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Evaluation of the radioactive waste characterisation at the Olkiluoto nuclear power plant

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ABSTRACT

The aim of this study is to evaluate the physical, chemical and radiological characterisation, handling and documentation of the radioactive waste packages to be disposed of in the VLJ-repository at the Olkiluoto NPP. A comparison with the current practices in Europe, based on information from Sweden, Spain and Czech Republic, is made.

The report presents recommendations for STUK to harmonise the LILW waste management practises in Finland with those in Europe.

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Avainsanat: radioaktiivisen jätteen karakterisointi, radioaktiivisen jätteen käsittely, jätteiden hyväksyttävyysvaatimukset, keski- ja matala-aktiivisten jätteiden huolto

TIIVISTELMÄ

Julkaisussa arvioidaan Olkiluodon loppusijoituslaitokseen toimitettavan radio-aktiivisten jätepakkausten karakterisointi-, dokumentointi- ja laadunvarmistusmenetelmät ottaen huomioon Euroopan maissa tällä hetkellä voimassa olevat vastaavat käytännöt ja vaadittavan turvallisuustason. Euroopasta mukaan valitut maat ovat Ruotsi, Espanja ja Tsekinmaa.

Säteilyturvakeskukselle tehdään esitys niistä toimenpiteistä, jotka arvion perusteella katsotaan aiheelliseksi keski- ja matala-aktiivisten jätteiden huollon turvallisuuden saattamiseksi eurooppalaiselle tasolle niiltä osin kuin poikkeamia havaitaan.

CONTENTS

1 INTRODUCTION

The low and intermediate level waste (LILW) repository at the Olkiluoto nuclear power plant (NPP) site has been in operation since 1992. During this time the waste handling, characterisation and Quality Assurance (QA) have been developed into well-established routines.

The aim of this study is to evaluate the physical, chemical and radiological characterisation, handling and documentation of each main category of the radioactive waste packages including a comparison with the current practices in some European countries. The selected countries for comparison with Finland are Sweden, Spain and Czech Republic. Like in Finland they all have active LILW final repositories and the waste forms are comparable with those produced in Finland.

2 LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE MANAGEMENT POLICIES AND STRATEGIES

Waste management comprises all administrative and operational activities that are involved in the handling, treatment, conditioning, transportation, storage and disposal of waste [1]. The objective of waste management is to deal with the radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burden to future generations [2].

The basic requirements of the radioactive waste management are:

- Identification of the parties involved in different steps of the radioactive waste management, including waste generators and their responsibilities;
- A rational set of safety, radiological and environmental protection objectives, from which standards and criteria may be derived within the regulatory system;
- Identification of existing and anticipated radioactive waste, including their location, radionuclide content and other physical and chemical characteristics;
- Control of radioactive waste generation;
- Identification of available methods and facilities to process, store, and dispose of radioactive waste on an appropriate time-scale;
- Taking appropriately into account interdependencies among all steps of the in radioactive waste generation and management;
- Appropriate research and development to support the operational and regulatory needs; and
- The funding structure and the allocation of resources that are essential for the radioactive waste management, including decommissioning and, where appropriate, maintenance of repositories and post-closure surveillance [3].

Radioactive waste in which the concentration or the quantity of radionuclides is above the clearance level established by the national regulatory authority, but which has a radionuclide content and thermal power below those of high level waste (HLW), is addressed as LILW [4]. LILW is often separated into short lived and long lived waste. The term “long lived” refers to radionuclides with half-lives usually greater than 30 years. Short lived LILW management is discussed in this report.

A nation’s waste management strategy is influenced by many factors, e.g. its degree of radioisotope utilisation; its policy with regard to developing domestic capability for various parts of fuel cycle; status of the fuel cycle; the number of fuel cycle facilities; the country’s geology, geography and population density, which determine the availability of potential waste disposal sites; the country’s other resources, such as technically trained personnel, financial strength and sophisticated manufacturing capability; and public and governmental attitudes towards nuclear power [5].

International consensus exists on the fundamental principles and basis of standards, which are to be used in the disposal of radioactive waste. International co-operation through the IAEA, the European Commission and the Nuclear Energy Agency of the OECD (OECD/NEA) has played an important role in the development of such a consensus by contributing to a better informed and more objective debate on such a sensitive issue. The way such principles and standards are translated into operational procedures and regulatory requirements differ from country to country [6].

Disposal to surface structures or using shallow land burial or deep underground repositories such as depleted mines is practised widely with no obvious technical problems. Nevertheless, many nations see disposal of LILW as their most pressing waste management problem—principally because of the waste volumes involved and the difficulties faced in siting disposal facilities [5].

This chapter gives an overview of the LILW management policies and strategies in Europe and in Finland.

2.1 Radioactive waste management in Europe

Regulatory and institutional aspects

The organisation of a national waste management and regulatory authority is a reflection of the legislative and governmental structure within a country. Usually, a single authority is in charge of all safety matters involved in the approval and control of national disposal facilities for radioactive waste [7].

In many countries in Europe, national non-governmental agencies have been established to take responsibility for implementing disposal programmes. This is the case notably for Belgium (ONDRAS/NIRAS), France (Andra), Netherlands (COVRA), Spain (ENRESA), Sweden (SKB) and the United Kingdom (Nirex). In other countries, government agencies (e.g. the Bundesamt für Strahlenschutz in Germany) or the waste generators themselves are generally responsible for the waste disposal. The overall waste management system is under the control of regulatory bodies.

The institutional responsibility for the conditioning and disposal of radioactive waste varies from country to country, but in general it is the waste generator who ultimately has the responsibility for characterising each of the waste forms and package types produced. In particular, the waste generator is responsible for identifying the radionuclide inventory of the waste, for detailing the presence of any toxic constituents and for identifying the chemical and physical properties of the waste. The waste generator is responsible for providing this information to other organisations, which may subsequently handle, treat, store or transport the waste. Ultimately, the organisation consigning the waste to disposal must provide the final characterisation to the operator of the repository facility. It is generally accepted that an independent party should certify the adequacy of this characterisation data. This certification should include the verification that the waste characterisation data satisfy all of the repository waste acceptance requirements and all assumptions of the repository safety analysis or performance demonstration [8].

Repository concepts and operation

Repositories for LILW can broadly be categorised into two groups: near surface disposal facilities and disposal facilities located at rock cavity. Extensive operational experience exists for disposal of LILW in near surface disposal facilities. Some experience has been accumulated on the operation of rock cavity repositories for LILW.

Shallow ground repositories can be split into two groups: repositories with and without engineered barriers. Several shallow ground repositories without engineered barriers are in operation. The waste disposed of this way consists of very low level waste (VLLW) such as paper, plastics, wood, packing material, protective clothing and metal scrap. Many repositories for VLLW disposal consist of earthen trench in which waste packages are placed and then backfilled and covered with earthen material. Examples for this type of repository are repositories in Sweden (at the site of Oskarshamn and Forsmark nuclear power plants (NPP) and one at the Studsvik nuclear research centre), in Finland (at the site of Olkiluoto NPP) and the older trenches in the Drigg site in the UK.

The Drigg facility (UK), the Centre de la Manche and l'Aube facilities (France), the Dukovany facility (Czech Republic) and El Cabril facility (Spain) are examples for engineered shallow ground repositories. There is a trend to increase the engineered barriers in new repositories as has been done in France where the repository in the Centre de l'Aube will be equipped with more advanced engineered barriers than the Centre de la Manche. Also there is a tendency to dispose of waste at greater depths, even low level waste (LLW). In the United Kingdom, for example, the policy regarding disposal of LLW has been changed. No further shallow land repositories will be constructed, except for the expansion of the Drigg facility. Future LLW shall be disposed of together with intermediate level waste (ILW) in deep repositories. German plans to place all categories of radioactive waste in deep repositories. In some countries, siting shallow ground facilities is meeting strong public and political opposition. This may induce waste-disposal organisations to turn to deep repositories for the disposal of all waste categories [6]. The type of repository ultimately selected depends upon each country's geological conditions, specific disposal requirements and regulatory approaches. All these factors are tied to the design of the facility.

Rock cavity disposal facilities for LILW have been or are being constructed and operated in some countries in Europe: Germany (Morsleben and Konrad), Sweden (Forsmark), Norway (Himdalen) and Finland (Olkiluoto and Loviisa).

The repository programmes have progressed to different stages in different countries. Some countries have well established routines for disposal of LILW while others are currently developing or applying siting techniques and conducting the necessary parallel research to complement the disposal. In the latter countries (e.g. Belgium and Netherlands) interim storage of waste packages is required and must be arranged in such a way so as to ensure the integrity of radioactive waste packages and their suitability for further disposal after retrieval from a storage facility. In such circumstances long-term storage of these waste packages is necessary, the storage facility must develop a set of acceptance criteria of their own for waste packages generated under these conditions. [9]

2.2 Radioactive waste management in Finland

In Finland, four nuclear power units have been in operation for 18–22 years and generated more than 5700 m^3 of LILW. The accumulation of LILW from other sources (e.g. universities, hospitals, industry etc.) is only about one percent of that from the NPPs [10]. The largest radioactive waste producer in Finland outside the NPPs is the Technical Research Centre of Finland (VTT). The waste is produced in the operation of the 250 kW Triga MK II research reactor and in the research with radioactive substances.

Regulatory and institutional aspects

The Finnish Government issues general and the Radiation and Nuclear Safety Authority (STUK) detailed regulations for radioactive waste management. STUK also monitors adherence to the safety regulations. The waste producers are responsible for the required measures at their own expense.

General requirements for waste package acceptance are included in the regulatory guide issued by the STUK. The repository specific waste package requirements, the so-called waste type description, are included in the Final Safety Analysis Report of the particular repository and subject to approval by the regulator. They address the type of waste and its conditioning and packaging method, its radiological properties (dose rates, nuclide specific activities) and its potential adverse characteristics (e.g. flammability, swelling capacity, gas generation potential, concentrations of chemically aggressive substances). Some of the requirements are waste package specific (e.g. each waste package shall comply with the dose rate constraint) while others are specific to a waste emplacement room (e.g. the average of gas generation potential shall comply with the constraint for the particular emplacement silo).

All waste management activities are subject to the quality assurance programme of the waste producers and to a similar regulatory control as all their other radiation activities. This regulatory control includes review of the relevant documents and inspections to the waste management facilities.

Repository concepts and operation

The Finnish waste management policy is based on the disposal of LILW into rock cavity repositories located at the NPP sites. The waste from the NPP is conditioned, packed, and stored both temporarily and finally at the plants or in their immediate vicinity. The construction of the repository at the Olkiluoto plant began in 1988 and the operation of the repository commenced in May 1992. The construction of the repository at the Loviisa plant started in 1993 and the Government granted the operation licence for part of it in April 1998.

The designs of the Olkiluoto and Loviisa repositories are somewhat different mainly because of the local geological conditions. At Olkiluoto the host rock massif favours a vertical silo-type cavern, whereas at Loviisa horizontal tunnels are more suitable. The repository of Loviisa consists of two tunnels for dry maintenance waste and a

Table I. *The inventory of waste in the VLJ final storage and in the interim storage at Olkiluoto, 31 December 1998. [11]*

Waste category	Package	Quantity	Volume (m^3)
Intermediate, bituminised waste	200-l steel drum	6136	1227
Low-active maintenance waste etc.	200-l steel drum or 200-l steel drum compacted to 100 l	5356	1009
Mixed maintenance waste and scrap	1.3 m^3/1.4 m^3 steel boxes	449	624
Mixed maintenance waste and scrap	5.2 m^3 concrete box	101	525
Mixed maintenance waste and scrap	Stored without packing		267

Table II. *The inventory of waste in the final storage and in the interim storage at Loviisa, 31 December 1998. [11]*

Waste category	Package	Quantity	Volume (m^3)	Activity (GBq)
Used resin	Tank for interim storage	1	369	23 317
Liquid evaporator waste	Tank for interim storage	1	624	727
Maintenance waste	200-l steel drum	5240	1089	243

cavern for immobilised wet waste. One of the tunnels for dry maintenance waste is in operation. The cavern has been excavated but the construction and installation works will be completed later. Ion-exchange resins and other intermediate level waste in Loviisa are placed in storage tanks. There has not yet been any final treatment of the resin. The design of a plant based on a cementation process is underway.

The accumulated amounts of waste from the nuclear power plants are presented in Tables I and II. The volumes 2948 m^3 and 547 m^3 of waste were disposed of (at the end of 1998) in the repositories at Olkiluoto and Loviisa, respectively.

The waste from other sources is taken care of by the STUK and is stored in an interim storage, which is located in the premises of the VLJ-repository at Olkiluoto. The interim storage contains only dry waste. Liquid waste is conditioned by the STUK before it is transferred into the interim storage. The main part of the waste, taken care of by the STUK, originates from industry and consists of radiation sources released from instruments for measuring thickness, density etc. A minor part of the waste consists of radiation sources from hospitals, universities and research laboratories. The annual waste production is 1–2 m^3. The total volume of waste in the STUK's interim storage was 35 m^3 at the end of 1998 (Table III.).

Table III. *Accumulated waste in the STUK's interim storage (VLJ-repository at Olkiluoto) 31 December 1998. In addition to the listed material, a minor amounts of isotopes with a half-life < 5 years (mainly ^{55}Fe, ^{147}Pm and ^{204}Tl) were also deposited in the storage. [13]*

Nuclide	Activity (GBq)
^{3}H	25700
^{60}Co	366
^{85}Kr	447
^{90}Sr	148
^{137}Cs	1072
^{226}Ra	229.4
^{241}Am	239

A limited amount of waste is produced during the operation of the 250 kW Triga MK II research reactor at the VTT. All waste is interim stored at the VTT. The spent ion resin is stored in the plastic drums that were used for the delivery of fresh resin. Other types of radioactive waste are mainly packed into 200-l steel drums. Work on active material from nuclear power plants is carried out at the VTT. The active waste thereof is returned to the NPP and handled in accordance with the regulations for the waste normally produced by the NPP.

3 OVERVIEW OF THE REQUIREMENTS TO AND THE CHARACTERISATION OF RADIOACTIVE WASTE FORMS AND PACKAGES

Radioactive waste suitable for disposal in shallow ground and rock cavity facilities are of various types and contain variable amounts of individual radionuclides with different half-lives and radiotoxicities, as well as non-radioactive components. The characteristics of waste play an important role in the performance of the disposal system. The relative importance of waste form and package-related characteristics depends on the waste disposal options selected (i.e. shallow ground disposal or rock cavity disposal), the physical characteristics of the site and any special design of the repository or operating procedures selected for the facility [14].

This chapter gives an overview of the requirements and characterisation process. It also briefly describes the main waste form and package parameters, which need to be taken into account in the entire waste management process. Sampling during the operation of a nuclear facility could be helpful in obtaining sufficient data for the characterisation of the waste.

3.1 Overview of the characterisation process

Safe disposal of LILW is in the interest of the regulatory body, the repository operator and the waste generator. Figure 1 provides a description of the process to achieve acceptable waste package quality. The main parties and their main areas can be identified as:

- *Regulatory body* has an overall responsibility in a country to ensure compliance with safety requirements in any nuclear facility. Similarly for radioactive waste repository the regulatory body oversees the overall safety for workers, public and the environment.
- *Repository operator* should establish site specific waste acceptance criteria that encompass the requirements of the regulatory body and the repository. Confirmation of compliance with the site specific waste acceptance criteria is implemented through inspection and verification. Suitably qualified personnel at the waste generator site or at the repository shall perform them.

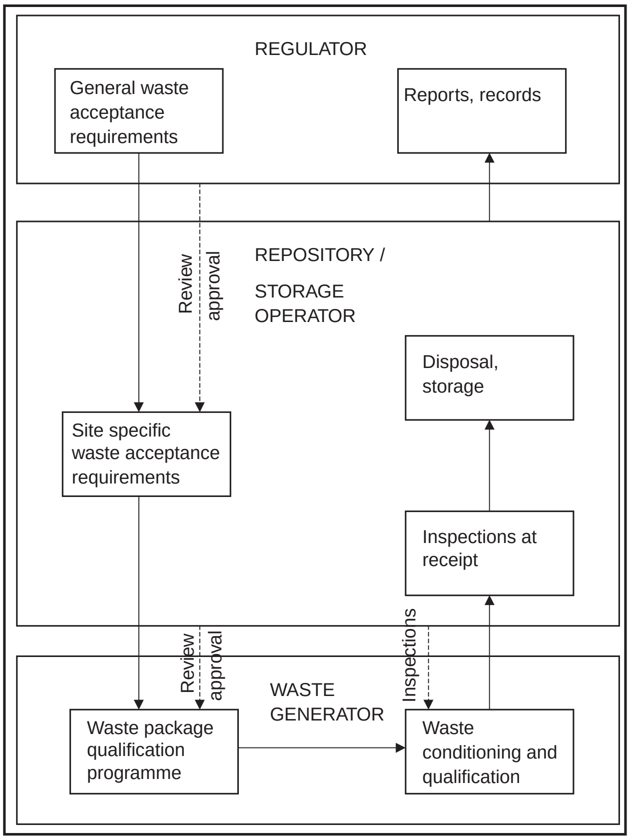

Figure 1. *The responsibilities of the regulator, waste generator and repository operator. [15]*

- *Waste generator* has the primary responsibility to meet the waste acceptance criteria established by the repository operator and the transportation requirements. Each waste generator shall establish a quality assurance programme describing his overall plan, approach, methodologies, and quality assurance methods that will be implemented to meet the repository operator's waste acceptance criteria. [15].

It should be noted that further waste conditioning may be performed by either the waste generator or the repository operator [16]. Regulatory authorities and/or other government agencies in individual countries grant the approvals required for disposal of radioactive waste. The overall safety of a near surface repository depends on the combined characteristics of the site, the repository and the waste package and is of concern to the regulatory body, waste generator, repository operator and also to the public.

The characterisation of waste is performed for numerous reasons by waste management organisations, including the need for a safety assessment of treatment, conditioning and interim storage facility. Waste characterisation is also necessary to qualify the treatment/conditioning processes and to perform quality control of waste forms and packages during conditioning. Ultimately, the organisations responsible for the disposal of radioactive waste use waste characterisation data as a basis for the acceptance of such waste in the repository [8].

There are three phases associated with the repositories, i.e. pre-operational, operational and post-closure. During the first two phases, the safety of the repository relies to some extent on its appropriate and specific systems for the waste package, including the associated quality management system. However, it is during the first phase that the waste acceptance criteria will be established although they should be reviewed and updated from time to time [15].

The waste acceptance criteria (WAC) will include general criteria issued by the regulatory body and the site specific criteria specified by the repository operator [17]. The WAC are the conditions imposed on a waste producer by the regulator and/or operator of a waste handling, transportation, storage, processing and/or disposal service. The WAC usually specify such things as the required physical form of the waste, the maximum levels of radioactivity, the packaging requirements, etc. as well as what kind of waste forms are excluded from their service.

Since a waste package consists of a waste form and a container, a specific set of technical requirements can be addressed to them separately and to the waste package as a whole [9]. For the waste form, these criteria concern, but are not limited to, the following:

- waste composition
- chemical durability
- immobilisation and/or stabilisation
- structural stability
- respirable fraction
- distribution of activity

The WAC for the waste containers may cover the following parameters:

- pressure strength
- mechanical integrity
- properties affecting primary confinement
- venting
- compatibility with the waste form

Each waste package must meet a general set of criteria in addition to the requirements specific to the waste form and the waste container. The WAC applied for the waste packages generally include the following:

- seal integrity
- free liquids
- gas generation
- flammability
- radionuclide inventory
- fissile mass
- decay heat
- radiation dose rate and surface contamination
- configuration and weight
- identification

3.2 Important parameters of waste packages

A waste package is the product of conditioning that includes the waste form and any container and internal barriers (e.g. absorbing materials and liner), as prepared in accordance with the requirements for handling, transportation, storage and/or disposal.

Table IV. Waste form requirements—properties related to radioactivity [8].

Property	Requirement
Total activity	Total activity of alpha, beta and gamma emitters in the waste package must be determined.
Radionuclide composition	Isotopic composition of radionuclides in the waste form must be determined.
Fissile mass and criticality safety	Inadvertent criticality must be prevented. Fissile mass and its distribution within the waste package must be determined.
Thermal power	Thermal power must be quantified for individual waste packages to ascertain of the physical integrity of the waste form and packages.
Radiation stability	Radiation stability of the waste form must be quantified if the waste form is required to remain stable in storage or disposal.
Homogeneity	Distribution of radionuclides may need to be determined for critically safety and stability of the waste form.
Surface dose rate and surface contamination	Must be characterised for ALARA reasons and in order to comply with transport regulations.

Safety must include both radiological and non-radiological aspects. Radiological safety includes radiation and contamination levels associated with the waste package and in particular the performance of the total waste package in terms of potential release of activity into the environment. Non-radiological safety aspects will include both conventional safety issues associated with mechanical handling and the safety of the package and its contents in terms of exclusion of dangerous materials such as explosives and significant levels of toxic waste [15]. It is to be emphasised that the parameters of key importance may be different for one disposal route than for the other. Ultimately, this is a matter that must be agreed upon between the waste generator, the repository operator, the regulatory authorities and/or other competent bodies [8].

If the repository site is located away from the generator site, transport of the waste from the waste generator to the repository may impose certain additional requirements on waste packages [15].

Properties related to radioactivity

Radiological waste characterisation involves detecting the presence of individual radionuclides and quantifying their inventories in the waste. This can be done by a variety of techniques, depending on the waste form, radionuclides involved and level of detail and accuracy required. For example, a simple radiation dose rate measurement will give an indication of the total quantity of gamma emitting radionuclides in a waste package, but will not identify individual radionuclides or their concentrations. Gamma spectroscopy will identify the individual gamma-emitting radionuclides and, when properly calibrated, their quantities as well. Other techniques, such as active or passive neutron interrogation, alpha spectroscopy, and liquid scintillation counting are used for other classes of radionuclides. The preferred methods are often referred to as “non-destructive” or “non-invasive”, since they do not involve opening the waste package to take samples. The properties related to radioactivity and the waste form requirements are listed in Table VI.

Chemical properties

In radioactive waste usually radioactive substances form only a minor component. Other components, the majority, are non-radioactive and harmless (e.g. concrete, water, steel) but some maybe chemotoxic. Chemotoxic substances in radioactive waste maybe:

Inorganic:

Heavy metal isotopes, some packaging material (Pb, Cu), neutron absorbing materials (Cd, B), special chemicals used in various steps of the nuclear fuel cycle or in research activities.

Organic:

Solvents, degradation products of plastics, decontamination chemicals, cyanides

Because these materials do not have finite half-life they may pose a greater long-term risk to the health and safety of the public than the radioactivity of the waste [14].

Table V. *Waste requirements, chemical properties [8].*

Property	Requirement(s)
Chemical stability	Leachability or solubility of the waste form must be assessed. Desirable durability of the waste form or waste package must be ensured.
Chemical composition	Chemical composition should be determined to identify hazardous or toxic substituents.
Pyrophoricity	Prohibited pyrophorics within the waste package items must be excluded or, if this is not possible, identified.
Ignitablity	Presence of ignitable materials should be verified by test, and treatment methods applied to eliminate this characteristic.
Reactivity	Potentially reactive chemicals or metals should be identified and treatment methods applied to eliminate this characteristics.
Corrosivity	Potential corrosivity should be determined by analytical characterisation and eliminated.
Explosivity	Explosive material should be identified and controlled.
Chemical compatibility	Individual waste chemical constituents should be analysed for compatibility prior to treatment and conditioning.
Gas generation	Potential for the generation of flammable gases should be identified and assessed.
Toxicity	Toxic elements or chemical constituents in the waste form should be determined and either eliminated or reduced.
Decomposition of organic waste	Complexing agents and cellulose in the waste package should be determined. Decomposition of organic waste should be avoided.

Table VI. *Waste form requirements, physical properties[8].*

Property	Requirement(s)
Homogeneity	Slurry type waste to be homogeneously dispersed in the product. Solid waste to be immobilised by the matrix with no regions of non-encapsulated material at the product surface.
Voidage	Minimise voidage to ensure that waste is immobilised and do not affect other properties, such as strength and permeability.
Permeability	Permeability should be sufficiently high to allow gases to be vented, but low enough to restrict the release of radionuclides.
Porosity	Low porosity is desirable to improve the microstructure and to minimise the release of radionuclides.

In disposal facilities (both near surface and geological), the ability of the waste form to resist solubility and leaching is important in predicting the repository performance. If the waste container or immobilisation medium is required to provide a long term confinement in the repository environment, then the container or waste form integrity needs to be quantified. Chemical waste characterisation involves the determination of the chemical components and properties of the waste. This is most often done by chemical analysis of a waste sample [7]. Table V summarises the requirements for chemical properties of the waste forms.

Radiological and chemical waste characterisation can also be inferred from process knowledge. For example, if you are a medical researcher who only uses a few particular radionuclides under controlled experimental conditions, or a manufacturer who uses a particular chemical, then you can determine from your “knowledge of the process” which radionuclide(s) and/or chemicals are present in your waste.

Physical properties

Some of the physical properties can be affected by such factors as the presence of free liquids, chelating and complexing agents and gas mixtures. Physical characterisation involves inspection of the waste to determine its physical form, strength, etc. Inspection of closed waste packages can be done using a variety of techniques, such as radiography (X-ray) and sonar. Table VI summarises the requirements for physical properties of the waste forms.

Table VII. Waste package requirements, mechanical properties. [8]

Property	Requirement
Mechanical strength	Should exceed a minimum for normal operations.
Dimensional stability	Dimensional changes should be minimised to ensure that the waste form maintains its integrity over a prolonged time.
Impact resistance	Waste package should be capable of withstanding an impact in an accident.

Table VIII. Waste package requirements, thermal properties [8].

Property	Requirement
High temperature stability	Waste package should withstand an external fire and fragmentation at high temperatures.
Activity release at elevated temperature	Activity release from the waste package as a result of an external fire should not exceed the safety limits.
Thermal cycling	Thermal cycling should not cause instability or significant reduction in the strength of the waste package.

Mechanical properties

The mechanical integrity of a waste package is important as it gives improved and more predictable performance under transportation, handling, storage and disposal. This is particularly important in relation to particulate activity releases from impacts and fires. Table VII summarises the requirements for mechanical properties of the waste package.

Thermal and biological properties

The best thermal performance would be obtained with a waste form of very low thermal conductivity. However, heat generated within a waste form, and overall heat, must be conducted to the exterior of the drum or box. The repository and the packages have to be designed according to the thermal conductivity of the waste form. Table VIII summarises the requirements for thermal properties of the waste package.

Although the nature of many waste forms, especially after conditioning, is hostile to biological activity, it is nevertheless a further factor, which needs to be taken into account. Waste forms with high organic content may undergo biological degradation. The possible effects on the physical and chemical properties need to be considered.

3.3 Sampling

It is frequently necessary to determine important properties of the waste by sampling. Sampling can be done at two stages of the conditioning process, i.e. during the actual conditioning or in the conditioned waste package. Usually, samples taken on-line are considered to be the most representative. Obviously, the investigation of samples, taken from full size waste packages, yields more relevant information than small samples taken on-line. The effects of specific thermal, mechanical or physical phenomena, e.g. settling, are not reproduced in the on-line samples [8].

For homogeneous waste such as evaporation concentrates or sludge, or bituminised or cementitious products, it should always be possible to take one sample which is representative of the waste provided it can be established that the process control consistently results in a sufficiently low product variance in the chemical and physical properties. Heterogeneous waste includes contaminated industrial waste (laboratory waste, leaded rubber gloves, combustible materials etc.) and waste from decontamination and decommissioning (contaminated piping, ductwork, concrete etc.). Heterogeneous waste streams present real difficulties in terms of statistical, analytical and radiological characterisation [8].

3.4 Quality assurance for radioactive waste packages

The waste packages should be prepared by the waste generator in a way to meet the waste acceptance criteria. On the other hand, it is essential for the repository operator to assure compliance of waste packages to be disposed of, with the waste

Table IX. *Examples of waste package inspection and verification at the repository, prior to final acceptance by the repository operator [15].*

Administrative checks	Visual checks	Direct measurements
Completeness of consignment record	Package labelling/identification	Radiological contamination survey
Package identification	Tamper seals	Radiation dose survey
Weight	Package closure	Weighing
Activity limits	External package condition	Tightness (torque) testing
Dose rate		Radiography/tomography
Surface contamination		Activity measurement
Shipment number		Container integrity survey
Special conditions		Destructive testing
Container type		
Fissile mass		

acceptance criteria set for a repository or disposal system [15]. The objective of a quality assurance programme (QAP) is to ensure that waste packages comply with the waste acceptance criteria and the disposal requirements as approved by the appropriate national authority. Assurance can be achieved through a systematic inspection of waste packages (checking documents, destructive and non-destructive examination of waste packages) based on a set of detailed specifications derived from the acceptance requirements. Key elements addressed in the QAP should be:

- strategy and quality assurance methods used to verify the compliance with the repository waste acceptance criteria
- organisational and management structure including the roles and responsibilities of the key personnel involved with waste generation, packaging and compliance activities
- overview of the waste generating and conditioning processes that must be performed to meet the repository WAC
- description of the waste generator quality assurance system and controls being implemented (internal surveillance, audits, calibration programmes, reporting of deficiencies) to meet the WAC and other operational requirements of the repository
- description of the methods used to characterise the final waste form for disposal (characterisation tests on non-radioactive mock-ups, sample collection methodologies, laboratory methods, data interpretation, record keeping)
- description of the methods used for the assessment of activity of each waste package
- description of the methods used for packaging the waste for transport and disposal. Procedures may be written according to general guidelines defined by the repository operator in the WAC
- description of the methods for transmitting data to the repository operator for review and approval prior to shipment
- description of the methods for records management including retention times [15].

Waste generators must prepare a QAP for each waste package category. In general authorisation for acceptance of each waste package category to the repository is the responsibility of the repository operator [15]. The majority of controls to ensure the acceptability of waste packages for disposal are generally carried out prior to receipt of the packages to the repository itself. These include those identified in the QAP and the inspections carried out at the waste generating system. However, it is important to carry out some inspections on receipt of waste packages at the repository, as this is the last opportunity to verify that the waste package meets the criteria for acceptance. Examples of these are given in Table IX. Inspections should be aimed at demonstrating to the repository operator and also to the regulator that the packages meet the required specifications. Independent and adequate inspection and verification of the waste generator's data for waste to be disposed of should be the responsibility of the repository operator.

4 REQUIREMENTS AND CHARACTERISATION AT OLKILUOTO AND REPOSITORIES IN SELECTED EU COUNTRIES

This chapter gives overview of the LILW requirements and characterisation at Finland (Olkiluoto), Sweden (SFR), Spain (El Cabril) and Czech Republic (Dukovany) repositories. The Finnish regulations for disposal of LILW include rather stringent requirements on the safety of a repository, as well as detailed guidance for the preparation of the Final Safety Analysis Report (FSAR) [10]. Most of the information from Sweden, Spain and Czech Republic were collected using a questionnaire (Appendix). A short description of each repository is given first.

Finland

The VLJ repository is an underground disposal facility for operational LILW generated by the Olkiluoto NPP and is located 700 m away from the NPP. The repository consists of two silos excavated at a depth of 60–100 meters in the bedrock. The silo for LLW is a shotcreted rock silo. For ILW a reinforced concrete silo has been constructed inside the rock silo.

Sweden

The Swedish system for management of radioactive waste consists of a ship based transportation system. The repository is SFR, the Swedish Final Repository for operational waste. The repository has been in operation since 1988 and receives short lived LILW from operation and maintenance of the Swedish NPPs. It also receives small quantities of similar waste from research, medicine and industry in Sweden. The main waste producers in Sweden are the NPPs (Forsmark, Oskarshamn, Barsebäck and Ringhals).

The repository is built in the bedrock under the Baltic Sea close to the Forsmark NPP. A 50-metre layer of rock covers the repository caverns under the seabed. The first stage of SFR, which is in operation, includes buildings on the ground level, tunnels, operating buildings and disposal caverns for 60 000 m^3 of waste. The second stage for approximately 30 000 m^3 is planned to be built and commissioned after the year 2000. By the end of 1997, the amount of waste disposed of was about 23 000 m^3 [18]. The waste materials are conditioned at the power plants, at the Central Interim Storage Facility for Spent Nuclear Fuel (CLAB) or at Studsvik [19]. There are different caverns for different ILW and LLW packages in the SFR. The ILW packages (drums and boxes) containing the most of the activity are disposed of in a concrete silo. Waste packages containing a minor part of the activity content are disposed of in 160 m long caverns: BLA for LLW ISO containers, BMA for ILW drums and boxes and BTF for concrete tanks.

Spain

The El Cabril facility has the capacity to meet the LILW disposal needs of Spain until the second decade of the 21st century. El Cabril has been in operation in Spain since October 1992. It is located on the site of a former uranium mine, where radioactive waste has been stored since 1961. As of December 1997 some 14 000 m^3 of conditioned waste had been disposed of. The repository receives on the average 2 000 m^3 of LILW per year. El Cabril is very similar in design to the French facility at Centre de L'Aube, with preformed cells, mobile crane and weatherproof disposal building. The 200-l drums go into concrete containers filled with mortar, the blocks go into the disposal cells, each having a capacity for 320 containers. The disposal cells have a drainage system into tunnels running underneath.

Czech Republic

At this time three near surface repositories are in operation. A shallow land disposal facility for LILW from Czech nuclear power plants is situated on site of Dukovany NPP. The other two subsurface repositories are for institutional radioactive waste from the hospitals, research institutes, industry etc. Both repositories are placed in abandoned mines.

Hostim repository was the first one in former Czechoslovakia, which was put into operation. Since 1953 till 1965 about 400 m^3 of institutional waste was placed into two galleries in an abandoned limestone mine several tens of metres below the surface.

In 1964 a new repository, Richard II, went into operation. It was also situated in an abandoned limestone mine 40–60 m below the surface. This facility is destinated for institutional waste with exception of those contaminated by natural radionuclides. The total volume of the repository exceeds 16 000 m^3. Out of this figure 8600 m^3 is available for disposal and the rest is used for communications (gangways and corridors). By 1995 about 5 200 m^3 were filled with waste so that about 2 800 m^3 still remain free for disposal.

The third repository, Bratrstvi, is used for waste contaminated only by natural radionuclides. The facility was built in an abandoned uranium mine. During its operation that started in 1974 about 250 m^3 of conditioned waste is disposed of. The remaining capacity of 40 m^3 of waste will be filled within 3–5 years.

Surface repository for LILW is in operation since 1993 at the Dukovany site. The operation of the Dukovany NPP produces annually 400–500 m^3 of conditioned LILW. The Dukovany disposal site is 500 m from unit 4. The repository spaces are formed by two double-rows of vaults, the dimension of each approximately 6 × 6 × 18 m^3. Sealed concrete walls are 70 cm thick. The base of the vaults is sloped to facilitate drainage into collecting reservoirs. Each vault has 1200 barrels (200 l). The existing volume, 112 vaults with a capacity close to 60 000 m^3, can be extended by constructing 8 new double rows. Its radioactivity shall not exceed 10^{16} Bq. Final multi-layer capping shall be installed after a double-row is filled with waste. Void spaces in vaults are to be grouted by a concrete mortar.

4.1 Waste acceptance requirements

The WAC are the most important technical requirements to be met by the waste packages. The acceptance criteria are specific to a disposal facility or long-term storage. They may either cover a broad range of different products or be established for individual types of waste packages. Typically the WAC will include aspects on radionuclide content, the physical, chemical and biological properties of the waste and on the nature of the waste containers [20].

4.1.1 Waste containers

The waste containers are primarily designed to provide a complete isolation of the waste matrix during a specified period of time. The container has a primary function to contain the waste during filling, storage, handling, transportation and emplacement in a repository, thereby avoiding the formation of respirable fines. The container has to resist repository and possible impact pressures which will be non-uniform and which could be concentrated around the void spaces in the containment [8]. In some cases the container is designed to fulfil the additional function of providing radiation shielding during emplacement operations. [6] The demands on the waste container for LILW are not particularly high. The safety of the system is ensured through the engineered and geological barriers of the disposal system and by the low activity contents in these types of waste.

Olkiluoto

The container types used for storage of LILW at Olkiluoto are mainly:
- plastic sacks
- bales (1.2 m wide, 0.7 m high and 0.7 m long)
- 200-l steel drums
- 1.3 m^3 or 1.4 m^3 steel boxes
- 5.2 m^3 or 3.9 m^3 concrete boxes for 12 or 16 drums (or 32 compressed drums).

Combinations of the various container types are also used, like plastic sacks in steel drums and steel drums in concrete boxes. The quality of steel drums is mainly the same for all drums, but minor variations occur. A typical drum is made of 1.0 mm thick, cold rectified Fe37B-steel covered with rust-preventing paint. The weight of the drum (empty) is 18.4 kg. The steel boxes are made of the same material with thickness of 2.0 mm. The weight of the empty box is 165 kg.

SFR

The container types used for storage of SFR at Forsmark are mainly:

- 200-l steel drums
- concrete box (cubic 1.2 m side length)
- steel box (cubic 1.2 m side length)
- concrete tank (1.3 m wide, 2.3 m high and 3.3 m long)
- standard freight containers (ISO)

El Cabril

The container types used for storage of El Cabril are mainly:

- 200-l steel drums
- concrete box (cubic 2.2 m side length)

Dukovany

To date the only authorisation is for the use of a 200-l steel drum.

4.1.2 Waste packages

The waste package, refers to a product of waste conditioning and packaging process, it includes the waste in any form (unconditioned or conditioned), the container and the possible internal barriers (e.g. absorbing materials and liner), as prepared in accordance with the requirements for handling, transportation, storage and disposal [1].

Olkiluoto

LILW is produced during maintenance work and in the purification system of process water. In Olkiluoto LILW is classified as follows:

- Maintenance waste (working clothes, papers etc.)
- Bituminised waste (ion exchange resins, evaporated slurries)
- Metal scrap
- Filter rods
- Solidified liquid waste
- Waste from other sources than NPP

SFR

- 200-l steel drums
  - Bitumen solidified ion exchange resins (I/X)
- concrete box (cubic 1.2 m side length)
  - Cement solidified I/X
  - Cement backfilled scrap material
  - Cement solidified sludges
- steel box (cubic 1.2 m side length)
  - Cement solidified I/X
  - Cement backfilled scrap material
  - Bitumen solidified I/X and evaporator concentrate
- concrete tank (1.3 m wide, 2.3 m high and 3.3 m long)
  - Dewatered low level I/X
- standard freight containers (ISO)
  - Low level scrap and trash

El Cabril

To date the only authorisation is for the use of a cube-shaped concrete container measuring approximately 2.2 × 2.2 × 2.2 metres where 200-l steel drums are placed. The waste is conditioned inside the drum. Different packages (200-l drum):

- homogeneous waste incorporated in a solid matrix (level 1)
- heterogeneous waste immobilised in a solid matrix (level 1)
- homogeneous waste incorporated in a solid matrix (level 2)
- heterogeneous waste immobilised in a solid matrix (level 2)

The drums are transferred to disposal units. Each unit (concrete container) has a capacity for eighteen 220-l drums. These are the basic disposal units at the El Cabril Centre. The disposal units are transferred to disposal cells (each cell has a capacity for 320 units).

Czech Republic

Three different repositories are in operation:

- URAO Dukovany, solid and solidified waste (bituminised)—only for NPP waste
- BRATRSTVI, solid and solidified waste (cemented)—natural radionuclides
- RICHARD, solid and solidified waste (cemented)—artificial radionuclides

4.2 Requirements for each waste package

4.2.1 Packaging and conditioning methods

Conditioning of LILW involves those operations that transform radioactive waste into a form suitable for handling, transportation, storage and disposal. The operations may include immobilisation of radioactive waste, placing the waste into containers and providing additional packaging.

Some waste types can be readily disposed of into shallow ground or rock cavity repositories in the form in which they are generated, while other waste types require some type of treatment, conditioning or packaging to make them acceptable for disposal [14].

Olkiluoto

The LLW is compacted in 200-l steel drums. Metal scrap is packed without treatment into concrete boxes. The intermediate level ion-exchange resins are bituminised and transferred to 200-l steel drum. The waste drums are transferred into concrete boxes of a size of either 5.2 m^3 or 3.9 m^3, containing 16 and 12 drums, respectively. Large building parts are transferred into steel containers, which can be placed in the above types of concrete boxes.

Maintenance waste

All the maintenance waste (i.e. protective plastics, filters, waste cloths, paper, wood, broken tools etc.) is packed in plastic sacks or directly into 200-l drums and transported to the waste facility of the power plant for measurement of radioactivity and further treatment. Large objects are packed into 1.3 m^3 steel boxes (earlier 1.4 m^3 steel boxes) or concrete boxes of either 5.2 m^3 or 3.9 m^3.

The maintenance waste is first packed into plastic sacks and the dose-rate at the sack surface is measured. If it is below 10 µSv/h the sacks will be further monitored for clearance. Sacks, which exceed the threshold value of the dose rate measurements, are stored in 16 m^3 containers until they are packed into 200-l steel drums. Before packing the non-compressible material (e.g. wooden structures and metal) is removed from the sacks. A hydraulic press with a maximum force of 120 kN is used in the packing. The drums are further pressed with another press of 200 kN, enabling to halve the original drum volumes (since 1995). The non-compressible materials are packed into 200-l steel drums, 1.3 m^3 steel boxes or concrete boxes.

Bituminised waste

The slightly active waters are purified in either ion exchangers or evaporators. Both the ion exchange resins and the evaporator concentrates are pumped into storage tanks for further treatment. After storage the intermediate level ion-exchange resins and liquid waste slurries are dried, bituminised and transferred into 200-l steel drums. The waste drums are transferred into concrete boxes of either 5.2 m^3 or 3.9 m^3, containing 16 or 12 drums, respectively.

Filter rods

The used filter rods were packed earlier into 200-l steel drums or 1.4 m^3 steel boxes without treatment. Now the filter rods are packed, without treatment, into 1.3 m^3 steel boxes.

Metal scrap

Metal scrap is packed without treatment into 1.3 m^3 steel boxes (earlier into 1.4 m^3 steel boxes) which are packed into 5.8 m^3 or 4.4 m^3 concrete boxes. The metal scrap can also be packed straight into concrete boxes without steel boxes. Also 200-l drums are used for packing small-size metal scrap. The pipes and other compressible materials

Table X. Some examples of waste packages accepted in the SFR repository [21]. In 1996, 39 waste types (of a total of about 50) were accepted for disposal. [18]

SFR Typ nr	B	F	O	R	S	Container and waste form
1				X		Concrete box (cemented LLW I/X)
2			X	X		Concrete box (cemented ILW I/X)
5	X					Steel drum (bituminised LLW I/X)
6	X					Steel drum (bituminised ILW I/X)
7	X		X			Concrete tank (dewatered LLW I/X)
12	X					Standard freight container (low level scrap and rubbish)
13					X	Steel drum (ash)
14					X	Steel drum (rubbish and scrap)
17		X				Concrete box (bituminised LLW I/X in steel drums)
18		X				Concrete box (bituminised ILW I/X in steel drums)
21	X	X		X		Steel drum (rubbish and scrap)
23		X	X	X	X	Concrete box (rubbish and scrap)

B= Barsebäck
F= Forsmark
O= Oskarshamn + CLAB
R= Ringhals
S= Studsvik

are compacted before packing. A hydraulic press with a maximum force of 200 ton is used in the packing. The metal scrap can be also cut into pieces with a cutting machine. The boxes are packed so that the metal scrap is of the same activity, type and origin in one box. Big pipes, which are contaminated only in the inner surface, are sealed in the ends and interim stored without packing.

Solidified liquid waste and slurries

The liquid waste is produced mainly during washing and decontamination operations. The liquid waste can be organic dissolvents, water, acids etc. The liquid waste and slurries are collected into the drums, stored, classified, combined and later bituminised with ion-exchange resins. The waste that can not be bituminised, is cemented or solidified using special solidification agents.

Waste produced outside the NPPs

The waste produced outside the nuclear power plants consists of small amounts of dry materials and originates mainly from various instruments for measuring thickness, density etc. in industry. A minor part of the waste consists of contaminated metal scrap, radioactive sources, and minor objects containing radioactive paint (e.g. compasses and emergency exit signs). Minor objects, like compasses, are packed into 200-l steel drums, or small steel or lead packages. Large metal objects, like parts from cobalt therapy instruments are stored without packing on floors or shelves [21]. There are five waste types that are interim-stored in the VLJ-repository.

- Cemented waste in the drum
- Waste in the drum
- Waste with radiation shields in the drum
- Waste with radiation shields in the steel boxes
- Large waste objects with radiation shields

The waste in the interim storage consists of solid material. The material is well documented and for every nuclide there is information about its chemical form.

SFR

The waste materials are conditioned at the NPPs, at the Central Interim Storage Facility for Spent Nuclear Fuel (CLAB) or at Studsvik. Ion exchange resins are incorporated in either cement or bitumen. Scrap from maintenance work is treated in the same way if needed.

The NPP waste consists of wet ion exchange and filter material, sludges, scrap and rubbish. The wet waste is solidified with cement or bitumen before transport to SFR. Rubbish and scrap are compacted, incinerated or molten before packing. Some examples of waste packages accepted in the SFR repository are given in Table X. A large part of the scrap can be exempted after decontamination. Material with very low activity content

Table XI. *Upper activity limits for the most important nuclides in different waste packages for Olkiluoto. The limits refer to the moment when the waste is transported to the VLJ final disposal. [23]*

	ILW-silo				LLW-silo		
	Bitumi-nised	Filter rods and metal scrap			Maintenance waste, filter rods, metal scrap and solidified liquids.		
Nuclide	Steel Drum (Bq)	Steel Drum (Bq)	Steel box (Bq)	Concrete box (Bq)	Steel Drum (Bq)	Steel box (Bq)	Concrete box (Bq)
^{14}C	$3 \cdot 10^{8}$	$3 \cdot 10^{7}$	$2 \cdot 10^{8}$	$3 \cdot 10^{8}$	$1 \cdot 10^{6}$	$5 \cdot 10^{6}$	$5 \cdot 10^{6}$
^{60}Co	$3 \cdot 10^{11}$	$3 \cdot 10^{10}$	$2 \cdot 10^{11}$	$3 \cdot 10^{11}$	$1 \cdot 10^{10}$	$5 \cdot 10^{10}$	$5 \cdot 10^{10}$
^{59}Ni	$2 \cdot 10^{8}$	$2 \cdot 10^{7}$	$1 \cdot 10^{8}$	$2 \cdot 10^{8}$	$1 \cdot 10^{7}$	$5 \cdot 10^{7}$	$5 \cdot 10^{7}$
^{63}Ni	$3 \cdot 10^{10}$	$3 \cdot 10^{9}$	$2 \cdot 10^{10}$	$3 \cdot 10^{10}$	$2 \cdot 10^{9}$	$1 \cdot 10^{10}$	$1 \cdot 10^{10}$
^{90}Sr	$3 \cdot 10^{10}$	$3 \cdot 10^{9}$	$2 \cdot 10^{10}$	$3 \cdot 10^{10}$	$1 \cdot 10^{8}$	$2 \cdot 10^{8}$	$2 \cdot 10^{8}$
^{99}Tc	$2 \cdot 10^{7}$	$2 \cdot 10^{6}$	$1 \cdot 10^{7}$	$2 \cdot 10^{7}$	$5 \cdot 10^{4}$	$1 \cdot 10^{5}$	$1 \cdot 10^{5}$
^{129}I	$9 \cdot 10^{4}$	$9 \cdot 10^{3}$	$6 \cdot 10^{4}$	$9 \cdot 10^{4}$	$5 \cdot 10^{2}$	$1 \cdot 10^{3}$	$1 \cdot 10^{3}$
^{135}Cs	$9 \cdot 10^{5}$	$9 \cdot 10^{4}$	$6 \cdot 10^{5}$	$9 \cdot 10^{5}$	$5 \cdot 10^{3}$	$1 \cdot 10^{4}$	$1 \cdot 10^{4}$
^{137}Cs	$3 \cdot 10^{11}$	$3 \cdot 10^{10}$	$2 \cdot 10^{11}$	$3 \cdot 10^{11}$	$1 \cdot 10^{9}$	$2 \cdot 10^{9}$	$2 \cdot 10^{9}$
^{238}Pu	$3 \cdot 10^{7}$	$3 \cdot 10^{6}$	$2 \cdot 10^{7}$	$3 \cdot 10^{7}$	$3 \cdot 10^{4}$	$6 \cdot 10^{4}$	$6 \cdot 10^{4}$
$^{239,240}Pu$	$3 \cdot 10^{7}$	$3 \cdot 10^{6}$	$2 \cdot 10^{7}$	$3 \cdot 10^{7}$	$6 \cdot 10^{4}$	$1 \cdot 10^{5}$	$1 \cdot 10^{5}$
^{241}Am	$3 \cdot 10^{7}$	$3 \cdot 10^{6}$	$2 \cdot 10^{7}$	$3 \cdot 10^{7}$	$6 \cdot 10^{4}$	$1 \cdot 10^{5}$	$1 \cdot 10^{5}$
$^{243,244}Cm$	$3 \cdot 10^{7}$	$3 \cdot 10^{6}$	$2 \cdot 10^{7}$	$3 \cdot 10^{7}$	$2 \cdot 10^{4}$	$4 \cdot 10^{4}$	$4 \cdot 10^{4}$

that can not be exempted is buried at Ringhals, Oskarshamn, Studsvik and Forsmark, while rubbish and scrap with higher activity are disposed of at SFR.

El Cabril

Waste treatment is normally performed in two stages:
- Segregation into types, depending on their characteristics.
- Volume reduction in order to concentrate all activity into a small fraction of the original waste (*Solids*: Decontamination, compaction, incineration, filtration etc. *Liquids*: Evaporation, ion exchange, precipitation, incineration etc. *Gaseous*: Filtration, absorption, adsorption etc.).

Conditioning implies incorporating the waste into a solid, stable matrix and packaging it into containers allowing for appropriate handling. Conditioning is done into matrices of concrete, asphalt or polymers in metallic or concrete drums or containers.

Dukovany

Liquid effluents are evaporated and bituminised while solid waste is segregated, compacted and, when suitable, super-compacted. Sludges and ion exchange resins shall be dried and disposed of in polyethylene High Integrity Containers. All these waste forms are disposed of in the surface repository at the Dukovany site.

4.2.2 Activity limitations, surface dose rates and final disposal

Olkiluoto

The surface dose rate limit for bituminised waste packages is 0.8 Sv/h in average with allowable local maximum of 3 Sv/h. The dose rate limit for other waste packages is 0.1 Sv/h [21]. The activity limits are seen in Table XI.

SFR

The surface dose rate limits are 2, 10, 100, 500 mSv/h for BLA, BMA, BTF and Silo, respectively. The activity limits are seen in Table XII.

El Cabril

Solid and solidified LILW, which after packing constitutes what is known as a package, is defined as the waste whose activity is mainly due to the presence of short and medium-lived (approximately 30 years or less) beta or gamma-emitting radio-

Table XII. Nuclide-specific activity limits for different disposal options of the SFR. The limits refer to the year 2010. [24]

Nuclide	Half-life (year)	BLA (Bq)	BMA (Bq)	BTF (Bq)	Silo (Bq)
^{3}H	12.3	—	—	—	$1.3 \cdot 10^{14}$
^{14}C	$5.7 \cdot 10^{3}$	$2.6 \cdot 10^{9}$	$2.9 \cdot 10^{11}$	$1.3 \cdot 10^{11}$	$6.8 \cdot 10^{12}$
^{55}Fe	2.7	$2.3 \cdot 10^{12}$	$1.0 \cdot 10^{14}$	$1.7 \cdot 10^{13}$	$7.1 \cdot 10^{14}$
^{59}Ni	$7.5 \cdot 10^{4}$	$2.3 \cdot 10^{10}$	$1.0 \cdot 10^{12}$	$1.5 \cdot 10^{11}$	$6.8 \cdot 10^{12}$
^{60}Co	5.2	$5.8 \cdot 10^{12}$	$2.6 \cdot 10^{14}$	$4.0 \cdot 10^{13}$	$1.8 \cdot 10^{15}$
^{63}Ni	100	$1.9 \cdot 10^{12}$	$8.8 \cdot 10^{13}$	$1.5 \cdot 10^{13}$	$6.3 \cdot 10^{14}$
^{90}Sr	28.8	$7.1 \cdot 10^{10}$	$6.5 \cdot 10^{12}$	$2.7 \cdot 10^{12}$	$2.5 \cdot 10^{14}$
^{94}Nb	$2.0 \cdot 10^{4}$	$2.3 \cdot 10^{7}$	$1.0 \cdot 10^{9}$	$1.5 \cdot 10^{8}$	$6.8 \cdot 10^{9}$
^{99}Tc	$2.1 \cdot 10^{5}$	$1.1 \cdot 10^{8}$	$8.8 \cdot 10^{9}$	$3.6 \cdot 10^{9}$	$3.3 \cdot 10^{11}$
^{106}Ru	1.0	$2.1 \cdot 10^{9}$	$1.7 \cdot 10^{11}$	$6.2 \cdot 10^{10}$	$6.1 \cdot 10^{12}$
^{129}I	$1.6 \cdot 10^{7}$	$6.4 \cdot 10^{5}$	$4.7 \cdot 10^{7}$	$2.2 \cdot 10^{7}$	$1.9 \cdot 10^{9}$
^{134}Cs	2.3	$2.6 \cdot 10^{11}$	$2.2 \cdot 10^{12}$	$1.1 \cdot 10^{13}$	$8.1 \cdot 10^{14}$
^{135}Cs	$3.0 \cdot 10^{6}$	$6.4 \cdot 10^{6}$	$5.3 \cdot 10^{8}$	$2.2 \cdot 10^{8}$	$1.9 \cdot 10^{10}$
^{137}Cs	30.2	$1.4 \cdot 10^{12}$	$1.3 \cdot 10^{14}$	$5.3 \cdot 10^{13}$	$4.9 \cdot 10^{15}$
^{238}Pu	87.7	$4.7 \cdot 10^{8}$	$3.1 \cdot 10^{10}$	$1.7 \cdot 10^{10}$	$1.2 \cdot 10^{12}$
^{239}Pu	$2.4 \cdot 10^{4}$	$1.9 \cdot 10^{8}$	$1.2 \cdot 10^{10}$	$6.9 \cdot 10^{9}$	$3.8 \cdot 10^{11}$
^{240}Pu	$6.6 \cdot 10^{3}$	$2.9 \cdot 10^{8}$	$1.9 \cdot 10^{10}$	$1.1 \cdot 10^{10}$	$7.8 \cdot 10^{11}$
^{241}Pu	14.4	$1.5 \cdot 10^{10}$	$9.4 \cdot 10^{11}$	$5.4 \cdot 10^{11}$	$4.2 \cdot 10^{13}$
^{241}Am	433	$3.8 \cdot 10^{8}$	$2.4 \cdot 10^{10}$	$1.3 \cdot 10^{10}$	$1.0 \cdot 10^{12}$
^{244}Cm	18.1	$4.4 \cdot 10^{8}$	$2.8 \cdot 10^{9}$	$1.5 \cdot 10^{9}$	$1.2 \cdot 10^{11}$
Total activity		$\mathbf{1.2 \cdot 10^{13}}$	$\mathbf{6.0 \cdot 10^{14}}$	$\mathbf{1.4 \cdot 10^{14}}$	$\mathbf{9.2 \cdot 10^{15}}$

Table XIII. Solid waste, or waste which has been solidified by being incorporated or immobilised in a characterised solid matrix, satisfying sufficient stability requirements and having specific activities below the following values[1]. [20]

	Bq/g [2]
Level 1	
Total alpha activity	$1.85 \cdot 10^{2}$
Individual beta-gamma emitter activity (nuclides with a half-life > 5 years, except tritium)	$1.85 \cdot 10^{4}$
Total radionuclide beta-gamma activity (nuclides with a half-life > 5 years)	$7.40 \cdot 10^{4}$
Tritium activity	$7.40 \cdot 10^{3}$
Level 2	
Total alpha activity	$3.70 \cdot 10^{3}$
^{60}Co activity	$3.70 \cdot 10^{5}$
^{90}Sr activity	$3.70 \cdot 10^{5}$
^{137}Cs activity	$3.70 \cdot 10^{5}$

[1] The activity is measured or calculated on the date of the package production
[2] The weight is the total weight of the waste, the container and the immobilisation or solidification material. The weight of shielding material is not included.

nuclides and whose long-term radionuclide content is low and limited according to the classification levels given below. Two levels of properties required of the package are defined in Table XIII. [20].

The technical conditions associated with the license issued by the Regulatory Authority place limits on the specific activity per disposal unit and the activity per disposal cell [25]. The activity limits are seen in Table XIV.

Dukovany

The surface dose rate limit for steel drums are 1 mSv/h. The activity limits are seen in Table XV.

The mobile activity limits are determined as

Beta, gamma radionuclides	$2.4 \cdot 10^{12}$	(Bq/vault)
^{90}Sr	$2.3 \cdot 10^{8}$	(Bq/vault)

4.2.3 Clearance levels

Olkiluoto

Maintenance waste and metal scrap are clearaed from regulatory control if they are below the indicated clearance levels. The maintenance waste is first packed into plastic sacks and the surface dose-rate is measured. If it is above 10 µSv/h the sack cannot be cleared. Sacks to be cleared are baled. The weight and dimension of the bale are about 300–400 kg and 1200 × 700 × 700 mm, respectively. The bales are measured with gamma-spectrometer and transported to the plant dump if the activity limits are below clearance levels (Table XVI).

Metal scrap is exempted if it is below 1 kBq/kg (beta and gamma activity) and 100 Bq/kg (alpha activity). The surface contamination should not exceed the limits 4 kBq/m^2 (beta and gamma activity) and 400 Bq/m^2 (alpha activity).

SFR

The waste is cleared if it is below 500 Bq/kg including maximum 100 Bq/kg of alpha activity. The surface contamination must be below 40 kBq/m^2 for beta and gamma and below 4 kBq/m^2 for alpha.

Dukovany

The waste is cleared if it is below 300 Bq/kg.

4.2.4 Other important characteristics

Olkiluoto

The following waste package characteristics are also reported: the amount of free water, combustibility, swelling capacity, gas generation potential, chemical stability and concentrations of chemically aggressive substances.

SFR

The following waste package characteristics are also reported: composition and structure, corrosion resistance, gas formation, combustibility and fire-resistance, chemical reactivity, leaching, mechanical stability and strength against external stresses.

Table XIV. *Maximum specific activity per disposal cell (vault) (Maximum average activity for the assembly of packages disposed of in any disposal cell) and the upper activity limits per disposal unit at El Cabril. [25]*

Radionuclide	Disposal cell (vault) Bq/g	Disposal unit Bq/g
^{3}H	—	$1.0 \cdot 10^{6}$
^{14}C	$6.1 \cdot 10^{4}$	$2.0 \cdot 10^{5}$
^{59}Ni	$1.9 \cdot 10^{4}$	$6.3 \cdot 10^{4}$
^{63}Ni	$3.6 \cdot 10^{6}$	$1.2 \cdot 10^{7}$
^{60}Co	—	$5.0 \cdot 10^{7}$
^{90}Sr	$2.7 \cdot 10^{4}$	$9.1 \cdot 10^{4}$
^{94}Nb	$3.4 \cdot 10^{1}$	$1.2 \cdot 10^{2}$
^{99}Tc	$3.0 \cdot 10^{2}$	$1.0 \cdot 10^{3}$
^{129}I	$1.4 \cdot 10^{1}$	$4.6 \cdot 10^{4}$
^{137}Cs	$1.0 \cdot 10^{5}$	$3.3 \cdot 10^{5}$
Total Alpha	**$1.0 \cdot 10^{3}$**	**$3.7 \cdot 10^{3}$**

Table XV. *The upper activity limits for the most important nuclides in steel drums to the surface repository at the Dukovany site.*

Nuclide	Package (Bq)	Vault (Bq)	Double row (Bq)
^{90}Sr	$7.8 \cdot 10^{11}$	$1.0 \cdot 10^{14}$	$1.1 \cdot 10^{16}$
^{137}Cs	$2.2 \cdot 10^{11}$	$2.9 \cdot 10^{13}$	$3.2 \cdot 10^{15}$
^{239}Pu	$3.9 \cdot 10^{6}$	$5.1 \cdot 10^{8}$	$5.6 \cdot 10^{10}$
^{241}Am	$2.3 \cdot 10^{6}$	$3.0 \cdot 10^{8}$	$3.4 \cdot 10^{10}$

Note: ^{54}Mn, ^{60}Co, ^{106}Ru, ^{144}Ce are not limited.

Table XVI. *Activity limits of various nuclides in a bale (max. weight 400 kg). [23]*

Nuclide	Activity (Bq/kg)
^{51}Cr	$1 \cdot 10^{5}$
^{54}Mn	$2 \cdot 10^{4}$
^{58}Co	$5 \cdot 10^{3}$
^{60}Co	$5 \cdot 10^{3}$
^{65}Zn	$2 \cdot 10^{4}$
^{134}Cs	$1 \cdot 10^{3}$
^{137}Cs	$1 \cdot 10^{3}$
^{95}Nb	$5 \cdot 10^{3}$
^{95}Zr	$5 \cdot 10^{3}$

El Cabril

The most important characteristics for waste package are the following [20]:
- The waste form must be well identified.
- The producer must indicate the activity of the “key nuclides” (Cs-137 and Co-60), and shall give the information necessary to calculate the tritium and alpha emitter content.

- Active test probes taken from a real package shall be submitted to compression, traction and immersion tests.
- Depending on technical criteria, a package on a 1:1 scale or test pieces extracted form a package shall be submitted to leaching tests having a minimum duration of 1 year.
- Standard drums must be used and they shall conform to the filling level limitations.
- The packages shall fulfil the requirements of the Road Transport Regulations.

4.3 Characterisation for each waste package

4.3.1 Determination of activity content

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

- Calculation or estimation based upon well known data.
- Measurement of the dose rate and calculation or estimation using reference samples if the isotopic composition is known and remains sufficiently constant.
- Measurement of the total activity or the specific activity of certain characteristic "key nuclides" and calculation or estimation of the radionuclide inventory using reference samples. Key nuclides are radionuclides which can be easily measured and for which scaling factors are evaluated. The scaling factors are used to calculate the activity of all other relevant radionuclides from the measured activities of the key nuclides [8].

Olkiluoto

Maintenance waste

The surface dose rate of the drum is measured after filling. The drums of similar waste are collected into groups of 1–10 drums, of which one drum is measured by gammaspectrometry. The measured drum represents all drums of the group. The selection is based on the original data of the waste and the surface dose rate of the drums. If the surface dose rate exceeds the value of 5 mSv/h the drum is measured separately.

Bituminised waste

The surface dose rate of the drum is measured after filling. The drums are collected into groups of 1–10 drums of which one drum is measured. The selection is based on the surface dose rate of the drum, which should not vary more than by factor 0.5 to 2. The activity of other drums in the group is calculated using the measured drum data and the difference of the surface dose rate factors of drums. The results of ^{60}Co, ^{137}Cs and ^{134}Cs are documented. Deviant drums are measured separately.

Metal scrap

The surface dose rate and the dose rate at 1 m distance of surface are measured for the steel boxes. The activity is calculated (assuming that the radiation is from ^{60}Co) for the boxes of low surface dose rate (< 2 mSv/h). Every fourth box is measured. All boxes of surface dose rate over 2 mSv/h are measured gamma spectrometrically.

The surface dose rates of drums are measured after filling. The activity measurements are mainly the same as for the maintenance waste. Nuclide specific attenuation corrections are used if the density of the drum is different from the reference maintenance waste drum which is used for calibration. If the surface dose rate exceeds the value of 10 mSv/h the drum is measured separately.

The activity of mixed metal scrap is determined using the calculation based on the surface dose rate, geometry and density data. The surface dose rate is measured in three different phase (1/3, 2/3 and 3/3) of the filling at the distance of 1.5 metres from the metal scrap surface. The values are used for determination of the attenuation correction.

Filter rods

The surface dose rate of the drum is measured after filling. The activity measurements are the same as for the metal scrap case in the steel drums and boxes.

SFR

Gamma measurements are done on individual packages or on waste sample. Two measurement methods are used concerning homogeneous waste packages: the measurement of a small sample of homogeneous waste form or the measurement of the waste package after filling. Inhomogeneous waste packages are measured after filling. The surface dose rate and dose rate at 1 m distance of the surface are measured.

El Cabril

The method adopted for the determination of the activity of waste packages coming from NPPs may be summarised as follows [25]:

A) Non-typified packages (historical) containing homogeneous and heterogeneous waste from the NPP.
   - ^{137}Cs and ^{60}Co: Producer information
   - Alpha emitters: Scaling factors
   - Other significant beta and gamma emitters: Scaling factors

B) Typified packages of homogeneous waste from the NPP.
   - ^{137}Cs and ^{60}Co : Producer information
   - Alpha emitters: Analysis of batch samples (ENRESA)
   - Other significant beta and gamma emitters: Scaling factors

Periodic documentary control relating to the factors affecting activity calculation. Analysis of actual waste package samples as quality control.

C) Typified packages of heterogeneous waste from the NPP.
   - ^{137}Cs and ^{60}Co: Producer information
   - Alpha emitters: Scaling factors
   - Other significant beta and gamma emitters: Scaling factors

Periodic documentary control relating to the factors affecting activity calculation.

Dukovany

The surface dose rate and dose rate at 1 m distance of the surface are measured. No gamma-spectroscopic measurements are made at the moment.

4.3.1.1 Detectors and calibrations for activity measurements

Olkiluoto

Gamma-spectroscopic measurements are made with a HPGe detector. Drums with low activity are measured with a scanning device. Drums with bituminized ion-exchange resin are measured from a distance of 10 m. Extra collimator can be placed between the detector and the drum in those cases where the activity is very high.

At Olkiluoto, there are several calibration sources used for different types of waste packages:

- Reference drum of maintenance waste (^{152}Eu homogeneously mixed into inactive maintenance waste).
- Reference bale of VLLW (^{152}Eu homogeneously mixed into inactive waste bale).
- Reference drum of intermediate level bituminised ion exchange resin waste [19].

SFR

Gamma-spectroscopic measurements are made with a semiconductor detector at the NPPs, at the CLAB or at Studsvik. The size of the detector is different in different measurement systems but otherwise the systems are more or less similar. The measurement system is calibrated with point sources [22].

El Cabril

Gamma-spectroscopic measurements are made with a segmented gamma scanner, designed for objects up to 1.2 m high, 0.76 m in diameter and up to 1 500 kg in mass. The collimator diaphragm can be changed. The detector-object distance is automatically controlled (range from 0.0 to 2.0 m). The gamma scanner includes a remote controlled arm with Geiger-Müller detectors to take segmented dose rate of the drum. The calibrations for the following measurement geometries are available [26]:

- Liquid and solid homogeneous cemented radioactive waste in a 220-l drum.

- Contaminated primary coolant cartridge filters in a special internally concrete shielded 220-l drum.
- Heterogeneous pre-compacted solid waste both in a 220-l drum pre-shielded with concrete and in a non-shielded drum.
- Metallic and heavily contaminated pieces in concrete matrix in a 220-l drum.
- Scrap metals from radioactive lighting rods decommissioning (^{241}Am determination) in a 220-l drum.
- Homogeneous 220-l drum reconditioned in a concrete pre-shielded 480-l drum.
- 2-l specimens from drilling of homogeneous 220-l drums.
- Supercompacted 220-l pellets in 290-l drum (in development).
- Non-compactable heterogeneous waste (e.g. metal scrap and decommissioning waste) in a 220-l drum with and without special lead and/ or concrete shielding (in development).

4.3.1.2 Sample homogeneity and screening effects

Olkiluoto

The scanning of rotating drums and other packed waste is made in order to obtain representative values for the total activity. In those cases where the dose-rate is too high, scanning is not possible. The measurements are then made in the direction of the highest dose rate, which is a conservative alternative for scanning.

The maintenance drum at the Olkiluoto contains only soft material. The self-absorption is minimal and is automatically considered via the reference sample, which has approximately the same screening effect as the sample.

El Cabril

The measurement is carried out with continuously rotation of the drum at 7 r.p.m., performing a multirotational gamma scanning in 8 segments and keeping constant height of each segment. Analysis is performed on the sum of the individual segment spectra once corrected for attenuation, self-absorption and dead-time. As a result, the activity of the radionuclides and the distribution of these nuclides per segment is determined [26].

4.3.1.3 The use of reference nuclides

The gamma-spectroscopic measurement makes it possible to observe nuclides with energies above 60 keV. However, nuclides at low gamma energy and pure beta and alpha active nuclides are not seen. It is often necessary to use destructive methods to complement the non-destructive data.

Use of the scaling factor is an established method for estimating the content of difficult-to-measure nuclides in waste. The long-lived pure beta emitters which are present in the waste from nuclear reactors are ^{63}Ni and ^{90}Sr. The actinides U, Pu, Am and Cm are alpha active and emit also relatively few gamma quanta with low energy. The reference nuclides are ^{60}Co and ^{137}Cs. The scaling factors vary somewhat between different types of reactors and also between individual reactors of similar type mainly due to fuel leakage.

Olkiluoto

Two sets of scaling factors are used, a primary water related set and a surface contamination related set. The power plants have an extensive bookkeeping for the waste. The scaling factors are updated for various nuclides and waste categories periodically. This makes it possible to adjust the total activity estimates in the storage, if, for example, it is found out that the previously used scaling factors were erroneous during a certain period. Some correlation factors are presented in Tables XVII and XVIII.

SFR

Beta emitters are estimated through correlation with the key nuclides (^{137}Cs and ^{60}Co). Estimation is done by SKB and not by the waste producer. Alpha emitters are estimated based on analyses on reactor water sample. Scaling factors are presented in Table XIX.

El Cabril

As regards the activity of the waste packages from the NPPs the producer is responsible for determining (by calculation or measurement) the activity of ^{137}Cs and ^{60}Co, as well as for analysing the average value of tritium in the coolant in different

Table XVII. *Scaling factors from Olkiluoto NPPs for bituminised waste. [23]*

	Nuclides	Factors
Key-Nuclide ^{60}Co	^{59}Ni/^{60}Co	$5.0 \cdot 10^{-4}$
	^{63}Ni/^{60}Co	$1.0 \cdot 10^{-1}$
Key-Nuclide ^{137}Cs	^{90}Sr/^{137}Cs	$1.0 \cdot 10^{-1}$
	^{99}Tc/^{137}Cs	$5.0 \cdot 10^{-5}$
	^{129}I/^{137}Cs	$3.0 \cdot 10^{-7}$
	^{135}Cs/^{137}Cs	$3.0 \cdot 10^{-6}$
	^{238}Pu/^{137}Cs	$2.3 \cdot 10^{-5}$ … $5.0 \cdot 10^{-7}$
	$^{239+240}$Pu/^{137}Cs	$6.0 \cdot 10^{-5}$ … $1.0 \cdot 10^{-6}$
	^{241}Am/^{137}Cs	$1.0 \cdot 10^{-6}$ … $6.7 \cdot 10^{-6}$
	$^{243+244}$Cm/^{137}Cs	$1.2 \cdot 10^{-5}$ … $1.0 \cdot 10^{-6}$

Table XVIII. *Scaling factors from Olkiluoto NPPs for maintenance waste. [23]*

	Nuclides	Factors
Key- Nuclide ^{60}Co	^{63}Ni/^{60}Co	$1.0 \cdot 10^{-1}$
	^{55}Fe/^{60}Co	$1.0 \cdot 10^{0}$
Key- Nuclide ^{137}Cs	^{90}Sr/^{137}Cs	$1.0 \cdot 10^{-1}$

Table XIX. *Scaling factors from NPPs in Sweden. [19]*

	Nuclides	Factors
Key-Nuclide ^{60}Co	^{63}Ni/^{60}Co	$5.0 \cdot 10^{-1}$
Key-Nuclide ^{137}Cs	^{90}Sr/^{137}Cs	$1.0 \cdot 10^{-1}$
	^{99}Tc/^{137}Cs	$5.0 \cdot 10^{-3}$
	^{129}I/^{137}Cs	$4.0 \cdot 10^{-7}$
Key-Nuclide $^{239+240}$Pu	^{238}Pu/$^{239+240}$Pu	$5.0 \cdot 10^{1}$
	^{241}Am/$^{239+240}$Pu	$5.0 \cdot 10^{0}$
	^{243}Cm/$^{239+240}$Pu	$1.0 \cdot 10^{0}$
	^{244}Cm/$^{239+240}$Pu	$1.0 \cdot 10^{0}$

months [25]. Scaling factors are presented in Tables XX and XXI.

For other types of waste from a radioactive facility, activity assessment is the responsibility of the respective producer, no method of correlation has been developed.

4.3.2 Determination of chemical content

Radioactive waste may contain both organic and inorganic chemically toxic substances. The contents are highly dependent on the type of waste. In the waste from nuclear research, the most likely toxic materials are the metals Be and Cd, used for various purposes in nuclear physics, and Pb, widely used as shielding material. Also Hg may occur in the research waste [12].

Most of the inorganic materials (e.g. heavy metals) can be considered as a minor problem because they tend to be converted into insoluble form. Behaviour of the organic components is more difficult to predict, but the content of organic chemotoxic components in the radioactive waste is normally small. Examples of these could be chlorinated aromatics, fungicides, pesticides and some organic metal-compounds [12].

Olkiluoto

The LILW from the NPPs is sufficiently well documented to ensure that its handling and conditioning can be made in a controlled manner. The need for detailed chemical analysis of waste is normally not very high. The present routines provide sufficient background information for correct decision-making concerning the safety aspects related to the handling, interim storage and final storage of LILW in Finland.

SFR

Acceptance test consists of measuring, analyses or indirect estimations of contents of different substances and materials.

Dukovany

No determination of the chemical content is done, only the declaration of the waste producer is checked.

4.4 Documentation and traceability

Olkiluoto

The following data are documented of each drum: the identifying mark of drum, the date of filling, the weight of the drum, the weight of the binding agent, the material of the binding agent, the contents, the surface dose rate, storage ID and the coordinates in the storage.

Table XX. *Scaling factors for PWR. [25]*

		Streams			
	Nuclides	**Bead resins**	**Evaporator bottoms**	**Cartridge filters**	**D.A.W.**
Key-Nuclide ^{60}Co	$^{14}C/^{60}Co$	$1.0 \cdot 10^{-2}$	$1.8 \cdot 10^{-2}$	$7.3 \cdot 10^{-3}$	$1.5 \cdot 10^{-2}$
	$^{59}Ni/^{60}Co$	$8.6 \cdot 10^{-3}$	$8.6 \cdot 10^{-3}$	$8.6 \cdot 10^{-3}$	$8.6 \cdot 10^{-3}$
	$^{63}Ni/^{60}Co$	$4.8 \cdot 10^{-1}$	$4.8 \cdot 10^{-1}$	$4.8 \cdot 10^{-1}$	$4.8 \cdot 10^{-1}$
	$^{94}Nb/^{60}Co$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$
Key-Nuclide ^{137}Cs	$^{90}Sr/^{137}Cs$	$3.2 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$1.8 \cdot 10^{-2}$	$4.7 \cdot 10^{-3}$
	$^{99}Tc/^{137}Cs$	$5.2 \cdot 10^{-7}$	$2.1 \cdot 10^{-7}$	$2.9 \cdot 10^{-6}$	$7.6 \cdot 10^{-7}$
	$^{129}I/^{137}Cs$	$2.7 \cdot 10^{-7}$	$2.7 \cdot 10^{-7}$	$2.7 \cdot 10^{-7}$	$2.7 \cdot 10^{-7}$
	$^{239}Pu/^{137}Cs$	$2.1 \cdot 10^{-5}$	$5.7 \cdot 10^{-5}$	$2.6 \cdot 10^{-3}$	$6.0 \cdot 10^{-4}$
	$^{241}Pu/^{239}Pu$	$9.7 \cdot 10^{+1}$	$9.7 \cdot 10^{+1}$	$9.7 \cdot 10^{+1}$	$9.7 \cdot 10^{+1}$
	α-total/^{239}Pu	$4.5 \cdot 10^{0}$	$4.5 \cdot 10^{0}$	$4.5 \cdot 10^{0}$	$4.5 \cdot 10^{0}$
^{3}H	$^{3}H/^{3}H$ coolant	$5.2 \cdot 10^{-2}$	$1.3 \cdot 10^{-1}$	$1.5 \cdot 10^{-1}$	$7.0 \cdot 10^{-3}$

Table XXI. *Scaling factors for BWR. [25]*

		Streams			
	Nuclides	**Bead resins**	**Evaporator bottoms**	**Cartridge filters**	**D.A.W.**
Key-Nuclide ^{60}Co	$^{14}C/^{60}Co$	$4.7 \cdot 10^{-4}$	$2.2 \cdot 10^{-4}$	$6.9 \cdot 10^{-4}$	$8.7 \cdot 10^{-4}$
	$^{59}Ni/^{60}Co$	$5.0 \cdot 10^{-4}$	$5.0 \cdot 10^{-4}$	$5.0 \cdot 10^{-4}$	$5.0 \cdot 10^{-4}$
	$^{63}Ni/^{60}Co$	$2.9 \cdot 10^{-2}$	$2.9 \cdot 10^{-2}$	$2.9 \cdot 10^{-2}$	$2.9 \cdot 10^{-2}$
	$^{94}Nb/^{60}Co$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$	$3.4 \cdot 10^{-5}$
Key-Nuclide ^{137}Cs	$^{90}Sr/^{137}Cs$	$5.4 \cdot 10^{-3}$	$8.1 \cdot 10^{-3}$	$7.3 \cdot 10^{-3}$	$5.0 \cdot 10^{-3}$
	$^{99}Tc/^{137}Cs$	$8.8 \cdot 10^{-7}$	$1.3 \cdot 10^{-6}$	$1.2 \cdot 10^{-6}$	$8.1 \cdot 10^{-7}$
	$^{129}I/^{137}Cs$	$2.8 \cdot 10^{-7}$	$2.8 \cdot 10^{-7}$	$2.8 \cdot 10^{-7}$	$2.8 \cdot 10^{-7}$
	$^{239}Pu/^{137}Cs$	$5.4 \cdot 10^{-5}$	$8.5 \cdot 10^{-5}$	$1.1 \cdot 10^{-4}$	$7.4 \cdot 10^{-4}$
	$^{241}Pu/^{239}Pu$	$6.4 \cdot 10^{+1}$	$6.4 \cdot 10^{+1}$	$6.4 \cdot 10^{+1}$	$6.4 \cdot 10^{+1}$
	α-total/^{239}Pu	$3.5 \cdot 10^{0}$	$3.5 \cdot 10^{0}$	$3.5 \cdot 10^{0}$	$3.5 \cdot 10^{0}$
^{3}H	$^{3}H/^{3}H$ coolant	$4.5 \cdot 10^{-1}$	$7.0 \cdot 10^{-1}$	$4.6 \cdot 10^{-1}$	$4.0 \cdot 10^{-1}$

Table XXII. *The parameters to be controlled or measured of the cement solidified ion exchange resin in a steel container. [28]*

Raw waste	**Origin and type, weight and resin/water ratio.**
Packaging	Production under the specifications given by the waste generator, visual control when received at waste treatment facility and sample tests for dimensional check.
Solidification process	Recipe within acceptable variation from the base recipe, functional test of equipment, weight of waste, water, cement and additives, mixer current, time of mixing and speed of mixer, dose rate, concentration of boric acids (PWR). Calculation of water/cement ratio and hydration.
Measurement on waste package	Contents of radionuclides measured by gamma spectroscopy, total gamma, visual check for geometry changes etc. Surface contamination is checked only if contamination is suspected.

SFR

Unique identity number on each package is documented. Each type of waste has its own waste type descriptions (WTD) in which all main data are documented. Results of tests and analyses are also documented in the WTD [28].

An example of the parameters to be controlled or measured is given in Table XXII.

The waste type is cement solidified ion exchange resin in a steel container (box).

El Cabril

The process book is a control document and consists of three different parts [20]:

Package Description Documents: Waste producer is responsible for the preparation of these documents. It contains the description of the raw waste, activity, evaluation method, conditioning system, package characteristics and quality control measures.

Package Characterisation Process: ENRESA is responsible for the co-ordination of the package characterisation process, the tests are performed in the Characterisation Laboratory. These tests are done either on real drums or samples.

Package Type Acceptance Agreement: ENRESA is responsible for the interpretation of the test results and their comparison with the acceptance criteria and the safety rules. After that a final agreement will be signed by ENRESA and the waste-producer.

Dukovany

The following data are documented of each drum: identifying mark of the drum, activity, composition, dose rate, weight, producer and location.

4.5 Transportation and disposal process

Olkiluoto

The waste packages are stored at the power plant in separate storage facilities for low-level waste and medium level waste. LILW waste packages are transported in the repository at Olkiluoto by a truck especially built for this purpose. All LILW that is transported to the repository is finally placed in concrete boxes. No measurements on the waste packages are done when they arrive at the repository. Only a document check is performed. All relevant information about each waste package is transferred to the repository data base.

SFR

The sea transportation system consists of a specially designed ship, M/S Sigyn, 27 IP-2 containers (ATB) for transport of LILW and 5 terminal vehicles. One of the vehicles is specially designed for operation in the SFR repository.

All relevant information about each waste package is documented and collected in a computerised waste register. Before the waste is transported to SFR, the contents of the waste register are transferred to the SFR-data base. No measurements on the waste packages are done when they arrive at the SFR repository. Only a document check is performed.

El Cabril

LILW waste arrives at El Cabril in 220 l drums transported by trucks especially built for this purpose, and are unloaded at the Conditioning building. The drums are sorted on the basis of their previous conditioning and the processing line is selected accordingly:

- Conditioned waste suitable for direct introduction into the Disposal Unit (part of this waste is supercompacted).
- waste destined for the laboratory, for quality checks and tests
- waste requiring specific treatment (waste from radioactive facility)

Once their radiological characteristics have been checked and they have been conditioned, the last two types are transferred to the Disposal Units (DU).

4.6 Quality assurance

Olkiluoto

All waste management activities are subject to the quality assurance programme of the NPP and to a similar regulatory control as all other activities at the NPP. The regulatory control includes the review of the relevant documents and inspections to the waste management facilities [10].

SFR

The Swedish Radiation Protection Institute, SSI, performs control measurements to waste packages produced by the Swedish NPPs. To verify some of the parameters measured by the waste generator, independent measurements on dose rate and activity contents are performed annually.

The quality systems are such that it has been found unnecessary to make any measurements on the waste packages when they arrive at the SFR repository. Only a document check is performed.

El Cabril

As regards the characterisation of waste packages, the producer performs a series of actions periodically in order to ensure the compliance with the limits established by the acceptance criteria [25]. The most important of these are the following:

Analysis of samples of non-conditioned waste: In relation to the different waste streams associated with an approved process book, the producer will be requested to provide waste samples prior to initiation of the corresponding conditioning campaigns. The frequency of these samples, which will be established beforehand, will depend on the type of waste stream and on the characteristics of the producing facility.

Production control: These control actions are carried out at the producing facilities themselves and are aimed at verifying the compliance of the producers with the conditions established in the appropriate documents as regards the assurance of the waste package quality and activity evaluation.

Supercontrol: It consists of destructive tests carried out on accepted packages already received at the disposal facility, and are aimed at verifying the compliance with the aspects relating to the safety criteria. These control actions shall be focused mainly on the determination of the waste package activity, using samples of powder extracted by perforation, and on the verification of other physicochemical characteristics of the matrices.

In order to ensure the compliance with the quality objectives required of the waste packages, ENRESA has developed a characterisation laboratory at the El Cabril disposal facility, equipped with all the resources and equipment required for performing the tests. The laboratory has two different areas in two buildings, the active one and the inactive one.

For the characterisation of the matrices of real packages, tests are to be carried out in the active laboratory. They can be divided into three main groups [25]:

A) Non-destructive testing. This includes package transport tests and spectrometry. They are carried out on the whole packages and do not generate secondary waste.
B) Destructive testing. This includes tests on the preparation and handling of real waste packages, physical-mechanical tests and homogeneity tests. They involve destructive treatment of the packages and therefore generate secondary waste.
C) Tests for microstructural characterisation of the matrix. These include leaching tests on real packages and test pieces.

The inactive laboratory runs characterisation tests on samples and inactive test pieces simulating the matrix of the packages. This requires the preparation of test pieces and samples reproducing the matrices of the package. The behaviour of these test pieces under thermal cycles, their mechanical resistance and certain physical properties are found out [25].

Dukovany

State Office for Nuclear Safety (SONS) is taking care of the QA of the waste packages.

5 DISCUSSION

Proven and safe technologies exist for managing LILW. These technologies have been used since the early 1970s. In France and the UK, shallow land burial has been used as long as nuclear energy has been produced. A modern variant taking advantage of the experience and technology progress is near surface disposal in engineered structures. This is now being implemented or planned more widely, e.g. in Spain, France and Czech Republic. Underground facilities are also existing, for example in Germany, Sweden and Finland. There is a tendency now to dispose of waste at greater depths.

Waste conditioning and packaging for disposal

A wide variety of waste packages are used in different countries to meet the needs of the nuclear industry and research. In most countries the 200-l drum is the standard container for LILW. In addition, many special types are designed and manufactured for the needs of a specific use.

Radioactive waste may exist in many forms when it passes through the treatment and conditioning process. It may exist sequentially as raw, treated, immobilised and fully conditioned waste. Many low-level solid waste forms may be packaged in the appropriate waste containers without requiring a matrix for immobilisation. For intermediate level waste a matrix will be required to improve both the mechanical properties of the waste package and its ability to hold the radionuclides. Matrices, which are being used for waste, include different cements, bitumens, polymers, ceramics, low melting point metal alloys and combination of these. National practices for treating and conditioning of LILW are summarised in Table XXIII.

In practice a large variety of packaging methods and containers are in use. The present practice of waste conditioning and packaging in Olkiluoto is considered to work well.

Characterisation of waste

The performance objectives such as dose limitations defined by the national authorities are the basic requirements. The waste acceptance criteria discussed in this document address only one component of the whole waste management. Unfavourable characteristics in this component may be compensated for by adding another component or by improving its performance.

The precise characteristics will vary between different type of waste. The type of disposal facility and the specific regulatory framework will influence the handling, conditioning, interim storage, transportation and disposal of the waste. The characteristics of waste to be considered cover their radionuclide content as well as their chemical, physical, mechanical, thermal and biological properties [8].

The chemical, physical, mechanical, thermal and biological characterisations are generally performed in the first place, in the laboratory and field tests, before the waste package type is accepted for the final disposal (e.g. the properties of bituminised spent ion exchange resins from Olkiluoto have been studied by VTT since the late 70s [29]). Routine waste characterisation involves only detecting the presence of individual radionuclides and quantifying their inventories in the waste. Gamma-spectrometry is the primary means in characterisation of the waste packages in countries considered in this study, except Czech Republic. But also at Czech Republic they are going to start these measurements in the near future. Significant work has been done for the development of waste characterisation techniques, especially for homogeneous waste. Heterogeneous waste present a particular challenge and currently available methods tend to be either costly or time-consuming [8].

The waste acceptance criteria and waste char-

Table XXIII. Current LILW management practices in some European countries. [7]

	Treatment process					Conditioning
	Evapo-ration	Ion ex-change	Precipi-tation	Compaction	Incineration	Matrix/process
Belgium	X		X	X	X	Bitumen, concrete
Bulgaria	X		X			Concrete
Czech Republic	X	X	X	X		Calcination, cement, concrete
Finland	X	X		X		Bitumen
France	X	X	X	X	X	Bitumen, cement + bitumen, concrete, polymer + cement
German	X	X		X	X	Concrete, drying, packing, high force compaction, thin film rotary evaporation
Hungary	Use of concrete for liquid waste					Concrete
Italy	X	X	X	X	X	Concrete
Netherlands			X	X		Concrete
Poland	X		X			Polymers
Romania	X	X	X	X	X	Cement, concrete
Spain	X	X		X		Concrete
Sweden	X	X	X	X	X	Bitumen, concrete
Switzerland			X	X	X	Bitumen, concrete, polymers
UK	X	X		X	X	

acterisation is more complicated in the SFR and the El Cabril repositories than in the Olkiluoto repository because the SFR and El Cabril have to meet the disposal needs of all LILW produced in these countries. Especially in El Cabril, which is an engineered shallow ground repository, many waste package characteristics have strict limitations given in the WAC. In order to ensure the compliance with the WAC, ENRESA has developed a characterisation laboratory at the El Cabril disposal facility.

The LILW from the Olkiluoto NPPs is sufficiently well characterised. The database concerning the physical and chemical properties of the waste types was obtained through analysis of test samples taken during commissioning and the early stages of operation of the conditioning process. The control of most of the relevant properties of waste packages is based on this database and on the control of the parameters of the conditioning process [10]. Routine waste characterisation involves only detecting of the presence of individual radionuclides and quantifying their inventories in the waste. The need for detailed chemical analysis of waste is not very high. It can be said that the present characterisation routines provide sufficient background information for correct decision-making concerning the safety aspects related to the handling, interim storage and final storage of LILW at Olkiluoto.

Quality Assurance

The objective of a quality assurance program is to ensure that the waste management system complies with the waste acceptance criteria and the disposal requirements as established by the appropriate national authority. In addition, may be important to verify independently some of the parameters measured by the waste generator. The regulator can charge independent experts to check the documentation of the waste deliverer and to assess the non-destructive and destructive tests. Belgium (SCK/CEN), France (CEA Cadarache), Germany (TUM/RCM), Italy (LNRR), Netherlands (KEMA Arnhem / ECN Petten), Spain (CIEMAT and ENRESA El Cabril laboratory) and United Kingdom (EA's Waste Quality Checking Laboratory in Winfrith) are examples of laboratories undertaking quality checking of waste packages in Europe. Waste packages to be checked are taken as random samples. Additional checks are performed e.g. if indications of faulty packages are found in visual inspection of the lot [30].

6 CONCLUSIONS AND RECOMMENDATIONS

The waste remains radioactive for hundreds of years after closure of the final repository and eventually the responsibility of the safety is transferred to the state. The safety of the disposal facility to the future generations is based fundamentally on the validity of the characterisation data produced by all parties involved in the waste management process. At Olkiluoto there is no clear separation between the waste generator and the repository operator but the NPP organisation is in charge of both duties. At the moment no independent quality-checking measurements of the waste packages are done. This is different from the practice of many European countries. Independent verification measurements to the waste packages to be disposed of should be done. Control measurements of waste packages must be understood as a complementary activity giving suitable assurance on the waste package quality and waste generator's compliance with the repository waste acceptance criteria. Measurements should be focused on randomly selected waste packages.

More common features than differences can be observed in the LILW management procedures in the selected European countries. Finland has a well-established national framework for radioactive waste management and substantial experience in regulating the disposal of solid LILW into deep disposal facilities. LILW are safely disposed of based on the WAC.

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APPENDIX

QUESTIONNAIRE

1.Organisation/repository

1.1 Please give your name, address, contact person name, tel., fax and Email?

2.Waste producers

2.1 Please give a list of the main waste producers?

3.Waste containers and waste packages

3.1 Please list the different waste container types used for disposal?

3.2 Please list the main categories of the waste packages accepted in the final LILW repository?

3.3 Requirement (criteria) **for each** main category of the waste package

3.3.1 Please give the title of the main category?

3.3.2 Please describe the packaging and conditioning methods?

3.3.3 What requirements there are for the waste container?

3.3.4 Please list the activity limitations of individual isotopes (Bq/package)?

3.3.5 What is the limit of the surface dose rate or the dose rate at a certain distance?

3.3.6 Any other relevant information? (e.g. chemical and physical properties etc.)

3.4 Characterisation **for each** main category of the waste package

3.4.1 Please describe the determination of activity contents by answering the following questions:
What kind of detector is used for activity measurements?
What kind of calibration is used?
How is taken care the sample homogeneity and screening effects?
What kinds of reference nuclides are used?

3.4.2 Please describe the determination of chemical content?

4. Documentation and traceability

4.1 Please describe the waste package identification?

4.2 What is documented for each waste package?

5. Free release

5.1 What is the free release level for each main category of the waste package?

6. Storage

6.1 How long the waste packages are stored in the interim storage before the final disposal?

7. Quality Control

7.1 Please provide a very brief description of the Quality Control for waste characterisation?