the plant.

The quality assurance programme also provides some form of routine review of the quality and completeness of the records, based on the information required.

Some details on organisation, responsibilities, the transfer of ownership, records with a special status and the loss of operational records are given in the following sections.

6.1.4.2 Organisation

During the design, construction, operation, shutdown and decommissioning of a facility a large amount of information will be generated and will require management. There is also a need to identify clearly the organisation(s) responsible for the records management system as early as possible in the life cycle of the facility.

Within some countries there may also be a requirement that the regulatory body approves the specification of the minimum records required, their content and the procedures for any records management system.

The organisation responsible for the records management system may be, depending on a number of factors, the operating organisation, a department of the country, an agency or any other appropriate organisation designated by law.

Figure 6.2 is presented as an example of an organisation chart for records management during decommissioning. Regardless of the decommissioning strategy or intended programme, it is essential that an organisation exists to administer and control the documents, as there will be a large volume of records at the time the facility is shut down.

It is useful to appoint a senior manager, with sufficient resources and authority, responsible for undertaking the management of the records and for administering the records archive. The procedures for records control should be overseen by a quality assurance department or records management officer. All trough-life auditing function should be ensured.

Further information on the practical aspects of records management during decommissioning is given in Sub-section 6.1.5.

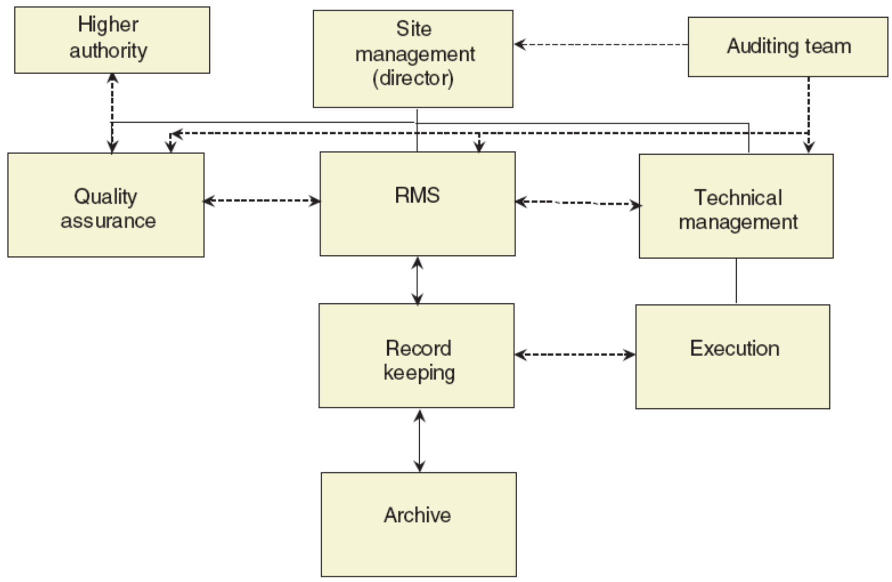


Figure 6.2 Organisation chart for the management of decommissioning records

In many facilities the records management organisation will already exist as a requirement of the licence for its operation. The organisation for the administration of its records during operation of the facility could be much larger and more complex than that required for decommissioning. In many cases this organisation could be transferred to the decommissioning organisation after shutdown of the facility. This is desirable in order to facilitate some continuity in its organisation and because the vast majority of the records for decommissioning will arise from the detailed design, construction and operating records archive. Where no proper records management organisation exists (e.g., for very small facilities), then an organisation will have to be created as a priority after shutdown to avoid the loss and dispersion of the records and to register the new records that will be generated.

6.1.4.3 Responsibilities

To ensure continuity of the management of the records through time it is essential that the line of responsibility from senior management be maintained (see Figure 6.2).

The administration and management of the records is a direct responsibility of the management of the site. The quality assurance responsibility is separate and, to give independence, encompasses all decommissioning activities, with the quality assurance unit reporting to the site management or directly to a higher authority (e.g., a corporate body). The site management identifies within the organisation who is responsible for defining, developing, operating and maintaining the records management system.

Different groups within an organisation may use different record management systems (see Figure 6.1), but there should be cross-references between them. Each record should be referenced uniquely, even if shared between systems.

The site manager ensures that appropriate quality assurance provisions are applied throughout all of these tasks and ensures that the defined goals are achieved.

The responsibility for identifying and selecting the records for decommissioning is a technical function and lies with those who plan and execute the decommissioning strategy and activities. The responsibility for collecting, indexing, cataloguing, recording and archiving the records is an administrative function and resides with the records management group.

6.1.4.4 Changes in ownership or management

An issue that may be encountered is the transfer of ownership or management of a facility during its operational life. This may also occur after a permanent shutdown or during any phased decommissioning. Recent developments in the nuclear industry aimed at enhancing competitiveness may lead to organisational changes, such as a merging of operating organisations or transferring ownership from one operator to another. In some countries a national authority (a national decommissioning operator) takes over from a former private operator at the time of shutdown of a facility. These ownership or management transfers produce new responsibilities for record keeping and are often subject to an in-depth regulatory scrutiny. Under these circumstances the safety and cost effectiveness of decommissioning can still be ensured as long as the new operator has full access to and an understanding of all the existing records and provided that a continuity of the records management is maintained.

6.1.4.5 Records with a special status

There could be records that have a special status that limits their availability or distribution. These are typically designated as classified (e.g., for facilities transferred from a military to a civilian usage), confidential or proprietary records. The issue of proprietary research records may be sensitive, and special provisions may be needed to address these concerns without hindering timely decommissioning. It is important that a record of the existence of these records be made in the records management system, with a brief description given.

6.1.4.6 Loss of operational records

Despite quality assurance provisions and other means intended to ensure records are preserved, accidents or incidents such as fires, floods, bankruptcy or human errors (e.g., the inadvertent deletion of electronic files) may result in a loss of records important for decommissioning planning. In such cases knowledge may be regained to some extent through mechanisms such as an additional characterisation of the site and facility, interviews with staff familiar with the affected systems or an assessment of systems at similar facilities.

In any case, it is felt that decommissioning planning and implementation may still proceed safely even under such circumstances. However, the caution needed in the planning of and in implementing decommissioning with unknowns will generally result in extra costs and delays. In some cases a trade off between extra characterisation efforts and the need to decommission with significant unknowns may be in order.

6.1.5 The records management system

A records management system is essential for the collection, cataloguing, maintenance and dissemination of records for the required timeframe, which could be several decades. The records management system needs to be established with written instructions, procedures or plans with quality assurance procedures, and regular independent auditing is necessary at all stages.

The primary focus of a decommissioning records management system is to ensure that the relevant records are selected to support decommissioning and that the data sources are validated, as appropriate. This may include the preservation of the necessary information for the duration of the active institutional control period and, where necessary, beyond this period. The information may exist in many media forms. Issues that need to be addressed through a system of documented instructions, procedures and plans, to ensure that the integrity of the information is preserved, may include:

- the requirements and responsibilities of all parties;
- the identification of records, including the validation of data sources;
- the transmittal, receipt and acceptability of the records;
- record indexing and retrievability;
- record retention classification;
- the record medium (e.g., paper, microfilm or electronic) and the primary and secondary storage locations;
- the protection of the records from adverse environments;
- access control;
- the control of modifications to the records;
- the periodic reproduction or transfer between record forms;
- the national and international archives requirements.

Details of these aspects are given in the following sub-sections.

6.1.5.1 Requirements and responsibilities

The organisation responsible for decommissioning (for example the site management) will generally be responsible for allocating record keeping responsibilities and approving the relevant procedures required by national legislation and/or regulations. Typical functions include:

- Taking decisions on which records are to be inserted in the records management system or on modifications to the existing records.
- Ensuring that access to the records management system is controlled.
- Maintaining the records management system and its records, including any required long term preservation.
- Providing input to and output from the records management system.

These functions will be carried out by various groups, including the decommissioning staff, quality assurance staff and information management staff.

The allocation of these responsibilities will be specified in formal procedures.

Depending on the subject, scope and strategic objectives of the actions associated with the records management system, the management may be involved in the decision making. Examples include decisions on granting access to specified organisations or establishing restrictions on access.

The group that manages the records management system includes those responsible for:

- establishing the requirements for managing the decommissioning records;
- managing the information, for example inputting record entries, modifying existing records or producing outputs upon request (usually information management reports);
- Maintaining the records management system.

It should be noted that, although the records management system group is responsible for managing the physical records, it is the technical decommissioning group that is responsible for generating the documents and records and ensuring that this information meets the regulatory requirements.

6.1.5.2 Identification of records, including validating data sources

The existing operational and technical staff should be used as much as possible in the identification and verification of the records for decommissioning prior to the start of the actual work. The records that are typically collected and archived for decommissioning are shown in Table 6.1. Sub-section 6.1.1.2 and Sub-section 6.1.2 also provide guidance on how to identify the records that will be important for decommissioning. Verifying the records is complex and needs to be established on a case-by-case basis by a decommissioning team, based on the type of information, the quality control applied to the original information and some verification, evaluation and review of the information. This verification process is normally supported by independent auditing.

6.1.5.3 Transmittal, receipt and acceptability of the records

Procedures for collecting, transmitting and incorporating information into an records management system should be established and should include provisions for verifying the acceptability of each record. It is important that each record be legible, official, accurate and complete.

6.1.5.4 Record indexing and informational retrieval

A detailed records index that captures information from the operational records management system should normally be established as early as possible. It should be maintained as records are inserted in the records management system throughout the decommissioning process. It is important that accuracy be checked and validated through quality assurance procedures.

The retrievability of information and hence keyword searching and record indexing are necessary components of a records management system. Indexing systems should link record attributes such as the title, date, subject, abstract, source of the record, keywords for the location of the record and other information. Information retrieval is taken to mean that once the record is located it can be accessed with existing tools and read. The timely retrieval of records management system information for decommissioning is directly related to the effectiveness of the selected indexing system. All these activities can be done manually without an automated system, but it is desirable to use automated systems for the enhanced search and access capabilities that they provide.

6.1.5.5 Record retention classification

Records management system records may be subject to varying statutory periods of retention, based on their expected future use, or may have long term value as historical records. Such requirements should be considered when assigning a record retention classification.

Each country typically establishes retention policies to ensure the availability and future use of information.

If records are classified with varying retention periods, the classifications and controls for assigning classifications are documented in instructions, procedures or plans. Controls may include periodic reviews to evaluate established classifications and, if necessary, to reclassify records. It is recommended and often legislated that duplicate back-up copies of records be maintained and securely protected in a separate location.

The disposal of redundant records should be subject to explicit, written procedures and controls to minimise the risk of inadvertently losing important information.

6.1.5.6 Record medium and location

The organisation responsible for the records management system normally establishes a set of documented instructions and procedures to control the process of the identification, collection and preservation of information. It is important that record archives be retained at least for the full decommissioning period in a secure system that minimises damage, deterioration and loss. The retrievability and usability of records may be dependent on the continual review of and migration or conversion to new technologies. Knowledge of the location of records, including back-up copies, is essential to ensure and demonstrate retrievability at any time.

The medium chosen to store the information should meet the following requirements:

- It should be capable of capturing and storing the required information.
- It should have physical and chemical stability so that legibility is preserved for the required timeframe.
- It should be capable of being easily copied or transferred to another medium, without loss of information.
- It should be retrievable over extended periods of time, as required.
- It should be readable and clear.
- It should be resistant to alteration by unauthorised individuals.

Most countries currently require the management and storage of the original hardcopy records to meet their legal and regulatory requirements.

6.1.5.7 Protection of records from adverse environments

Based on the record medium selected, appropriate controls can be established to protect records from deterioration due to, for example, temperature, humidity, light and micro-organisms.

The objective of storage is to give protection against loss due to a single event such as a fire, flood, tornado or earthquake. This protection can be accomplished by engineered protection such as a vault and/or the separate storage of a duplicate set of records in a secure separate location. Two independent and separately located archives are desirable (e.g., one at the plant and one at the company’s headquarters). The consistency of the contents of each archive is crucial and should be ensured by regular reviews.

It is important that records be both protected and yet easily available to the decommissioning staff when needed. One possible method of achieving this is by having a tightly controlled and managed master or original hardcopy archive of all the required records, as well as another information copy that may be a hard copy or in an electronic information system.

6.1.5.8 Access control

A formal control and access process can be established to obtain hard copies of drawings and documents. Methods of controlling access to records are established and documented to prevent the loss, destruction or unauthorised alteration of records. Controls include the identification of organisational responsibility for authorising and controlling access to the records.

6.1.5.9 Control of modifications, revisions and the archiving of records

Controls should be established to identify the personnel authorised to make modifications to records and the conditions under which modifications may be made.

Records should be distributed according to predefined distribution lists.

During decommissioning, certain records may no longer be required or be beyond their retention period. Procedures should be in place to demonstrate, to the site manager and possibly the regulator, that these records or documents, including quality assurance or operating procedures, are no longer required.

6.1.5.10 Periodic reproduction or transfer between media

Procedures for periodically ensuring the physical durability of the information contained in a records management system should be established and based on the record storage media used. The expected life for each record should be established and controlled to ensure that it is reproduced or the information transferred to another medium prior to the end of its expected life. Controls to ensure and verify the legibility and integrity of reproduced or transferred information should be established. Appropriate corrective actions should be taken to restore deteriorated records. For long term retrievability, procedures should be established to ensure that the tools necessary for reading the records (for example microfilm readers and computer software and systems) continue to be available.

It is important that any loss of information during the reproduction of records be documented. This document may determine or estimate the extent and content of the lost data.

It should be noted that at the present time many regulatory authorities are reluctant to allow records to be stored solely on electronic media. Where hard copies exist there is currently a reluctance to allow primary paper sources of information to be destroyed.

6.1.5.11 National and international archive requirements

Depending on the national archiving requirements of a country, it may be required to provide copies of specific documents to the national archives and/or follow other national and international archive guidelines.

6.1.6 Management of new records

6.1.6.1 Management of new records

New records will be generated continually during decommissioning through to the final facility or site release or reuse. Some records may have to be retained after the release of the site, for example waste disposal and health records. Typical records arising from decommissioning are shown in Table 6.2.

Based on the appropriate regulations, some records will be only temporary, for example work schedules and permits to work, while others will be permanent, for example radiological survey completion reports or health records.

One important issue for the management and organisation of decommissioning projects is the interaction of all the parties involved. These parties include the operator of the facility, the regulatory body, contractors, the public and other stakeholders. Records are generated, requested or required by each party in the course of decommissioning. Figure 6.3 shows the typical relationship between decommissioning related activities and the regulatory body in the course of the life cycle of a facility, including the submission and approval of documents, where appropriate. Managing this cross-flow of information and the related records is a key responsibility and an essential part of a records management system.

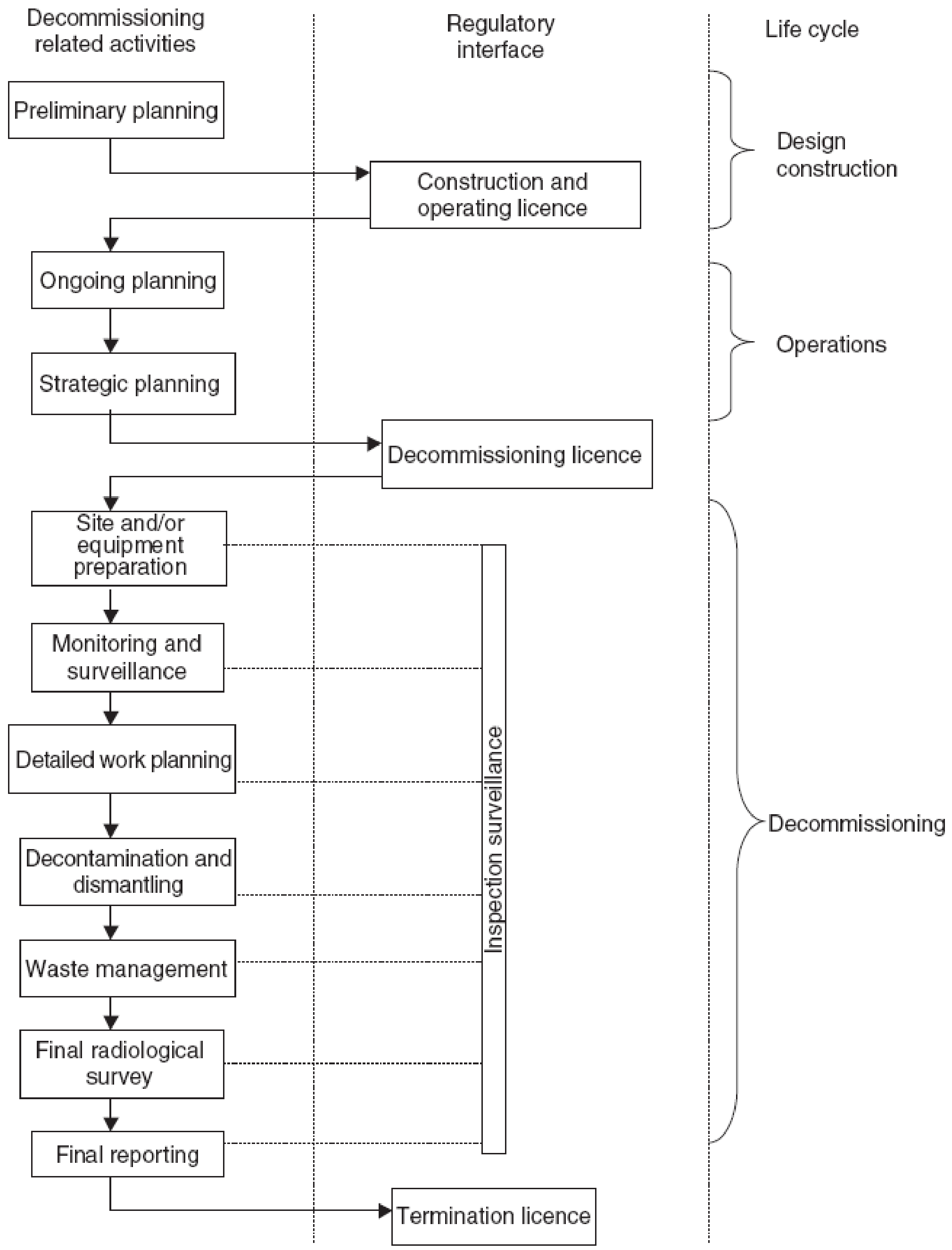


Figure 6.3 The decommissioning process

The progress of decommissioning is documented by the responsible management organisation. All waste materials (i.e., radioactive, hazardous and non-hazardous) that were present at the beginning of the decommissioning activities should be properly accounted for and their ultimate destination identified. After each phase of decommissioning, as required, the operating organisation may report to the regulatory body on the management and disposal of the waste generated during that phase. The report should also provide the current status of the decommissioning work at the facility or site and identify any anomalies observed during the phase. Moreover, information such as radiological surveys and personnel monitoring data should be reported to the regulatory body, as required.

Additional reasons for creating and maintaining the records generated during decommissioning include the potential legal and/or regulatory aspects, litigation concerns and information for other ongoing or future decommissioning projects.

New records should be typically managed in the same manner as historical records. The computer assisted management of decommissioning records will aid their real time acquisition.

6.1.6.2 Management tools and experience in the management of records for decommissioning

A few examples of recent experience in the management of records are presented in this sub-section. The best system for the management of records is that which works for the particular approach the project management finds is best suited to its needs. These examples are merely case studies of systems that have worked successfully:

- The United States Department of Energy (USDOE) has developed a data information management tool for use on selected decommissioning projects. The tool is called the System of Tracking Remediation, Exposure, Activities and Materials (STREAM). This technology is a multimedia database that consolidates project information into a single, easily accessible location for decommissioning work tracking. Information inputs can range from procedures, reports and references to waste generation logs and manifests, photographs and radiological survey maps. The STREAM system was successfully demonstrated at C reactor at Hanford, USA, together with other software tools. Especially when incorporated early in project planning, it is a systematic and cost effective tool for controlling and using project information.
- A code system for the management of a decommissioning project has been developed in Japan. Various data about the Japan Power Demonstration Reactor dismantling have been accumulated in a database. These data are being used for managing ongoing dismantling activities and verifying the code system for the management of reactor decommissioning (COSMARD) developed for forecasting management information and planning the future decommissioning of commercial nuclear power reactors. The components that make up the data sets are radiation control data, dismantling operations data and waste management data.
- A data management system was set up for the decommissioning of the main process building of the Eurochemic Reprocessing Plant in Belgium, which is able to process, for example, working hours, production factors and budget data for performance.
- At the Greifswald nuclear power site in Germany, a data management system called the Project Information System has been set up successfully to perform and control the ongoing decommissioning project. This information system comprises about 500 work packages and contains their required capacities and costs, the masses to be handled and radiological data. Logistics are important to maintain the complex material flow. The PC programme ReVK has been developed to represent material and waste flows at the Greifswald site, exercise data control within administrative constrains, maintain bookkeeping, generate reports and manage transport and storage options. For radioactive waste and its final disposal, ReVK includes two other computer programmes, AVK and AVK-ELA. The first is for controlling radioactive waste flows, the second is for final disposal purposes. Other software tools have been developed for the assessment of the required volumes and related costs for the disposal of decommissioning waste.
- One example of record keeping on a specific US decommissioning project is a project involving the decontamination and decommissioning of a plutonium fabrication facility at the Nuclear Fuel Services, Inc., facility in Erwin, Tennessee. In order to provide an accurate history of the decommissioning activities, every opportunity has been taken to utilise electronic monitoring, recording, retrieval and reporting. Waste items are tagged and tracked by a barcode from the moment they are removed from the process line to the time they are placed in waste drums for disposal. This audit trail provides a validation of facility characterisation as well as a real time material accountability control and assists in the management of the decommissioning effort.

The records required for waste shipment, storage and disposal are generated by tracking based on information in the database.

6.1.6.3 Decommissioning reports

During decommissioning planning and in the preparation of decommissioning planning reports, several new supporting reports and records may be required, for example environmental assessment documents, project plans, waste management plans and safety reports. As noted, in addition to new information, significant supporting historical records will also be required. Supporting records can be summarised and referenced in planning reports and managed in a records management system.

At the completion of decommissioning, a final decommissioning report that includes appropriate supporting records should be prepared. In accordance with the national legal framework, these records should be held and maintained for purposes such as the confirmation of the completion of decommissioning activities in accordance with the approved plan, recording the disposition of waste, materials and premises, and responding to possible liability claims. The records to be assembled should be commensurate with the complexity of the facility being decommissioned and the associated hazards.

The final decommissioning report, supported by the records assembled, provides the confirmation of the completion of the decommissioning activities. Any remaining restrictions on the site should be registered, as required by national regulations.

6.2 Waste/material management information systems

To manage radioactive wastes over a long time period, there is a need to compile, manage and maintain the variety of records that are generated. The long term management of these records benefits on a hierarchical records management system, and discusses the concepts of high level information (HLI), intermediate level information (ILI), and primary level information (PLI) [27].

Within the primary level information, a wide variety of information in support of radioactive waste management may be generated and managed. An important component of the primary level information is a waste inventory record keeping system, which provides a comprehensive and detailed description of waste repository inventories.

Figure 6.4 illustrates the elements of a hierarchical records management system. Prior to closure of a repository, records are generated from activities such as:

(1) waste generation, processing, and transportation;

(2) monitoring (such as operational control, health protection, environmental);

(3) repository site selection/characterisation;

(4) repository design, construction, operation, closure and performance assessment; and

(5) repository inventory management.

These pre-closure records comprise the primary level information set, which is generated and managed principally to support waste management facility licensing, operation and closure. Among other records, the primary level information includes waste inventory records as well as documentation such as facility license applications, which may include repository performance assessments (PA) and environmental assessments (EA). The record keeping system for waste inventory records is the waste inventory record keeping system, which is a subset of the overall primary level information.

Many records may be generated during pre-disposal waste management (generation, handling, processing, storage) prior to repository operation. As such, a waste inventory record keeping system is likely to be needed prior to repository operation, notably to support storage operations. Additionally, storage operations may be integrated with other operations, such as waste processing. Therefore, the implementation of a waste inventory record keeping system is needed as early as possible even if disposal has not yet been implemented.

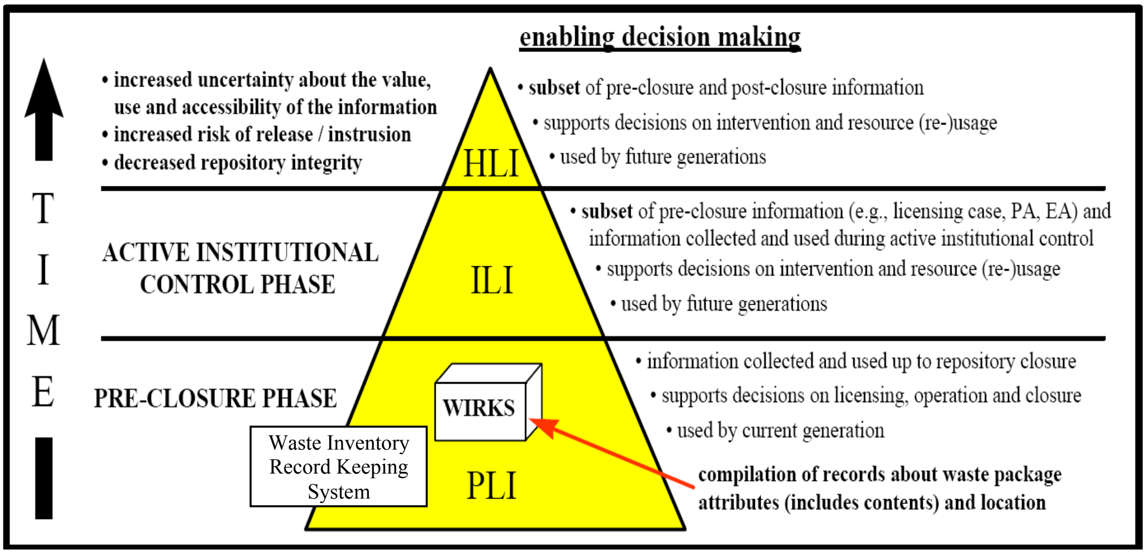


Figure 6.4 Elements of a records management system

After closure, a country may implement an active institutional control phase. Prior to this phase, a country may decide that some of the pre-closure information that is compiled may not need to be retained after closure. Much of the information that may be suitable for retention after closure may be in documentation such as the license application, the performance assessment and the environmental assessment for a repository. A principal use of this information, called the intermediate level information set, is to assist in repository intervention activities (remediation or waste retrieval) should this be deemed desirable or necessary.

If there is no active institutional control phase or after this phase has been concluded, a country has to decide what records will need to be retained for future generations since it is important that future generations are aware of the potential hazards involved. This will allow them to make informed decisions concerning the safety of the repository, to avoid inadvertent intrusion and to assist decision making on the possible reuse of the site, its contents and surrounding controlled areas. This last set of records is called the high level information set.

A waste inventory record keeping system represents a subset of the primary information level information.

The information accumulated in a waste inventory record keeping system is the primary information for a disposal facility and arises from data created for individual waste packages. A significant step is the formation of individual waste package records, which are compiled in a waste inventory record keeping system. Systems need to be in place to create the necessary data at the appropriate time, which may be before or at the time of waste packaging as well as at the time of transfer to a repository.

6.2.1 The need for a waste inventory record keeping system

To support the implementation of a radioactive waste repository, where the intention is to isolate wastes from humans and the environment for hundreds to thousands of years, there is a need to create records and, therefore, there is a need for record keeping systems. Since repositories may be subjected to either inadvertent or intentional future human actions after they are closed, the expected long term performance of a repository may be impaired. The likelihood of inadvertent human actions disturbing the repository system can be reduced by the long term maintenance of records that provide warnings and information regarding the presence of the waste and its potential hazard. Such information could also facilitate a possible retrieval of repository contents if future societies determine that retrieval is desired or warranted.

Assuring the transfer of information to future societies enables them to make informed decisions regarding the repository design and contents. Present day societies should facilitate the possibility for future societies to make their own judgements about a repository and the continued management of its contents.

Adequate information about repositories should exist at the time of repository closure. In addition, assurance is needed that some of this information will be retained for a long period of time following repository closure.

Waste inventory record keeping system records in support of one repository can assist with the provision of information to waste inventory record keeping system at other repositories or to other waste management related information systems, such as national or regulatory information systems within a country. The implementation of multiple waste inventory record keeping system within a country should consider the exchange of information between these systems.

The implementation of a waste inventory record keeping system in a country can support the exchange of information with other countries. Consideration of issues related to information exchange during waste inventory record keeping system implementation can reduce or eliminate redundancies in the reporting of information and, therefore, can reduce or eliminate inconsistencies. They can also reduce the costs associated with data reporting.

The following sub-parts list specific needs for implementing and maintaining a waste inventory record keeping system in support of radioactive waste disposal, including pre-disposal activities.

6.2.1.1 Planning and repository operations

A repository may have limitations reflected in waste acceptance conditions that are based on a safety assessment or other means. For example, there may be limitations on the quantity of waste, the quantities of specific radio-nuclides, toxic/hazardous materials, and chelating agents that are authorised for disposal or for other considerations, such as operational constraints. Therefore, repository operators maintain detailed records of the waste that has been disposed to help optimise operations for their repositories. Operators may also use these records, in conjunction with surveys of predicted future arisings and knowledge of the quantities of wastes in storage, to forecast when their repositories will be filled and to plan future repositories.

A waste inventory record keeping system can be used to plan for waste receipts and to track wastes from the agreement to accept them from a generator through to their final disposal. Tracking could include the recording of changes made to wastes such as compaction, concentration, repackaging or grouting, as well as the waste's final location. Some data that are recorded at the time wastes are placed into a repository may have been generated at earlier times, for example at or before the time that the waste was processed.

Even if a country's disposal strategy may not include provisions for waste retrieval, that is, there may be no intention of retrieving waste, waste retrieval may become necessary. For example:

(1) prior to closure, ongoing assessments of the repository may lead to retrieval of some of the waste (either permanently or to perform remedial actions, such as providing new containers);

(2) during the active institutional control phase, intervention may be required, which may involve modification to the repository and this may also include waste retrieval; and

(3) future societies may decide to retrieve part or all of the repository inventory.

Regarding (1) and (2) in the above list, a waste inventory record keeping system needs to accommodate the accumulation of waste as well as the removal of waste from a repository in order to maintain an up-to-date, actual repository inventory. It is unlikely, but still possible, that future societies will use a present day waste inventory record keeping system to adjust a repository's inventory.

6.2.1.2 Licensing

Countries may have legal or regulatory requirements to implement and maintain a waste inventory record keeping system as a prerequisite to obtaining a disposal facility license and for maintaining the license. Such a system should keep records of the amount and activity of all radioactive wastes at storage and disposal facilities. These records should be based on the waste inventory record keeping systems maintained by the operators of storage and disposal facilities. An operator's waste inventory record keeping system may have to be approved by the authorities.

6.2.1.3 Approval/compliance records

A waste inventory record keeping system can be used to record the approval of a generator's waste conditioning processes (for example, vitrification) or waste streams as a means to pre-approve waste packages for acceptance. It can also be used to record the approval of individual waste packages or consignments of waste packages.

To ensure that waste packages conform with acceptance requirements, facility operators may perform a number of routine quality controls on packages. These controls result in a variety of records, which can include:

(1) controls versus limits for radiation field, contamination, heat generation, etc.;

(2) non-conformance records, corrective action records.

6.2.1.4 Reporting

A waste inventory record keeping system could serve as the basis for producing reports that may be required by regulatory or license conditions or created to support operations and planning. For example, periodic reports can provide volume and activity totals in storage or disposal facilities.

Other reports, based on information recorded in a waste inventory record keeping system, could include lessons learned and could cover experience gained with the operation and monitoring of repositories to provide feedback about how to improve both current and future repository operations, including waste acceptance.

6.2.1.5 Inputs for performance, safety and environmental impact assessments

To assess the performance and safety of repositories, which can span very long time periods (e.g., geological scale), computer models may be used. An essential input to these models is a repository's inventory. Prior to operation of a repository, safety assessments may use estimates of the repository's inventory based on waste inventory record keeping system data for stored waste. As waste is received into an operating repository, a waste inventory record keeping system may be used to record the emplacements, which can be used to provide data for operational and post closure assessments. For example, it can be used to add up and correct for decay the activities of the radio-nuclides that are in the waste that is actually placed into a repository.

6.2.1.6 Remediation or selective retrieval activities

The objectives of radioactive waste disposal are to remove wastes from the human environment and to ensure that it remains isolated from that environment and inaccessible to humans until the radioactivity has decayed away. This may be impossible to achieve for very long lived radio-nuclides. Therefore the intention is to design repositories that ensure that any radioactivity that enters back into the environment in the future does so at levels resulting in acceptable risks to humans and the environment.

Repositories may not perform as expected. For example, there may be unexpected package or engineered barrier failures, resulting in a need or a desire to perform remedial actions, which may include waste retrieval. Retrieval requires a knowledge of not only the overall inventory of a repository but also a knowledge of the contents of individual packages, or groups of packages, and their locations in the repository, which requires a knowledge of the structure of the repository.

6.2.1.7 Information for other repository owners

In some jurisdictions, the performance and safety assessments of one repository may require a consideration of the impact of other nearby repositories. As such, it may be necessary to provide information about the inventory of one repository to another repository proponent in line with paragraph 2.26, Part (b) of the Basic Safety Standards.

6.2.1.8 Safeguards

Safeguards are essentially a technical means of verifying the fulfilment of political obligations undertaken by countries in concluding international agreements relating to the peaceful uses of nuclear energy. The technical objective of safeguards is the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection.

Final disposal of spent fuel will accumulate large inventories in a disposal site creating a long term proliferation risk. Safeguards for spent fuel disposal in geological repositories, therefore, have to be continued even after the repository has been backfilled and sealed. The effective application of safeguards must assure an unbroken continuity of knowledge that the nuclear material in the repository has not been diverted for an unknown purpose. For effective and efficient application of safeguards, the IAEA requires vital information on facility design and operation. Part of the required information will also flow from the other obligations, for example, safety, waste disposal, environmental protection, etc. An integrated approach to document all required information will be an advantage to all concerned. Safeguards confirmation that the material has not been diverted, established by confirmation of integrity of containment, can also ensure that safety criteria have not been breached. The IAEA has proposed requirements for records and reports related to safeguards for geological repositories.

6.2.1.9 Financial

A waste inventory record keeping system can assist with invoicing generators for cost recovery for disposal. It can also be used to assess future liabilities for wastes that have not yet been emplaced into a repository.

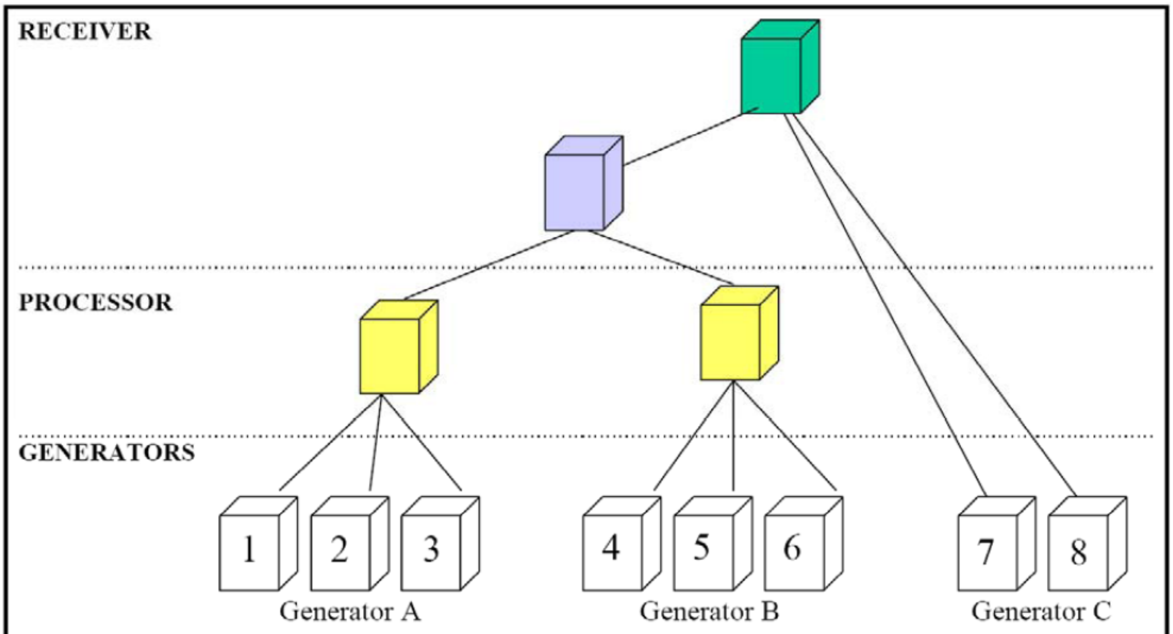
6.2.2 Waste inventory record keeping system information

The information that is to be managed by a country within a waste inventory record keeping system should be identified based on consultations with the organisations or groups that will use the waste inventory data, such as regulators, generators, the public, computer modellers and repository operators. The identification of information should also be considered in the context of information exchange with international organisations.

A waste inventory record keeping system represents what is commonly referred to as a waste inventory database, which is used to record the properties of waste packages and their locations, which, in turn, are used to compile a repository inventory. With regards to package properties and location, a single waste package may contain other waste packages that originated from one or more generators, as illustrated in Figure 6.5.

In some countries, a waste receiver only needs to track packages back to the organisation that created the package, which is considered to be the sole generator of the waste. In other countries, the receiver must track the waste back to each generator if a package contains waste from more than one generator. For the latter case, there are various models for implementing the requirement to track back to each generator. Two possible implementation models are described in Figure 6.6 and Figure 6.7.

In these models, the important factor to note is that tracking can be performed only at the waste package level. If a package contains nested packages from more that one generator, the waste can be tracked back to each generator only if (a) prior to nesting, each package only contains waste from a single generator and (b) each nested package can be clearly identified. Tracking cannot be performed at the sub-package level, that is, it cannot be performed on individual items within a package.



In this example, three generators have waste. Generators A and B transfer their waste packages to a processor, which processes their wastes separately and creates two new packages. These new packages are transferred from the processor to a waste receiver, which puts these new packages into another new package. The third generator transfers packages directly to the waste receiver.

The receiver places all waste packages that it created or that it received into another new package. If all packages use the record model shown in Figure 1, then all packages can be tracked from point of generation to final disposition.

Figure 6.5 Schematic representation of waste package nesting

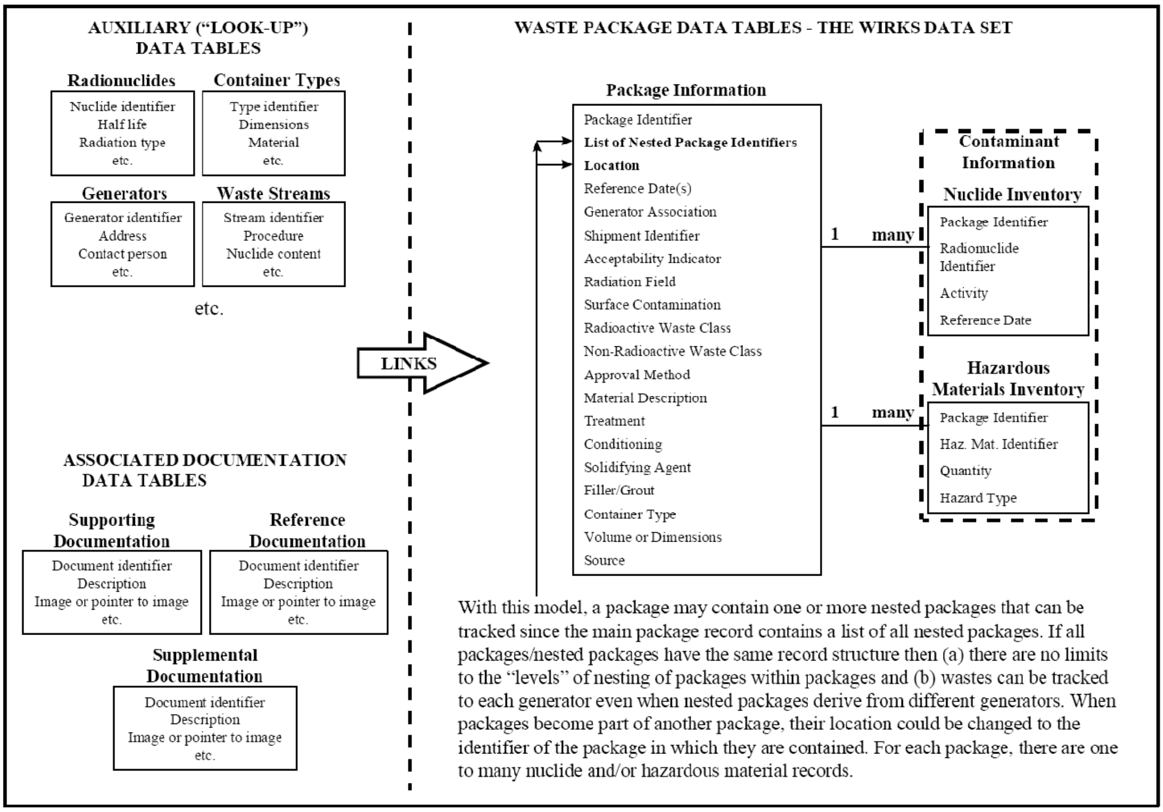


Figure 6.6 Example of a schematic representation of possible waste inventory record keeping system information (Model 1).

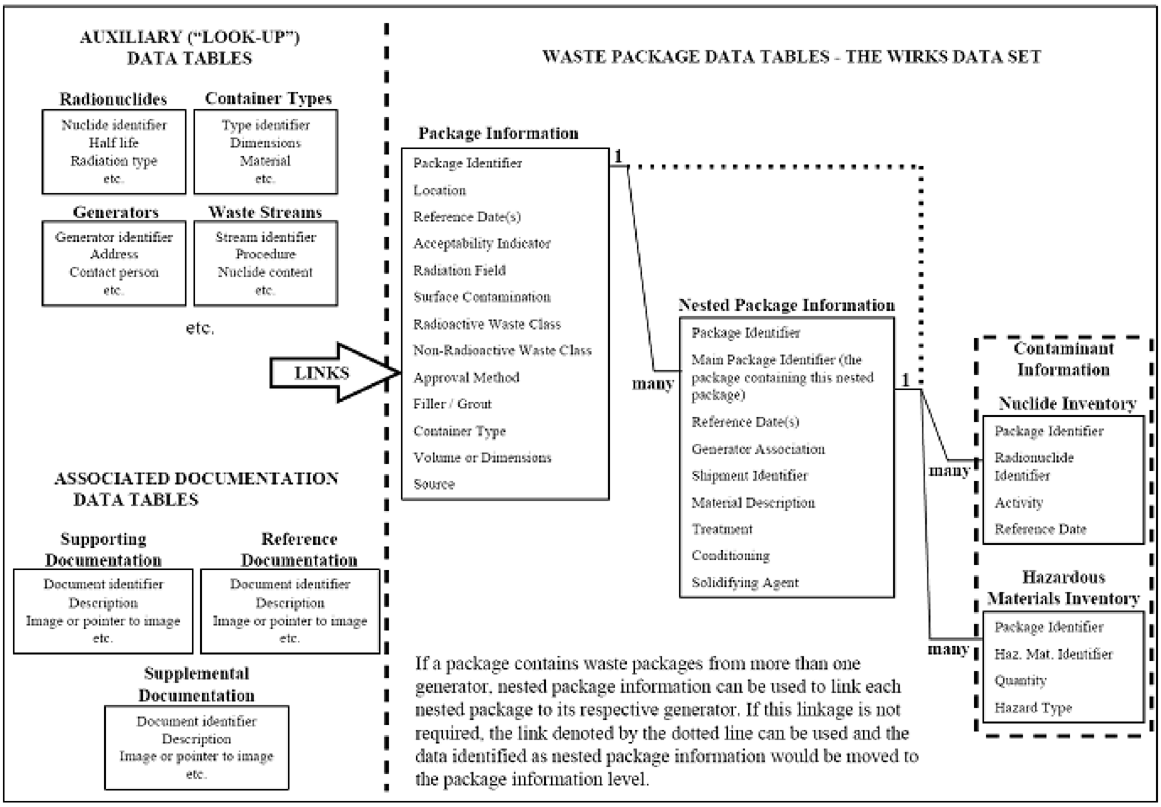


Figure 6.7 Schematic representation of possible waste inventory record keeping system information (Model 2).

The models are similar in that they both can enable the tracking of nested packages. However, for Model 1:

(1) all waste package records have the same structure;

(2) there is no limit to the levels of nesting of packages (since all packages have the same record structure); and

(3) the model can be used whether or not tracking is required for nested packages.

For Model 2:

(1) there are two waste package record structures if nested packages have to be tracked, one for the main package (which contains the nested packages) and one for nested packages;

(2) there can be a single waste package record structure if nested packages do not have to be tracked, which is comprised of information that would have been otherwise contained in separate waste package record structures;

(3) the model is best suited to cases where a single level of nesting is used; and

(4) if the tracking of nested packages is not required initially and a single record structure is implemented, the database structure would have to be redesigned to implement a dual record structure if a decision is made later to track nested packages.

For Model 1 (see Figure 6.6), the package information table contains a 'List of nested package identifiers' field. This field is used to record the identifier numbers of all packages that are placed into another package. For Model 2 (see Figure 6.7, the nested package information table contains a 'Main package identifier' field to record the identifier number of the package that contains this nested package.

Nested packages would no longer be physically managed as individual packages since they would be located inside another package that is handled as a single entity. However, the original data records for the nested packages would be retained as separate records in the waste inventory record keeping system because the information for these packages needs to be retained. Since the physical location of nested packages would be identical to the physical location of the package in which they are placed, the location field in Model 1 could be updated for nested packages to contain the identification number for the package in which they are contained.

Model 1 allows the greatest flexibility for package nesting and it is simpler to implement since a single record structure can be used for the waste package information table. In addition, Model 1 applies whether nested packages are tracked or not, whereas Model 2 would require either (a) modification of an existing waste inventory record keeping system to add nested package tracking; or (b) implementation of a dual record structure model to allow for the addition of nested package tracking, whether or not it is implemented. It is worth noting that waste inventory record keeping systems have been implemented in some countries using a dual record structure to track nested packages, per Model 2, and these waste inventory record keeping systems have been reported to work well.

A waste inventory database is not a stand-alone information set. While it lists package properties and locations, normally it would not include information about how those properties were derived. In addition, while an inventory database may define where a waste package is located within a repository, it is unlikely to provide information about the repository's structure, which would be needed by someone unfamiliar with the repository design if that person wanted to locate a package. As such, a waste inventory record keeping system can be considered to be comprised of a data set (see Sub-section 6.2.2.1) and associated documentation (see Sub-section 6.2.2.2), as represented in either Figure 6.6 or Figure 6.7.

6.2.2.1 Example waste inventory record keeping system data set

An example waste inventory record keeping system data set is represented by different tables of data. The package information tables contain data that are uniquely associated with an individual waste package. The contaminant information tables identify the radioactive and non-radioactive contaminants in a package, which are used to determine the radioactive and non-radioactive hazard classes. The package information tables are summarized in Table 6.4.

Table 6.4 Possible waste inventory record keeping system package information tables with example data set fields

Model 1 (see Figure 6.6) Package Information Table	Model 2 (see Figure 6.7) Package Information Table	Nested Package Information Table
Package Identifier	Package Identifier	Package Identifier
List of Nested Package Identifiers		Main Package Identifiers
Location	Location	
Reference Date(s)	Reference Date(s)	Reference Date(s)
Generator Association		Generator Association
Shipment Identifier		Shipment Identifier
Acceptability Indicator	Acceptability Indicator	
Radiation Field	Radiation Field	
Surface Contamination	Surface Contamination	
Radioactive Waste Class	Radioactive Waste Class	
Non-Radioactive Waste Class	Non-Radioactive Waste Class	
Approval Method	Approval Method	
Material Description		Material Description
Treatment		Treatment
Conditioning		Conditioning
Solidifying Agent		Solidifying Agent
Filler/Grout	Filler/Grout	
Container Type	Container Type	
Volume or Dimensions	Volume or Dimensions	
Source	Source	

Prior to the nesting of packages using Model 2, all of the information in the package information table and in the nested package information table would be in only the package information table. When nesting occurs, this information would be split between the package information table and the nested package information table.

The source means the source of the waste, represented by:

- Radionuclide Identifier;
- Activity;
- Reference Date;
- Package Identifier;
- Hazardous Material Identifier (for example mercury);
- Quantity;
- Hazard type (toxic, flammable, non-hazardous).

Additional data to be recorded for each package identifier could be:

- Nuclear Fuel Cycle Waste;
- Reactor Operations;
- Reprocessing;
- Fuel Fabrication, Fuel Enrichment;
- Decommissioning, Remediation;
- Non-Nuclear Fuel Cycle Waste;
- Nuclear Applications;
- Defence;
- Decommissioning, Remediation.

6.2.2.2 Associated documentation

6.2.2.2.1 Supporting documentation

In addition to recording and managing waste package information within a waste inventory record keeping system data set, it is necessary to record and maintain descriptions of how this information was derived in order to provide a basis for assessing or reassessing the performance, safety and environmental impact of repositories, not only by current societies, but also by future societies. Supporting documentation either describes how the average characteristics of a waste stream were determined or it describes how one or more waste packages were characterised on a case-by-case basis.

Typically, supporting documentation is used by a waste receiver to qualify the waste for acceptance into a repository. A waste inventory record keeping system data set should provide a link to supporting documentation (see 'Links" in Figure 6.6 and Figure 6.7).

Examples of how waste inventory record keeping system data set values could be documented include:

- information on a manifest, shipping record, data sheet, or disposal record/form;
- descriptions of how raw data were collected:
  * methods used to determine radionuclide activities in packages;
  * methods to treat waste (e.g., evaporation) which affects contaminant concentrations;
  * methods to assess non-radioactive hazards of wastes (e.g., leach rates of contaminants from waste forms);
- descriptions of quality assurance/quality control mechanisms:
  * inspections (e.g., methods to measure wall thickness of waste packages);
  * calibrations and standards used;
  * limits of detection of instrumentation;
  * calculation of data variability;
  * approval of a waste generator's waste management quality assurance system;
- descriptions of how raw data were processed/manipulated:
  * identification of numerical algorithms used;
  * list of assumptions and parameter values used in calculations.

Raw and processed data would be collected/derived by both generators (e.g., waste characterisation) and waste receivers (e.g., inspection, compliance/receipt monitoring).

Supporting documentation may not be the same for all wastes. As an example, some countries have found that they need to manage supporting documentation from routine and non-routine wastes differently:

- *Routine wastes:*

  Some processes or activities generate radioactive wastes that have consistent characteristics within some defined envelope and these are commonly known as routine wastes or waste streams. For these wastes, it is often possible to derive and document their average characteristics. In addition, some waste processing activities, e.g., vitrification, use process control to establish the average characteristics of the product. The product itself is not characterised - the product's characteristics are inferred (estimated) based on process knowledge.

  In these cases, waste streams or waste products can be pre-approved for acceptance by the receiver. Documentation is created as a result of this approval process and it should be linked to the data set.

- *Non-routine wastes:*

  In some cases, average characteristics cannot be established for wastes (e.g., clean-up waste, decontamination waste, waste that does not conform to a receiver's general specifications, etc.). In these cases, generators are generally required to provide supporting documentation, case-by-case, which describes the waste characteristics and how they were determined. This documentation should also be linked to the data set.

The type and quality of documentation may be the same for routine and non-routine wastes. The difference is typically how the information is presented.

A single document or set of documents may be used as a basis for routinely accepting routine waste. Waste shipments could merely reference the supporting documentation. For non-routine wastes, the format of supporting documentation may be the same as used for routine wastes; however, for individual waste shipments, information such as who characterised the waste, when it was characterised, how it was characterised, etc. would be provided on a case-by-case basis.

The preparation of supporting documentation for the case-by-case characterisation of waste may be much more costly than establishing the average characteristics for routine waste. As such, it is likely to be advantageous for a waste management organisation to identify and establish the average characteristics of routine wastes to the greatest extent practicable.

6.2.2.2.2 Reference documentation

Supporting documentation can indicate the parameters and assumptions that were used in determining waste inventory record keeping system data set values, such as radionuclide activities. However, these parameter values and assumptions are typically not described in detail. For example, a supporting document may state that waste was 'decay corrected' to a specific date; however, a supporting document is unlikely to list the radionuclide half-lives or describe the algorithms used for the calculations.

Future societies may use different approaches to data manipulation and, therefore, maybe unable to understand how the work that is described by supporting documentation was performed unless relevant information is provided. Reference documentation provides a means to record and maintain such information.

6.2.2.2.3 Supplemental documentation

Supplemental documentation has additional information related to the waste inventory record keeping system data set but it typically would not be a part of either supporting or reference documentation. As an example, a repository owner may have detailed drawings showing chambers or sub-sections within a repository. This information would be needed, along with a description of a package's location in the repository (which chamber or sub-section) for someone to locate the package.

6.3 Summary overview

There is a need for identifying the content, structure and maintenance of a records management system necessary prior to, for and after decommissioning. Important records from prior to and after decommissioning should be preserved to assist future planning, safety assessments and remedial actions.

The major observations for records management system are:

- The main goal of a records management system is to provide the necessary, sufficient and up-to-date information for those who implement the decommissioning and other parties to make informed decisions on planning and the implementation of decommissioning actions. There will be significant financial consequences if there is inadequate documentation to support decommissioning.
- The main sources of decommissioning records are the records of the design, construction, modification, operation and shutdown of a facility. Keeping these records typically will be the responsibility of the operator.
- Planning for eventual decommissioning should be considered during the design and operation of a facility, as by doing so the information will be readily available and transferable when needed.
- The preservation of the necessary information for the duration of the active post-shutdown phase, safe enclosure and final dismantling requires the early establishment and maintenance of a records management system.
- A records management system is desirable to facilitate safe and efficient decommissioning.
- Throughout the lifetime of a facility the records archive should be frequently and independently audited, with decommissioning as a primary focus.
- An auditing process should identify gaps in a records management system and address the usefulness of the archives for decommissioning.
- Since technologies may change and knowledge of a facility may diminish, information may be less understood over time. It is therefore important that the information transferred to the future users be usable. Keeping control of records (and institutional knowledge) is necessary for the whole decommissioning process.
- Redundancy and diversity in a records management system are necessary for the effective management of the records.
- The media used need to be selected to ensure the durability, readability and retrievability of the information they contain.

The major observations for waste inventory record keeping systems are:

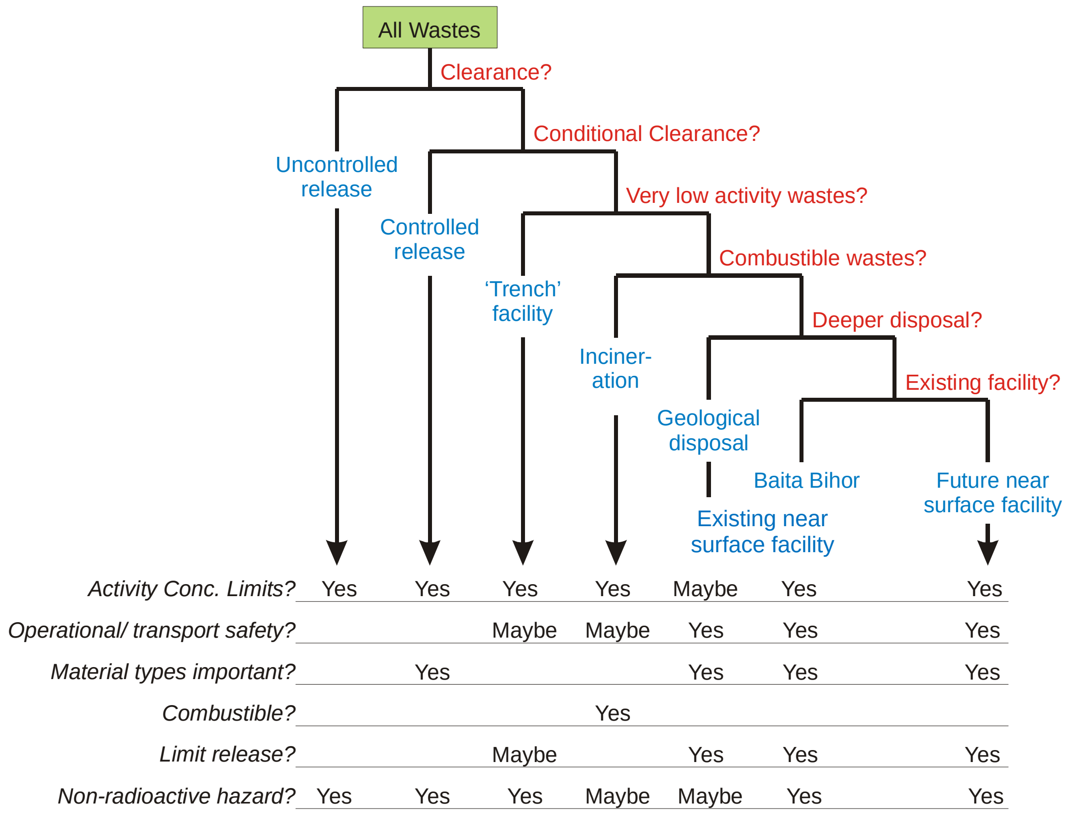
- A waste inventory record keeping system represents only part of the overall primary level information (PLI) set in support of radioactive waste disposal, which includes pre-disposal waste management activities.
- The fundamental unit is the waste package. A waste inventory record keeping system is principally a database for recording, maintaining and reporting on the characteristics and locations of waste packages.
- Some countries have multiple waste inventory record keeping systems that do not have consistent structure, format or content.
- Waste inventory record keeping system implementation is country specific. However, the type and the quality of the data should be examined in the context of international perspectives, such as waste classification systems, the IAEA Waste Management Database, the possibility of regional repositories and the possibility of a future, international archive for repository information.
- If multiple waste inventory record keeping systems are implemented in a country, consideration should be given to the exchange of information between these systems and between any existing planned or possible central, national radioactive waste management information system.
- Countries should implement a waste inventory record keeping system in support of disposal during the predisposal phase. It is better to take a series of small, incremental steps than to wait and take one big step. Even without the actual implementation of a waste inventory record keeping system, an implementation plan should be prepared as soon as practicable.
- A lead organisation in each country should define the goals, minimum record content and procedures of a waste inventory record keeping system. This lead organisation could be a regulatory body.
- An organisation should be identified to ensure that the waste inventory record keeping system is properly maintained, which, in turn, will ensure data integrity, security and accessibility during facility operation and upon closure.
- Physical, hard-copy records should be organised in preparation for waste inventory record keeping system implementation. International experience has shown that early planning and implementing for a waste inventory record keeping system can result in lower overall effort and cost to implement a waste inventory record keeping system.
- Countries planning a waste inventory record keeping system should consult with organisations that have extensive experience with such systems to obtain advice on what to do and what not to do.
- The identification of organisations with responsibility for waste inventory record keeping systems needs to consider the long time scales of radioactive waste management. Document/information management in support of radioactive waste management can consider time scales on the order of many decades or centuries. It may be prudent to assign the responsibility for a waste inventory record keeping system to a radioactive waste management organisation that has longer term goals than most other organisations.

7. Waste Characterisation

Characterisation of radioactive waste arising within decommissioning activities is a key point in its consecutive management. In fact, the need to characterise radioactive waste before its consecutive management steps on the disposal route represents, among others, a practical application of the principle number 8, ‘Radioactive waste generation and management interdependencies’, of the safety principles of radioactive waste management [28]. Characterisation consists of the determination of the essential physical, chemical, biological, mechanical and radiological properties of the radioactive waste with the aim of:

1. establishing the need for further adjustment, treatment and conditioning of the wastes.
2. establishing the suitability of the wastes for further handling, processing, storage, transportation and disposal.
3. demonstrating that the assumptions made during laboratory investigation and process development are confirmed in full scale waste treatment and conditioning processes.
4. validating the assumptions in safety analyses and performance assessments regarding waste form limits, and ensuring that properties such as radionuclide inventory and stability and waste package integrity are valid;
5. confirming that the limit values of evaluation criteria, including specific regulatory requirements, continue to be valid with time.

Figure 7.1 Relationship between end-point waste management and general characteristics of wastes



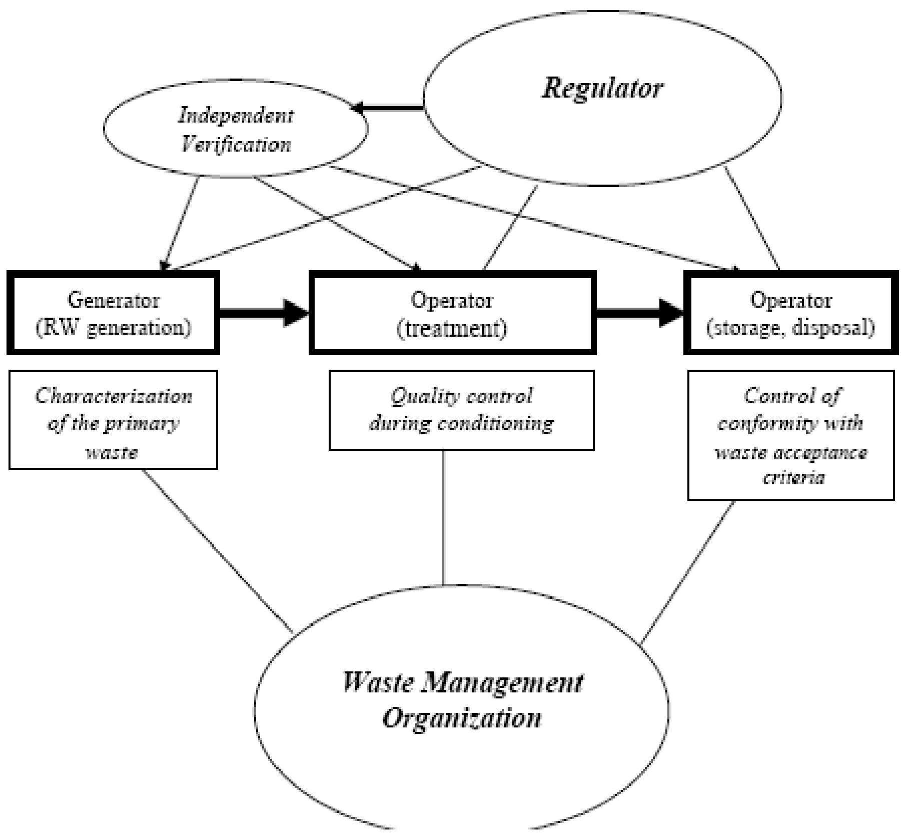
Potential end points for radioactive wastes, being considered within decommissioning, that have been identified include:

- unconditional clearance;
- conditional clearance (including disposal, re-use and recycling of specific materials according to specific practices);
- incineration of combustible wastes and oils;
- disposal of mildly contaminated very low level waste to a simple trench;
- disposal of low/intermediate level waste to an existing near-surface engineered disposal facility;
- use of future near surface and/or geological disposal facilities.

The extent to which these end-point options are applied to decommissioning wastes determines the key criteria on which the wastes need to be segregated and characterised. Figure 7.1 shows that, depending on the extent to which the options are used, wastes will need to be segregated and streamed according to activity concentration, operational/transport safety consideration (dose rate and surface contamination), material, combustibility, characteristics that could lead to release (e.g., gas generation, corrosion) and non-radioactive hazards [29].

In fact, there is no principal difference regarding approaches to characterisation of wastes for the various disposal routes described in Figure 7.1. With some level of simplification, it could be generally stated that the difference between the various waste streams (streamed according to disposal route) rests in various concentrations of values which are the subject of characterisation, i.e., in various requirements for methods and instrumentation of their determination.

Figure 7.2 Diagram showing the responsibilities of the various players in a waste characterisation programme



Regarding the organisation and responsibilities within the waste characterisation processes, Figure 7.2 indicates the participants in the processes, their responsibilities and possible interactions.

7.1 Techniques

For the purpose of establishing a waste characterisation strategy, waste streams may by ease of measurement and sampling ability be classified into four groups [19]:

- ‘simple and stable’,
- ‘complex and stable’,
- ‘simple and variable’, and
- ‘complex and variable’.

Conservatively, in a first approach to a characterisation strategy, decommissioning wastes are generally mentioned as an example of ‘complex and variable’ wastes. Nevertheless, it is immediately noted that decommissioning wastes and historical wastes are unusual cases that, depending on the circumstances, could fall under any one of the four sub-categories.

As a first objective, a characterisation strategy should define what characterisation details are required. With too little data, the most appropriately engineered disposal may not be selected. Too much data (and the associated data collection costs) will result in a waste of resources. Characterisation must be carried out correctly, with acceptable level of precision. If inaccurate data are obtained, the resulting packages may be found to be non-compliant late in the life cycle, which will be expensive to correct. If errors go undetected, the result may be unsafe disposal or release with potentially serious long term effects on people and the environment. A quality assurance programme should be implemented as well as a system of records for keeping all the waste data.

The decontamination and dismantling of a nuclear facility and the clean-up of the site may produce radioactive wastes with great variety in terms of type, activity, size and volume that may be activated or contaminated. Waste streams produced by decommissioning activities are often similar to the waste produced during operation of nuclear facilities when only dry decommissioning processes are used.

If a chemical process is used (for example, chemical dissolution), there will probably be a partitioning of radio-nuclides, and the final radionuclide composition will be different from the initial one. In such a situation, several sampling programmes may be necessary to determine if the radionuclide composition has changed or not, and perhaps more than one scaling factor may be necessary.

An initial sampling is often necessary to clarify the different possibilities. The main questions are:

- what were the operations performed over the lifetime of the facility? Answering this question may require consulting archives and/or retired workers to develop a comprehensive history.
- what kind of wastes will be produced (surface or deep disposal, clearable, very low level, etc.)?
- will the waste stream be stable? If not, how many sub-waste streams will be produced, from which type (simple and stable)?
- is it technically and financially possible to decontaminate part of the waste stream to avoid producing high level waste, or waste non-acceptable in near surface type repositories, respectively?
- which waste form will be produced?

- what are the required correlation and scaling factors[2]?

Conceiving such an exhaustive characterisation programme is not easy. Analytical measurements are expensive and time consuming. For example, if the decommissioning strategy leads to the decision to build a decontamination workshop, new information may be required by the design engineers (for safety or physical parameters). It might require some re-sampling. When re-sampling is too difficult and/or expensive (dose rate, time), a sampling strategy will become necessary.

If the long term safety of the last step of the disposal route depends on total or mean values of the radioactivity or other characteristics, the level of confidence and the accuracy gained for the composition of a single package should be of minor importance. In other words, the primary requirement of the methods used to declare the radio-nuclides and environmentally important non-radioactive elements is that they must be free from any significant systematic errors. In this case, accurate results can be obtained for the entire lot using rapid and inexpensive measurements of each container.

The main sources of uncertainty arise in checking the homogeneity of a waste stream and sampling, selecting the key nuclides and non-radioactive elements, measuring the easy and difficult to measure radio-nuclides (by using non-destructive analysis or destructive analysis including a dissolution step), and calculating the impossible to measure radio-nuclides.

7.1.1 Sampling

The type of waste sample, depending upon the consistency and complexity of the waste properties, will influence the methods of sampling. Another important consideration that will affect sampling will be the occupational dose uptake to plant operators and laboratory personnel, which must satisfy the ALARA principle. The wastes may be characterised by the statistical evaluation of test results obtained for different samples or by testing one or several representative samples. Generally speaking, it is recommended that representative samples be taken if possible [30]. Statistically based sampling must always consider the level of confidence and the accuracy required for the waste data.

Sampling events, quantities and amount of analyses measured will also depend on the data need and usage, e.g., for primary waste stream characterisation, process control, process monitoring, contractual compliance, environmental compliance, safety basis compliance, technical investigations.

For these reasons, an integrated sampling and analysis written procedure should be developed early in the process and agreed upon in conjunction with the waste generator, the treatment and conditioning plant owner and operators, the storage and disposal personnel, and the regulators.

Based upon negotiations between the participants, sampling and analysis by an independent laboratory may also need to be included in the plan.

Before defining a sampling strategy, it is necessary to know the complexity and history of the facility to be decommissioned. If this knowledge is not available, a non-destructive analysis map of the facility may be made using a gamma camera or a gamma spectrometer.

For decommissioning waste characterisation, the following sampling strategy is recommended when sampling is unsafe, difficult to perform, or expensive:

- Define an initial sampling programme regarding the representativeness of determinations, taking into account the future necessity to realise complementary analytical determinations (number and weight of the samples).
- Clearly identify the traceability records.

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[2] See footnote 1.

- Send the samples to the laboratory in order to realise representative sub-samples (necessity to clarify before the representative criteria such as key nuclides measurements, granulometry).
- Archive sufficient quantities of representative sub-samples.
- Process the initial analytical characterisation programme to get the data required to define the decommissioning strategy.

It may be possible to use aliquots of representative archived sub-samples for the following needs:

- to get more information relative to the waste stream to be decommissioned (special measurements required for safety, for design engineers).
- to qualify the final waste form.
- to determine the ratios for the difficult-to-measure scaling factor.

Regarding the sampling techniques, many procedures have been developed and practiced to take samples during pre-disposal stages of the waste management chain. The size and quality of the samples depend on the purpose of sampling, e.g., for checking chemical or radiochemical composition, for leach tests, for determining homogeneity, etc. For homogeneous wastes, it should always be possible to take one sample which is representative for the provided wastes. It can be established that process controls consistently result in sufficiently low product variance in the chemical and/or physical properties. Sometimes a minimum volume of the sample must be prescribed in the sampling scheme, especially if the wastes are homogeneous only in macroscopic sense (e.g., spent ion exchange resins) or if it is required for proper characterisation.

Heterogeneous waste streams which mostly arise from decontamination and decommissioning activities present real difficulties in terms of statistical, analytical and radiological characterisation. Sometimes it is possible to obtain homogeneous samples from such wastes by shredding or compaction of the material or other treatment processes which lead to a reduction in volume of the wastes amenable to representative sampling. Heterogeneous wastes present a significant challenge for non-destructive assay systems as well. Calibration standards can only marginally represent the attenuative and moderating effects of such waste forms. Consequently, large uncertainties are inevitably introduced in the assay of heterogeneous wastes, if destructive radiochemical analysis is performed on samples from the waste package, because they can only be marginally representative of such wastes. If the isotopic composition of heterogeneous wastes can be measured by sampling, it is often possible to measure the activity of gamma emitting radio-nuclides by non-destructive methods (e.g., dose rate measurements or gamma scanning) and to determine the radionuclide inventory by calculation.

Determination of the total number of samples required for a characterisation effort depends on the accuracy and precision required of the data to be reported. Statistic texts contain tables that give the confidence intervals for various degrees of freedom, based on the assumption that the data are, or have been transformed to, normally distributed data. If the waste data are not normally distributed and cannot be transformed to the normal distribution, it is recommended that the main waste streams which are combined in the waste package, are identified and that they are characterised separately by taking at least one representative sample per waste stream. The properties of the package may then be assessed taking into account the waste fractions contained in the package.

7.1.2 What to determine – identification of key characteristics

The key characteristics of the radioactive wastes must be identified through consideration of the relationship between the physical, chemical and radiological properties of the waste and its long-term safety. This must give consideration to various long-term waste management practices, but is primarily focused on the internationally accepted practice of disposal of low and intermediate short-lived level wastes in engineered facilities, disposal of very low level waste in less-engineered facilities, and the clearance of materials or wastes with radiologically insignificant concentrations of radio-nuclides (see Section 2).

7.1.2.1 Radiological characteristics

Consideration of existing waste acceptance criteria (such as those for existing facilities or for hypothetical repositories) and the key attributes of typical repository systems, indicates that the main radiological characteristics are related to the dose rate, concentration of various groups of radio-nuclides and their distribution in the wastes. The reason why each characteristic is important to be considered is indicated in Table 7.1. As can be noted, some characteristics are important because of operational, storage and transport safety reasons, whilst others are related to long-term safety.

Table 7.1 Key radiological characteristics of wastes

Characteristic	**Attribute**	**Reason for importance**
Dose rate	Waste safety in operational/ storage phase	External gamma dose rate from waste packages (contact, and potentially also at 1 m) is a key consideration in the operational safety of wastes and the safety of wastes in storage. In the long-term, dose rates from wastes can be assessed with reference to the radionuclide inventory, with allowance for radioactive decay and in-growth.
Package surface contamination	Waste safety in operational/ storage phase	Potential surface contamination of waste packages is an important issue for the safe handling and storage of wastes. It is unlikely to be a significant issue for long-term safety of the wastes as such contamination will be far less significant than the concentration of radio-nuclides in the waste package itself.
Radionuclide activity concentration	Radiological health risks	The type and concentration of radio-nuclides present ultimately determine the radiological hazard of the waste.
Relative concentration of short-lived radio-nuclides	Radiological health risks (operational and post-closure)	Shorter-lived radio-nuclides present initially in high concentrations are important with respect to operational/storage safety, whilst lower concentrations of longer-lived radio-nuclides generally dominate the safety of disposed wastes. The proportion of short- and long-lived radio-nuclides is important in determining the rate of decrease in waste hazard over time.
Homogeneity	Ease of assessment of radiological health risks	The homogeneity and reproducibility of the radiological characteristics reduce uncertainties in inventory estimation. It also allows wastes with particular concentrations (e.g., dominated by tritium) to be treated, conditioned, packaged, stored and disposed of in a manner that reflects the particular characteristics of the radio-nuclides present (e.g., high gamma dose rate wastes may benefit from additional shielding).
Nature of contamination	Limitation of contaminant mobility	The location of radio-nuclides in or on waste materials determines the rate of release from the wastes itself. For example, surface contamination is relatively easily available, whereas activation products in the matrix of materials such as stainless steel will only be released slowly by corrosion and diffusion. It is therefore important to understand the nature of radionuclide contamination in the context of safety assessments.

7.1.2.2 Non-radiological characteristics

A careful analysis of the reported non-radiological acceptance criteria should be used to determine the most important non-radiological characteristics. It is needed to note, that not all of them represent values derived by methodology analogous, for instance, to radiological safety assessments. In fact, there are very limited documented approaches how to derive such criteria [29]. Some of these reflect requirements of non-nuclear legislation in a given country, some follow criteria of analogous disposal facilities and routes. Key generic physical characteristics are reported in Table 7.2, whilst the most important chemical characteristics are detailed in Table 7.3. In each case, the reason why each characteristic is important to be considered is indicated by reference to an 'attribute' that is important to the safe future management of the waste.

Table 7.2 Significant physical characteristics of the waste

Characteristic	Attribute	Reason for Importance
Origin of the waste	Information on source of radio-nuclides	Important historical record that can be used to determine other un-recorded waste characteristics if required in the future.
Radioactive waste volume and dimensions	Package design and capacity	Determines the need for particular disposal packages. If all wastes are to be pre-treated (size reduced) for standard packages, then this characteristic is only likely to be relevant to specific waste arising (e.g., large redundant pieces of equipment).
Radioactive waste state (solid, liquid, mixture)	Waste conditioning to limit contaminant mobility **and** package requirements to limit contaminant release	Solid wastes are almost universally preferred for radioactive waste disposal. The state of the waste determines the need for potential treatment (e.g., to remove or stabilise liquids) and/or packages (e.g., to avoid adverse reactions of packages with liquids that could lead to a release of liquid from the package).
Package volume and dimensions	Physical requirements of disposal facility	The package must be consistent with the design of the waste repository.
Homogeneity	Ease of assessment of waste behaviour	Homogeneity of group of waste packages (waste streams) means that the behaviour of the group of wastes is easier to assess and therefore more predictable. Uncertainties in the waste behaviour are reduced. This has benefits in many aspects of subsequent waste management, but does rely on good quality assurance/quality control.
Density	Long-term waste package stability	Waste density (or the mass of waste packages for disposal) has implications for the structural strength of wastes in terms of storage and the long-term emplacement of the wastes in a repository.
Voidage and swelling potential	Long-term waste package stability	The potential for wastes to either decrease or increase in volume over time has important consequences for the integrity of the waste packages over long periods of time, and consequently the capability for contaminants to be contained by the package.

Table 7.3 Significant chemical and other characteristics of the waste

Characteristic	Attribute	Reason for Importance
Presence of reactive chemicals	Long-term waste package stability	Potentially reactive chemicals can generate heat, gas and other chemicals that might in turn affect the integrity and long-term stability of the wastes. It is therefore important to consider potential combinations of (otherwise innocuous) substances in waste packages. Limiting such potential combinations is an important driver for waste segregation into physically similar and compatible materials.
Presence of chemicals hazardous to health	Non-radiological health risks	Some non-radioactive materials may pose a significant hazard to health (e.g., asbestos). The presence of such materials in the wastes should therefore be identified and controlled to limit non-radiological health risks from potential exposure to the wastes.
Presence of flammables and/or explosives	Non-radiological health risks **and** long-term waste package stability	Some materials may pose the potential to ignite or explode. This could constitute a risk to waste package stability and the potential release of contaminants (both rupturing a package and leading to a gaseous release). An explosion could also directly injure a person. The presence of such materials in the wastes should therefore be identified and controlled to limit the potential for fire or explosion.
Presence of organics	Non-radiological health risks **and** limitation of contaminant mobility	Some organic chemicals pose a significant hazard to health. Organics also have the potential to influence the mobility of some radio-nuclides, and furthermore offer the possibility for transport of contaminants in non-aqueous form.
Presence of corrosives	Limitation of contaminant mobility	Corrosive agents have the potential to both degrade waste packages, and degrade the wastes themselves, both of which can enhance the release of contaminants from waste packages. Corrosion can also generate gas that increases the possibility of loss of waste package integrity.
Presence of complexing agents	Limitation of contaminant mobility	Complexants, such as EDTA, can increase the mobility of contaminants significantly, increasing the rate of release and subsequent transport from a disposal facility.
Presence of volatiles	Limitation of contaminant mobility	Volatile chemicals can result in pressurisation of sealed disposal packages (increasing the possibility of failure), and the potential for releases of radio-nuclides in gaseous form.
Presence of biological material	Limitation of contaminant mobility	Biological material can generate gas that increases the possibility of waste package failure and contaminant release. Other products such as organic acids may also be important for the overall chemical stability of the wastes.

7.2 Methods and instrumentation

7.2.1 Characteristics related to radioactivity

The radionuclide inventory of wastes can be determined using at least one of the following characterisation methods:

a) Calculation or estimation based upon well known data (e.g.; activation calculations for parts from the reactor core).

b) Measurement of the dose rate and calculation or estimation using accepted standards if the isotopic composition is known and remains sufficiently constant.

c) Measurement of the total activity or the specific activity of certain characteristic ‘key’ nuclides and calculation or estimation of the radionuclide inventory using accepted standards. Key nuclides are these which can be easily measured and for which scaling factors are evaluated, relating their activity to the activity of all other radio-nuclides that are relevant for the next stages of the waste management chain.

In some cases, it is possible to assess the total activity and isotopic composition of the wastes by calculation, using a computer programme (e.g., ORIGEN). In other cases, when the isotopic composition of the wastes is well known and if unknown changes in this composition do not occur, the characterisation of the nuclide content of the wastes is often possible by dose rate measurements. In this case, mathematical modelling or calibration procedures are needed taking into account the properties of the waste form and the spatial distribution of the inventory.

If relationships exist between different radio-nuclides in the wastes and if at least one of these nuclides denoted as a key nuclide can be easily measured, then the waste characterisation is simplified. The inventory can also be characterised by measurement of total values of activity (beta/gamma or alpha).

7.2.1.1 Non-destructive assay

Non-destructive assay may be defined as the observation of spontaneous or induced radiation, interpreted to estimate the content of one or more nuclides of interest in the item without affecting the physical or chemical form of the material. In case of low or intermediate level waste, where the radioactive content is mostly due to the presence of beta-gamma emitters, the activities per unit of mass can be directly measured by gamma spectrometry. While in the case of homogeneous distribution of the nuclides in wastes the spectrometry of the waste package as a whole can be adequate, the accuracy of the activity determination can be improved by scanning. Currently used devices for segmented gamma-scanning include high resolution, transmission corrected scanning in which semiconductor detectors (high purity germanium or Ge(Li) detectors) are used. To minimise assay errors owing to axial inhomogeneities, assays are performed in segments along a waste package vertical axis, with the help of various types of collimators. The effects of radial inhomogeneities are minimised by rotating the drum during the assay measurement. The measurement arrangement and modern nuclear electronics allow performing the measurements with dynamic counting rate ranges (more then 10^5) and with dead time corrections. A prime factor that determines the applicability of the gamma scanning method is gamma ray transmission through the package. Segmented gamma scanning is in many cases therefore restricted to the measurement of penetrating, highly energetic gamma rays from key nuclides (e.g., ^{60}Co). Higher uncertainties must be accepted in the measurement and evaluation of low energy gamma rays (e.g., ^{241}Am).

Other factors affecting assay measurements include particle self-absorption and non-homogeneity of the assayed item. Two conditions must be met to optimise assay results. First, the particles containing the nuclide must be small to minimise the self-absorption effect. Second, the mixture of material within a package segment must be reasonably uniform to apply an attenuation factor computed from a single measurement of item transmission through the segment or from an average density measurement. This problem should be minimised through strict waste segregation procedures. Nevertheless, some waste forms are inherently unsuitable for segmented gamma scanning measurements due these problems.

It should be noted that recent developments allow performing gamma spectrometry measurements using the convenience of semiconductor detectors also with portable instruments, sometimes in combination with dose rate or surface contamination measurements. Another option is to use a scintillation detector system, often equipped with a NaI(Tl) crystal. Lower resolution and higher efficiency are the basic properties of such systems. Recent developments in nuclear electronics allow for combining advantageous properties of both detector systems, often in measurements using various types of coincidence arrangements.

For detection of spontaneous neutron emitting radio-nuclides the neutron assay systems could be used, in passive or passive-active arrangement. The passive coincidence measurement provides quantitative information on event isotopes (such as ^{240}Pu) present in the waste package. The passive single neutron count rate (i.e., the difference between the total neutron rate and the neutron rate due to spontaneous fission events) provides semi-quantitative information on alpha particle emitters (such as ^{241}Am) present in the waste package. The active assay provides quantitative information on ^{239}Pu and other fissile isotope constituents. The method has serious limitations due the effects of the matrix. The approach to matrix corrections has been based on basic corrections on measured quantities determined as adjuncts to the primary active and passive assay measurements. The systematic matrix correction algorithm is based on an analytical fit to assay measurements obtained for different positions of the source within a waste package.

Application of computed tomography in a given area represents a result of the last developments. There are two types of tomography: emission and transmission. Emission tomography visualises the activity distribution in a sample, while transmission tomography visualises the density distribution. The method appears to overcome the uncertainties associated with changes in matrix density and radionuclide distribution which are common to gamma scanners. It may also allow relatively easy identification and removal of 'hot spots' which may permit reclassification of the waste package.

The uncertainty in a non-destructive measurement is the composite error, including both the precision and accuracy (bias, systematic error). Estimates of precision can be calculated by standard error propagation techniques. The precision of a non-destructive assay measurement is not strongly related to the adherence of the measurement item to ideal matrix and nuclide density assumptions. For the segmented gamma scanning systems, the measurement bias depends primarily on the adherence of the measurement item to the assumptions of small particle size and homogeneity. When systematic matrix correction formalism is used, the corresponding systematic uncertainty in the passive assay (gamma or neutron) can be decreased to less than 5 %. For the active assay, the uncertainty could reach more than about one order of magnitude higher values.

A detailed overview of non-destructive assay methods, i.e., gamma measurements and scanning methods including matrix correction methods, and neutron measurements methods including matrix correction methods, is shown in tables of [19].

Non-destructive assay is good for some applications but severely limited for others. In cases when it is not so effective, it is usually better to avoid non-destructive assay in favour of more reliable techniques (calculation or a combination of destructive assay and calculation). Non-destructive assay does not by itself provide conclusive characterisation. It always requires a minimum amount of process knowledge to interpret the results. The better the process knowledge means the more accurate the results.

There has been a recent trend to develop large, complex and expensive all purpose non-destructive assay systems for use on wastes during later phases of the waste life cycle. This approach may have been desirable for waste managers because all waste can be routed through a single characterisation process, thereby making the process flow chart simpler. This logic, however, is often flawed. Leaving characterisation activities until later in the life cycle is giving up the opportunity to get the best quality information in the most cost effective way. Although this type of approach may be necessary to characterise historical wastes, the best approach for new wastes involves the simplest methods designed to match the given waste streams, with measurements taken as close to the time of waste generation as possible.

When there is an abundance of process knowledge and the characteristics are stable and reproducible, the simplest (and cheapest) non-destructive assay methods (e.g., package dose rate) may be used with a high degree of accuracy. If there is little prior knowledge or the package characteristics exhibit a high degree of variability, more sophisticated and complex (and hence, more expensive) methods must be employed, and the accuracy will be severely limited. The best and most accurate results occur when the process is highly controlled, the waste characteristics are uniform, and the simplest (and least expensive) methods are fully adequate. When waste characteristics are relatively uniform and a high degree of process knowledge exists, non-destructive assay may be effective with larger package sizes. When there is a high degree of variability or little specific process knowledge, accuracy may be limited by package size. This may have significant throughput considerations for a waste operations process. This may be less of a concern for a quality assurance control process where processing volumes and production demands may not be as severe.

The more sophisticated and complex the method means the higher knowledge and experience requirements for the operator. In addition, there will be increased maintenance and operational limitations – in short, the greater the potential for things to go wrong, both regarding accuracy of the measurements and in keeping the process operating. Highly complex solutions are difficult to manage and generally only employed when there has not been an opportunity to gain the appropriate information in advance (e.g., historical wastes).

The simper methods tend to be used for routine waste processes. More complex methods tend to be used for research and development, for quality control laboratories, or for very complicated and difficult waste streams such as historical wastes. When the highest degree of accuracy is required for each individual package, the more complex methods may be required. When the accuracy of a large population is more important than that of each individual package, simpler methods may be adequate. The statistics of averaging works is in favour of this regard. The best approach is to try to develop a programme whereby detailed process knowledge is obtained and preserved and the simplest, highest volume processes adequately meet accuracy requirements.

7.2.1.2 Radiochemical methods

In case of pure beta emitters and alpha emitters of long half-life, which are important for the long term safety of disposal, it is usually necessary to use destructive methods. These methods can be used in three steps:

a) Dissolution of samples;

b) Use of specific methods of separation;

c) Measurement depending upon the chemical and radioactive properties of the nuclides.

A general overview of radiochemical analysis techniques for some safety-significant nuclides is shown in Table 7.4 [30].

A more exhaustive overview of radio-analytical methods is given in [19].

Table 7.4 Radiochemical determination of some safety-significant nuclides

Nuclide	Description of analytical technique	Final counting method
^{3}H	Separation by distillation in oxidising environment. Preparation of distillate for liquid scintillation counting.	Liquid scintillation counter.
^{14}C	Oxidising distillation converting the carbon species to CO_2. Absorption in the liquid scintillation cocktail.	Liquid scintillation counter.
^{55}Fe	Dissolution with strong acids. Gamma spectroscopy of ^{59}Fe in sample to determine the amount of tracer ^{59}Fe. Precipitation of the ferric hydroxide and preparation of sample for measurement of X rays of ^{55}Fe.	Thin window NaI(Tl) (^{55}Fe) and Ge spectrometers (^{59}Fe).
^{59}Ni,^{63}Ni	Dissolution with strong acids. After precipitation of the ferric hydroxide in ammonia environment, nickel remains in solution. Specific precipitation by dimethylglyoxine. Destruction of organics by strong acid, dissolution, determination of yield, preparation of samples for measurements.	Thin window NaI(Tl) (^{59}Ni); liquid scintillation counter (^{63}Ni).
^{89}Sr,^{90}Sr	Dissolution or leaching with strong acids. Combination of precipitation and scavenging techniques. Separation of yttrium, precipitation of strontium as carbonate. Counting. Two week yttrium in-growth period, dissolution of carbonate, separation and counting of yttrium. The ^{90}Sr is derived from ^{90}Y counting. The ^{89}Sr concentration is calculated by subtracting the ^{90}Sr and ^{90}Y contributions to the counting rate of strontium carbonate. For older samples, the first counting of carbonates ends the analysis.	Gas proportional counter.
^{94}Nb	Dissolution with strong acids. Carrier is added and niobium is precipitated with ammonia hydroxide. After washing the precipitate with hot nitric acid, niobium is dissolved; cobalt and barium is used to scavenge contaminants. Solution is purified by anion exchange. Niobium is precipitated as the oxide.	Ge spectrometer.
^{99}Tc	Rhenium is substituted for technetium as a carrier. Alkaline fusion with subsequent dissolution with concentrated nitric acid and boiling to remove iodine. Cobalt is repeatedly used as a scavenging agent to remove radio-cobalt and other transition metal radio-nuclides. Precipitation as a complex with tetraphenyl arsonium chloride.	Gas proportional counter.
^{129}I	Alkaline fusion and subsequent dissolution. Stirring with anion exchange resin in batch form. Liquid-liquid extraction, precipitation as CuI and preparation of sample.	Thin window NaI(Tl) spectrometer.
Alpha and transuranic	Alkaline fusion followed by dissolution in HCl. Co-precipitation with barium sulphate. Dissolution and liquid-liquid extraction separating U, Th, Pu, Np, Am and Cm. Each fraction is electroplated and analysed.	Alpha spectrometer.
^{241}Pu	A fraction of the Pu separated from the U-TRU products is counted in a liquid scintillation counter. ^{241}Pu is counted in the tritium window and alpha emitting Pu is counted in a window above tritium. The ^{236}Pu tracer yield is determined from the ratio of ^{236}Pu to total alpha/plutonium applied to the liquid scintillation alpha counting.	Liquid scintillation counter with pulse height analysis.

7.2.2 Chemical characteristics

Following characteristics can be determined by various analytical methods [30]:

- Leaching;
- Pyrophoricity;

- Content of explosive mixtures of flammable gases;
- Corrosion (both container and waste);
- Chemical compatibility;
- Head space gas components in waste packages;
- Chemical reactivity;
- Content of hazardous or toxic components.

Regarding leaching, data on the chemical durability or leach rate will serve following purposes:

a) Development and characterisation of waste forms;

b) Analysis of the safety of waste management alternatives;

c) Quality of the waste immobilisation product.

A principal problem of the various leaching rate analysis techniques is interpretation of the results from the extrapolation in time point of view, or simulation of a long term repository condition in relatively shorter term measurement, respectively. The problem is solved pragmatically by using standardised conventional measurement methods.

Phyrophoricity is analysed, if needed, by thermo-gravimetric analysis. The same method may also be used for the determination of the free water content of the waste form or the burnable fraction of incinerator ashes.

Some waste forms can, in the long term condition within disposal structures, generate hydrogen or methane by radiolysis or by anaerobic corrosion, or have the volatile compounds as waste constituents. It can be estimated by calculations or empirically, by analysis of the void space air of the waste package.

Another type of waste, waste form and waste package study is the estimation of corrosivity. Various immersion test methodologies can be used for this study, followed by examination of the nature of the corrosion processes by visual inspection of surfaces, mass loss measurements, metallographic investigations, or other types of surface analyses.

7.2.3 Physical, mechanical and thermal characteristics

The physical, mechanical and thermal properties of waste packages are better understood by considering structural features, and mechanical and thermal characteristics. Following characteristics may be studied if such study is required by the safety analysis approaches for the next waste management stages pending disposal:

- Porosity and permeability for gases and water;
- Homogeneity;
- Dimensional stability;
- Content of free liquids;
- Diffusion measurements (for cementitious materials);
- Compressive strength and load resistance;
- Drop tests;
- Fire resistance and thermal conductivity, freeze/thaw stability.

Biological properties represent the last group of characteristics, the study of which may be, similarly to the characteristics above, subject of standardisation.

7.2.4 Summary overview

1. The characteristics, which need to be determined, will vary between different types of wastes. Basically, it can be influenced by the waste acceptance criteria for disposal of the waste (see Section 7.5) and by other pre-disposal management stage requirements. For wastes, the characteristics to be considered cover their radionuclide content as well as their chemical, physical, mechanical, thermal and biological properties.
2. Significant progress has been made in the development of waste characterisation techniques, especially for homogeneous wastes. Heterogeneous wastes present a particular challenge and currently available methods tend to be costly and time consuming.
3. Some waste streams also contain non-radioactive toxic constituents, such as heavy metals, which may be more hazardous than the radioactive constituents. Special attention needs to be given to ensure that such constituents are adequately characterised and appropriate treatment and conditioning methods are utilised.
4. The characterisation techniques in use vary widely in both application and method, both nationally and internationally. Serious efforts have been made for standardisation of the characterisation methods. A good overview of the relevant standards is listed in a particular annex of [19]. The same effort lead to establishing the European network of testing facilities for the quality control of radioactive waste packages (EN-TRAP) 10 to 15 years ago [31].

7.3 Monitoring of difficult to measure nuclides

The inventory of the radio-nuclides to be declared with each waste form is numerous and varied (alpha and/or beta emitters). Some of these radio-nuclides are easy to measure using non-destructive or destructive assays, but most of them are difficult to measure and need destructive analysis in a laboratory or a calculation using special codes (e.g., pure beta emitters). Some of these are impossible to measure even in a laboratory.

The scaling factor methodology [30] can determine the radioactivity of difficult and impossible to measure radio-nuclides using correlations between them and key nuclides chosen among the easy to measure radio-nuclides. Specifically, the difficult to measure nuclides are predicted from a gamma nuclide easily measured by multiplying the concentration of this key nuclide by the scaling factors calculated from the radioactivity of nuclides obtained through appropriate radiochemical analysis or through modelling code calculation, and which represent the average relationship of the difficult to measure nuclide to the key nuclide. The impossible to measure nuclides are also predicted from an easy to measure gamma nuclide, which best is done by calculation using a model such as ORIGEN.

Each waste package can be measured with a non-destructive assay system, which provides the key nuclide concentrations. By using the scaling factors, the associated difficult-, impossible- and easy-to measure nuclides are then calculated and declared for each package.

Generally, the relationship between specific radionuclide (RN) to a key radionuclide (KN) can be expressed by the following equation:

$$a_{RN} = R\,(a_{KN})^b$$

where: a is the activity of the radionuclide considered,

R is the correlation factor,

b is the regression coefficient,

and log-log plots are used to present the different data. A meaningful correlation between RN and KN is reached for b values between 0.5 and 1.5.

Such correlation is typically determined through destructive analysis of a representative number of samples belonging to a specific waste stream, e.g., evaporator concentrates from a single nuclear power plant. The key nuclides are ^{60}Co and ^{137}Cs. The corrosion products (for example, ^{14}C, ^{59}Ni, ^{63}Ni, ^{94}Nb) are correlated with ^{60}Co and the fission products (for example, ^{90}Sr, ^{99}Tc, ^{129}I, ^{135}Cs) and actinides are correlated with ^{137}Cs.

It is important to analyse a sufficiently large number of waste samples prior to the attempt to establish correlations. It is also imperative to take into account the typical waste origin (reactor type, burn-up, chemistry of the reactor operation, type of waste, etc.) in advance to possibly use the same correlations for several waste streams.

When non-destructive assay measurements provide the key nuclide and other easy to measure nuclides, these data are kept for periodic control of the validity of the scaling factors used for the inventory. This methodology shows good applicability for stable waste streams. It is also applicable for simple and variable waste streams with higher correlation uncertainty. Its applicability for complex and variable waste streams is difficult. A scaling factor methodology requires sampling, destructive analysis, modelling, non- destructive analysis, and calculation.

7.4 Characterisation of historical wastes

Historical wastes are defined as wastes that have been generated without a complete traceable characterisation programme or quality management system in place [19]. According to this definition, these wastes may, in fact, be generated today. For example, the continued collection and placement in storage of raw wastes without robust characterisation and segregation, pending retrieval and full characterisation/treatment, etc. at some time in the future, is actually continued generation of historical wastes.

Key characteristics of historical wastes are:

- may be conditioned, partially treated, or raw;
- poor or no information/traceability;
- cannot conclusively identify originating process/location; and,
- waste streams may be mixed.

The primary identifiers of historical wastes are:

- incomplete history;
- incomplete or improper characterisation/treatment; and
- quality system does not cover the whole life cycle at the time of generation or does not meet the modern standards for the whole life cycle.

Different types of historical wastes can be distinguished:

- tanks with liquids;
- waste to be decontaminated before decommissioning;
- waste from workshops to be decommissioned;
- waste sites (near surface, interim storage sites).

For *tanks containing one liquid phase*, it should be recommended, at first, to try confirming the homogeneity by interviews with operational personnel with the knowledge of history and processes. Mixing is recommended, if possible, before sampling. In case of two or more immiscible phases, before sampling, the following steps should be recommended:

- determine the interface(s) and the volume of each phase;
- assume each phase is homogeneous;

- sampling has to be made in the middle of each phase.

Mixtures of liquid and solid phases in tanks represent a very frequent initial situation when characterising historical liquid wastes. In this case, the ratio solid/liquid should be determined, and each phase must be characterised. If the tank can be mixed, sampling must be conducted during the mixing process at different depths. At each depth, a sample must be made consisting of several sub-samples, one of which is to be sent to the laboratory. After that, all other sub-samples are mixed, filtered and the ratio of solid to liquid and its composition are determined. Consequently, for all samples taken at different depths, the laboratory performs a first evaluation. If chemical and physical analytical results are similar, it is assumed that the mixing process was correct and it is possible to calculate easily the solid/liquid ratio and the composition of each. If not, it is necessary to change the mixing process, requiring additional energy and time. If it is not possible to mix the tank, then recognised specialists should study this difficult case to determine the risks and the difficulties of further processing.

Regarding *waste decontaminated before decommissioning*, the first step is to measure the efficiency of the decontamination process by:

- taking samples of the decontamination liquid waste and performing a destructive analysis, and/or
- performing a non-destructive assay.

In case of difficulties, the object (e.g., walls, counters, drains, piping) to be decontaminated should be observed with a camera and/or non-destructive assay device such as a γ-camera to determine the specific location and type of contamination. It is advisable to use historical records (process data sheets, chemical inventory lists, purchase records, interviews with operational personnel) and/or non-destructive assay measurements (spectrometry, neutron counting), and determine risks due to fissile materials, decide for mechanical decontamination in order to reduce the final waste volume, use previous data to determine the sampling plan, make the chemical and radiochemical analysis and determine the first (chemical) decontamination process.

The aim of the characterisation of *wastes arising from the decommissioning of workshops* is to determine the scaling factors with sufficient accuracy together with the key nuclides. Controlling for homogeneity and representation of the sampling and sub-sampling process are therefore crucial points. It could be recommended to use non-destructive and destructive assays, calculation and modelling, which is necessary for radio-nuclides below detection limits. Three complementary steps are proposed. The first deals with measurements made in-situ, the second with analytical measurements, and the third is related to the calculation and modelling step.

If the historical data are accurate and well known, it will be possible to start the first two in-situ actions:

- Determine and locate hot spots by using a γ-camera, (or dose rate measurement);
- Control the consistency of the nuclide vector to decide about the location and the choice of the representative active samples by using a gamma spectrometer with a counting system software;

and then take representative samples and classify these by type (metal, concrete, chemo-toxic elements).

Analytical measurements (for each type of sample) may consist of:

- dissolution by qualified procedures and control the representativeness of the solutions by processing standard samples of similar matrices;
- chemical separation needed prior to the measurement, depending on the measurement device (liquid scintillation, ICP-MS);

- measurement and calculation for each radionuclide of the mass and radioactivity;
- considering the appropriate chemical toxicity testing.

Final control and calculation of the difficult to measure radio-nuclides means:

- Validation of the different destructive assay results:
  * coherence between different samples;
  * coherence between destructive assay results and non-destructive assay results from the in-situ measurements.
- Calculation of the difficult to measure radio-nuclides:
  * study of the workshop history (where was the fuel processed, …);
  * selection of a key nuclide for each hard to measure radionuclide;
  * calculation of the ratios between difficult to measure radio-nuclides and the chosen key nuclide using computer codes (ORIGEN, CESAR) on fuels processed in the workshop before decommissioning;
  * considering the process chemistry that may alter these calculated ratios and make corrections.
- Fingerprint calculation:
  * determination of the associated fingerprint for each type of sample;
  * establishing the global fingerprint of the workshop by calculating it proportionally to the weight of each type of waste inside the workshop.

The last historical waste types are the *historical waste sites* (e.g., near surface, interim storage sites). Several questions are to be answered and information must be provided when preparing the characterisation strategy for such wastes:

- What is the extent (site/volume) of the waste? Within this question, it is needed to:
  * determine the physical boundary of the waste site that will be retrieved and processed by physical sampling and analysis (refer to sampling standards).
  * use any historical information and interview individuals who can provide information.
- Non-destructive assay should serve to assess safety requirements for workers.
- What is the condition of the waste?
  * drummed;
  * intact;
  * breached/corroded;
  * dry or wet solid or liquid;
  * contaminated equipment/supplies;
  * contaminated soil;
  * use information to determine retrieval, transport, characterisation plans.

Issuing from the answers, the next strategic characterisation steps need to comprise the following actions:

- Determine the best methods to sort the waste into similar waste streams based upon waste minimisation, disposal path and anticipated waste form.
- Determine a representative sampling plan to include witness samples for archival purposes (based upon heterogeneity of waste stream). Samples should be archived until there is an approved waste conditioning process, waste form and disposal site.
- Apply detailed non-destructive/destructive assay characterisation of waste sorted into similar waste streams for safety, transportation and general knowledge of nuclear, chemical and physical properties.
- Based upon characterisation data, determine if further processing of sorted waste streams (or established parts of a site) would be beneficial for safety, waste minimisation or regulatory purposes.
- Determine final waste conditioning process, waste form and disposal site.

7.5 Characterisation of wastes for disposal

The importance of specific radio-nuclides is a function of many factors. Because of these factors, which are dependent on the particular characteristics of the waste disposal route, it is not possible to prejudge conclusively which radionuclide will be the most significant in terms of human health and environmental impact in the long-term. However, an analysis of radio-nuclides for which specific waste acceptance criteria are defined in other countries reveals a degree of consensus on the most important radio-nuclides, as demonstrated for repositories in Table 7.5 [32].

The following radio-nuclides feature in more than eight of the ten criteria reviewed:

- ^{3}H, ^{14}C, ^{59}Ni, ^{63}Ni, ^{60}Co, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I and ^{137}Cs.

It is notable that these are all beta- or beta-gamma-emitters, and that generally the specification of alpha-emitters is less common. Often the approach adopted is to define a total alpha activity limit on the waste. Whilst this is convenient, it misses highly important aspects of the relative importance of different radio-nuclides (particularly uranium, plutonium and americium isotopes). Given the importance of these radio-nuclides to long-term radiological safety of disposed wastes, it should also be recommended that key isotopes of these elements should be specifically determined, as well as total alpha and so-called total beta activity. Table 7.6 shows a complete generic set of the safety-significant radio-nuclides.

The considerations above deal exclusively with the waste acceptance criteria - radionuclide inventory and concentration. The criteria are corresponding to the repository safety assessments [33]. Similar approaches can be applied in the selection of other, for instance non-radiological criteria. Their general list is shown in the previous Section 7.2.

Generally, the characterisation of radioactive waste performed for the purpose of acceptation of wastes or waste packages for disposal addresses the question of whether the waste characteristics conform to the waste acceptance criteria of the given disposal facility. The waste acceptance criteria could match:

- the results of the safety assessment of repositories, especially when determination of waste acceptance criteria is an objective of such assessment,
- parameters of waste package and form taken into account within the safety assessment of repositories,
- regulatory requirements, both nuclear and non-nuclear (e.g., content and inventory of non-nuclear hazardous and toxic components),
- criteria qualitatively or semi-quantitatively reflecting a 'good practice' in waste acceptance criteria of well-functioning disposal facilities and their corresponding processes controlling the waste package characteristics compliance with the waste acceptance criteria.

Table 7.5 Considerations of radiologically significant radio-nuclides in various waste acceptance criteria

Radio-nuclide	Half life (years)	IAEA Generic WAC (Operational)	IAEA Generic WAC (Disposal)	Mohovce WAC	Dukovany WAC	El Cabril WAC	Drigg WAC	SFR WAC	US WAC (10 CFR 61)	Romanian Phare WAC*	OPG Waste Inventory*
H-3	12.4										
Be-10	1.6E+6										
C-14	5730										
Na-22	2.6										
Cl-36	3.0E+5										
Ca-41	1.4E+5										
Mn-54	0.856										
Fe-55	2.7										
Ni-59	7.54E+4										
Ni-63	96										
Co-60	5.27										
Zn-65	0.668										
Se-79	1.0E+5										
Sr-90	29.1										
Zr-93	1.53E+6										
Mo-93	3500										
Nb-94											
Tc-99	2.13E+5										
Ru-106	1.01										
Pd-107											
Ag-110m	0.684										
Sn-121m	55										
Sb-125	2.77										
Sn-126	1.0E+5										
I-129	1.57E+7										
Cs-134	2.06										

Radio-nuclide	Half life (years)	IAEA Generic WAC (Operational)	IAEA Generic WAC (Disposal)	Mohovce WAC	Dukovany WAC	El Cabril WAC	Drigg WAC	SFR WAC	US WAC (10 CFR 61)	Romanian Phare WAC*	OPG Waste Inventory*
Cs-135	2.30E+6			■				■			■
Cs-137	30	■	■	■	■	■		■	■	■	■
Ce-144	0.779	■									
Pm-147	2.62	■									
Sm-151	90	■	■	■							■
Eu-152	13.3	■									■
Eu-154	8.8	■									■
Tl-204	3.78	■									
Pb-210	22.3	■									
Ra-226	1600	■	■				■			■	
Ra-228	5.75	■	■								
Ac-227	21.8	■									
Th-232	1.40E+10	■	■				■				
U-234	2.45E+5	■	■				■				■
U-235	7.04E+8	■	■				■				■
U-238	4.47E+9	■	■				■				■
Np-237	2.14E+6	■	■								■
Pu-238	87.7	■	■	■			■	■			■
Pu-239	2.41E+4	■	■	■	■		■	■			■
Pu-240	6540	■	■				■	■			■
Pu-241	14.4	■	■					■			■
Am-241	432	■	■	■	■						■
Cm-244	18			■				■			■
Total α						■	■		■	■	
Total β-γ							■				

* Demonstrating CANDU-type (Canada Deuterium Uranium) waste

Table 7.6 Key radio-nuclides for characterisation in terms of disposal

Radio-nuclide	Half life (years)	Comments
H-3	12.4	Highly mobile and therefore potentially important during the first hundred years or so of radioactive waste management. Although low-hazard beta emitter, potential present in large quantities and therefore important for long-term waste safety.
C-14	5730	Generally mobile and with relatively long half-life. Although low-hazard beta emitter, potential present in large quantities and this, coupled with mobility and potential to accumulate in foodstuffs, makes this a key radionuclide for the safety of disposed radioactive waste.
Ni-59	7.54E+04	Beta-emitter of low-medium mobility in the environment, but generally present with considerable activity due to nickel in steels and structural components. Long half life means that the radionuclide persists in disposed waste, and can be important due to its relatively significant internal dose coefficient.
Ni-63	96	Beta-emitter of low-medium mobility in the environment, but generally present with considerable activity due to nickel in steels and structural components. Relatively high half life and a key radionuclide for the same reasons as Ni-63.
Co-60	5.27	Short half life radionuclide, which is important in terms of gamma dose rates during storage and potentially the early period of waste emplacement in a repository. Not significantly mobile before radioactive decay reduces concentrations to trivial levels.
Sr-90	29.1	Intermediate half-life beta emitter, which is particularly important due to its relatively significant activity concentration and high internal dose coefficients. Most likely to be of concern in terms of intrusion/disruption pathways in the long-term, after waste disposal.
Nb-94		Long-lived radionuclide that is relatively immobile, but which has high energy gamma emissions and can be important for external irradiation by contaminated environmental materials, or waste itself. Usually found to be one of the dominant gamma-emitting radio-nuclides in safety assessments of disposed radioactive wastes.
Tc-99	2.13E+05	Relatively mobile and long-lived radionuclide, which can migrate from a disposal facility relatively easily and which can give rise to amongst the highest future doses from disposed radioactive wastes.
I-129	1.57E+07	Relatively mobile and long-lived radionuclide, which can migrate from a disposal facility relatively easily and which can give rise to amongst the highest future doses from disposed radioactive waste.
Cs-137	30	Radionuclide often dominant in terms of activity (along with Sr-90). Can be moderately mobile in environmental media under certain conditions, but intermediate half life means that it is a key radionuclide in disposed radioactive wastes for the first few hundred years after disposal.
U-234	2.45E+05	Constituent of nuclear fuel that has a long half life and can give rise to radiologically significant progeny such as Th-230, Ra-226, Pb-210 and Po-210. Both the parent and daughter radio-nuclides are commonly found to be amongst the radio-nuclides resulting in the highest doses from disposed radioactive waste.
U-235	7.04E+08	Constituent of nuclear fuel that has a long half life and can give rise to radiologically significant progeny such as Pa-231 and Ac-227. Both the parent and daughter radio-nuclides are commonly found to be amongst the radio-nuclides resulting in the highest doses from disposed radioactive waste.
U-238	4.47E+09	Constituent of nuclear fuel that has a long half life and is the parent (following several short-lived radio-nuclides) of U-234. A key radionuclide in its own right, and as a precursor to U-234 and its progeny.
Pu-238	87.7	Intermediate half life radionuclide, highly immobile in the environment but parent to the radioactive decay chain. Alpha-emitting plutonium isotopes are characterised by their high internal hazard, which means they are significant radio-nuclides in disposed radioactive wastes.

Radio-nuclide	Half life (years)	Comments
Pu-239	2.41E+04	Long-lived radionuclide with high radio-biological hazard, and parent to a range of radioactive progeny. Highly immobile in the environment. These factors mean that disposed waste containing Pu-239 is a persistent hazard in the environment, and as a result it is a significant radionuclide.
Pu-240	6540	Long-lived radionuclide with high radio-biological hazard, and parent to a range of radioactive progeny. Highly immobile in the environment. A significant radionuclide for the same reasons as Pu-239.
Pu-241	14.4	Short-lived radionuclide that is principally important as a parent to Am-241, and therefore significant. Highly immobile in the environment.
Am-241	432	Long-lived radionuclide with high radio-biological hazard, and parent to a range of radioactive progeny. Highly immobile in the environment. This means that Am-241 is a persistent hazard in the environment, and as a result it is a significant radionuclide.

An approach analogous to the above may also be applied in still frequent cases where wastes are conditioned according to criteria established in some instances by waste operating organisations in the absence of a repository and its approved waste acceptance criteria. National criteria need to be established based on repository (conceptual, real) design and corresponding performance assessment. Existing historical waste inventories also need to be characterised against such criteria.

8. Management of Special Wastes

8.1 Aqueous waste

8.1.1 Characteristics of aqueous waste

In most cases, the general approach to the treatment of wastes from this group is to develop a sequence of treatment steps that would in the first step(s) render the waste amenable to standard treatment technologies applied in the final stage(s) of the sequence. It comprises boric acid removal and organic complexants degradation in the first two steps followed by a standard step of radio-nuclides separation by sorption onto inorganic ion exchange sorbents. Four categories of ‘problematic’ wastes have been identified in this group, comprising:

- decontamination liquids;
- waste containing oxo-anionic contaminants;
- aqueous waste containing large amounts of suspended matter, oils and other organic substances;
- acidic transuranium elements containing waste.

In the category of decontamination liquids, four specific waste streams have been addressed as examples, the major radioactive contaminants in all streams having been activated corrosion and fission products:

- Liquid waste from the MEDOC process (Metal Decontamination by Oxidation with Cerium) based on the use of Ce (IV) as a strong oxidising species in sulphuric acid, and developed and used at the SCK-CEN for the decontamination of stainless steel or carbon steel pieces. The effluents of this process represent diluted sulphuric acid mixed with cerium IV sulphate and all the dissolved metallic elements coming from the corrosion of the stainless steel material. The salt content of about 22 g/l gives a mixture of hydroxide sludge of about 40 g/l after neutralisation with sodium hydroxide.
- Spent decontamination solutions with novel composition based on the use of nitrilotriacetic acid (NTA) as a substitute for the corrosive oxalic acid. The spent solution used contained various quantities of nitrilotriacetic acid, hydrazine, Fe^{3+}, Mn^{2+}, Ni^{2+} and Cr^{3+} ions.
- Spent decontamination formulations generated at different stages of the nuclear fuel cycle that contained various organic complexants and/or anionic surfactants, e.g., ethylenediaminetetraacetic acid (EDTA), citric and ascorbic acids, sodium salt of dodecyl benzene sulfonate or sodium lauryl sulfate.

  Some hazardous organic materials may be separated from the liquid waste streams, purified, and recycled. Assuming that the recycling of such materials has been maximised, the remaining organic hazardous materials are best treated by a destruction technology, commonly by a thermal treatment. It is important to demonstrate that the thermal treatment process is capable of destroying the hazardous organic material which may be present.

  Organic sludges are heavy, highly viscous oil or grease, or may be organic liquids adsorbed onto inorganic materials such as vermiculite, clay or onto organic adsorbent materials. Some inorganic stabilising agents will solidify oil with as little as approximately 25 wt % added inorganic. Processing of organic sludge is more difficult than the treatment of liquid wastes discussed above. The objective is to remove or destroy organic material so that the remaining inorganics may be stabilised for disposal.

- Acidic decontamination solutions based on oxalic, citric, formic, sulphuric or nitric acid that may also contain some reducing agents like hydrazine, Na-dithionite, or ascorbic acid.

  In the category of oxo-anionic contaminants containing waste, fuel pond water as a waste stream was addressed, that contained sulphates, oxalates and chlorides at µg/l level and that was contaminated by ^{125}Sb, ^{137}Cs and ^{60}Co at 101-102 Bq/l level.

  The aqueous waste streams studied, contained large amounts of suspended matter, oils and other organic substances, comprising, e.g., 5.85 g/l NaCl and 50 mg/l with tens of grams of salts or silts per liter.

  The acidic transuranium elements containing waste stream was a simple diluted nitric acid solution (pH = 1.1) contaminated with ^{241}Am and ^{236}Pu. This stream was generated during the selective sorbents development and testing in the laboratory [34].

8.1.2 Processing of aqueous waste

8.1.2.1 Decontamination liquids

In [34], it is indicated that two different types of spent decontamination liquids have been addressed by a total of four different groups. The first type of this waste - spent decontamination liquids containing organic complexing agents and/or anionic surfactants - is a waste stream that is widely recognised to be 'problematic' because it is not amendable to standard high volume reduction waste treatment processes. A two-stage process has been developed comprising photocatalytic degradation of the organic components of the waste followed by treatment of the resulting liquid by a standard method (e.g., sorption).On the other hand, highly selective inorganic ion exchangers have been developed that would allow direct separation of the radio-nuclides from the organic components containing waste.

A photo-catalytic organics degradation process was applied in the first stage of a two stage process for the treatment of spent decontamination liquids. In photo-catalysis, degradation of organic compounds occurs mainly due to the reactions with OH-radicals generated by interaction of ultraviolet radiation with a catalyst. In the presence of heterogeneous photo-catalysts (e.g., TiO_2), the electron-hole pairs are formed that can then produce the OH-radicals in reaction with the molecules of water adsorbed at the surface of the catalyst.

In the case of spent decontamination solutions containing nitrilotriacetic acid (NTA), it was demonstrated that most radio-nuclides can be effectively removed from these solutions even in the presence of nitrilotriacetic acid, e.g., by means of a choice of inorganic absorbers or Purolite NRW160 resin. The two-stage process was needed only for efficient separation of radioactive cobalt that was not satisfactory from the original spent solution. Laboratory-scale experiments, performed with a simulant of a solution of the Temelin nuclear power plant, revealed that the efficiency of the separation of radioactive cobalt is significantly increased after nitrilotriacetic acid photo-degradation; however, total degradation of the nitrilotriacetic acid is mandatory. Purolite NRW 160 resin or sodium titanate inorganic ion exchangers show the highest distribution coefficients for radioactive cobalt uptake.

Another approach to the treatment of spent decontamination solutions comprised the development and the study of a novel highly selective inorganic ion exchanger for direct separation of radioactive cobalt from media containing organic complexing substances and/or acids commonly used in decontamination [34]. The new manganese antimonate (MnSb) material belongs to the group of novel developed mixed oxide materials with pyrochlore structure. These materials allow varying the acidity and micropore size of the material by substitution of different metals in the pyrochlore structure thus affecting its ion exchange selectivity.

The results achieved demonstrated good prospects of these novel materials. In the experiments with oxalic, citric, formic, sulphuric and nitric acids, uptake of radioactive cobalt was more dependent on solution pH than type of acid. Picolinic acid decreased the uptake of radioactive cobalt significantly, obviously due to strong complexation. Generally, the distribution coefficients of radioactive cobalt were high (5’000-10’000 ml/g) at pH 3-2 but decreased clearly as the pH approached the value of 1. In a test conducted at pH = 2, reducing oxidising agents (hydrazine, Na-dithionite, ascorbic acid) did not decrease the uptake of radioactive cobalt.

A second type of spent decontamination liquids represents a rather specific case of liquid effluents from the MEDOC© process. The efforts in this field concentrated on developing an alternative to the process of effluents treatment that involves a costly and energy- and space-demanding freeze/thaw cycle making the separation of sludges easier, i.e., electro-electrodialysis (EED), the latter allowing the removal of 90 % of the 1M sulphuric acid present [34]. More than 99 % of the cerium IV, which is a high cost product, could be recuperated as $Ce_2(SO_4)_3$ with a very high selectivity (impurities < 0.1 %) after reduction of Ce(IV) to Ce(III) by H_2O_2.

An effective removal of sludges from the waste was achieved by optimising both the solids separation steps. For the reduction of the volume of settled sludge and easing the filtration, an optimum coagulants-flocculants mixture and the conditions of their application were selected in extensive tests from the chemicals regularly used in the treatment of industrial waste water. The best results were obtained with the organic coagulant EPIDMA and a mixture of coagulants-flocculants $FeCl_3.6H_2O$/C587/A370 L. As an alternative to the freeze/thaw treatment, the use of a pre-coat (perlite) was proposed to improve the performance of the filtration step. The use of such pre-coat could enhance the maximum load capacity on each filter cartridge by about a factor of two; further amelioration was achieved by addition of a coagulant (EPIDMA) during the filtration.

8.1.2.2 Waste containing oxo-anionic contaminants

Two different technologies were proposed for the treatment of oxo-anionic contaminants (e.g., ^{99}Tc or ^{125}Sb species) containing waste [34]. A first technology is based on the development and application of novel highly selective inorganic ion exchange materials. The newly developed material is based on a tin dioxide modified by doping by other metal cations. In tests with simulated solutions, this material exhibited very high uptake for ^{99}Tc - values of the distribution coefficients (Kd) over 106 ml/g - in solutions with ionic strengths of 1 mol/l, and the uptake level remained high over a broad pH range from pH 2 to 11. Two different uptake mechanisms were identified for the technetium uptake in acidic and near neutral to basic solutions; however, the mechanism of uptake by ion exchange was questioned and the exact sorption process was still left open. The results of the column tests performed with 1 M $NaNO_3$ indicated that there is at first a rapid uptake at the surface of the material, followed by a slower process inside the granules of the material.

A second technology is based on the exploitation of the interaction of some anions with poly-nuclear hydroxo-complexes of polyvalent metals. The basic part of the research performed demonstrated that the anions investigated can be sorted according to their ability to influence formation and state of iron (III), chromium (III) and thorium (IV) poly-nuclear hydroxo-complexes.

8.1.2.3 Aqueous wastes containing large amounts of suspended matter, oils and other organic substances

For this type of waste, a multistage process consisting of three basic steps has been developed and demonstrated [34]:

- removal of suspensions, mineral oil emulsions, or colloid particles with their simultaneous concentration to the highest achievable level in secondary waste;
- separation of radio-nuclides and other toxic impurities by membrane and/or sorption technologies;
- concentration of the aqueous secondary waste producing concentrates with salt content up to 300 – 350 g/l;

They differ by the technology used for the treatment of the liquid waste after separation of the particulates, colloids and/or emulsions. These options are:

- the AQUA-EXPRESS technology using a cascade of sorption filters with various inorganic sorbents, which have selectivity to different radio-nuclides, and a polishing stage of ultra-filtration. This set-up is suitable for cleaning of liquid waste with a chemical composition similar with natural waters, and a limited set of radio-nuclides (for example, ^{137}Cs, ^{90}Sr, ^{60}Co).
- RO-technology that uses reverse osmosis with a subsequent additional treatment of the permeate on selective inorganic sorbents. In comparison with the AQUA-EXPRESS, this technology is characterised by a smaller volume reduction of the waste but it allows cleaning of the waters with complex chemical and radionuclide composition.

The new developments pertain to the first and the third of the above specified stages of this new process. The efficiency of the filtration stage could have been significantly improved by:

- replacement of standard filtration devices with cross-flow micro- or ultra-filtration. For this purpose, 'DISKVIKIG' and 'CYLINDERVIKIG' devices equipped with rotating disk- or cylinder-shaped ceramic, cermet or metallic filtration membranes were developed.
- introduction of strong oxidising agents (e.g., ozone, hydrogen peroxide, oxo-chlorides) into the filtered waste. By this addition, reduced membrane clogging is achieved by organic substances decomposition that is further facilitated by the prolonged contact of the organics with the oxidising agents in the cross-flow filtration reject stream. In addition, some contaminants (e.g., Fe^{2+}) will precipitate after the oxidation and will be filtered off.

For the concentration of the secondary aqueous waste, an original method has been developed and tested on real waste for the evaporation of water from the wastes at ambient temperature (+15 to +45 °C) in a filter-press device with porous polymeric plates. The pilot device used had a productivity of up to 10 l of evaporated water per hour.

The above-described set-up has been optimised and applied for the treatment of several real waste streams in full scale.

8.1.2.4 Acidic transuranium elements containing wastes

This is an example of wastes that have been defined as 'problematic' because the presence of alpha-nuclides in the waste disqualified it for acceptance for processing with other liquid wastes by means of locally available technologies [34]. For the removal of ^{241}Am and ^{236}Pu, a commercial highly selective inorganic ion exchange material, Quasar N (antimonysilicate - $SbSiO_2$), was proposed. In two 10 ml columns, two batches of the waste with total volumes of 5 l and 1 l, contaminated up to about 8 or 71 kBq/l, respectively, were treated. The total decontamination factors reached in these runs were about 30 or 73, respectively. The decontaminated acidic solution was released into the sewer after neutralisation.

8.2 Aqueous concentrates

8.2.1 Characteristics of aqueous concentrates

A variety of aqueous concentrates are generated during the operation of reactors and other nuclear fuel cycle facilities. Such solutions contain high concentration (10 - 500 g/l) of dissolved salts. The evaporation of low level effluents is a very common operation that gives rise to such waste in the form of evaporator concentrates. Reverse osmosis is another process which leads to the generation of a concentrated ‘reject’ stream. The aqueous raffinates generated during reprocessing operations contain substantial quantities of dissolved salts which get further concentrated in the evaporators. Though direct immobilisation is possible and is practiced in some cases, it is desirable to have a management scheme which leads to substantial volume reduction. Implementation of such a scheme would depend on the availability of highly efficient processes for radionuclide separation from the concentrates.

The difficulty of such an approach can be appreciated from the fact that many of the non-radioactive ions, present at high concentration levels, can compete with the radioactive ions which are present at much lower concentrations. The situation is further complicated when the concentrates also contain strong complexing agents which bind the radioactive ions and must be destroyed before separation can be achieved.

Concentrated aqueous waste streams from a number of sources and having a variety of chemical/radiochemical composition have been taken up for study [34]. The radio-nuclides present in these streams included fission products (^{137}Cs, ^{90}Sr, ^{106}Ru, ^{125}Sb, ^{99}Mo, ^{99}Tc, etc), activation products (^{60}Co) and actinides (U, Pu). The waste streams in this category belong to four groups depending on their origin: reactor operation, reprocessing, decommissioning and isotope production:

- The problem of removing ^{137}Cs and ^{90}Sr from WWER-type (water-water energy reactor) nuclear power plant evaporator concentrates was solved earlier using selective inorganic sorbents like Cs-Treat and Sr-Treat. The removal of ^{60}Co was evaluated based on a simulant containing sodium tetraborate hexahydrate (95 g/l), potassium nitrate (20 g/l), sodium nitrate (127.5 g/l), sodium hydroxide (30 g/l) and various concentrations of ethylenediamine-tetraacetic acid (EDTA). The isotope ^{57}Co was used as a tracer for radioactive cobalt.
- Alkaline evaporator concentrates of nuclear power plant origin contained alkali metal salts of borate, nitrate, citrate and oxalate ions. The waste was similar to the one noted above except that instead of ethylenediaminetetraacetic acid (EDTA), citrate and oxalate were present resulting from the use of the Citrox process for decontamination applications. Similar to the above described waste, the problem of removing ^{137}Cs and ^{90}Sr from such waste was solved earlier; the investigated process was based on the use of selective inorganic-organic sorbents with a poly-acrylonitrile (PAN) matrix and active components based on current CRP; the target radionuclide was ^{60}Co.
- Acidic waste was generated during the recovery of ^{99}Mo from nitric acid solution resulting from the dissolution of irradiated low enriched UO_2 targets. The waste contained nitric acid (<1 M), uranyl nitrate (20 - 200 g/l) and fission products. Given that the related waste had been produced and accumulated since the late 1960s, the activity in the stored liquid was dominated by ^{137}Cs and ^{90}Sr.
- A saline waste would result from the proposed dissolution of Na/K liquid metal coolant during decommissioning of fast reactors, e.g., experimental (60 MWt) and prototype (600 MWt) fast breeder reactors. The projected volume and compositions were 15’000 m^3 of 4M NaCl + 0.15M KCl and 800 m^3 of 2.2M $NaNO_3$ + 0.7M KNO_3, respectively. The target radio-nuclides included ^{137}Cs, ^{90}Sr, ^{60}Co and Pu.
- Another WWER-type (water-water energy reactor) nuclear power plant evaporator concentrate was considered including 250 - 300 g/l of sodium-potassium metaborate,

sodium nitrate (up to 75 g/l), substantial concentrations (up to 2 - 3 g/l) of Trilon B (disodium salt of ethylene-diaminetetraacetic acid - EDTA) and up to 2 g/l of complex salts of iron, nickel and chromium, pH 9-12. The major problems of this waste were concerned with Cs and Co radio-nuclides.

- The decommissioning of nuclear submarines and associated decontamination activities resulted in a mixed saline (sea-water) waste containing oxalates and about 10^{-3} mCi/l of ^{137}Cs and ^{90}Sr. The salt content was 0.1 g/l and the pH was 7 - 8. This waste was subjected to treatment using a reverse osmosis unit (DalRAO), resulting in a concentrated stream having 30 - 50 g/l salts, with calcium content more than 0.5 g/l. The activity was in the range of 10^{-1} mCi/l. The removal of ^{137}Cs from this waste was easily achieved with the application of composite ferrocyanide sorbents. The removal of ^{90}Sr was difficult and had been further addressed.
- The use of ferrous sulphamate in the past in reprocessing operations based on the PUREX (plutonium-uranium extraction) flow sheet resulted in high level waste containing substantial concentrations of sulphate ions (up to 10 g/l). Problems associated with phase separation of a sulphate-rich layer were encountered during vitrification of such sulphate-bearing stored high level wastes in lead borosilicate glass. Extensive work was carried out to overcome this problem by adopting two approaches: direct immobilisation in an alternative matrix without phase separation and chemical pre-treatment for removal of sulphate from the waste followed by vitrification.
- Ruthenium-bearing waste streams from reprocessing operations presented a special challenge for treatment because ^{106}Ru was present as numerous nitrosyl complexes which were difficult to remove by conventional ion exchange or chemical precipitation methods. The presence of high concentrations of dissolved salts further complicated the problem. In such streams, the activity of ^{106}Ru could vary from 10^{-3} to 1 mCi/l, depending on whether the source was intermediate level waste from reprocessing operations or from the pre-treatment of sulphate-bearing high level waste at a waste immobilisation plant.
- Plutonium reconversion operations at reprocessing plants have resulted in oxalate bearing acidic waste containing 0.01 M oxalic acid, 3.0 M nitric acid and traces of plutonium. The available practice of destruction of oxalic acid using $KMnO_4$ leads to additional salt loading in the solution. The possibility of an alternative improved technique for destruction of oxalic acid was addressed.

8.2.2 Processing of aqueous concentrates

A number of processes including selective ion exchange, chemical precipitation, advanced oxidation methods, etc. were developed for the effective treatment of the various ‘problematic’ concentrated waste streams described above [34]. The scale of work ranged from laboratory scale using simulated waste spiked with radiotracers to full scale plant operation with actual waste.

8.2.2.1 Reactor operation

For WWER-type (water-water energy reactor) evaporator concentrates of composition similar to that from the nuclear power plant Loviisa, Finland, a three-step process was developed and demonstrated at the laboratory scale. The first step deals with the boric acid removal from the solution. On decreasing the pH value of the solution from 12.5 to about 9 by addition of concentrated nitric acid, the vast majority of boric acid is separated due to solubility limits and is filtered off. The separated boric acid can be purified and reused in the nuclear fuel cycle. The second step aims at limiting the negative influence of the presence of ethylenediaminetetraacetic acid (EDTA) in the solution to radioactive cobalt sorption on inorganic ion-exchangers. Photo-catalytic degradation of ethylenediaminetetraacetic acid (EDTA) using the photo-Fenton system was carried out in a small immersion well photo-catalytic reactor of 100 ml capacity. A 6 W low pressure mercury lamp was used for irradiating the samples by the short-UV range (254 nm) radiation. The method was shown to be promising for the destruction of the ethylenediaminetetraacetic acid (EDTA). After removal of both boric acid and ethylenediaminetetraacetic acid (EDTA), the last step is the removal of the radioactive cobalt. This was tested using Co-Treat (titanium oxide-based inorganic absorber. The result showed that sorption of radioactive cobalt improved after ethylenediaminetetraacetic acid (EDTA) removal by photocatalytic degradation. However, better sorption performance is required for practical application of this process. It is possible that further optimisation of the ethylenediaminetetraacetic acid (EDTA) degradation process would lead to total ethylene-diaminetetraacetic acid (EDTA) removal and more effective separation of radioactive cobalt by the sorption process.

8.2.2.2 Reprocessing

For the sulphate-bearing high level waste of reprocessing origin, barium borosilicate was found to be a suitable matrix which did not result in phase separation. Full scale operations have been carried out in an induction-heated metallic melter to vitrify nearly 20 m^3 of high level waste [34].

Waste loading in the direct vitrification process is limited, however, on account of the limited solubility of sulphate (≤ 1.5 % w/w) in glass. Hence, an alternative approach leading to sulphate separation from high level waste was studied at the laboratory scale using simulated and actual waste and at the pilot scale using inactive simulated waste. In this process, acidic high level waste is made alkaline by addition of sodium hydroxide. This leads to the precipitation of a sludge which carries down a number of waste constituents: uranium, plutonium, lanthanides, strontium and substantial quantities of ruthenium. The alkaline supernatant stream contains most of the Cs activity, some portion of ^{106}Ru as well as sulphate and aluminate. A process flow sheet was developed to manage the sulphate-free sludge and alkaline supernatant and verified experimentally at pilot scale using simulated solutions. The sludge could be dissolved in concentrated nitric acid and vitrified in a conventional sodium borosilicate matrix.

The removal of uranium from the solution by solvent extraction using Tri-n-Butyl-Phosphate (TBP) in a hydrocarbon diluent was found to lead to increased waste loading. The alkaline sulphate bearing supernatant can be decontaminated and converted to low level waste by removing ^{137}Cs using a Resorcinol Formaldehyde Polycondensate Resin (RFPR) column and ^{106}Ru using a zinc-charcoal column. The recovered ^{137}Cs can be utilised as a radiation source.

8.2.2.3 Decommissioning

For the treatment of highly saline wastes that would result from the dissolution of Na/K liquid metal coolant after decommissioning of a fast breeder reactor, tests were conducted using a broad range of commercial inorganic and proprietary composite absorbers. These tests were conducted on simulated wastes. Cs-Treat was confirmed to be good for Cs removal, KNiFC-PAN was proposed as an even more efficient alternative. Three absorbers, i.e., IONSIV IE-910, sodium titanate and hydrous titanium dioxide were identified as prospective candidates for Sr and Co removal [34].

8.2.2.4 Isotope production

The acidic intermediate level waste resulting from the processing of ^{99}Mo has been processed to a crystalline hydrated uranyl nitrate solid form by a combined evaporation-chemical process. This solidification process leads to a 97 % reduction in volume but is an interim step only as the solid crystalline product is unsuitable for long term disposal due its solubility in water. Therefore, incorporation of the waste into a titanate based ceramic waste form (SYNROC) was identified as an appropriate strategy for immobilisation of the waste before ultimate disposal. The process involves liquid impregnation into a precursor, drying and subsequent calcination for canning and hot isostatic pressing [34].

8.3 Spent ion-exchange resins and radioactive sludges

8.3.1 Characteristics of spent ion-exchange resins and radioactive sludges

The waste streams discussed in this section can be divided into two categories, a major one including spent ion-exchange resins (SIER) and a small one related to low-level sludges of specific type.

The radioactively contaminated ion-exchange resins comprise a specific type of radioactive waste produced at different stages of the nuclear power cycle. Most of the spent ion-exchange resins are produced at nuclear power plants and spent nuclear fuel reprocessing plants. During the operation of a nuclear power plant, the contaminated spent ion-exchange resins emerge in the following processes:

- Clean-up system for the primary coolant;
- Condensate decontamination of single-circuit reactors;
- Decontamination of water from spent nuclear fuel storages (cooling ponds);
- Decontamination of de-activation, trap and other water in special water decontamination systems of nuclear power plants.

The estimated total volume of ion-exchange resins at a nuclear power plant is 0.02 – 0.025 m^3/MW of nominal power for two-circuit reactors of the WWER-type (water-water energy reactor) and 0.08 – 0.025 m^3/MW of nominal power for single-circuit reactors of the RBMK-type (high power channel reactor).

The spent ion-exchange resins from other production cycles substantially differ on contamination degree and radionuclide composition. For example, the spent resins from the decontamination systems of primary circuits contain significant amounts of ^{14}C isotopes.

For interim storage, when short-lived radio-nuclides decay, the spent ion-exchange resins are usually sent to liquid waste storage facilities where they are kept underwater for some time period. Before transferring to long-term storage, these resins are subject to processing, consisting in conditioning and volume reduction of radioactive wastes sent for final disposal.

As for the second category, the problem of managing low-active sludges produced during the operation of nuclear power plants, reprocessing plants and research reactors is in a way similar to the spent ion-exchange resins treatment problem. The concentration and transformation of sludges into a form suitable for disposal requires special research activities.

As regards the first category, the following real and simulated spent resins and respective storage facilities have been studied and tested [34]:

- Radioactive spent resins of the primary circuit of a heavy water reactor with specific activities for ^{137}Cs, 2.87×10^3 Bq/g (dry) and ^{60}Co, 5.89×10^4Bq/g.
- Radioactive spent resin storage and in-station resin storage tanks (mixed cation exchanger and anion exchanger of the resin Amberlite IRN-150) at CANDU-reactors (Canada Deuterium Uranium) with activities for ^{14}C from 170 GBq/m^3 up to 380 GBq/m^3.
- Spent resins located in a pressurised heavy water reactor with activities for ^{60}Co, 6.7 x 10^{12} Bq, for ^{137}Cs, 2.7 x 10^{13} Bq and for total alpha, 3.5 x 10^{10} Bq.

- Poly(styrene-divinylbenzene) resins – cation exchangers (–SO_3H) and anion exchangers (CH2N+$(CH_3)_3$OH-) with activities from 0.1 mCi/l up to 10 mCi/l (radio-nuclides present: ^{60}Co, ^{137,134}Cs, ^{90}Sr, ^{106}Ru, ^{65}Zn, ^{54}Mn, ^{125}Sb).
- Amberite IRN 77 as a simulant of spent resins containing heavy metals and non-active cesium and strontium for incineration.

The second category of wastes, sludges, were specially simulated to test a respective system at an effluent plant. The batch simulated test material had the following composition to test the most severe cases for the thickened sludge (total volume of the tank3000 l) [34]:

- NaCl (11 kg), $NaNO_3$ (4.5 kg), Na_2SO_4 (3.3 kg), $Al(OH)_3$ (125 kg), Na_2CO_3 (1 kg), water (made up to 3000 l).

8.3.2 Processing of spent ion-exchange resins and radioactive sludges

The methods for processing spent ion-exchange resins (SIER) for long-term storage can be divided into non-destructive methods, in which spent ion-exchange resins undergo minimum destruction in some preliminary procedures and are immobilised into an inorganic or organic matrix, and destructive methods concerned with complete or partial destruction of the organic ion-exchange resins.

Non-destructive methods of processing spent ion-exchange resins are characterised either by an increase of the total volume of radioactive waste (cementation) or by a slight decrease of this volume (bituminisation, polymerisation). In this case there are no problems of decontamination of large amounts of radioactively contaminated gases and spreading of contamination of ^{14}C radio-nuclides that occurs with carbon oxides emitted during the incineration of resins.

Spent ion-exchange resins have been fixated in composite cement materials containing high-efficiency cements (ASC) with fillers (zeolites) substantially reducing the radionuclide leachability from cement composites formed. Studies of composites properties with using high-efficiency cements were performed in comparison with regular Portland cements. Optimisation of the cement mixture composition was performed in regard to composite mechanical properties and their changes in the process of many days’ water immersion and freeze-thaw cycles. The obtained cement compositions showed good mechanical properties, performance stability and low diffusivity of ^{137}Cs and ^{60}Co radio-nuclides [34].

Destructive methods of spent ion-exchange resins processing, including various incineration methods, wet oxidation, pyrolysis, acidic and alkaline destruction are usually accompanied by a substantial (sometimes tens-fold) reduction of the volume of conditioned solid radioactive waste formed during processing. One of the possible ways of this volume reduction is the application of spent ion-exchange resins incineration in molten salt, in so-called molten salt oxidation reactors. The installation may comprise two molten salt oxidisation reactors, a primary reactor heated up to 700 - 900 ^{O}C; a secondary reactor heated up to 900 - 950 ^{O}C. The conditions of reactor operation in the process of destruction and accumulation of heavy metals, strontium and cesium in the system and their transformation into non-volatile carbonates have been determined. It was shown that the use of the secondary reactor allows reducing the carbon monoxide emission by up to 100 ppmv. Sulfur-containing compounds were completely retained in the reactors. Sulfur dioxide emission was not observed at all [34].

The problems of ‘dry’ destructive methods are related to noticeable emissions of radioactively-contaminated gases and a large content of ^{14}C radio-nuclides in the carbon dioxide formed at spent resins incineration. The problem of ^{14}C is becoming more urgent due to stricter requirements on environmental protection. Thorough filtration and absorption of all the emitted carbon dioxide by alkaline absorbers with formation of insoluble carbonates reduces significantly the total volume of solid radioactive waste produced at the end of such processes. That is why recently the methods of preliminary deactivation of spent ion-exchange resins have been developed intensively. This is especially crucial in regard to ^{14}C radio-nuclides, since the very necessity of localisation of the whole amount of carbon dioxide emitted at destruction imposes limitations on applications of many oxidation methods.

In principle, a separate group of technologies may be defined dealing with preliminary decontamination of spent ion-exchange resins and transferring of radio-nuclides into the liquid phase from which they could be conditioned with using of conventional methods. As an example, the radionuclide removal may be carried out through the consecutive treatment of a resin by sulfuric (10 %) and, after washing with water, oxalic acid [34]. Metal radio-nuclides (^{60}Co, ^{137}Cs) and α-emitters from deactivation solutions are removed by fixation in the process of electrolysis in aqueous (Co) and alcohol (^{137}Cs) media. The ^{14}C radionuclide emitted, after acidic treatment, as carbon dioxide, is fixated by gas infiltration through a sodium hydroxide solution. Secondary decontamination from cesium and cobalt radio-nuclides is conducted by filtration of the deactivation solutions through zeolites. The system with acidic decontamination of spent ion-exchange resins through consecutive treatment by sulfuric and oxalic acids was tested on real spent ion-exchange resins of a nuclear power plant in a pilot-plant variant, and it was shown that less than 0.7 % of the initial cesium activity and less than 2 % of the initial cobalt activity remained after the treatment.

The problem of spent ion-exchange resins treatment is similar to that of the processing of radioactively contaminated sludges. As an example, a simulated waste stream containing aluminium hydroxide, sodium carbonate, nitrate, sulfate and chloride has been analysed and comprehensive tests of the approach were performed to estimate the applicability in the management of radioactive waste sludges produced in an effluent plant [34]. Optimal regimes of drum dryer performance were defined and the approach is commissioned for treatment of real wastes.

8.4 Site-specific and miscellaneous waste

8.4.1 Characteristics of site-specific and miscellaneous waste

Some problematic waste may be very site-specific, requiring consideration not only general, but also of some special properties or characteristics of this waste, taking into account some particular local conditions and options locally available for managing the concrete waste problem. Another specific and problematic waste may be the result of some combination of waste characteristics not typical for more common waste streams, which also may require ‘non-conventional’ approaches or some special technologies for treatment and conditioning of this waste. Some examples of such waste streams are [34]:

- A variety of radioactive wastes generated in the nuclear industry and at different nuclear applications may contain substantial portions of different organic components, which may be flammable, toxic, chemically aggressive, etc., which make them difficult to handle and process by conventional treatment and conditioning technologies.
- In existing installations and in future fusion installations, various categories of tritiated wastes are and may be produced, which could be considered as problematic for conventional processing to reduce or eliminate potential impact on the environment and human health. In particular for organic liquids containing tritium the objective is to destroy organics and to separate and fix tritium for further immobilisation, storage or reuse.
- Substantial portions of historic or legacy wastes could now be considered as problematic for their relocation and reconditioning because of their complex composition, pure initial characterisation, and inappropriate conditions for long-term storage or disposal.

Old biomedical wastes contaminated with ^{14}C, ^{3}H and other radio-nuclides is one example of such waste types. ^{14}C, ^{3}H isotopes found in healthcare wastes are mostly contained in a liquid solution. The contaminated liquid is either inside a syringe or still in the glass vial used for different medical tests. Some other materials used in these procedures are also discarded into waste containers and can also be contaminated with these or other isotopes. The potential bio-hazardous nature of these wastes has proved to be an additional complication in the development and implementation of appropriate characterisation and conditioning methods for such wastes.

- Spent liquid scintillation cocktails containing long lived alpha isotopes such as Am, Pu, etc. could also be classified as problematic waste for further processing both because of the specific radioactive contamination and the organic nature of these wastes. Special options should be developed to address both these characteristics of these wastes [34].

8.4.2 Processing site-specific and miscellaneous waste

Some conventional technologies for the processing of organic wastes, as for example incineration, require substantial investments, sophisticated off-gas cleaning and off-gas control systems, and are now hardly acceptable for the public. Therefore, new methods, more tolerant to the environment, for organic waste destruction and conditioning are needed that can provide high-destruction efficiencies with low capital and operating costs for the safe treatment of these types of 'problematic' radioactive wastes. New non-thermal methods for the destruction of organic components of radioactive waste are now being explored [34]. A molten salt oxidation (MSO) process has been studied for the safe and effective destruction of organic components of radioactive wastes. This work involved a laboratory-scale molten salt oxidation system where solid or liquid organic waste is injected into a bed of molten carbonate salt in the presence of an oxidising gas. The relatively simple molten salt oxidation process with carbonate salt completely destroys organic compounds such as contaminated oils. A major advantage of the process is that any acid gases generated are neutralised, so that no off-gas scrubbing system is required. All radioactive contaminants are retained within the bed of molten salt. In the past, high melting point salts have been used with air oxidation. In new evaluations, the use of low melting point salts is being investigated along with stronger oxidising agents, such as Fenton's Reagent and manganese (VII), oxidant concentration and temperature. The use of lower melting point salt mixtures appears to be an effective improvement to the molten salt oxidation process in order to lower consumption of energy required.

To process tritiated organic liquid waste a thermal/catalytic oxidation technology has been developed and demonstrated [34]. The heart of the system is a two-stage high temperature Hastelloy reactor. Organic liquid is injected into a heated cavity using oxygen as the propellant. Oxidation products and excess oxygen are re-mixed in an alumina diffuser zone downstream of the cavity and passed over heated 0.5 % platinum on an alumina catalyst. Gases leave the reactor and are separated into condensable (water) and non-condensable (CO_2) fractions. Typically, the wall temperature of the cavity is set at 750 °C while the catalyst temperature is fixed at 700 °C. A nitrogen gas cooling circuit is installed under the cavity heater to remove reaction heat when the need arises. The installation can accept liquid flows of up to the order of 1 l/h with activities up to 0.2 TBq/l. The gases leaving the reactor are excess oxygen, carbon dioxide and water vapour. Tritium is exclusively present in the water phase as HTO (tritiated water). Condensed tritiated water could be conditioned and stored or its tritium content could be recovered for recycling.

Investigations have been carried out to establish procedures for the characterisation and conditioning of old biomedical waste contaminated with ^{14}C, ^{3}H and other radio-nuclides [34]. Three potential options for this waste characterisation and conditioning were considered for evaluation of their efficiency, safety and cost:

(1) Evaluation of the gamma nuclides content of drums by a segmented gamma scanner. Sterilization of waste after recovery from the drum by the Stericycle Electro Thermal Deactivation (ETD) process. This consists of a mechanical shredding and mixing of the waste in a negative-pressure environment, after which it is transferred through the Stericycle Electro Thermal Deactivation (ETD) process. After deactivation, the waste is reloaded into the drum for further conditioning by super-compaction and cementation.

(2) Initial evaluation of the gamma nuclides content in the drum as above. Sterilisation of the drummed waste using a very high activity gamma source. Super-compaction, collection and analysis of the released liquid phase (if any), followed by conditioning of the compacted waste and released liquids into cement (in over-pack container). Model experiments have shown that present liquids in this case could almost completely be released from the compacted waste. This method has an advantage, as the Stericycle Electro Thermal Deactivation (ETD) process requires shredding and repacking of the wastes.

(3) Evaluation of the gamma nuclides content in the drum as above. Placing the waste container inside an over-pack and filling the voids with a water-absorbing material (calcium bentonite clay), which has relatively low swelling characteristics and would therefore not damage the outer container. A conservative estimation of the ^{14}C content in the drums is made in this case, which could not exceed 8.3×10^{10} Bq per drum. This option does not require the waste drums to be opened and therefore no sterilisation is required.

The advantages and the disadvantages of each option have been evaluated against the existing waste acceptance criteria for disposal, cost involvement and impact on the long term safety of the facility.

The treatment of spent liquid scintillation cocktails (Wallac HISafe 3) used for the measurement of ^{241}Am has been evaluated [34]. A large number of ion exchange materials was screened for the removal of ^{241}Am from the scintillation cocktails. The aquatic phase in the cocktails was either 0.2 - 0.4 M $NaNO_3$ or HNO_3. The cocktails were studied as such or mixed with methanol, ethanol or propanol diluents. The tentative test comprised the measurement of the distribution coefficient K_d of the tracer batch-wise in the various liquids for the different ion exchange materials. In general, the americium uptakes were low (K_d < 500 ml/g) for all ion exchangers studied for the HiSafe-cocktail. However, addition of alcohols as a solvent increased the uptake significantly. Especially ethanol produced a very high uptake (K_d > 10’000’000 ml/g) for Sr-Treat material. Further work was carried out with best materials (Sr-Treat-titanate, antimonysilicate, manganese oxide) to optimise the amount of diluents and to conduct column experiments. It was found that the ^{241}Am uptake in the materials increased with increasing dilution until the dilution ratio reached about 1:2.

Columns (10 ml) packed with MnO_2 and $SbSiO_2$ removed ^{241}Am efficiently from diluted HiSafe-liquid with a maximum decontamination factor of 60 - 250. There was no sign of column exhaustion when 2.5 l (250 BV) of liquid had been processed. A total of 8 l of scintillation liquids were purified in the column tests. In addition, americium-bearing nitric acid waste liquids (pH = 1.1) were decontaminated successfully using $SbSiO_2$-columns and the purified acid solution was disposed of in the sewer after neutralisation.

8.5 Beryllium

8.5.1 Characteristics of beryllium waste

Beryllium metal or beryllium oxide (BeO) are used as moderators and as reflectors to enhance the thermal neutron flux densities in the cores of many research reactors. The design and geometry of the beryllium components are specific to the needs of each individual reactor. Depending on the design, different types of cladding of beryllium elements can be used, such as aluminium and zircaloy. In some countries beryllium has been an important metal component in nuclear weapons production and has been used in the fuel rods of special reactors.

Soluble beryllium compounds are toxic if incorporated or absorbed through the skin. Beryllium is accumulated permanently in tissues, with no known self-regulating excretory mechanism. The principal concern is the toxic and carcinogenic effects caused by the inhalation of airborne particulates of beryllium or beryllium compounds, which include inflammation of the respiratory system and lung disease. There is sufficient evidence to classify beryllium and its compounds as carcinogens.

Beryllium is not radioactive in its commonly found form. However, after use in a reactor core it may contain significant quantities of tritium. Activation of impurities within beryllium material can also lead to significant levels of other radio-nuclides, the most important of which is ^{60}Co. This needs to be taken into account in the handling and processing of all beryllium components during facility decommissioning [35].

8.5.2 Processing of beryllium waste

Beryllium oxide may swell when irradiated. Irradiated beryllium metal will become brittle and crack due to the production of helium by neutron irradiation. Owing to these features, beryllium components from reactor decommissioning are not commonly reused.

Current practice for the management of the beryllium metal and beryllium oxide components generated during decommissioning is their interim storage awaiting final disposal. These components are placed in cans and the void between the component and the can is filled with quartz sand, which ensures the mechanical stability of the packaged waste and minimises voids around the component. The cans are placed in an interim storage container.

Natural convection by ventilation will allow cooling of the containers and prevent any tritium build-up within the containers (studies have been performed with the aim of holding the tritium in the waste package).

After intermediate storage, to allow the decay of ^{60}Co to a suitable level and the consequential diffusion of the tritium, the void between this canister and the primary package can be filled with cement and the primary package then closed and stored in an interim storage facility to await final disposal. The behaviour of beryllium and its components in the repository conditions should be considered in the safety analysis of the repository.

It is not recommended to directly immobilise beryllium in cement, as done with many other waste types, since it reacts with water in the basic cementitious matrices, leading to high levels of hydrogen gas generation and volume expansion. Owing to its chemical properties, beryllium requires extensive research and development work to investigate its compatibility with the encapsulation material and to identify the optimal conditions for storage and disposal.

8.6 Sodium and sodium potassium alloy

8.6.1 Characteristics of sodium and sodium potassium alloy

Arisings of sodium or sodium-potassium alloy (NaK) waste in the nuclear industry, including decommissioning activities, are closely associated with the development of the liquid metal fast breeder reactor (LMFBR), in which they are used in the liquid form as a low moderating effect coolant to extract thermal energy from the core. NaK alloy is sometimes preferred to sodium, because it is liquid at ambient temperature. Owing to their unique chemical and physical properties, sodium or NaK alloys are also used in other nuclear fields as a high performance heat transfer medium and oxygen trap. These applications will not be considered here specifically because in terms of the management of the associated sodium waste they are the same as for liquid metal fast breeder reactor activities [35].

8.6.1.1 Bulk sodium and sodium-potassium from main circuits

After draining from the main circuits of the reactor or the facility to be decommissioned, bulk sodium or NaK is considered as a waste. The quantities of such alkali metal waste are generally large, for example 1’600 – 5’500 t for commercial sized liquid metal reactor power plants. This waste constitutes a significant hazard, especially in the course of decommissioning-dismantling activities.

Several sodium cooled reactors also have a relatively small inventory of NaK used in auxiliary circuits.

Smaller quantities of bulk sodium are associated with critical or non-critical research and development facilities; these quantities range from a few kilograms to a few tonnes.

8.6.1.2 Residual coolant in reactors

After draining of the main circuits in the course of decommissioning operations, some coolant can remain trapped inside. The capability of reactor circuits to be completely drained depends on the general concept (loop or pool type) and on the particular design features of the reactor. Depending on the construction of the reactor, the total volume remaining after draining will vary significantly.

8.6.2 Characterisation of sodium waste

The radioactivity of bulk sodium or NaK from primary circuits has several causes:

i) Activation by neutron flux from the core and formation of radio-elements, mainly ^{22}Na and ^{40}K (^{24}Na is also produced by activation, but it rapidly decays due to its short half life).

ii) Contamination by fission products from the fuel due to initial external contamination, fuel cladding failures during operation or use of leaking fuel pins: typically radioisotopes of actinides (plutonium, americium, curium and uranium), cesium and tritium.

iii) Contamination by activated corrosion products from fuel cladding and primary circuit structures, such as ^{54}Mn and ^{60}Co.

The specific activity of radioactive products in the primary coolant is highly dependent on the operational history of the reactor and may vary from 2 to 15 kBq/g after the reactor has been shut down for two to five years. Major contributors to the radiological inventory, as indicated above, are generally ^{137}Cs, ^{22}Na (^{40}K in NaK) and tritium.

Besides the risk associated with the manipulation of radioactive products, nuclear alkali metal waste generates specific risks due to the chemical properties of alkali metals. Sodium and potassium are very reactive. Reactions with water, air and oxygen are generally violent and produce hazardous by-products such as hydrogen and caustic products. Sodium can burn in air at temperatures above 115 - 130°C, but also at lower temperatures, depending on the amount and physical condition of the surface exposed and other factors such as humidity. When finely divided into particles, for example aerosol deposits, sodium can burn at room temperature in humid air. It is therefore recommended to store alkali metal waste under an inert atmosphere and to avoid transfer and manipulation in the open air.

8.6.3 Processing of sodium and sodium-potassium alloy

The potential for recovery and reuse of sodium and NaK is limited. The most likely potential use would be as reactor coolant, but there are currently no liquid metal fast breeder reactors being constructed that would use these alkali metals.

There are two types of bulk alkali metal streams requiring treatment:

a) Alkali metal from the reactor main circuits, for which the difficulties encountered are in general associated with the large quantities involved. As there is a minimal possibility of recycling, it must be transferred into a chemically stable form, with the resulting question of how to handle the large quantities of the resulting reaction products (release to the environment, in accordance with existing authorisations, or conditioning for final disposal).

b) Alkali metal from research and development and test facilities and specific reactor auxiliary fuel storage, which is generally present in smaller quantities compared with reactor circuits but is sometimes much more radioactive and may contain other contaminants. The configuration of these sodium quantities may also present an issue as to the methods of removal from the containers and introduction into the treatment process.

It may be beneficial to reduce the radioactivity level of the sodium by pre-treatment processes, in order to facilitate subsequent treatments and to simplify the required equipment. This could also have a bearing on the final effluent. Tritium and ^{137}Cs are the elements that are generally considered in sodium decontamination processes. In general, ^{137}Cs is the main gamma contaminant and tritium is the main contaminant of the gaseous effluent resulting from sodium treatment.

Bulk sodium oxidation processes to obtain a stable product at ambient conditions are described in detail in [35]. Two continuous processes for bulk alkali metal treatment are the NOAH process and the Argonne process. The NOAH process is based on the injection of small amounts of liquid sodium into a large flow of aqueous sodium hydroxide. The Argonne process is based on a two stage process involving a sodium hydroxide forming process step and a carbonate forming process step.

As indicated in Section 8.6.1, there is a relatively small inventory of NaK in sodium cooled reactors. The most common practice is to dilute this NaK into the bulk sodium before treatment. In this case the small proportion of potassium does not affect the physical and chemical properties of the sodium.

As previously described, following drainage of the primary vessel, there will be some residual alkali metal that cannot be removed without additional engineering work. The treatment of residual sodium or NaK in liquid metal fast breeder reactor main circuits in most cases must be performed in-situ (and sometimes reaction products must be eliminated). This is necessary in order to eliminate the potential hazard due to alkali metal chemical reactivity and associated surveillance requirements, or to allow in-air operations without fire and caustic hazards. This may be also the case when the decommissioning strategy is to keep the circuits or the vessels under a care and maintenance regime for a long period, in order to benefit from radioactivity decay or to defer the dismantling effort.

When NaK systems are drained, the empty system must be maintained under an inert atmosphere in order to prevent the ingress of air, which can potentially form unstable potassium superoxide.

The main treatment process for the residual coolant in reactors is the water vapour nitrogen (WVN) process, which is based on the circulation of a nitrogen carrier gas containing a small proportion of water vapour (1 - 6 vol. %). Care has to be taken with hydrogen production; hydrogen must be safely vented from the reactor circuit.

8.7 Contaminated graphite

8.7.1 Characteristics of contaminated graphite

The radioactive graphite coming from nuclear installations has different characteristics than other radioactive wastes due to its physical and chemical properties and also because of the presence of ^{3}H and ^{14}C. Even after many years of irradiation, graphite retains most of the good mechanical properties and is relatively insoluble and not otherwise particularly chemically reactive. It appears therefore 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 graphite details applied in nuclear reactors show that this graphite cannot be accepted by existing disposal sites without particular conditioning.

Different options have been studied for the management of the radioactive graphite, but the final and generally accepted solutions for its conditioning and/or disposal have not been decided yet. 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:

- 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.

Both disposal options, i.e., near surface repositories and also deep geological formations, have been evaluated and disposal on the seabed was also analysed.

The great majority of the radioactive graphite arising from nuclear power plant decommissioning is associated with the bulk moderator and reflector graphite in these reactors, together with shield-wall graphite (or other carbon-bearing material) in certain cases. In the largest reactor designs (later Magnox, i.e., magnesium non-oxidising reactors) this can amount to over 2’000 tonnes of graphite per reactor.

Permanent moderator and reflector blocks are present in all reactor designs already mentioned in this review, including high temperature reactors (HTR).

In addition, a number of water-moderated reactors including research reactors include graphite reflectors, some of this graphite containing significant quantities of boron.

The moderator and reflector components will mainly consist of large graphite blocks, e.g., 200 mm x 200 mm x 750 mm for the earlier Magnox (magnesium non-oxidising) reactors, through 250 mm x 250 mm x (200, 300, 500 and 600 mm) for RBMK (high power channel reactor) reactors to approximately 460 mm diameter x 900 mm long in the advanced gas-cooled reactors (AGR).

The reflectors in some of the high temperature helium cooled reactors have massive wedge shaped graphite blocks of high-density graphite.

The radioactivity associated with the graphite components arises both from the activation of the initial impurities and from subsequent contamination arising within the reactor circuit. For items such as fuel sleeves that have been pond stored, the inventory may have been modified by the immersion in the aqueous solution. The latter may be as solid materials arising as corrosion products from reactor steel work, etc. or as a consequence of fuel-element failures, or from gas-phase activation (e.g., ^{14}C arising from activation of ^{14}N) followed by subsequent incorporation into the graphite or into carbonaceous material deposited upon it. The reactor atmosphere may also influence the final radioactive inventory in other ways, such as providing a pathway for the removal of ^{3}H (arising both from ^{6}Li in the graphite and from fission events in the fuel) by exchange with gaseous and adsorbed compounds containing inactive hydrogen.

The initial impurities in different types of nuclear graphite differ significantly, and in components such as fuel sleeves, removed from irradiation after relatively short periods, radioactive inventories may be quite different. After longer irradiations, such as are experienced by moderator blocks, more isotopes reach equilibrium specific activities and some with short half-lives may even ‘burn out’ entirely.

A comprehensive analysis of the entire issue of graphite waste disposal is presented. The estimates of residual radioactivity and subsequent decay are, however, based upon the presumption, then current, that the reactors would operate for 40 years at a 70 % load factor and that this would be followed by 10 years ‘storage’ (within the reactor vessels) before the core graphite enters a final disposal route. Sleeve materials, irradiated for much shorter times, are not specifically addressed.

^{14}C, ^{3}H and ^{36}Cl (beta emitters) are the most significant isotopes likely to be present in the graphite from Magnox (magnesium non-oxidising) and advanced gas-cooled reactors (AGR) which need to be considered in terms of possible entry to the food chain, whilst ^{60}Co, ^{94}Nb, ^{152}Eu and ^{154}Eu are the most significant gamma emitters leading to shielding and handling requirements. ^{152}Eu is an exceptional case in that, in a Magnox (magnesium non-oxidising) reactor, most of the activity will remain in the outer reflector region because of burn-out in regions of higher flux.

8.7.2 Processing of contaminated graphite

A basic condition for initiation and progress in graphite parts dismantling or removal from interim storage (silos) and further graphite waste processing is the availability of the disposal option, or as a minimum a clear decision on the final direction of the processed graphite waste. Since such information is in most cases still pending, the most convenient option is safe storage of graphite in existing facilities.

Prior to proceeding to the dismantling and the retrieval of graphite moderator from a reactor or waste storage it is necessary:

- to get a good radiological characterisation of the graphite waste. This characterisation could be obtained through the knowledge of the operation and irradiation history, fuel failures, channel blockages, and significant operational incidents. Moreover, the radiological characterisation needs the knowledge of graphite chemical impurities or contamination at the time where the graphite was installed in the reactor.

  In addition, the radiological characterisation based on calculation has to be controlled by sampling of blocks and validated by experimental radioactivity measurements of radio-nuclides. However, the sampling method must be conducted carefully specially in the case of low temperature irradiated graphite in order to present a sudden release of Wigner energy during the trepanning operation of graphite samples.
- that the disposal route of graphite prior processing (incineration, impregnation, coating, decontamination) is carefully identified and studied, taking into account the practical feasibility and the performance of these operations at a large scale and probably during a long period of time. In some cases, retrievability of disposed graphite waste should be assured.
- that environmental impact studies will be carried out by the organisation responsible for the processing or for the long term disposal storage, and that these studies will be carefully reviewed by the safety authorities taking into account the application of national and international regulatory requirements and recommendations.
- that the radiological impact of the operations related to graphite retrieval, treatment, transportation and storage will comply with the recommendations established by the

International Commission, on Radiological Protection (ICRP) such as the ICRP 60 applicable for both the nuclear workers and the general public using the critical group approach on both the short term and the long term and taking into account all the possible pathways of the contamination to mankind.

8.8 Asbestos

8.8.1 Characteristics of asbestos

Asbestos is the name of a family of naturally occurring minerals that consist of silicates (chemical fragments of silicon and oxygen - a very common component of many minerals) and varying amounts of aluminium, calcium, iron, magnesium, manganese, potassium, and sodium. Each asbestos mineral forms long, thin needle-like fibres. The most common minerals are chrysotile, crocidolite and amosite. Chrysotile is the type of asbestos most widely used within the industry. Amosite is the major type of asbestos used in construction materials within buildings. Further details on asbestos minerals and properties of asbestos are provided in [35].

Nuclear laboratories and other nuclear facilities have long used asbestos cement board where its strength, chemical and heat resistance made it a popular choice for the interior walls and backs of older style fume hoods. It is usually easy to decide if a fume hood contains asbestos cement board. The board has a very hard and smooth, grey, low lustre finish. The board is normally a medium grey colour. Newer fume hoods use plastic or stainless steel panels, and it is easy to distinguish these materials from asbestos cement board. The same asbestos board was sometimes used as a counter top in laboratories or workshops to protect the cabinet from chemicals or heat.

Asbestos cement tiles were used until the late 1960's on building exteriors. The tiles come in various sizes but again, the tile is very hard and it sometimes has a textured surface. Because they are usually painted, colour is not usually helpful in identifying asbestos cement tiles. Occasionally, asbestos mixed with cement to form asbestos mortar was used as an exterior decorative finish on buildings. The mixture was trowelled or sprayed to form a continuous textured surface.

Heating pipes are usually covered with pipe insulation to prevent heat loss. In buildings, glass fibre insulation normally insulates straight runs of heating pipe. However, asbestos is often present at elbows and around joints and valves. Both fibreglass and asbestos pipe insulation were invariably covered with a paper or a fabric covering. In turn, the covering is often painted. It is fairly easy to assess whether pipe insulation is fibreglass or asbestos. Fibreglass pipe insulation is fairly pliable and 'gives' slightly when squeezed. In contrast, the asbestos insulation, which is applied to elbows and joints, is very firm.

Reactor pressure vessels and the reactor containment building were commonly insulated with material containing asbestos. For example, the asbestos may have been tied to the pressure vessel by an outer covering of chicken wire, which was fixed to tangs on the pressure vessel exterior.

Asbestos is only a concern when the fibres become airborne, because it is only when the fibres are present in the air that people can inhale them. Although there have been concerns raised about taking in asbestos fibres in food or water, there is no evidence that ingestion exposure to asbestos poses any risk. Neither is skin contact a concern. Some types of asbestos can cause skin irritation. But skin contact is not a direct cause of serious illness. Only inhalation of asbestos fibre presents a health hazard. If asbestos fibre is loose or can be crushed by hand, the risk is high that the fibre can get in the air. In such cases, a serious health risk might be present particularly if a person were to be exposed to a lot of dust or be exposed over a period of years. Asbestos fibres are extremely small and can remain airborne for an extended period of time. Of the asbestos fibres that can be inhaled, the most hazardous fibres are about 5 to 8 microns in length and about 1.5 microns in diameter.

Asbestos that can be easily broken or crushed is called friable asbestos. In contrast, when asbestos is present in products where the fibres are effectively bound, there is normally no hazard. However, drilling or cutting can free fibres even from these bound products. Under such situations, even tightly bonded asbestos-containing materials can present a hazard.

The human respiratory system has evolved ways to deal with inhaled dusts. The body can remove the large dust particles that become trapped in the upper parts of the respiratory system. And in addition, the body has other ways to capture and remove the very finest dust particles that can reach well down into the lungs. Unfortunately, the body's systems that clear dust from deep within the lungs do not work very well with asbestos. This poor performance of the deep lung dust clearing systems may be partially responsible for the illnesses that asbestos fibres cause.

The most significant medical implications that can be caused by the inhalation of asbestos fibres are asbestosis, lung cancer, mesothelioma and other cancers.

8.8.2 Processing of asbestos

Because of its highly dangerous nature and its specific physical/chemical properties there are no possibilities for recovery or reuse of asbestos. In many countries the use of asbestos is now forbidden.

The removal of contaminated asbestos from piping, vessels, walls, etc. has to be done by an authorised specialised firm. Preference will be given to companies that have experience of working in nuclear installations. For most other staff, observing the following rules will minimise the chances of a significant asbestos exposure:

- Do not work with loose asbestos without training, authorisation and without taking precautions to prevent release of fibres that might expose you or others to airborne fibres. Use appropriate personal protective equipment.
- Do not damage or remove any pipe wrapping without authorisation from the supervisor.
- Do not cut, sand, drill or break products that contain asbestos including asbestos cement insulation, asbestos cement board, asbestos cement tiles or vinyl asbestos tiles.
- To clean-up in areas where asbestos dust may be present, use a wet mop to prevent the fibre from becoming airborne or use a high efficiency particulate air (HEPA) vacuum cleaner. High efficiency particulate air filters are high efficiency filters that trap even the fine particles that pass through the filter of a normal vacuum.
- Exercise care in areas where asbestos is (or is likely to be) present including:
    * boiler or furnace rooms, around boilers, furnaces or heating pipes or other heating equipment;
    * above the drop ceiling or on the steel beams;
    * in electrical workshops, laboratory fume hoods or walls.
- If you unexpectedly encounter asbestos or an asbestos containing material, avoid any work that could create an airborne dust. Report the situation immediately to the supervisor.

The safety precautions to be observed by fully trained outside contractors for asbestos removal are more stringent than for the decommissioning of a nuclear facility. The site will have to be completely isolated from the rest of the building; the ventilation system will have to be equipped with supplementary pre-filters and absolute filters; an encapsulant coating may also have to be used to contain any fibres: the personnel and material air locks will have to be equipped with showers to avoid any transfer of asbestos fibres in the rest of the building.

Workers carrying out asbestos removal must observe many rules to ensure their own safety and that of other building occupants. Among the rules is the wearing of both protective clothing and respirators. Whenever possible, the asbestos is pre-treated with water so the work produces less dust. The removal of the contaminated insulation greatly improves the work environment for further decommissioning work; i.e., the respirator should only be requested when decommissioning works in the presence of asbestos are performed.

After removal, the contractor places the asbestos and any contaminated material in sealed heavy-duty, labelled plastic bags. Sometimes there is a need to store waste asbestos for a time until it can be shipped to a landfill site approved by the authorities to accept asbestos waste. Radioactive asbestos is treated as radioactive waste and in some cases is super-compacted along with other inorganic wastes to produce a waste product for disposal.

'Wrap and cut' refers to a method of asbestos abatement. This method is used when a building or facility component, such as a length of piping with asbestos containing material on it, is first wrapped in plastic sheeting. The entire wrapped component is then removed from the building. Sometimes asbestos containing material must be removed from part of the component to free it from the items around it. This method can also be used to wrap components with asbestos containing materials inside them. This method can be used instead of full containment procedures only when the asbestos containing material in or on the component is in good condition. Other restrictions for this process also apply.

Although vitrification of asbestos is safer than incorporation in concrete, the latter is industrially used for asbestos originated in non-nuclear applications. Generally asbestos originated by decommissioning of nuclear facilities is only slightly contaminated and is incorporated in concrete after compaction. However, if necessary, high temperature technologies are available for the vitrification of highly contaminated asbestos. The choice between solidification with cement and vitrification is dependant on the amount of asbestos to be solidified.

Acidic decomposition of asbestos to an amorphous silica suspension has also been considered as a method for treating asbestos prior to disposal. This allows for the complete destruction of the asbestos fibres. The amorphous silica solution is solidified by the addition of lime, sodium silicate, and possibly other reagents in preparation for disposal.

A mineral conversion process has been developed which changes asbestos to stable, non-hazardous minerals at temperatures significantly below those required for melting. The asbestos minerals, chrysotile and the amphiboles, are known to convert to other mineral phases, such as pyroxines and olivines, at high temperatures. In the patented conversion process, known as Asbestos Conversion Process, proprietary chemical conversion agents are added to the asbestos containing material. These additives greatly increase the rate and completeness of conversion. Sintering agents then cause the fine material particles to combine into hard, durable masses which are free of fibres.

This conversion process has been tested on a variety of asbestos containing materials from actual abatement sites. A commercially viable transportable production system is presently in operation, capable of successfully processing tons per day. A volume reduction by a factor of ten is achieved.

In addition to this asbestos conversion process, a number of high temperature technologies are available for transformation of radioactively contaminated asbestos to the form most appropriate for safe storage and subsequent disposal. For instance, traditional furnaces with direct-arc heating and induction melters could be used to obtain products with a leachability rate of around 10^{-7} g.cm^{-2} per day for strontium and cesium [35].

8.9 Cadmium

8.9.1 Characteristics of cadmium waste

Because of its high thermal neutron cross section cadmium (Cd) is used in reactor control rods to control the criticality. There are three functional classes of control rod: regulating rods, used for fine adjustments; shim rods, used for coarse adjustments; and safety or scram rods, used for rapid or emergency reactor shutdown. Because of its low melting point and soft nature cadmium was mainly used in research reactors in the early years. Pure cadmium is usually clad in stainless steel or aluminium.

In power reactor applications cadmium is alloyed with elements such as silver and indium, which are more effective neutron absorbers in the intermediate and higher neutron energy ranges.

Cadmium is also used in spent fuel racks to increase the spent fuel storage capacity. Cadmium plate is sandwiched between two stainless steel tubes surrounding the fuel assembly [35].

8.9.2 Processing of cadmium waste

Owing to the nature of the use of cadmium in reactor control rods, and to the comparatively small volume of cadmium components, there is no economic need and little opportunity for recovery or reuse of this material. The cadmium and associated cladding will be highly activated and in most cases will require disposal as intermediate level radioactive waste.

The cadmium used in spent fuel racks may be suitable for reuse or recycling, depending on its specific activity.

During decommissioning of cadmium-containing equipment, attempts should be made to avoid exposing workers to the cadmium. Therefore, contrary to how work is typically performed, cadmium-containing equipment should not be cut or sheared unless this is unavoidably necessary. Leaving the cadmium contained within its stainless steel or aluminium cladding is preferable. If control rods are broken, special attention must be paid to avoiding contact of the workers with the cadmium contamination.

The cadmium control rods or cadmium-containing component can be directly immobilised in a cementitious matrix and stored pending final disposal. The behaviour of cadmium and its components in the repository disposal conditions is considered in the safety analysis of the repository [35].

8.10 Mercury

8.10.1 Characteristics of mercury waste

Liquid mercury (Hg) was used as a coolant for early experimental fast reactors. The advantages of Hg relative to other liquid metal reactor coolants were its low melting point (minus 38.8ºC) and the fact that it is chemically inert to air and water. The disadvantages of using liquid Hg were its toxicity and high density (hence heavy pumping load) [35].

Mercury has also been used:

- as a liquid seal for rotating shields above the core of fast reactors;
- for radiological shielding;
- as a catalyst in the dissolution of uranium alloy fuels; and
- in lithium isotope separation [35].

8.10.2 Processing of mercury waste

Contaminated mercury can be recovered and reused through a process of distillation. The principle of operation is to distil the mercury under vacuum, thereby reducing the boiling temperature from 356°C to about 200°C. This allows the distillation rig to operate with lower power consumption and reduces the need for thermal lagging. The rig is designed to allow the vacuum pump to be removed once the rig is fully primed. The rig is effectively a single piece of glass, constructed from about ten elements, to avoid the need for greased joints, since the operating temperature of the rig would render such joints useless for maintenance of a vacuum.

The operating principles of the distillation rig require that the mercury be free from high levels of volatile material, since this would degas from the metal in the first boiling chamber and pressurize the chamber. Distillation will remove the contamination from the mercury and leave clean mercury for reuse or clearance.

Activated mercury cannot be treated in this way due to its radioactive mercury isotopes, which cannot be separated during distillation. It is treated and disposed of as radioactive waste.

If distillation is not applicable, mercury has to be immobilised. Most of the information available on the immobilisation/stabilisation of mercury arises from studies and trials carried out in the USA. The methods employed are varied and deal with all types of mercury waste, from the treatment of metallic mercury to the immobilisation of aqueous solutions of mercury compounds.

Some technologies for the immobilisation of metallic mercury and mercury solutions are described below (these have not been demonstrated on the industrial scale).

An amalgam is defined as an alloy of mercury and at least one other metal, such as copper, zinc, nickel or silver. In general, an amalgam is formed when metal powder is mixed with liquid mercury. The process is fairly simple, and involves the mixing of elemental mercury with an amalgamating material.

The mixture is then agitated/stirred in batches for a prescribed period of time. The most common metal used for amalgamation for the purposes of waste management is copper, principally because of its relatively low cost. Zinc is also used to form an amalgam, although this amalgam can be attacked by water, forming hydrogen and zinc hydroxide [35]. Investigations have been undertaken on the direct amalgamation of mercury with copper in the form of a fine powder. It has been found that a copper/mercury ratio of 3:1 (by weight) is required to produce a satisfactory amalgam at room temperature, with a mixing time of 40 min. Other trials have investigated pre-cleaning of the copper to remove the non-reactive oxide layer prior to mixing with the mercury.

The mixing times and mercury waste loading of the amalgam using pre-cleaned copper have been investigated. Two methods of cleaning have been identified:

(a) Cleaning with a solution of ammonium chloride and hydrochloric acid;

(b) Heated hydrogen gas at temperatures higher than 600°C.

It is claimed that mixing times can be reduced to three to five minutes and the copper/mercury ratio reduced to 1:1 (by weight). Waste loadings of up to 85 wt % have been achieved.

Amalgamation has the additional benefit of producing a solid waste product requiring no further stabilisation. Amalgamation is the recommended technology in the USA for the immobilisation of radioactive mercury waste.

Mercury sulphide is exceedingly insoluble in water and un-reactive chemically, being attacked only by concentrated HBr, HI or 'aqua regia'2. Therefore, mercury sulphide formation by the reaction: Hg(1) + S(S) → HgS(α or β), could be used for mercury immobilisation. This is a relatively simple process involving the mechanical blending of mercury with sulphur. Although sulphur is not a metal, this process is categorised in the USA as amalgamation. Mercury (II) sulphide exists in two forms, a red hexagonal α-HgS (cinnabar) form and a black β-HgS (metacinnabar) form. Some investigators have evaluated a high shear blending technique for the production of both cinnabar and metacinnabar mercury sulphide. Zinc amalgam has also been produced as a means of comparison [35]. It has been found that metacinnabar (black) is produced at low mixing speeds (< 1000 rpm) and cinnabar (red) at high shear (mixing speed of 19’000 rpm) and high temperature (270 - 300°C). It has therefore been concluded that the production of HgS via a low shear process (metacinnabar) is preferred to a high shear process (cinnabar), as it is safer and easier.

The sulphur polymer stabilisation method is used for the stabilisation of mercury contaminated mixed waste and uses heated sulphur polymer cement and small amounts of unspecified additives to convert metallic mercury into HgS. The thermoplastic mixture is melted and poured into a mould with the mixed waste to be immobilised, where it cools and solidifies. The process is claimed to be very effective and is used on a commercial scale in the USA [35]. The process has a further stage, in which the solidified waste is subsequently encapsulated within a casing of clean sulphur polymer cement (i.e., a form of macro-encapsulation). The process has not been proven for the immobilisation of elemental mercury. The process is costly and substantially increases the volume of the final waste form.

Immobilisation in Portland cement has been undertaken in which Portland cement was doped with 10 wt % aqueous mercuric nitrate - $Hg(NO_3)_2$ [35]. It is believed that the final chemical state of the dopant is mercuric oxide (HgO). The mercury was added to the grout mix in the form of 10 wt % aqueous $Hg(NO_3)_2$. The grout samples were left to set over a 28 day period at room temperature and a relative humidity of 85 %. The waste product was then analysed via X-ray photoelectric spectroscopy to identify the oxidation state and the chemical form of the mercury in the Portland cement, which showed it to be mercury oxide. The study was concerned solely with the form of the mercury in the final waste form. However, no data are available on the viability of the final waste form. As the mercury is immobilised in an aqueous form, the potential for metallic mercury to be present within the waste form is greatly reduced.

The study concluded that:

a) HgO was the final state of the mercury ion contaminant;

b) The oxidation state of the mercury was 2+;

c) There was no chemical bonding of the mercury to the cement components.

8.11 Lead

8.11.1 Characteristics of lead waste

Lead (Pb) is widely used in nuclear facilities as a shielding material in the form of bricks, sheets, wool or lead shot. The physical form of the lead is dependent on the nature of its use. In addition, lead-based paints and primers have been used during construction of many facilities. Initially, the toxic properties of lead were not fully understood and as a result, lead became widely used and the various sources of lead are generally not enclosed, encapsulated, or labelled.

In nuclear facilities, lead is predominantly contaminated rather than activated [35].

8.11.2 Processing of lead waste

There are many possibilities for the recovery and reuse of lead inside and outside of the nuclear industry. For reuse outside of the nuclear industry decontamination of the lead is generally required. There are many different methods that can be employed for this purpose. The method used is dependent on the form of the lead (i.e., mechanical methods are preferred for lead blocks) and the form of the contamination. The methods include mechanical and non-mechanical means such as:

a) Shaving;

b) Sand blasting;

c) Grinding;

d) Chemical methods;

e) Lead melting.

Mechanical treatment methods involve the physical removal of a layer of the lead surface and the associated contamination. This can be done by shaving or planning, by sand blasting or by grinding. However, sand blasting can have the effect of pushing the contamination further into the lead block.

Chemical methods are also applicable for the decontamination of lead; for example, the DECOFOR process, based on a patented formic acid process, can be used for chemically decontaminating lead to clearance levels.

The reuse of lead in nuclear facilities is encouraged, especially if the waste is activated and the clearance of this material is difficult to demonstrate. As indicated above, it is difficult to confirm whether contamination has been incorporated into the matrix, because of the density of the lead. The history of the previous use of the lead must be known in order to ensure that the contamination is only a surface effect, otherwise samples of the lead need to be taken to ensure that it meets the clearance levels.

If lead cannot be decontaminated for recovery and reuse, it is disposed of as radioactive waste. As a solid, it can easily be incorporated into a cement matrix for disposal or used as shielding in a waste container. The total quantities of lead in the repository are considered as part of the repository post-closure safety case.

8.12 Cyanide

8.12.1 Characteristics of cyanide waste

Cyanide is a carbon-nitrogen chemical unit, which combines with many organic and inorganic compounds. Different cyanide containing materials are quite often used in the nuclear industry, in particular in waste management practices. Cyanides are important components of different inorganic absorbents used for the selective removal of cesium from different liquid waste streams.

In the nuclear industry, cyanides are mainly found in stored sludges or liquids arising at research and development facilities. In some cases, cyanide containing liquids and sludges have been cemented by the waste producer and have been stored on-site for a long time. Reprocessing tests, metal-plating operations, etc. have generated wastes contaminated with cyanide. Generally they are mixed with other chemicals in complex wastes that must be carefully characterised prior to treatment for storage and disposal

Cyanides can also be found in ion exchange resin media. They occur as insoluble transition metal hexacyanoferrates. These have been used for decades for the removal of radioactive cesium and other radio-nuclides from solutions. Any decommissioning activity of a research reactor complex could have to deal with these types of ion exchange media. The issues associated with the management of spent ion-exchange resins has been discussed in Sub-section 8.3. The cyanide complexes should be stable during interim storage and processing (if stored in controlled conditions, e.g., non-oxidising media), but degradation of these resins is possible in the deep geological disposal environment with the liberation of free cyanide [35].

8.12.2 Processing of cyanide waste

Owing to the chemical complexity of cyanide-containing waste, the possibility for recovery and reuse is excluded.

Alkaline chlorination is the most widely used method for the destruction of cyanides and other chemical wastes. This process, which has been in commercial use for over 35 years, is suitable for destroying free dissolved hydrogen cyanide and for oxidising all simple and some complex inorganic cyanides in aqueous media. The cyanide in very stable complexes such as ferrocyanides or ferricyanides is basically unaffected by chlorination. Cyanide that is complexed with copper, nickel and precious metals is amenable to chlorination but reacts more slowly than free cyanide and therefore requires excess chlorine for efficient cyanide destruction. If properly designed, maintained and operated (good pH and oxidation-reduction potential control), the process will oxidise cyanides that are amenable to chlorination.

Traditionally this process involves successive oxidation of cyanides to cyanates, and then to nitrogen and non-toxic chlorides and carbonates by chlorine gas or sodium hypochlorite solution under high pH conditions (to avoid the formation of toxic CNCl or HCN gas):

$Cl_2 + 2NaOH \rightarrow NaOCl + NaCl + H_2O$

$NaCN + NaOCl \rightarrow NaCl + NaOCN$

$2NaOCN + 2NaOCl + 2NaOH \rightarrow N_2 + 2Na_2CO_3 + 2NaCl + H_2$

Excess hypochlorite is neutralised by adding dry sodium thiosulphate in stoichiometric amounts. The final pH is adjusted to neutral (pH 6 - 7).

Further chemical treatment is not necessary before the waste is discharged (assuming other waste types are not present). Any solids recovered are immobilised in a drum using an appropriate solidification process. The treated waste solution may be solidified in cement and land filled or sent for storage or disposal as radioactive waste.

The advantages of direct alkaline chlorination for treating process effluents are its relative simplicity, high efficiency (approaching 100 %), operation at ambient temperature (except for complexed cyanides), suitability for automatic control and low cost.

An obvious disadvantage of the above described method is that it requires the purchase and storage of large quantities of hazardous chlorine gas or hypochlorite solutions. However, the risks associated with these chlorine sources can be essentially reduced or fully excluded by the employment of innovative approaches and technologies.

8.13 Polychlorinated biphenyls

8.13.1 Characteristics of polychlorinated biphenyls

A commonly realised problem is the appearance of polychlorinated biphenyls (PCB) due to its superior technical properties as components in some organic materials. These were used in technical installations including old nuclear facilities. When used in controlled areas they can be radioactively contaminated [35].

The following list gives an impression about the materials/equipment where polychlorinated biphenyls have been used:

- Electrical transformer oil;
- Oil impregnated electrical cable insulation;
- Hydraulic oil;
- Machine cutting oil;
- Lubricant;
- Epoxy paint;
- Impregnation;
- Additives;
- Dielectric in capacitors.

When characterising a facility for decommissioning, possible polychlorinated biphenyl contamination has to be given careful consideration [35].

8.13.2 Processing of polychlorinated biphenyls

The inherent hazards associated with this class of compounds do not allow the future reuse of polychlorinated biphenyl-containing material. In many countries the production and use of polychlorinated biphenyl is now forbidden.

During decommissioning the initial removal of polychlorinated biphenyl-containing material and its processing are the biggest problems with respect to the personal protection of workers from exposure to these types of materials. Once polychlorinated biphenyl-containing waste is removed, it is recommended to incinerate it in an appropriate incinerator at temperatures exceeding 1’200°C. Destruction by incineration is the preferred treatment method; however, if incinerated improperly, another toxic component, dioxins, can be formed.

An advanced process for polychlorinated biphenyl treatment has been developed [35]. This powerful process, based on a low temperature, atmospheric pressure reaction between a chlorinated contaminant and finely dispersed metallic sodium, is equally efficient in treating low level polychlorinated biphenyl contaminated oils as it is in treating decontamination liquids and solid waste containing high levels of polychlorinated biphenyls. The toxic contaminant is converted to $NaCl$, $NaOH$ and an organic byproduct (polyphenols).

If an appropriate incinerator is not available, polychlorinated biphenyl-containing waste must be embedded in an inert matrix such as cement and isolated from the environment. The way to achieve this depends on the technical and physical properties of the waste.

8.14 Transport, storage and disposal of conditioned waste

Prior to transportation and disposal, waste arisings require packaging and conditioning into package/containers of adequate sizes. In general, increasing container size offers the potential to minimise the size reduction needed for solid waste with reduction in operational decommissioning costs, secondary waste generation, and dose uptake. Limitations to size increase are dictated by transportation on road or railways and movement of packages within interim store or final repository. Temporary packaging may be employed to allow component transfer to the disposal container. Usually, the cutting of contaminated items will be optimised to accommodate pieces into available waste containers.

Existence of temporary storage for packaged waste is necessary to have some flexibility in the transfer to the repository and even more when a disposal route does not exist. In this last case the interim storage facility must be capable of accommodating all waste generated during decommissioning. In a storage facility, provisions should be made for separate storage of waste with different activities, varying exposure rates or different radionuclide content. Under storage conditions, the waste packages must be clearly identifiable and must be protected against potentially adverse atmospheric conditions.

Waste material which can be exempted from regulatory concern can be disposed of according to national regulations to, for example, a sanitary landfill. Radioactive waste falling into a low level category can go to a licensed near surface disposal facility, if accepted. However, because part of the waste generated during decommissioning of non-reactor nuclear facilities contains long lived alpha emitters (in particular plutonium) and, also in the case of reprocessing plants contains fission products, depending on national circumstances, deep geological disposal of these contaminated materials has to be considered.

Discussion of regulations for the transport of radioactive materials is outside the scope of this document. However, it is important to recognise that these regulations are demanding in terms of both cost and the time taken to implement these. Therefore, an additional monetary benefit derived from waste minimisation is the concomitant reduction in the costs of transporting radioactive waste from the site of arising to the site of disposal.

The methods of 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 ratio for accomplishing the disposal. The type and specific activity of the radioactive material present in the wastes are the two most important factors used in selecting the storage and disposal method. Other important factors are the size of the package and the difficulty in handling the package during disposal.

The principal methods of disposal are near surface disposal and emplacement in rock cavities or repositories within deep geological formations. Near surface disposal is generally employed for low and intermediate level radioactive wastes. Rock cavities can potentially be used for all kinds of solid, low and intermediate level waste. Disposal in deep geological formations is envisaged for high level wastes having significant quantities of long lived radio-nuclides.

The choice of disposal method is dependent on the conditions prevailing in the country and on many other factors specific to the disposal system to be developed. Generally, near surface disposal and rock cavity concepts appear to be the most viable for disposal of decommissioning waste [35].

9. Conclusions and recommendations

This document provides a comprehensive overview of the current status and available experiences relating to the management of materials from the decommissioning of nuclear installations, including the issues characterisation of materials and possible ways to release materials into the environment. The document has been divided into five main sections which may be summarised as follows:

- *The management of materials arising during the decommissioning process*

  The main characteristics of materials resulting from the decommissioning of various nuclear facilities covering the whole nuclear fuel cycle (refining and conversion, enrichment, fuel fabrication, nuclear power plant operation and decommissioning and spent fuel reprocessing) have been discussed. The possible strategies and the potential approaches for the final disposition of the decommissioning materials including factors, constraints and waste minimisation principles that may influence the decision process in this field have been considered. In addition, the most commonly used procedures and technologies for radioactive waste processing, i.e., pre-treatment (including decontamination), treatment, conditioning of wastes and possibilities for storage and final disposal at suitable repositories have been characterised.

- *The clearance process*

  The methodology and the main characteristics of the clearance process relating to materials from decommissioning have been described. The analysis has been directed to a description of detection devices used and material monitoring systems. Scaling factors, statistical background for material activity evaluations, and quality systems in clearance have been considered as well. The issues described may lead to the demonstration of compliance with clearance levels for the given way(s) of release from regulatory control into the environment.

- *The organisation of material management and information systems*

  The requirements to have an effective on-site organisational system for material management including identification, processing, administration, retention of important and relevant design and operational data necessary for the decommissioning process, and management of new records from the decommissioning activities are the main issues considered in this section. In addition, the main features and requirements for radioactive material and waste management information and inventory record keeping systems have been discussed.

- *Radioactive waste characterisation process*

  The common used techniques, methods, instruments for waste characterisation and for identification of the important radiological and non-radiological characteristics of radioactive materials and wastes have been described. Challenges and problems relating to monitoring and characterisation of difficult to measure and historical wastes have been discussed as well. An overview of important and radiologically significant radionuclides in terms of final disposal has also been given.

- *The management of special wastes from decommissioning*

  The characterisation of physical, chemical and toxic attributes of special waste types has been addressed, e.g., concentrates, sludge, ion exchange resins, beryllium, sodium, graphite, asbestos, mercury and lead. The methodology and the technologies leading to safe processing and disposal at radioactive waste or conventional waste repositories has been discussed for each type of these special wastes.

Based on the reviewed state-of-the-art in the area of decommissioning material management, it should be emphasised and recommended that:

- Before starting decommissioning activities, it is necessary to analyse and evaluate the possible material management scenarios with emphasis on final disposition options: release of materials, waste disposal in various types of repositories, …
- Advanced analytical approaches applied to model and evaluate the material and waste management scenarios, could increase the predictability of the real process, and improve and make the decision making process easier in the period of preparation of nuclear facility decommissioning activities.
- Precise characterisation of radioactive materials and wastes in the pre-treatment phase is one of the most important issues and should simplify the further process and lead to a reduction of the amount of materials that should be disposed in radioactive waste repositories.
- Proper definition and application of limits and conditions for the clearance of materials from decommissioning could increase the volume of releasable materials, reduce the capacity requirements for repositories and save financial means needed for waste processing and final disposal.
- Conditional clearance (release) could generate new possibilities for recycling and reuse of originally non-releasable materials. Scenarios for further use of materials outside controlled areas should be developed, evaluated and strictly adhered.
- A well-developed material management organisation, information and record keeping system is necessary for proper controlling and managing the process and for avoiding unexpected situations during the process.
- In addition to the radiological aspects, processing of special wastes from decommissioning should also consider the toxic characteristics of the materials that could involve risks to the personnel and to the environment.

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Annex 1

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