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Disposal Aspects of Low and Intermediate Level Decommissioning Waste

Results of a coordinated research project 2002–2006

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FOREWORD

The decommissioning of nuclear facilities is a subject of growing importance in many IAEA Member States, due to the large number of facilities that have attained or soon will attain the end of their service life. As a result of decommissioning operations, a wide range and quantity of radioactive materials will need to be managed. Some of these materials may be recycled or reused if they continue to have an economic value, but most must be managed as radioactive waste. Thus, the development and implementation of appropriate strategies for processing and disposal of decommissioning waste has become an important issue.

In recognition of the importance of this subject, the IAEA decided to conduct this Coordinated Research Project on Disposal Aspects of Low and Intermediate Level Decommissioning Waste. Based on waste category data and their inventories, the specific objectives of this project were to outline appropriate strategies for decommissioning waste disposal, to assess the performance of the typical waste streams arising during decommissioning of nuclear facilities, to promote R&D activities relevant to the disposal of waste derived from decommissioning activities and to exchange and discuss information available on this topic with the countries participating in this project. The goal of the project is to contribute to a better understanding of decommissioning waste, its behaviour and its influence on the design and performance of appropriate disposal facilities.

The IAEA wishes to express its appreciation to the international experts who took part in this project, as well as to all those who contributed to the preparation of this publication. The IAEA officers responsible for this publication were L. Nachmilner and R. Dayal of the Division of Nuclear Fuel Cycle and Waste Technology.

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SUMMARY

This publication presents the result of a five-year study performed within the Coordinated Research Project (CRP) on Disposal Aspects of Low and Intermediate Level Decommissioning Waste, in which institutions from fourteen Member States assessed the link between decommissioning processes and waste disposal issues. The following countries have taken part in this research project: Argentina, Canada, China, Germany, Hungary, India, the Republic of Korea, Lithuania, Russian Federation, Slovakia, Sweden, Ukraine and the USA. In addition, the UK has participated as an observer.

The problems associated with decommissioning waste disposal are rather variable; thus only certain selected topics could be covered regarding managing different kinds of radioactive waste arising during the decommissioning of nuclear facilities. In particular, strategic and technical planning, facility-specific considerations of fuel cycle facilities and research and energetic reactors, performance and safety assessments of disposal facilities, and their conceptual design were among the selected topics. Emphasis was also placed on the specification of waste characteristics and formulation of appropriate strategies for their management.

Access to this information may be of use to those countries preparing and initiating radioactive waste disposal projects, namely those for whom guidelines and/or requirements for waste conditioning and disposal have yet to be prepared. These countries may benefit from existing knowledge and experience and use it, at their own discretion, as a reference for their national programmes.

1. INTRODUCTION

1.1. Background

Decommissioning is the final phase in the life-cycle of a nuclear facility after siting, design, construction, commissioning and operation. It is a process that involves decontamination, dismantling of plant equipment and facilities, demolition of buildings and structures, and management of the resulting materials. The decommissioning of nuclear facilities has become a topic of great interest to many Member States of the IAEA due to the large number of facilities that have approached or are approaching the end of their operational lifetime.

Nuclear facilities are predominantly comprised of large facilities such as nuclear power plants, fuel processing and reprocessing plants including their associated nuclear chemical facilities, relatively large prototype, research and test reactors, and waste storage facilities. However, a much larger number of facilities of smaller size and complexity are also to be decommissioned at the end of their service life. These facilities are located within research establishments, biological and medical laboratories, universities, medical centres, and industrial and manufacturing premises. As a result of the decommissioning of these facilities, a large and diverse range of radioactive waste materials will need to be managed.

The handling, treatment, conditioning, storage, transport and disposal of radioactive waste are generally well regulated within the nuclear industry. There are numerous publications available that provide guidance, for example, on the conditioning of waste for storage and disposal. The methods applied to decommissioning waste are, in general, similar to those used in other parts of the nuclear industry during the operation, management and refurbishment of facilities. However, decommissioning waste originating from smaller nuclear facilities may differ in size and chemical and radiochemical composition from operational waste, and may require different handling, processing and disposal techniques.

This publication focuses globally on disposal aspects of decommissioning radioactive waste, and addresses issues that may influence the planning, design and performance of a suitable disposal facility that need to be known and understood.

1.2. Objective

Radioactive waste originating from decommissioning of nuclear facilities and installations needs to be managed in a manner that is compatible with internationally recognized principles and standards. The conditioning and disposal methods chosen should, in particular, be commensurate with the specific characteristics of this type of waste. Currently, repository sites, both in near-surface and geological formations, are already in use or are planned to be used in a number of IAEA Member States that represent suitable disposal options.

To address the specific needs of national decommissioning programmes, the IAEA decided to conduct the Coordinated Research Project on Disposal Aspects of Low and Intermediate Level Decommissioning Waste. Information and experiences obtained from past and on-going decommissioning projects in certain countries were compiled and evaluated. The following countries have taken part in this research project: Argentina, Canada, China, Germany, Hungary, India, Republic of Korea, Lithuania, Russian Federation, Slovakia, Sweden, Ukraine and the USA. In addition, the UK has participated as an observer.

The objective of this publication is to provide information on certain national approaches in managing different kinds of radioactive waste arising during the decommissioning of nuclear facilities. Emphasis was placed on the specification of waste characteristics and formulation of appropriate strategies for their management. In addition, the connection between waste, safety assessment and repository design was highlighted and discussed. This information is particularly important for the planning and design of disposal facilities for decommissioned waste.

Access to this information may be of use to those countries preparing and initiating radioactive waste disposal projects, namely those for whom guidelines and/or requirements for waste conditioning and disposal have yet to be prepared. These countries may benefit from existing knowledge and experience and use it, at their own discretion, as a reference for their national programmes.

It should be emphasised that the aim of this publication is to assist; the statements given are not intended to be prescriptive.

1.3. Scope

This publication consists of a general introduction to issues associated with decommissioning waste disposal and includes annexes that summarize the experience gained and approaches proposed in participating countries.

Section 2 of this publication provides a description of typical waste arising from the decommissioning programmes of various participating countries. Based on this, the behaviour of waste materials and packages under repository conditions are described in Section 3. Section 4 identifies factors of relevance to repository design and safety assessments. Section 5 presents an overview of available disposal systems. Section 6 contains salient features of the work performed by various CRP participants; their detailed contributions are provided in Appendixes.

2. DESCRIPTION OF DECOMMISSIONING WASTE

2.1. Sources of waste

When a nuclear facility is permanently shut down, a transition phase from operation to decommissioning begins. During the transition phase, depending on the type of facility, fuel is removed from the core, and liquids and other operational waste are taken out from all systems. In fuel cycle facilities, the nuclear materials being processed or generated are removed from the installation. Decommissioning itself starts after this transition phase is completed.

This publication analyzes management aspects — from generation to disposal — of low and intermediate level waste resulting from decommissioning of nuclear facilities in the various participating countries; spent fuel and HLW disposal is not addressed. Decommissioning waste generation depends, among others, on plant size and design, construction materials used, operational history, and activities performed. In addition, management of this waste is affected by national legal and regulatory frameworks, including clearance limits, as well as national radioactive waste management strategies considering disposal options for different waste categories.

2.1.1. Nuclear power plants

Table 1 identifies the number of nuclear power reactors that are currently operating, shut down or under decommissioning, and under construction, in those countries represented in this CRP (participant countries) [1].The reactors identified in Table 1 include light water reactors (LWR), heavy water reactors (HWR), gas cooled reactors (GCR), high temperature reactors (HTR) and fast breeder reactors (FBR). It should be noted that the current tendency is to extend operating life of nuclear power reactors, which may defer decommissioning, but will not preclude it.

Table 1. Nuclear power reactors in participant countries

Country	NUCLEAR POWER REACTORS		
	In operation	Shut down or under decommissioning	Under construction
Argentina	2	0	1
Canada	18	7	0
China	10	0	3
Germany	17	19	0
Hungary	4	0	0
India	16	0	7
Korea, Rep.	20	0	0
Lithuania	1	1	0
Russian Fed.	31	5	4
Slovakia	6	1	0
Sweden	10	3	0
UK	23	22	0
Ukraine	15	4	2
USA	103	29	0
Total	**276**	**91**	**17**

2.1.2. Research reactors

A research reactor is a generic name for a wide range of non-power producing reactors. It includes large plants for radioisotope production or materials irradiation, with power outputs in the range of up to a few tens of MW, to small critical assemblies with negligible power outputs of a few W. Table 2 identifies the number of operating reactor facilities, permanently shutdown research reactors, or those that are at different decommissioning stages, fully decommissioned research reactors, and research reactors being constructed in participant countries [2]. Further information can also be found in [3].

Some research reactors have already been shut down, but have neither entered safe enclosure nor been decommissioned. Final decisions regarding these facilities will depend on political, technical and financial issues, among others.

Table 2. Research reactors in participant countries

Country	Research reactors			
	In operation	Shut down or under decommissioning	Decommissioned	Under construction
Argentina	5	2	0	0
Canada	8	5	3	2
China	14	2	0	2
Germany	13	11	22	0
Hungary	2	0	1	0
India	5	0	4	1
Korea, Rep.	2	2	0	0
Lithuania	0	0	0	0
Russian Fed.	49	36	11	1
Slovakia	0	0	0	0
Sweden	0	3	1	0
UK	3	6	27	0
Ukraine	1	2	0	0
USA	41	117	69	0
Total	**142**	**186**	**138**	**6**

2.1.3. Nuclear fuel cycle facilities

Fuel cycle facilities include, among others, reprocessing plants, enrichment plants, fuel fabrication plants, conversion plants and spent fuel storage facilities. Their size and complexity ranges from large reprocessing and enrichment plants, to small laboratories and pilot plants. In most cases, decommissioning waste from fuel cycle facilities contains a significant inventory of alpha-emitting radionuclides.

Table 3 identifies the number of nuclear fuel cycle facilities in operation, shut down or under decommissioning, fully decommissioned, and under construction in the participating countries [4].

Table 3. Nuclear fuel cycle facilities in participant countries

Country	Nuclear fuel cycle facilities			
	In operation	**Shut down or under decommissioning**	**Decommissioned**	**Under construction**
Argentina	8	9	1	0
Canada	20	7	13	0
China	11	2	0	2
Germany	11	7	13	12
Hungary	1	1	0	0
India	19	3	2	1
Korea, Rep.	4	2	0	0
Lithuania	1	0	0	0
Russian Fed.	24	2	0	2
Slovakia	1	0	0	0
Sweden	3	1	0	0
UK	16	20	8	1
Ukraine	3	0	1	0
USA	35	40	42	5
Total	157	94	80	23

Other facilities

In addition to nuclear fuel cycle facilities, radioactive decommissioning waste is also generated by laboratories and institutions, typically:

(a) Radioactive waste processing and storage facilities.
While technological equipment and surfaces in waste processing facilities are surface contaminated, the inventory of waste from these facilities is minimal

(b) Radioisotope production facilities.
Radioisotopes are produced in irradiation facilities (reactors or accelerators) and include installations that perform radiochemical separation of radionuclides, as well as those that produce sealed sources for shipment to various customers. Generation of decommissioning waste is strongly dependent on the type of the facility.

(c) Medical facilities
Radioisotopes are used in a wide variety of medical facilities. They are widely spread around the world, even in countries without other nuclear installations. They are often part of larger medical institutions, and include both diagnostic and therapeutic equipment. With the exception of sealed therapeutic sources, the volume and activity of radioactive waste generated from their decommissioning is usually small, and may appear only as a consequence of accidental or unintentional contamination.

(d) Radioisotope and radiation application facilities
Facilities for radioisotope and radiation application include, among others, irradiation plants, using either radioactive sources or accelerators, for sterilization of a wide range of products. Accelerators and radioactive sources are used as part of industrial processes for measurements and calibrations, non-destructive testing, quality control, and the

manufacturing of specialised products (e.g. smoke detectors).These facilities also support the production of radiopharmaceuticals.

(e) Research facilities
Other research facilities exist in universities, non-nuclear R&D institutions, and industry. They include accelerators as well as research and development laboratories that are typically equipped with fume hoods and glove boxes; this may require special consideration when dismantling and decommissioning the facility.

There is also a multitude of small facilities that may require decommissioning. Although it is difficult to establish the number of these facilities, since no comprehensive register is available, a short overview is provided in [5, 6].

2.2. Types of waste

Decommissioning of a nuclear facility results in both non radioactive wastes and three different types of radioactive wastes, in particular:

- Primary waste
- Secondary waste, and
- Contaminated tools and equipment.

Primary decommissioning waste refers to waste generated during dismantling activities. Depending on the type of reactor system, this waste can include plant system components, such as the pressure vessel and associated internal components from a pressurised water reactor; graphite from a graphite moderated reactor system; and primary circuits, in particular steam generators and the concrete biological shields that surround the vessels. Typically, primary waste consists of construction materials, such as steel, aluminium, reinforced concrete, and graphite or zirconium alloys; however, metal and concrete rubble often constitutes the bulk of primary waste.

Primary waste varies widely in terms of type, activity, size and volume, and consists of both activated and contaminated components. Activated components (usually construction materials) are characterised by the presence of both short lived and long lived radionuclides resulting from the activation of the construction material. In addition, most surfaces, including building materials and process equipment and components, are contaminated by radioactive surface deposits. These deposits result from the transport of neutron activated corrosion products and fission products released from the fuel assemblies during reactor operation, usually due to fuel failure. Items such as glove boxes and hot cells can be significantly contaminated by long lived alpha-emitting radionuclides.

Secondary waste refers to waste generated during various decontamination and dismantling activities, e.g. decontamination of metallic components or flushing of systems to reduce the amount of primary waste. Secondary waste consists of liquid waste, spent ion exchange resins, spent filters, and dry active waste. These wastes are processed and conditioned using procedures and facilities available for primary waste. The radionuclides present and their activity levels in secondary waste correspond to those of the decontaminated and/or dismantled components.

Contaminated tools and equipment refers to materials employed during decontamination and dismantling of a nuclear facility that become contaminated during use. As with secondary waste, the type and level of radioactive contamination reflects the pollution of components which were decontaminated and/or dismantled. Contaminated tools and equipment may be decontaminated and subsequently re-used to minimise radioactive waste generation.

Inactive (non-radioactive) solids and liquids may also be generated during the decommissioning of nuclear facilities, and can compose a significant part of the waste. Typically, non-radioactive solid materials include items such as piping, pumps, tanks, duct work, concrete rubble, structural equipment and electrical equipment. Inactive liquids and solid materials can be reused, recycled, or disposed of using conventional methods in accordance with applicable regulations. For some inactive materials, the physical and chemical risks associated with their disposal needs to be addressed [7-10]. Examples of such materials include lead, beryllium and asbestos.

In addition to the above, appropriate segregation and decontamination procedures shall be implemented to reduce, as much as practicable, the volume of radioactive materials requiring treatment [11].

2.3. Waste classification

The classification of radioactive waste is of particular importance when decommissioning nuclear facilities. The existence of guidelines for waste segregation and conditioning, in particular waste acceptance criteria for storage and disposal, can have a significant impact on the planning of decommissioning activities, particularly cost estimates and the selection of decontamination and dismantling activities [12, 13].

A widely used classification system based on dose rates identifies waste as one of the following: low level waste (LLW), intermediate level waste (ILW), and high level waste (HLW). However, this system serves mainly to support waste handling and storage activities. For the long term management of decommissioning waste, it may be more desirable to employ a classification based on the activity levels and half lives of the radionuclides contained in the waste, as defined by the IAEA [11] (see Fig.1).

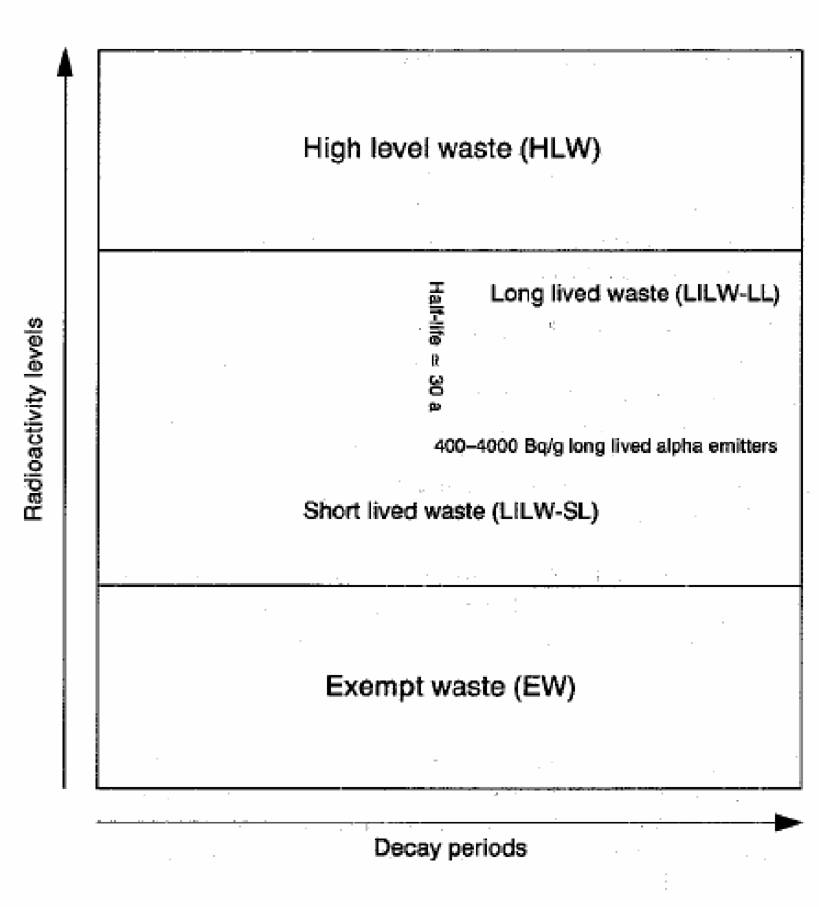

Fig. 1. IAEA waste classification system [11]

Decommissioning waste has a wide range of characteristics. Generally, with the exemption of high level waste, it falls into all categories. However, introduction of additional waste categories may be beneficial. An example is waste contaminated by very short lived radionuclides, called "transition waste" in the European Commission classification system. It is believed that, after storage for a sufficient period of time to allow for decay (referred to earlier as the "delay-decay" option), this waste can be cleared (exempted) from nuclear regulation.

Most decommissioning waste is short lived (LILW-SL) and may be disposed of in near-surface facilities. Long lived waste (LILW-LL) generated during the decommissioning of fuel cycle facilities or in the event of incidents/accidents at a reactor, needs to be disposed of in geological disposal facilities.

Some decommissioning waste may be classified as "very low level waste (VLLW)" as defined by some countries. This waste would still be under nuclear regulatory control, although it does not need to be disposed of in sophisticated engineered facilities; typically, trenches with appropriate isolation layers consisting of plastics and natural materials (clays) are sufficient.

Large amounts of materials generated have such a low radionuclide content they can be released as "exempt waste" [13, 15]. Such materials may be recycled, reused or disposed of as non radioactive waste.

In addition to classification based on radiological characteristics, decommissioning waste should also be classified in terms of its physical and chemical properties. This would allow for the development of acceptable waste forms and waste packages that can meet the waste acceptance requirements for particular disposal facilities.

2.4. Waste characteristics

Activation/contamination decommissioning waste to be disposed of consists of activated and/or contaminated material.

Stable atoms can be converted in a reactor or an accelerator into radioactive isotopes, in particular by irradiation with neutrons (activation). Activation is limited to the reactor core or the target area of an accelerator and is a "bulk phenomenon". The generation of radionuclides by activation can vary greatly as it depends, for example, on the neutron flux and the particular materials of construction, such as those containing Na, Co or Ni [16, 17].

Radionuclides generated or handled in a nuclear facility contaminate all surfaces they contact. Such contamination may be completely or partially removed by chemical, physical or mechanical means. For contamination that penetrates into the bulk material (e.g. porous cement), removal of the whole contaminated layer is required.

2.4.1. Half-lives of radionuclides

One of the most important waste characteristics is the half-life of a radionuclide. It is relevant to the classification of waste and its assignment to a disposal option, in particular those wastes containing long lived alpha-emitters or beta-emitters (such as ^{36}Cl, ^{59}Ni, ^{63}Ni, ^{99}Tc, ^{129}I and ^{14}C) that would have to be restricted to acceptable levels before they could be disposed of in a near surface repository [12]. In addition, long lived beta-emitters such as ^{36}Cl, ^{59}Ni, ^{63}Ni, ^{99}Tc, ^{129}I and ^{14}C may also have to be limited with respect to disposal in a near surface repository.

On the other hand, decay waste or materials containing very short lived radionuclides that decay in a short period of time may also be released from nuclear regulatory control.

2.4.2. Waste with unique characteristics

Due to the individual design of many nuclear facilities, waste materials with unique characteristics may be generated. For example, large amounts of radioactive graphite are expected during the decommissioning of reactors that use graphite as a moderator (including research reactors, and gas-cooled and water-cooled power-producing reactors). These materials require special attention if they are classified as radioactive waste. Graphite as a radioactive waste presents a unique set of technical issues, including both surface contamination and bulk activation, accumulation of energy (Wigner energy), and difficulties in volume reduction and processing into a chemically stable form. Currently, experience in the conditioning of graphite from large scale facilities is not yet available, even though a number of techniques for the treatment of radioactive graphite have been developed. These include encapsulation of solid graphite, incineration, pyrolysis, and recycling [18-21].

2.5. Conditioning of radioactive waste

Radioactive waste generated during decommissioning of nuclear facilities must be processed so as to comply with requirements for storage and disposal. Well-proven methods and installations (stationary and mobile) are available.

Most of the primary waste generated during decommissioning is in solid form and may be activated and/or contaminated. It can be partially decontaminated in order to reduce surface activity levels, allowing easier handling (from a radiological viewpoint) or even release from nuclear regulatory control (clearance). Decontamination may be performed both before and after segmentation of components that are subject to dismantling.

Waste that cannot be cleared has to be processed as radioactive waste. Processes to be applied in the conditioning of solid radioactive waste are determined by the type of waste and include, for example, size reduction, segregation, compaction, incineration, melting and immobilization (usually cementation).

The decommissioning of nuclear facilities is also associated with the generation of liquid waste as secondary waste, such as when cutting metallic components and during decontamination of solid materials. Processes that can be applied in the treatment and conditioning of liquid radioactive waste include collection, evaporation, drying, and immobilization (for example, cementation and bituminization) [22].

Waste that has been processed needs to be packaged for disposal. Such packages are very diverse in terms of dimensions, volumes, shapes and materials. Their volumes may range from several tens of litres to a few tens of cubic meters. Typical construction materials include carbon or stainless steel, cast iron, and reinforced concrete. The design of packages and their properties are largely dependant on the waste forms, available handling/transportation equipment and the site-specific conditions and requirements of the storage facility and repository. Storage periods for conditioned wastes before final disposal vary according to the individual national policy, and can be as long as several centuries.

Some countries intend to dispose of large components in one piece, without segmentation. In such cases, the component is handled and transported to the disposal facility without a container. If necessary, the item may be shielded or covered with a plastic sheet or by other means. However, in other countries such as Germany, large items are always size reduced to minimise waste volumes. Size reduction can be achieved using a number of techniques [23].

The main objective of waste conditioning is to limit the potential for the mobilization of radionuclides contained in the waste (via dispersion or dissolution) and to reduce the voids within the container in order to improve the integrity and stability of the waste package. Different types of encapsulation materials are used to immobilize waste, with cement being the most commonly used. Nevertheless, in many cases decommissioning waste is packaged without any sophisticated conditioning, provided that this solution complies with the repository acceptance criteria.

In any case, the waste package, which consists of the waste form and the container, must comply with the waste package specifications and the waste acceptance requirements for the particular storage and/or disposal facility. It should be noted that these requirements are based on safety assessments and are thus specific to the individual facility [24].

2.6. Waste package characteristics

Packaging is required to provide safe containment of the waste during handling, storage, transportation and disposal. Waste packages shall meet the waste acceptance requirements of the individual storage and/or disposal facilities. They can vary widely in their design complexity. In the context of decommissioning waste, packaging also includes large non-fragmented components.

As discussed earlier, radioactive waste generated during decommissioning consists of metallic waste, concrete rubble, liquid waste, sludge, filters, ion exchange and dry active waste. Most of these waste streams need to be appropriately conditioned and packaged for storage, transportation and disposal. Desirable characteristics of waste packages include the following:

- Limiting of the gross mass (important for both handling and stacking in engineered structures).
- Acceptable compressive strength and minimal void space to ensure the stability and stackability of waste packages, and the stability of the storage/disposal facility against subsidence.
- Absence of free liquids to prevent contamination and activity release, to prevent damage to the containers during handling, and to prevent corrosion of the waste packages.
- Restriction of the fissile material content to prevent nuclear criticality incidents, particularly in the event that the packages are exposed to water during storage or disposal.
- Absence or treatment of organic substances (e.g. decontamination solvents) that exhibit chelating or complexing behaviour.
- Physical and chemical compatibility between waste and immobilization/ encapsulation materials.
- Ability of the package to withstand various accident conditions, such as fire or drop/impact events.
- Immobilization of radionuclides consistent with the requirements of waste acceptance criteria for the disposal facility.

2.7. Behaviour of waste materials and packages

The choice of decommissioning strategies [25] may have an important effect on the type and inventory of wastes to be disposed of. Namely, the residual activity of short lived nuclides can vary notably depending on the duration of the deferring period between the termination of facility operations and start of dismantling activities.

Radiological and chemical characteristics, as well as physical properties of waste materials generated during decommissioning can be highly variable. Consequently, compatibility issues between the wide variety of wastes, as well as appropriate conditioning options need to be considered. The nature and extent of solid waste can range from neutron-activated reactor components to debris contaminated by fission products and/or transuranic elements. Various radioactive liquids and sludge are generated during decontamination and flushing of contaminated systems. Unlike LILW generated during routine operation of nuclear power plants and from radioisotope applications, a large proportion of decommissioning waste can be extremely heterogeneous in terms of both the radioactivity levels and the size of individual components. This heterogeneous nature may require application of a more conservative approach during both the safety assessment and the repository design.

Waste material behaviour influences both the operational and post-closure safety of the repository. During the operational phase, which may last several tens of years, the dose and operational impacts of potential accidents such as fire, dropping of a waste package, release of volatiles etc., need to be considered. Also, potential exposure of personnel during handling, transport and disposal operations needs to be addressed, taking into account usually inhomogeneous radiation fields. The performance of a waste package filled with heterogeneous waste during a fire or drop accident may differ significantly to that of a homogeneous waste package. For example, high activity waste items may be hulled from binding material, resulting in increased dose rates; or particulates from corrosion products or debris may be generated in an easily respirable form.

Differences between operational and decommissioning wastes need to be addressed during the source term assessment (the near field analysis), which is an important component of the site-specific safety assessment for a disposal facility. Typically, safety assessments for decommissioning waste should address the following waste characteristics: increased content of mobilization agents, such as surfactants and chelating compounds; greater waste heterogeneity; higher potential for non-uniform distribution of activity; and large component size, often with void spaces.

2.8. Waste form and waste package degradation

Waste containers contribute to waste package and repository performance by delaying the release of radionuclides, thereby allowing most of the short-lived radionuclides to decay prior to their mobilization. The estimation of container lifetime is necessary to predict long-term releases of radioactivity to the environment, particularly for near surface repositories.

While LILW waste packages are primarily selected for handling and transportation purposes, their degradation may become the principal concern for long-term safety considerations. As a result, the resistance of these packages to degradation is becoming one of the principal characteristics of the waste package. The confinement of radionuclides and structural integrity of the waste package, in other words its durability, is generally required for a period that is sufficient to reduce the risks to the public from disposed of LILW to acceptable levels. Factors affecting the waste container's durability during disposal need to be studied in the context of the design life of the whole repository barrier system, and its host environment.

Mechanisms that impact durability are identical for decommissioning and operational waste packages. However, due to the differences between the two types of wastes, the relative importance of the particular mechanisms may change. For example, decommissioning wastes will contain much more metal than operational wastes and, therefore, corrosion and gas generation are considered to be more important.

For metallic containers, corrosion performance is an important indicator of container integrity and lifetime. Therefore, it is essential to establish underlying corrosion scenarios that contribute to container failure for the various types of material. Corrosion-induced gas generation requires important consideration during the repository safety assessment. Carbonation rate, degradation due to chemical and mechanical attack, and corrosion of reinforcing metals need to be considered in order to estimate the lifetime of concrete containers. Polymer-container materials (HDPE), on the other hand, are not susceptible to corrosion; although creep, embrittlement, and irradiation-induced degradation can affect their durability.

The waste form itself provides certain radionuclide containment. The physical-chemical properties of the waste form, including the nature of the contaminant, and its compatibility with other engineered barriers and specific environmental conditions, will determine the rate of radionuclide release. Once the container degrades, giving rise to access of water to the waste, releases of radionuclides are determined primarily by the properties of the waste form.

In addition to contributing to radionuclide containment, waste form stability is also important to the overall integrity of the disposal facility. Together with the container, it should have sufficient mechanical strength to withstand loads associated with container stacking and backfilling. Impact resistance and compressive strength are important properties that are repeatedly tested and assessed to ensure structural integrity is maintained under anticipated repository conditions. Special consideration may be needed in analyzing structural integrity when disposing non-fragmented large equipment components. Their corrosion-induced collapse can contribute to the instability of the disposal vault and its cover, resulting in increased water inflow into the disposal spaces.

Degradation of packages and other engineering barriers can also have both positive and negative affects on the properties of buffer and backfill materials. The former is represented by the backfill self-sealing capability (backfilling of the pores by degradation products, such as iron hydroxides from steel and calcium hydroxide from cement). This self-sealing phenomenon is caused by the deposition of material from precipitation reactions that are driven by differences in pH and/or eH and solubility constraints However, self-sealing may also decrease the permeability of the backfill, thereby reducing its drainage capacity and retarding its mass transport properties. Evaluation of the interactions between waste packages, concrete structures, and backfill materials should also take into account the swelling and shrinkage properties of the materials that may also be negatively affected by degradation products.

Cellulose materials, such as paper, cloth, cotton, and wood that may be part of decommissioning wastes are susceptible to microbial attack, resulting in degradation and gas generation. In contrast, non-cellulose materials such as plastics, rubbers, metals, and cements generally exhibit high resistance to microbial attack. Degradation products of cellulose materials may contribute to leaching by chelating and complexing mechanisms. All of these characteristics need to be considered when selecting appropriate conditioning procedures for decommissioning waste, with the desired end product being solid, stable and inert waste forms.

2.9. Radionuclide release

Waste package performance refers to the combined performance of the waste form, container, and liner or overpack (if present) [26]. From a safety assessment point of view, the essential features of waste package performance are structural integrity and radionuclide containment. Waste package performance is used to determine the source term of the repository, i.e. the radionuclide flux leaving the disposal units. Releases of radionuclides from a disposal unit result from a number of physical and chemical processes that occur primarily in the presence of water. Generally, the infiltration of water initiates container degradation and leaching of the waste form, giving rise to the release of radionuclides from the disposal units. In addition, corrosion-induced hydrogen may be generated as a result of radiolysis of the waste form and container degradation in the presence of water. However, although the generation and build-up of gases can affect both waste package and repository integrity, when assessing repository functions, the existence of open pathways for water is generally much more important than escape of the gas itself.

Both the behaviour and performance of a waste package depends on the nature of the waste package and the repository site-specific conditions. Decommissioning waste packages are expected to exhibit a large degree of variability in their behaviour and performance due to the diversity in waste types, waste forms, containers, site-specific conditions and the complex chemical, physical, and microbiological processes and interactions that are likely to take place in the disposal units.

Depending on the type of waste form, radionuclide release mechanisms can vary significantly. For example, cemented waste forms exhibit diffusion-controlled release for many radionuclides, although for some radionuclides, the release mechanism is solubility-limited. In the case of activated metals in reactor decommissioning waste, the release is likely to be controlled primarily by the corrosion rate of the metal matrix, assuming the activation products are distributed uniformly. In contrast, simple surface contamination can be released instantaneously.

2.10. Modelling waste package performance

Within the safety assessment, major differences between decommissioning waste and operational waste are reflected in the performance assessment of the waste package. One of the unique challenges presented by decommissioning waste lays in the equalised parameters and homogeneous media typically used in safety assessment models, which may be inappropriate for extremely heterogeneous waste streams. This problem is, among others, tackled within the Applied Safety Assessment Methodology (ASAM) project sponsored by the IAEA [27]. Specific procedures need to be developed for modelling the non-uniform distribution of radiocontaminants or activation products and their release from large equipment components that have been disposed of without fragmentation.

Certain models used to assess the performance of waste packages during the operational period of a repository may be affected by heterogeneous waste streams. These typically include potential radiological impacts associated with fires and drops of packages that could either result in immediate releases of respirable particles or degradation of the package's long-term performance. However, many modelling approaches, such as corrosion and gas generation, will be conceptually similar for the operational and post-closure phases.

Prior to disposal, an assurance is required that waste acceptance into a repository can be accomplished with due protection of health and safety of the public. The long time frame over which public safety must be assured requires that computer simulations of events and processes consider the far future. Detailed guidance for performing these analyses can be found in [28, 29].

Models used for safety assessments are based on complex mathematical simulations that are highly sensitive to inputs. These inputs include information on the repository engineered barrier system, its construction, the host environment and, not least, the waste to be disposed of, including its characteristics and packaging. Therefore, it is important to collect and keep detailed information about the waste streams so that that model inputs are accurate and reliable, and so that a defensible decision can be made regarding the disposal system's compliance with regulatory requirements.

3. ISSUES RELATED TO REPOSITORY DESIGN AND SAFETY ASSESSMENT

As mentioned in Section 2, there are some important differences between decommissioning and operational waste that are likely to have an impact on the design and safety assessment of a disposal facility. The relative proportions of metal and concrete in decommissioning waste are significantly larger in comparison with standard operational waste. In many cases, waste packages and intact items (components without any packaging) are much larger than those present in operational waste.

3.1. Basic considerations

A comprehensive inventory of present and anticipated decommissioning waste is an important input to the design and safety assessment of a waste repository. The inventory should include predicted future arisings in order to allow for the development of optimal decommissioning waste disposal solutions. The following recommendations are based on the national radioactive waste management programme:

- Waste categories shall be clearly identified, e.g. short-lived LILW, long-lived LILW and heat generating waste (if applicable); waste acceptance criteria for existing facilities may provide guidance on the definition of activity limits;
- Pre-treatment, treatment and conditioning for waste storage and/or disposal shall be defined;
- Technical options for disposal operations and the size of the repository need to be determined.

The usefulness of waste inventory data depends on the reliability and completeness of the information provided for the various waste items. Characteristics and information being important for the design of a repository and the performance of a safety assessment typically include:

- Type of primary waste (e.g., solid, liquid, compactable, non-compactable, combustible);
- Size, weight and volume;
- Radionuclides present;
- Radioactivity per nuclide;
- Chemical and physical properties;
- Treatment and conditioning procedures;
- Type, material and construction of a waste package;
- Disposal-related characteristics of the waste package.

Co-disposal of decommissioning waste with operational waste can significantly increase the final disposal volumes. This has a direct impact on facility design, as the volume capacity has to be extended adequately to accommodate the anticipated waste inventories. Depending on the individual national waste management strategy, new repositories for decommissioning waste may be required. In some cases, the construction of a single purpose disposal facility may be beneficial, provided that the waste volume of is sufficient. Thus, disposal facilities are being considered or even operated in some Member States for decommissioning very low level waste, radium waste or irradiated graphite.

An important aspect of co-disposing decommissioning waste with operational waste is the significant increase of the total as well as the individual inventory of radionuclides. Namely, the activity of long lived nuclides, such as ^{14}C, ^{36}Cl, ^{59}Ni, ^{63}Ni, ^{94}Nb, ^{99}Tc, ^{126}Sn, and ^{135}Cs may become a critical issue, since the site capacity for these nuclides can be easily exhausted. The higher potential for releases of radionuclides from unconditioned and unpackaged decommissioning waste needs to be especially considered within respective safety assessments. This may result in limitations on allowable radionuclide-specific activities for these wastes.

3.2. Large components

In addition to the decommissioning waste that is packaged using conventional technologies, another type of waste is comprised of large waste components such as steam generators, control rods, pumps, and motors. These components are removed in one piece or a few large pieces, and serve as self-containers for disposal, assuming all openings, access ways, penetrations, etc. are properly sealed. Low density grout can be used to stabilize internal contamination. The dimensions of these components needs to be considered when designing a repository, and needs to address proper sizing of entrances, vaults, and handling areas; means of transportation; and equipment capacities. The safety assessment for disposing large components that are neither conditioned nor packaged needs to address easier water penetration into the system and mobilization of radio contaminants with consequently less restricted release [30].

3.3. Source term considerations

Confinement of radionuclides within the disposal facility prevents radionuclides from being released into the environment, allowing the radionuclides to decay. The performance of isolation barriers is dependant on the structural integrity of the facility and any backfill, the waste container, waste form, or a combination of these components. Thus, void spaces in packages need to be limited and their construction needs to consider resistance to the compressive stress induced by overlaying waste and construction materials. Understanding the deterioration of the waste package structure may also be required to demonstrate that the repository can provide adequate protection of human health even under degraded conditions.

The period during which the designed characteristics of the waste package should be maintained is determined by the site-specific repository safety assessment. Experiments on the deterioration of waste packages provide essential input data. Waste containers can contribute to overall waste package and repository performance by delaying the ingress of water, thereby allowing the short lived radionuclides to decay to insignificant levels. Estimation of container lifetime is required to establish how much credit can be assigned to the waste container for radionuclide containment. The lifetime of impermeable containers (metallic or HDPE) is determined as a function of the container design, material degradation mechanism and rate, environmental conditions and groundwater chemistry. These systems are generally modelled to be completely effective in isolating the waste from contact with water for the duration of their planned lifetime. After this period, the containers are assumed to be completely ineffective at protecting the waste form from interacting with water.

For concrete containers, water may percolate through the container; thus, the failure mechanism differs from impermeable barrier materials. Testing the confinement function of concrete containers requires a determination of the concrete's hydraulic conductivity and radionuclide diffusion coefficients, including sorption effects. It is important to consider the time-dependency of these properties as micro-cracks are formed and propagated in the concrete wall.

Most decommissioning wastes belong in the low level radioactive waste category, and can be packaged in simple containers without a confinement function as identified in safety analysis space, or disposed of unpacked. A decision on the appropriate disposal option is based on the performance assessment, and needs to consider handling and transportation issues, both at the decommissioned facility and at the repository.

Mobilization of radionuclides from activated metallic components is governed by the corrosion rate of the metal. Depending on the repository conditions (low pH, anaerobic atmosphere), the susceptibility of the metal to degradation may differ significantly; if appropriate, a safe chemical environment may be established with appropriate filling material. Design of disposal spaces needs to consider gas generation and allow for sufficient convection or diffusion to prevent overpressure and consequent uncontrolled damage to the isolation layers.

3.4. Potential gas generation issues and impact

Decommissioning waste often includes contaminated or activated steel and concrete. Gas generation due to metal corrosion is the most important phenomenon affecting repository design, and needs to be addressed in the site-specific operational and post-closure safety assessment. Biodegradation of organic waste materials, another potential mechanism for gas production, is of much less importance for decommissioning waste.

For disposal units that are located below the water table, corrosion of metallic waste and iron-based containers is initially aerobic. However, after a period of time (typically tens of years), following the consumption of all available oxygen, anaerobic conditions will develop and will generally prevail for the remainder of the post-closure phase. Anaerobic corrosion of iron and some other metals can then result in the generation of large amounts of hydrogen gas, which may have an adverse effect on repository performance. Volatile radionuclides such as ^{3}H and ^{14}C - if present - could escape from the repository along with hydrogen generated from metal corrosion. Pressure build-up inside the containers or the repository as a result of gas generation could have an adverse impact on the integrity of the waste packages, engineered barriers of the disposal facility, and the host geological environment. Gas generation is much less of an issue if the disposal units are located in the vadose zone, because the engineered barriers are generally not designed for tightness but rather to facilitate drainage.

Some disposal facilities place restrictions on waste streams that generate significant volumes of gas. As corrosion is a leading mechanism for gas generation, measures may need to be applied when conditioning and disposing metallic waste in order to meet facility waste acceptance criteria regarding gas generation.

In summary, a key characteristic of decommissioning waste relative to operational waste is the large inventory of metallic constituents. Thus, the potential for hydrogen gas generation and subsequent pressure build up is an important consideration in assessing the design and performance of a repository. The safety assessment needs to evaluate the potential impact of gas generation and subsequent gas pressure build up on the integrity and stability of waste packages, as well as on repository safety, including radiological and flammability hazards. Results of such an assessment may call for the venting of waste containers or for more sophisticated design solutions of the repository or its affected components.

3.5. Waste acceptance requirements

Waste acceptance requirements are qualitative or quantitative criteria by which the acceptability of waste packages for storage or disposal can be judged in relation to the specific conditions of a given waste management facility [24, 31]. Preparation of these requirements occurs in three stages that are distinguished from each other by the level of available information regarding waste package characteristics, facility design and site characteristics:

- Generic requirements are initially defined based on (i) the national radioactive waste disposal policy, (ii) general information on the type and quantities of waste packages expected to be generated, and (iii) the availability of certain sites.
- Site selection and site characterization follow, providing information on the characteristics of the potential disposal site.
- Once the actual characteristics of the entire disposal system have been identified, the formulation of specific waste acceptance requirements is completed.

Application of this stepwise approach allows for the development of general requirements prior to the availability of a waste storage or disposal facility; as the level of information regarding the storage/disposal system increases, the acceptability criteria can be more preciously specified. Therefore, waste acceptance requirements in their preliminary or first draft form generally only specify qualitative measures to be taken in order to achieve the objective of disposal (protection), and define the principles demonstrating that this objective can be reached. The final, quantitative formulation is based on a complete safety assessment and is issued as a part of the operating license for the facility.

In general, waste acceptance criteria distinguish between waste that can be safely managed in a facility and waste that must be rejected during routine operations. But even rejected waste can be later accepted into the facility, provided that additional compensatory activities, either technical (re-packing) or administrative (safety re-assessment) are implemented.

3.5.1. Radiological criteria

General radiological criteria for the protection of the workforce and the general public are established in national regulations. More detailed criteria for packaging and for repository design may be proposed by the implementing organization but must be subjected to regulatory approval. These criteria should take account normal operations and accidental situations that encompass all phases of the repository life cycle.

For operational purposes, the dose rate may be limited by virtue of the design of handling and transport equipment. Dose rate restrictions can also be achieved when using standard packages and waste forms by limiting the activity of certain radionuclides.

For post-closure safety, upper bounds are defined by the concentrations of selected short-lived and long-lived radionuclides. However, these restrictions are specific to a particular disposal facility; they cannot be specified in a universal manner. Often, specific activity limits are higher for a single package than for the whole disposal facility, in order to allow acceptance of a more contaminated item or waste without deteriorating the safety of the facility. In practice, admissible concentrations of radionuclides supporting post closure safety result from safety-related investigations and calculations [24].

In order to comply with activity limits for managing decommissioning wastes, a credible determination of the activity of the treated materials is required. Typically, the measurement of inner surface contamination of dismantled equipment requires special care and the development of efficient radioanalytical methods.

3.5.2. *Non-radiological criteria*

Disposal of materials that could present chemical or biological hazards needs to comply with relevant national regulations and addressed in the safety assessment. Such hazards may arise from the presence of the following materials:

- Free liquids, both aqueous and organic, even if they are retained on absorbents (solvents, oils and paints);
- Aggressive, chemically reactive and corrosive reactants (decontamination acids);
- Surfactants and complexing chemicals (decontamination solutions);
- Products capable of reaction in the presence of water (metallic Na, K);
- Explosives or products capable of strong exothermic reactions;
- Putrescible matter;
- Chemically or biologically toxic material (asbestos, lead, beryllium; infectious samples and waste from biologic experiments);
- Materials generating significant volumes of gas.

Oxidising and aggressive agents, as well as surfactants and complexing compounds are often used during decontamination. Chemicals of uncertain origin may be found in older facilities. Cutting of certain metals may produce pyrophoric filings. Medical facilities can potentially handle biotoxic materials. All these are examples that require conditioning and appropriate neutralization.

3.5.3. *Waste package considerations*

The waste to be disposed of shall be solid or in a solidified form, usually in an appropriate and undamaged package with handling or handling/confinement functions.

Waste packages are designed and fabricated to have sufficient mechanical strength to bear loads after repository closure and to be capable of withstanding accidents during the operational phase, as well as to comply with requirements for waste handling, stacking, transport and storage. Accordingly, consideration shall be given to waste package design and fabrication, in particular:

- Weight, volume or dimension limits;
- Waste confinement features;
- Stress and corrosion resistance.
- Performance of waste packages during handling, transportation, storage, receipt and accidents at a disposal facility, with particular attention paid to radiation protection;
- Mechanical properties compatible with the stability requirements of the disposal facility;
- Material properties, including application of paint, and contribution to the prevention or reduction of radionuclide mobilization and subsequent escape to the environment.

During decommissioning, large volumes of heterogeneous or bulky waste such as contaminated soil, dismantled components or demolition rubble is produced that could be disposed of without packaging, or using temporary or single use type packages. Conditions for accepting these wastes must be addressed in the waste acceptance criteria and approved by the regulatory body.

3.5.4. Administrative requirements

Radioactive waste generation, processing and disposal tracked in many countries in order to allow for the planning of waste management activities and facilities, and to preserve information on the final destination of radioactive material. The tracking system requires the identification and characterization of waste streams, together with an estimation of their inventories. Once the waste has been packed, the package shall receive a unique identification. Waste package characteristics, such as the activity of major and critical radionuclides, surface contamination, dose rate, waste form specification, weight, producer, and destination, shall be inscribed in a package passport issued for each handled item, both standard and untypical. The passport shall go with the package/item from the generator to the final disposal site and shall become a part of repository system's documentation.

Verification of proper package labelling and accuracy of all information in the passport is a part of the acceptance procedure. Verification of certain parameters (activity, contamination, and weight) may also be performed.

4. DISPOSAL OPTIONS FOR DECOMMISSIONING WASTE

The IAEA radioactive waste classification system, based on waste characteristics and radionuclide content, provides a framework for defining a generic approach to radioactive waste management. The system can be linked to potential disposal options for various waste categories based on their specific characteristics, with specific activity and longevity of radioactive constituents being the key distinguishing parameters. Accordingly, very low level waste, as well as short-lived low and intermediate level radioactive waste (i.e., decommissioning waste) containing radionuclides that decay to insignificant radiation levels within a few decades or centuries, can be disposed of near the surface. High-level and long-lived radioactive wastes require a higher degree of isolation and should be predominantly disposed of in geological formations (i.e., emplacement in engineered structure at depths of hundreds of meters). In principle, the higher the activity and the longer the half-live of major radiocontaminants, the deeper the facility should be. In addition, some national approaches to disposal prefer the emplacement of all types of radioactive waste (short and long-lived, low and high level) in geological formations.

4.1. Generic features of disposal systems

4.1.1. The multiple barrier concept

In developing any disposal system concept, reliance is placed on a multibarrier system approach in which both the site characteristics and the engineered (technical) barriers, namely the waste form and the packaging, together contribute to the isolation of the radioactive waste from the environment for time periods that are sufficiently long enough for radionuclides to decay to acceptably low levels. This general approach has been technically elaborated and adopted for all types of disposal facilities; naturally, the barrier system compositions differ accordingly. The use of multiple barriers provides reasonable assurance of adequate performance of a repository system and thus, its ability to achieve the protection objectives of radioactive waste disposal.

4.1.2. Institutional control

Institutional control consists of an active phase that includes monitoring and maintenance of repository fences and capping and, if needed, remedial actions, and a passive phase that is limited to application of restrictions on the uses of the repository area.

For near surface repositories in which the disposal units are within a few metres of the surface, institutional control provides, among others, assurance of the design performance of engineered barriers. This occurs during the initial period after repository closure when the activity of short lived radionuclides is still high. For disposal spaces that are greater in depth (tens of meters or more), such as in rock cavity repositories or geological repositories, less reliance is placed on institutional control.

The duration of post-closure institutional control can only be reasonably expected to last a few hundred years at the most; the exact period is a matter of the national strategy for radioactive waste management. Periods from 50 years for very low level waste to 300 years for low and intermediate short lived waste are used in certain countries.

4.1.3. Monitoring and surveillance

Monitoring a disposal facility begins with its siting. The near surface disposal concept usually envisages continued monitoring and surveillance of the site as a part of active controls after repository closure [32]. During this period it represents additional safety measure and contributes to confidence in the satisfactory performance of the disposal system.

The acquisition of data from monitoring also contributes to general scientific and technical knowledge that can be used in future repository planning work. Post closure monitoring and surveillance needs for geological repositories are comparatively minimal.

4.1.4. Human intrusion

Radiological acceptance criteria are determined as a result of analysing two main types of scenarios: (i) the mobilization of radionuclides and their transport to the environment and (ii) human intrusion in the facility. The latter distinguishes between intentional and unintentional disturbance of the repository system. Nevertheless, it is not anticipated that disposal facilities below 30 m could be intruded into without the intention to reach them. Therefore, radioactive waste that requires longer period of containment due to its activity and/or half-life of radionuclides needs to be disposed of below this depth limit.

4.2. Use of existing facilities for decommissioning waste

Existing disposal facilities, designed for the disposal of operational waste, may also be used for the disposal of decommissioning waste. If the operational permit does not consider decommissioning waste, extension of the licence is required.

4.2.1. Near surface facilities for LILW

A range of technical solutions exist for the emplacement of radioactive waste in near surface environments. The selection of a disposal option depends on many factors, both technical and administrative, such as the radioactive waste management policy; national legislative and regulatory requirements; waste origin, characteristics and inventory; climatic conditions and site characteristics; public opinion; etc.

Near surface disposal options include two main types (basic concepts) of disposal systems:

- Shallow depth facilities consisting of disposal units that are located either above (mounds, etc.) or below (trenches, vaults, pits, etc.) the original ground surface; they are suitable for low and intermediate short lived decommissioning waste.
- Facilities in which the waste is emplaced at greater depths in rock cavities or boreholes intended for decommissioning waste with increased activities of long lived radionuclides.

In the former case, the thickness of the cover and overlying material is typically a few metres, while in the second case, the layer of rock above the waste can be tens of metres thick.

The typical near surface disposal concept involves engineered structures. However, for very low level waste with activities only slightly above clearance levels, facilities with simple isolation barriers are used.

Some examples of existing near surface repositories accepting decommissioning waste include Püspökszilagy in Hungary, Mochovce in Slovak Republic, Trombay and Tarapur in India, Drigg in the UK, and Barnwell and Richland in the USA.

Rock cavity repositories can be built in existing mines or in intentionally excavated caverns in various geological formations. An example of the former facility is Richard II located in an abandoned limestone mine near Litomerice in the Czech Republic. This repository, in operation since 1964, has disposal rooms located in the vadose zone several tens meters above the water table and contains decommissioning institutional waste. Manmade caverns were built for LILW e.g. in Himdalen (Norway), Olkiluoto and Loviisa (Finland) and SFR Forsmark (Sweden). The last three caverns were constructed in crystalline rock, several tens of metres below the bottom of the Baltic Sea. They consist of different types of mined chambers and silos adapted for the disposal of ILW. The Swedish facility will accept decommissioning waste into a new segment extended for that purpose.

4.2.2. Geological disposal

Although this option is generally intended for high level waste and waste containing significant activities of long-lived radionuclides, depending on the national radioactive waste management policy (Germany), it may also be used for LILW. This option is suitable for the disposal of decommissioning waste as it can also accept without restrictions components containing high-activity or long-lived radionuclides.

Geological formations of very low permeability or covered by overburdens of very low permeability (e.g., thick clayish layers) offer the potential for long term waste isolation. Different disposal designs concepts have been developed, in different host rocks (granite, clay, tuff or salt) and at minimum depths of a few hundred meters.

Geological disposal programmes are in progress in various countries. Technological capability for the construction of geological repositories is available, as it is based on current drilling, tunnelling, backfilling and sealing technologies. Experience is being gained in underground laboratories and pilot facilities. At present, the Waste Isolation Pilot Plant in the USA is the only repository in a deep geological formation in operation.

For the disposal of operational and decommissioning LILW the Morsleben repository in Germany was used by 1998. Among others, after proper processing respecting Morsleben waste acceptance requirements, decommissioning waste originating from shut down nuclear power plants Greifswald, Gundremmingen A, Niederaichbach, Wuergassen and Rheinsberg was disposed of in this facility.

4.3. New facilities

4.3.1. New facilities for operational and decommissioning waste

Most new facilities that are in the planning stage for operational LILW are also designed to accommodate decommissioning waste, specific features of which have been taken into account in licence applications. Some countries are considering the construction of repositories for disposal of operational and decommissioning waste in surface engineered constructions (Romania, Lithuania, Iran, USA), and in intermediate depths of a few tens to a few hundred metres. Such repositories are in different stages of development in Canada, Hungary, and UK. These facilities are designed to provide greater confinement compared to near surface disposal facilities.

The German radioactive waste disposal policy has always been based on the decision that all types of radioactive waste (short-lived and long-lived) are to be disposed of in deep geological formations [33]. The planning work for the Konrad repository in Germany, which is to be constructed and operated at depths between 800 and 1300 m, addresses disposal of decommissioning waste as well. The license for this repository was granted in 2002 but the facility commissioning has been delayed due to court examination.

4.3.2. Single purpose facilities

New facilities that are specifically designed for the disposal of decommissioning wastes are planned in Japan and France. They will be of intermediate depth (many tens of metres) and are intended to accept radium waste and irradiated graphite from the decommissioning of gas-cooled nuclear reactors [19, 34].

4.3.3. Disposal of very low level waste

Some countries have introduced the very low level waste category (VLLW) within their national radioactive waste classification systems, typically containing material with the activity of some 2 orders above the exempt region. Disposal facilities for such waste do not need a high level of containment and isolation, and a near surface landfill with limited regulatory control is generally suitable. Typical waste would include soil and rubble with activities sufficiently low enough not to require shielding.

Some countries have considered or initiated the construction of this type of facility (Spain, Japan and Sweden). The French Morvilliers facility, in operation since August 2003, is designed to accommodate 650 000 m^3 of VLLW in excavated cells within a clay formation [34]. This new surface repository accepts waste originating from the dismantling of nuclear facilities as well as, from a few chemical and iron-steel industries and cleanup and rehabilitation activities.

Depending on the specific conditions of a site, surface facilities with simplified isolation structures are also suitable for large, unsegmented items of technological equipment, as there are fewer limitations regarding the size of disposed of items. For that purpose, operation procedures were adapted to accept heat exchangers, 6 m long containers, spent fuel transportation containers and, in the future, steam generators in Morvilliers facility [30]. US Ecology accepted a number of large items, including an entire reactor from Traian NPP, to its Richland repository.

4.3.4. On-site disposal/entombment

The on-site disposal option is considered in some national concepts for major reactor components and their associated equipment and materials. In addition to "sinking" the reactor vessel, the concrete reactor vault and the below-ground structures may be filled with LILW radioactive materials. The feasibility of such a solution is strongly dependent on site characteristics.

In such cases, the following should be ensured [9, 35]:

- The radionuclide activity complies with the acceptance limit for the given site;
- All equipment with unacceptable levels of contaminants is removed;
- Voids remaining after emplacement of contaminated materials are filled;
- Provisions for the construction of additional isolation barriers are made.

If entombed, a nuclear facility is structurally encased so that it remains intact until the radionuclides decay to a level that allows the facility to be released from nuclear regulatory control, typically a few hundreds of years. Such a facility becomes a de facto near surface repository.

5. NATIONAL CONTRIBUTIONS

Decommissioning of nuclear and radioisotope facilities consists of a series of activities that are described in more detail in a number of IAEA publications [36–39]. These publications provide insight into planning, strategic, managerial, regulatory, economical, personnel, technical, design and radioactive waste management issues. This publication presents the result of a five-year study performed within this Coordinated Research Project (CRP), in which institutions from fourteen Member States assessed the link between decommissioning processes and waste disposal issues.

The problems associated with decommissioning waste are so highly variable that, even with the number of participants in this CRP, only certain selected topics could be covered. In particular, strategic and technical planning, facility-specific considerations of fuel cycle facilities and research and energetic reactors, performance and safety assessments of disposal facilities, and their conceptual design were among the selected topics. The scope of each project is briefly characterised in the following paragraphs.

The objective of the Argentinean project was to analyze disposal schemes of waste arising from the total dismantling of national research reactors, starting with the oldest one that had been in operation since the late 1950s. In order to estimate characteristics and volumes of decommissioning waste, data was collected and sorted from both archive files (operational history and tracing of operational incidents) and field experiments carried out for purposes of this study (area monitoring and sampling). Measurements were complemented by neutron activation calculations. Due to their low specific activities, no disposal problems are foreseen for metals (aluminium, steel, lead) or concrete. In contrast, as the country has no experience in managing graphite radioactive waste, work was concentrated on that material. Due attention was devoted to the determination of stored (Wigner) energy and its annealing. HEPA filters were found to be effective enough for capturing gaseous emissions containing in particular ^{14}C and ^{36}Cl.

Ontario Power Generation (OPG) is a government-owned electrical utility operating in the province of Ontario, Canada. It owns five 4-Unit Nuclear Generating Stations. For planning purposes, OPG stations are assumed to be shut down after 40 years of operation and then decommissioned in accordance with a delayed dismantling strategy. An overall analysis, considering both waste volume arisings and radionuclide inventories, was performed to develop a reference database on the inventory and characteristics of potential low and intermediate level waste generated during decommissioning. This information, in turn, has contributed towards the development of a reference disposal plan for the decommissioning waste.

On-site disposal was selected for decommissioning waste arising at fuel fabrication plant ER in the North-West China. The governing reasons for selecting this strategy were the large volumes and very low-level activities of the disposed material, consisting mainly of contaminated soil and demolition debris. The disposal facility consists of pits isolated by compacted soil and clay. This system, together with a proper capping and surface water drainage, provides sufficient protection of the environment. This has been proved by measuring surface dose rates and by environmental samples.

Over the past two decades, Germany has gained substantial experience in the decommissioning and dismantling of nuclear facilities of different types and sizes: more than 60 of these facilities have been in varying post-operational stages. Based on their experience, the research project provides a systematic description of the national approach to safe management of waste generated during this period. Radioactive waste, from operations as well as from decommissioning, is to be conditioned so as to comply with the waste acceptance requirements of the repository. In Germany, all types of radioactive waste (i.e. short-lived and long-lived) are to be disposed of in deep geological formations. Currently, only one facility, the Konrad mine, has been licensed, but its operation is pending. Compliance with its waste acceptance requirements resulting from the licensing procedure must be demonstrated through waste package quality control, even if the waste cannot be disposed of until a future time.

The Nuclear Safety Directorate of the Hungarian Atomic Energy Authority requires that a preliminary Decommissioning Plan and a valid strategy of decommissioning exist for each nuclear facility in the country. The Plan provides an estimated inventory and source term for the decommissioning wastes. Radioactive waste, from operations as well as from decommissioning activities, is to be conditioned in such a way as to comply with the waste acceptance requirements of the existing or future repository. The general scope of the Project was to collect and treat data about the types and amounts of wastes to be generated during decommissioning, to improve the calculation of radionuclide inventories, and to assess the long term performance of waste packages (including corrosion and gas generation).

The Indian nuclear power programme is approximately five decades old and has many aging nuclear facilities: approval for extension of operational life for some of them has been obtained from regulatory bodies provided that their critical systems are refurbished. The experience gained during these reconstructions is described in the report. The data obtained have been employed when developing decommissioning plans required by the authorities for these facilities. Examples of waste managed in these campaigns include coolant channel replacement, decommissioning/disposal of contaminated equipment from waste immobilization plants and decommissioning of thorium processing radio-chemical plant.

The decommissioning project for two TRIGA type research reactors in Korea began in 1997, but a decommissioning plan including waste disposal scheme is yet to be submitted to the regulatory authority for each nuclear facility. A national radioactive waste repository will be operational from 2008, and it requires, among others, determination of the concept for packaging of decommissioning waste. A survey was conducted for this study on the status of decommissioning activities of nuclear facilities. Waste packaging and source-term considerations together with decommissioning waste characterization were investigated. Special attention was devoted to quantification of gas generation by metal corrosion and its following release from vaults: both metallic waste and its packaging were assessed. Finally, a conceptual design and preliminary safety assessment for the proposed disposal facility for decommissioning waste were also developed.

The Lithuanian project collected information on the main principles, criteria and methods for estimating inventories of contaminated and activated radioactive waste to be generated during dismantling of technological installations at Ignalina NPP. An updated version of the computer code “DECOM” was used to record the necessary information, process initial data and define contaminated waste based on their dose rates. Detailed information about the Ignalina NPP controlled area was used as the basis for performing analysis of possible waste generation during decommissioning of the facility. Modelling was performed of the activation components of the RBMK-1500 reactor core, and preliminary specific activity limits were derived, based on water pathway analysis, for packages with activated reactor components, such as the shielding and support plates of graphite stack, for disposal in planned near surface repository in Lithuania.

The study performed by the research group from St. Petersburg State Institute of Technology focused on the development of an integrated approach to decommissioning waste management, based on an assessment of inventory of decommissioning waste streams liable to disposal, development of advanced technologies for predisposal treatment and conditioning of decommissioning waste, investigations of barrier properties of materials for isolation of decommissioning waste, and development of indicators of reliability for barrier materials (both natural and man-made) intended for disposal of radioactive waste. As a result of the study, these issues have become a subject of constructive discussions at representative scientific and technical forums and in responsible organizations of the Russian Federation.

A safety re-assessment study was performed for the Mochovce disposal facility as part of the decommissioning project of Slovak NPP A-1. The purpose of this study was to update the already existing safety study and to demonstrate that acceptable levels of protection of human health and environment could be achieved when disposing both operational and decommissioning waste. Estimated inventories and characteristics of potential radioactive waste arising from the decommissioning of both A-1 and WWER reactors indicate some significant differences between the decommissioning and operational waste. Their impact on the safety of the disposal facility was assessed. Using the ISAM methodology, proposed by the IAEA, new features, events and processes (FEP´s) and scenarios were added to the existing analysis to accommodate the new waste forms in the inventory. The study summarizes safety assessment aspects related to co-disposal of operational and decommissioning waste in Slovakia.

The objective of the SFR project was to evaluate the inclusion of short-lived waste from decommissioning of the Swedish Nuclear Power Plants into the existing repository, SFR. Currently, the repository is licensed for short-lived low- and intermediate-level waste from operation and maintenance of the power plants. The scope of the feasibility study was to evaluate the impact of extending the existing repository to also include decommissioning waste. The decommissioning waste evaluated is similar to the operational, with the majority of the waste being contaminated scrap metal and concrete. Based on present plans to continue operating the power plants for 40 years followed by an early dismantling, the extended part of the facility should be in operation around the year 2020. The total volume of radioactive short-lived decommissioning waste from the 12 Swedish nuclear power plants has been estimated to be 150 000 m^3, mostly consisting of low level waste packed in freight containers. It is anticipated that some large components, e.g. steam generators and reactor pressure vessels, could be handled without packaging. The intermediate level waste could be disposed in remaining spaces of the existing silo, which has the most sophisticated engineered barriers. To allow for such a mix of waste, a new license needs to be granted for the extended repository. The modelling results indicate that the safety requirements of the extended repository could be met with simple disposal tunnels. Introducing chemical and engineered barriers would further reduce the peak release during the first years after closure.

The UK report provides a review of the extensive UK experience on corrosion of metals in a cement matrix. A large amount of metallic material present exists in decommissioning wastes both within the nuclear industry in the UK and world-wide that will have to be treated for disposal. In particular, the UK needs to treat steels, aluminium, Magnox (a magnesium aluminium alloy) and uranium metals. The preferred process for the treatment of these wastes in the UK is to encapsulate them in a matrix consisting of ordinary Portland cement that is typically blended with blast furnace slag or pulverised fuel ash. As water is present in the cement matrix even after hydration has occurred, corrosion reactions can take place. This has significant consequences, which include generation of gases (hydrogen and hydrocarbons) and expansive corrosion products; both of which may have a negative impact on the engineered barrier integrity. In particular the report addresses the corrosion behaviour of the different metals, and provides a discussion of general corrosion measurement techniques, long term extrapolation of behaviour and accelerated testing, and modelling of corrosion reactions and waste form evolution.

The Ukrainian study aimed at optimizing decommissioning scenarios for each reactor being operated in existing nuclear power plants. The following parameters were considered: decommissioning waste generation; time-dependent expenditures for decommissioning; and the number of personnel necessary for NPP decommissioning. Inventories of WWER-440 and WWER-1000 decommissioning waste were calculated for each NPP in Ukraine. Expenses are estimated for NPP decommissioning, including the disposal of decommissioning waste. Necessary annual deductions are proposed corresponding to uniform accumulation of costs for decommissioning.

The US paper discusses the differences between D&D wastes and routine operational radioactive wastes and provides a comparison of a source term analysis between operational and D&D waste streams. Understanding releases (source term) from decommissioning wastes is essential for disposal of these wastes in a cost-effective manner that is protective of human health and the environment. Decommissioning wastes often include surface contaminated building materials, activated metals, and large pieces of equipment that differ from traditional low and intermediate-level wastes in their origin, radionuclide content, and physical and chemical form. The report recommends that the characteristics of dismantling and decommissioning (D&D) wastes be incorporated into safety assessments.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Ali, S.S.	NRGP (K), India
Borrmann, F.	Bundesamt für Strahlenschutz (BfS), Germany
Carlsson, J.	Nuclear Fuel and Waste Management Company, Sweden
Dayal, R.	International Atomic Energy Agency
Godfrey, H.	British Nuclear Fuels Ltd, United Kingdom
Hanušik, V.	VUJE, Trnava, Slovakia
Harriague, S.	Comisión Nacional de Energía Atómica, Argentina
Husain, A.	Kinectrics Inc. Canada
Kim, C-L.	KHNP, Republic of Korea
Koryakovski, Y.	St. Petersburg State Institute of Technology, Russian Federation
Kumar, S.	Bhabha Atomic Research Centre, India
Lobach, U.	State Scientific Engineering Centre of Control Systems and Emergency Response, Ukraine
Nachmilner, L.	International Atomic Energy Agency
Nechaev, A.	St. Petersburg State Institute of Technology, Russian Federation
Poskas, P.	Lithuanian Energy Institute, Lithuania
Purtov, O.	State Scientific Engineering Centre of Control Systems and Emergency Response, Ukraine
Sullivan, T.	Brookhaven National Laboratory, United States of America
Takáts, F.	TS Enercon Kft., Hungary
Warnecke, E.	Bundesamt für Strahlenschutz (BfS), Germany
Wen, Z.	Beijing Research Institute of Uranium Geology, China

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