TECHNICAL REPORTS SERIES NO. 433

Upgrading of Near Surface Repositories for Radioactive Waste

IAEA
International Atomic Energy Agency

UPGRADING OF NEAR SURFACE REPOSITORIES FOR RADIOACTIVE WASTE

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BANGLADESH
BELARUS
BELGIUM
BENIN
BOLIVIA
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BULGARIA
BURKINA FASO
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHILE
CHINA
COLOMBIA
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ETHIOPIA
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LATVIA
LEBANON
LIBERIA
LIBYAN ARAB JAMAHIRIYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MOROCCO
MYANMAR
NAMIBIA
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
PAKISTAN
PANAMA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
SAUDI ARABIA
SENEGAL
SERBIA AND MONTENEGRO
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TUNISIA
TURKEY
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VENEZUELA
VIETNAM
YEMEN
ZAMBIA
ZIMBABWE

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

TECHNICAL REPORTS SERIES No. 433

UPGRADING OF NEAR SURFACE REPOSITORIES FOR RADIOACTIVE WASTE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2005

Sales and Promotion Unit, Publishing Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna
Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
http://www.iaea.org/books

Printed by the IAEA in Austria
July 2005
STI/DOC/010/433

IAEA Library Cataloguing in Publication Data

Upgrading of near surface repositories for radioactive waste. — Vienna : International Atomic Energy Agency, 2005.
p. ; 24 cm. — (Technical reports series, ISSN 0074-1914 ; no. 433)
STI/DOC/010/433
ISBN 92–0–112704–9
Includes bibliographical references.

1. Radioactive waste disposal. 2. Overview of near surface disposal systems. 3. Radioactive waste disposal in the ground. 4. Radiation — Safety measures. I. International Atomic Energy Agency. II. Technical reports series (International Atomic Energy Agency) ; 433.

IAEAL 05–00409

FOREWORD

Providing guidance on the disposal of radioactive waste constitutes an important and integral component of the IAEA programme on radioactive waste management. Low and intermediate level waste (LILW), even though it contains a small fraction of the total activity of all radioactive waste produced globally, represents more than 90% of the total volume of radioactive waste. Most of the radioactive waste produced in many developing Member States is primarily LILW. The IAEA has received many requests from Member States for technical assistance in the safe management of low and intermediate level radioactive waste. As a result, a number of activities have been initiated by the IAEA to assist Member States in the disposal of LILW, focusing on both technology and safety aspects.

Many existing disposal facilities were developed and began operation long before current regulatory requirements took effect or more recent site suitability guidance, technological advances, safety assessment methodologies and quality assurance systems became available. National laws, regulations and disposal methods have evolved and improved with time. Various Member States have ongoing programmes both to upgrade these facilities and/or to develop new near surface disposal facilities.

Recently, a binding international regime for radioactive waste management was established through Article 12 of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. This article states that "each contracting party shall in due course take the appropriate steps to review the results of past practices in order to determine whether any intervention is needed for reasons of radiation protection". It is anticipated that the Joint Convention will result in safety reassessments of certain repositories and that corrective actions will be pursued based on these assessments.

Upgrading measures are being implemented or planned at a number of disposal facilities in numerous countries. Examples range from adoption of new waste acceptance criteria and container specifications to building additional engineered barriers, installing hydrogeological cut-off walls, improving cover systems, waste retrieval and other measures. Extensive experience and information are therefore available on actions that may be employed to upgrade disposal facilities. This publication draws on such international experience.

The objective of this publication is to provide guidance to Member States with near surface disposal facilities on approaches and technologies that can be used to: (a) identify potential corrective action needs; (b) assess options and select appropriate corrective actions; and (c) plan and implement the corrective actions selected to enhance repository performance and safety both before and after closure.

This report considers a variety of circumstances that may require potential upgrading measures to be assessed or implemented at near surface disposal facilities. The circumstances leading to the corrective actions, or the corrective actions themselves, may be of either a technical or non-technical nature. Methodologies that can be employed to implement effective solutions to problems are discussed, including assessment of alternative options prior to selecting corrective actions and the planning, implementation and verification of the specific measures adopted. Examples are provided of approaches and technologies that may be used to improve repository performance and safety. Information is also provided on experience in various Member States in upgrading disposal facilities.

It is anticipated that this publication will be useful to scientists, engineers, managers and others involved in assessing and improving the performance of near surface repositories and related regulatory compliance, as well as public acceptance issues. The lessons learned from the application and evaluation of corrective actions may also be relevant to the development of new repositories or facilities that require additional disposal capacity.

This report was developed with the help of consultants and through a Technical Committee meeting in Budapest in August 2003. The IAEA officer responsible for this publication was R. Dayal of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Large volumes of low and intermediate level waste (LILW), containing a wide range of radionuclides, are produced in the nuclear industry from activities such as research, uranium enrichment, fuel fabrication, reactor operations, isotope production and fuel reprocessing. In countries with operating nuclear power plants, fuel cycle waste represents a large fraction of the overall volume of waste generated. These wastes include ion exchange resins, concentrates from liquid waste treatment by chemical processes or evaporation, contaminated equipment, tools and rags, and workers' protective clothing. Decommissioning of nuclear facilities is another major source of LILW. Examples include contaminated concrete, reactor components, building debris, pipes, wiring, contaminated tools, protective clothing, etc. It is generally estimated that up to 95% of the total volume of radioactive waste generated in reactor decommissioning can be categorized as LILW.

Medical and research applications of radioisotopes are common worldwide. These activities generate solid radioactive wastes such as laboratory media and glass, syringes, plastic bottles, paper and rags, animal carcasses, contaminated blood and liquid wastes (e.g. organic solvents and liquid scintillation media). Disused radioactive sources are generated by medical, research and industrial applications.

Recognition of the potential impact of radioactive waste on human health and the environment has led to the development of national and international standards and guidelines for radiation protection and radioactive waste management including disposal [1–11]. These standards and guidelines are based on scientific and technical knowledge gained worldwide from many years of experience in research, development and operation of radioactive waste management facilities [12–45].

The IAEA defines disposal as the emplacement of waste in a repository with no intention of retrieving it in the future. Disposal is different from storage, which implies an intention to retrieve the waste in the future [46].

The IAEA Radioactive Waste Safety Standards (RADWASS) classification system [27] provides a generic approach to radioactive waste management and identifies suitable disposal options for different waste categories, based on their specific characteristics. For example, geological disposal is required for high level waste (HLW), spent fuel and certain long lived radioactive waste. Near surface disposal is a suitable option for short lived LILW, i.e. waste containing mainly radionuclides that will decay to radiologically insignificant levels within a few centuries. Limited quantities of long lived radionuclides may be accepted for near surface disposal [29, 30]. In the past, however, significant volumes of longer lived and relatively high activity waste were placed in near surface repositories.

Safety assessments are used to derive both generic and site specific waste acceptance criteria (WAC) [30] to identify acceptable waste characteristics, place limits on concentrations and inventories of radionuclides in the waste and potentially in the repository as a whole, and to specify waste form or package requirements.

Near surface disposal of radioactive waste has been carried out for more than 50 years. There are more than 80 near surface repositories around the world [26]. Other near surface facilities previously defined as 'storage facilities' are considered in this publication to be disposal facilities since the intention to retrieve the waste was not clearly established and they are similar, both in design and operation, to many existing disposal facilities.

The majority of near surface repositories place waste in disposal units located close to the ground surface. Disposal units are either excavated below the original ground level (e.g. earth trenches or concrete cells) or built above the original ground level (e.g. mounds and concrete structures). In either case, after operations, the disposal units are generally isolated from the surrounding environment by an engineered cover system several metres thick. Cover systems may comprise multiple layers designed to limit moisture infiltration and control biotic intrusion (e.g. plant roots or burrowing animals). Barriers to impede human intrusion may also be included. Near surface disposal also includes emplacement of waste in rock cavities or other disposal units (e.g. boreholes) at depths of up to several tens of metres. Repositories of this type exist in some Member States.

Many of these repositories were developed and began operations long before current regulatory requirements took effect or more recent site suitability guidance, technological advances, safety assessment methodologies and quality assurance systems became available. National laws, regulations and disposal methods have evolved and improved with time. Various Member States have ongoing programmes both to upgrade these facilities and/or to develop new near surface disposal facilities.

The term 'corrective action', as used in this publication, comprises all activities and measures undertaken to:

(a) Achieve compliance with modified regulatory requirements;
(b) Rectify an existing unsafe condition;
(c) Prevent an unsafe condition from occurring in the future;
(d) Respond to societal demands.

A binding international regime for radioactive waste management has been established through Article 12 of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [3]. This article states that “each contracting party shall in due course take the appropriate steps to review the results of past practices in order to determine whether any intervention is needed for reasons of radiation protection”. It is anticipated that the Joint Convention will result in safety reassessments of certain repositories and that corrective actions will be pursued based on these assessments.

While specific regulatory requirements and approval processes vary among Member States, regulatory oversight is an integral component of the corrective action planning and implementation process. In addition to regulatory body involvement, members of the local community, local government officials, non-governmental organizations, the general public and others may be involved in the corrective action process. Experience indicates that stakeholder interest may be an important consideration in the planning and implementation of corrective action. Many Member States have addressed this by planning and implementing stakeholder involvement programmes as part of the overall corrective action process. Examples of stakeholder involvement activities include interaction with existing or specially formed local community groups, public meetings, tours of the disposal facility, distribution of information material and press releases, public document repositories and Internet sites.

Quality management systems (QMSs) also play a role in the process. These programmes vary from country to country and have undergone substantial changes in recent years, reflecting increased adoption of ISO 9000 and ISO 14000 approaches [47]. Components of QMS programmes generally include controls for management organization, design, procurement, procedures and processes, documentation, inventory, inspections, tests, equipment calibration improvements in the event of non-conformance, and audits.

Upgrading measures are already being implemented or planned at a number of disposal facilities in numerous countries. Examples range from adoption of new WAC and container specifications to building of additional engineered barriers, installing hydrologeological cut-off walls, improving cover systems, waste retrieval and other measures. Extensive experience and information is therefore available on actions that may be employed to upgrade disposal facilities. This publication draws on such international experience.

1.2. OBJECTIVE

The objective of this publication is to provide guidance to Member States with near surface disposal facilities on approaches and technologies that can be used to:

(a) Identify potential corrective action needs;
(b) Assess options and select appropriate corrective actions;
(c) Plan and implement those corrective actions selected to enhance repository performance and safety both before and after closure.

It is anticipated that this publication will be useful to scientists, engineers, managers and others involved in assessing and improving the performance of near surface repositories, related regulatory compliance and public acceptance issues. The lessons learned from the application and evaluation of corrective actions may also be relevant in the development of new repositories or facilities that require additional disposal capacity.

1.3. SCOPE

In the context of this publication, the terms repository or disposal facility include facilities defined as such in the IAEA Radioactive Waste Management Glossary [46], and facilities that may have been designated as storage facilities but where the intent for future retrieval of the waste was not clearly established.

This publication discusses potential initiating events for considering or undertaking corrective actions. These events may include inadequate repository performance, evolving regulatory and legislative requirements, technological developments, societal concerns and other factors. Methodologies that can be employed to implement effective solutions to problems are discussed, including assessment of alternative options prior to selecting corrective actions, and the planning, implementation and verification of the specific measures adopted. Examples are provided of approaches and technologies that may be used to improve repository performance and safety. The appendix provides information on experience in various Member States on the upgrading of disposal facilities.

The scope of this publication includes near surface disposal facilities for low and intermediate level radioactive waste, including borehole facilities, rock cavity repositories and other near surface or deeper confinement facilities. Technical and non-technical issues for both operational and post-closure repository phases are covered. Disposal facilities for uranium mining and milling waste are beyond its scope. Similarly, this publication does not address remediation of contaminated sites unrelated to near surface disposal, or facilities designed specifically for discharge of radioactive liquids.

1.4. STRUCTURE

Section 2 provides an overview of near surface disposal system concepts and components. This includes short descriptions of different disposal system components, the stages in the development of a repository, and related monitoring and surveillance activities. Section 3 discusses the conceptual framework and sequence of the steps involved in the planning and application of corrective actions to near surface repositories. Section 4 describes examples of corrective action approaches and technologies. Section 5 summarizes the overall conclusions of the report. Case studies involving corrective actions planned or carried out in various Member States are presented in the appendix.

2. OVERVIEW OF NEAR SURFACE DISPOSAL SYSTEMS

This section gives an overview of disposal system concepts and components. It also summarizes the main phases of the repository development process, site monitoring and surveillance, and the use of safety assessment methodology.

The disposal system includes the near field and the surrounding geological media. The near field consists of the waste, engineered barriers (including waste form and containers) and the adjacent geological media disturbed by excavation and other construction and operational activities.

The natural barrier system consists of the geological media hosting the repository and any other geological formations contributing to waste isolation.

The biosphere is that part of the environment inhabited by living organisms. Radionuclides released from the repository or through the geological barriers may be diluted, retarded or concentrated before causing any radiological impacts to humans and other species. The purpose of a disposal facility is to limit radiological impacts to acceptable levels.

Experience with safety assessments of existing near surface repositories shows that the radiological impact of near surface repositories is generally linked to radionuclides that are long lived and/or highly mobile. However, radiological impacts involving short lived, high activity radionuclides may be a significant factor in both operational exposure and inadvertent human intrusion scenarios.

2.1. MULTIPLE BARRIER CONCEPT

The primary safety related objective of near surface disposal is to provide effective isolation of the wastes from the surrounding environment. Achieving this objective generally requires that disposal facilities be sited, designed, constructed, operated, closed and maintained to prevent or control the release of radionuclides to acceptable levels. In order to ensure that the disposal system is robust, a multiple barrier concept which utilizes the properties of the waste form, engineered barriers and the site's natural barriers to prevent or restrict the release of the radionuclides from the facility [31, 32] is generally selected. The relative contributions of the various barriers to the overall safety of the disposal facility will depend upon the characteristics of the waste, the site conditions and the disposal system concept, and will vary with time.

2.2. DISPOSAL SYSTEMS

This section discusses basic concepts and components of near surface disposal systems. Additional information is provided in Refs [31–45].

Selection of a particular repository design depends on many factors, including national radioactive waste management policies, waste characteristics and inventories, available site characteristics, climate conditions, technology and resource availability. Two main methods have been used for near surface disposal: shallow facilities close to the ground surface, or deeper facilities.

Shallow facilities consist of disposal units located either above (e.g. mounds) or below (e.g. lined or unlined trenches, vaults, boreholes, bunkers) the original ground surface. The cover over the waste is typically several metres thick and may consist of multiple layers engineered to limit moisture infiltration, control biotic intrusion and promote vegetation growth. Deeper confinement facilities allow waste to be emplaced in rock cavities or deep boreholes. The thickness of the soil and/or rock above the waste can be up to several tens of metres. These depths contrast with geological repositories for long lived radioactive wastes, where the wastes are emplaced at depths of hundreds of metres.

Near surface facilities are generally located above the water table. However, local conditions may allow or require the disposal modules to be constructed in the saturated zone. In both cases, the disposal units need to be designed and constructed to limit the flow of water into the repository and subsequent radionuclide migration. Major disposal system components generally include the following:

— The waste form;
— The waste container;
— Any additional engineered barriers;
— The natural barrier system.

The waste form may involve a solid matrix in which radionuclides are immobilized through treatment and/or conditioning prior to packaging. Some wastes may not be conditioned, in which case the waste form will consist of the originally contaminated material (e.g. paper, glassware, plastic, wood, animal carcasses), possibly in a compacted form to reduce void space. Different types of materials can be used to stabilize waste, e.g. cement [37], bitumen and polymers. Combustible wastes such as contaminated clothing, plastics, paper, wood and other organic matter may be incinerated and the ashes incorporated in a solid matrix [38].

The waste package, which consists of the waste form and container, may be designed to meet requirements for handling, transport, storage and disposal [31, 37–43]. Alternatively, waste may be transferred to different containers prior to disposal. To limit the release of radionuclides and other contaminants, some packages include additional features such as absorbents and impermeable liners. Concrete, polymer coated concrete, carbon steel, high density polyethylene and other engineered materials are also used for containers. Gas vents may be necessary if the disposal units incorporate impermeable barriers. The integrity of waste packages is important if they represent an engineered barrier or source of structural stability that is important to the safety case, and/or the ability to retrieve the waste is considered.

Additional engineered systems may consist of structural walls, solid or free draining backfill materials placed around the waste packages, chemical additives, low permeability soil or synthetic liners and covers. Depending on the disposal concept, these may be supplemented by other engineered components, including leachate collection and drainage systems, and impermeable subsurface cut-off walls. To ensure that the engineered barrier system is robust enough to perform as specified in the design, materials can be used that will maintain their function and integrity under anticipated repository conditions for the required period of time.

2.3. REPOSITORY DEVELOPMENT

The three main repository development phases are pre-operational, operational and post-closure [40, 44]. The pre-operational phase includes development of the disposal system concept, siting and design, licensing and construction. The operational phase includes waste emplacement and subsequent repository closure. The post-closure phase comprises maintenance and institutional control activities following repository closure. Activities related to each of these phases are described below.

The pre-operational phase begins with a conceptual design and siting stage. It starts with the development of the initial disposal concept, based on the nature and estimated quantity of waste requiring disposal, regulatory requirements, site availability and environmental constraints, availability of resources, waste transport routes, cost, and societal considerations. Construction of the repository may be phased to provide additional disposal capacity for waste received over time at the facility.

The operational phase generally comprises commissioning, waste receipt and emplacement, backfilling, covering of disposal units, operational monitoring and surveillance, and closure. Emplacement of waste comprises both the physical placement in the repository and subsequent management until that part of the repository is covered or sealed. The repository may have a number of units progressively constructed and used for disposal. As soon as a particular part of the repository is filled to capacity, the voids around the waste packages may be backfilled. It may also be necessary to protect that part of the repository with a temporary cover to limit infiltration of water and provide radiation shielding prior to closure.

The operational period of the repository generally lasts up to some tens of years. Closure begins after waste emplacement operations have been completed. Closure is generally carried out in accordance with an approved plan that includes an updated safety assessment and a description of the institutional controls intended for the post-closure phase. Typically, a final cover system is emplaced to:

(a) Control erosion and ensure the physical integrity of the repository;
(b) Minimize infiltration of water;
(c) Control plant, animal or human intrusion, which is particularly important for shallow or above ground disposal units, since the waste is emplaced relatively close to the surface.

The post-closure phase includes institutional controls as an integral part of the overall waste isolation system. The controls may include both active and passive measures. Active institutional controls comprise maintenance, monitoring and surveillance of the disposal site. These generally include:

(1) Cover system inspection and any needed repairs;
(2) Environmental monitoring;
(3) Maintenance of fences, signs and other physical components.

These active controls are conducted for periods ranging from several decades to a few hundred years. Ongoing monitoring and surveillance data can also be used to update safety assessments, as discussed in Section 2.5.

Passive measures may include disposal unit and site markers, land use and other legal restrictions, and archived records of waste inventories and their location within the repository. The purpose of passive controls is to preserve relevant operational records and reduce the likelihood of the wastes being disturbed. Provisions for ensuring that funds will be available for the post-closure phase are also important. Figure 1 shows the phases of the repository's life cycle, including the key activities during each phase.

2.4. MONITORING AND SURVEILLANCE

Monitoring is the continuous or periodic observation and measurement of radiological, environmental, engineered barrier performance and other relevant parameters. Surveillance consists of periodic inspections to verify that structures, systems and components relevant to the safety of the repository continue to function or are in a state of readiness to perform their functions. Monitoring and surveillance of existing disposal sites has provided valuable data on the performance of near surface disposal facilities and can contribute significantly to the upgrading of repositories.

Environmental monitoring covers a broad range of media, including atmosphere, surface and groundwater, soils, and plant and animal species that may be part of the food chain or indicators of radionuclide releases. Groundwater and vadose (unsaturated) zone monitoring may provide an early warning of the potential release of contaminants, including mobile radionuclides.

Pre-operational site environmental monitoring involves the collection of data, particularly those expected to vary seasonally or with time, to characterize the site and define ambient conditions. These data are used in initial safety assessments and engineered barrier design, evaluation of potential impacts due to construction, and identification of preferential water flow

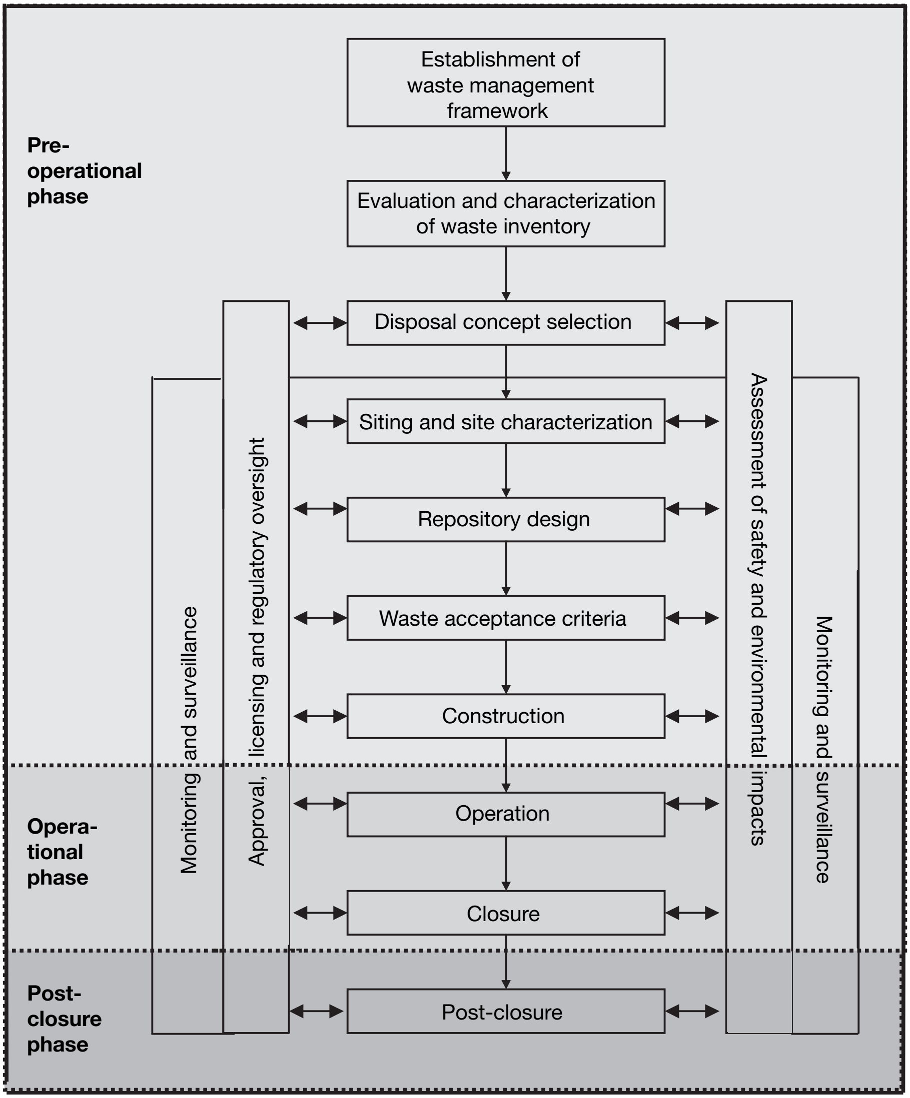

FIG. 1. Life cycle phases and activities of a near surface repository.

pathways. They may also serve as a benchmark for the testing of safety assessment models. The data also provide the necessary baseline for comparison with measurements taken during repository operation and following closure.

Monitoring during the operational phase is intended to confirm facility performance and provide data to refine assessment of repository impacts on the surrounding environment. Visual inspections and physical surveillance of disposal unit covers may be conducted to determine if their integrity has been compromised by erosion, cracking, subsidence, action of burrowing animals or deep rooted plants, and other processes. The covers may also be monitored to detect increases in water content or infiltration phenomena.

Technical requirements for monitoring during the post-closure phase are not expected to differ significantly from those during the operational phase. However, additional monitoring may be required to ensure the performance of cover systems and any other barriers installed at the time of closure. Monitoring frequency generally decreases with time after closure if the repository performs as expected.

Post-closure surveillance may include inspection of the repository cover, surface drains and monitoring systems. In addition, disposal facilities with leachate collection systems and drains may require additional surveillance. Fences and warning signs prohibiting access to the site may also need to be maintained. Periodic inspections will help ensure that land use restrictions and prohibitions are complied with.

2.5. SAFETY ASSESSMENT METHODOLOGY

In line with internationally agreed upon principles of radioactive waste management, the safety of near surface disposal facilities needs to be ensured during all stages of their life cycle.

In 1997, the IAEA launched a Coordinated Research Project (CRP) on Improvement of Safety Assessment Methodologies for Near Surface Disposal Facilities (ISAM) [48]. The main outcome of the project, completed in 2000, was the development of a harmonized, iterative methodology (ISAM methodology) for carrying out post-closure safety assessment of near surface disposal facilities. The methodology has since gained widespread acceptance and is to be published in a series of reports dealing with scenario development, modelling and confidence building, together with three documented test cases for vault, borehole and radon-type disposal facilities. Upon completion of the ISAM project, a need was recognized to further apply the ISAM methodology to a range of practical issues.

Accordingly, the IAEA organized a follow-up CRP on Application of Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities (ASAM) [49], built on the experience gained with the ISAM programme. This effort places special emphasis on application of the ISAM methodology to problems of topical interest, to:

(a) Explore practical application of the ISAM methodology to a range of near surface disposal facilities for a number of purposes, e.g. design concepts, safety reassessment and upgrading of existing facilities;
(b) Develop practical approaches to assist regulators, operating organizations and other specialists reviewing safety assessments.

The emphasis of the ASAM project is on evaluating the post-closure safety of radioactive waste disposal facilities. Where appropriate, operational safety may also be assessed. The ASAM programme is also intended to provide information and recommendations on:

(1) Development of an approach and methodology to assist in the selection of corrective action alternatives;
(2) Development of an approach to balancing radiological and non-radiological risks;
(3) Comparison of different options for corrective actions to assist in decision making for future repository development.

3. CORRECTIVE ACTION PROCESS

As stated in Section 1, corrective actions may be undertaken for one or more of the following reasons:

(1) Compliance with regulatory requirements;
(2) Prevention of an unsafe condition;
(3) Rectification of an unsafe condition;
(4) Response to societal demands.

This section provides information on the corrective action process, which includes definition of the initiating event, root cause analysis, potential corrective action identification and assessment, preferred action planning and implementation, and confirmation of effectiveness. There are many examples of successful repository development, but inadequate repository performance has also occurred, particularly in older repositories developed before substantial experience and international guidance were available. Examples of initiating events that could necessitate corrective actions are provided in the appendix, in Sections 3.1, 3.2 and 3.4, and in Refs [44, 45, 50–63].

Corrective actions may have different objectives, depending on the status of the disposal facility. Specific actions may be undertaken to:

(a) Achieve regulatory compliance;
(b) Respond to an accident or incident;
(c) Permit continuing operation;
(d) Improve current operational practices;
(e) Apply new technological developments;
(f) Expand disposal capacity;
(g) Restart disposal facility operations following a suspension;
(h) Prepare for final closure of the facility;
(i) Improve the performance of a previously closed facility;
(j) Address public and other stakeholder concerns.

Guidance on approaches to safety and dose assessment is provided in Refs [4, 11, 64].

A systematic process, such as that illustrated in Fig. 2, may be followed to ensure that appropriate corrective actions are identified and effectively implemented. The process illustrated is generic and may require modification to address each specific situation. The implementation process for each step may also vary. For many of the steps, the process may be iterative.

3.1. INITIATING EVENTS

Initiating events are defined as circumstances at a specific repository that may require corrective actions. For disposal facilities, the initial design and operational procedures may have been considered adequate for the protection of human health and the environment, but circumstances may have developed during the operational and/or post-closure stages that necessitate corrective actions.

While not mutually exclusive, initiating events may generally be grouped into the following categories:

(a) Changes in regulatory standards and requirements:
 - Changes in international standards and practices;
 - Member State regulatory changes, e.g. new regulatory and/or legislative requirements, including repository design requirements;

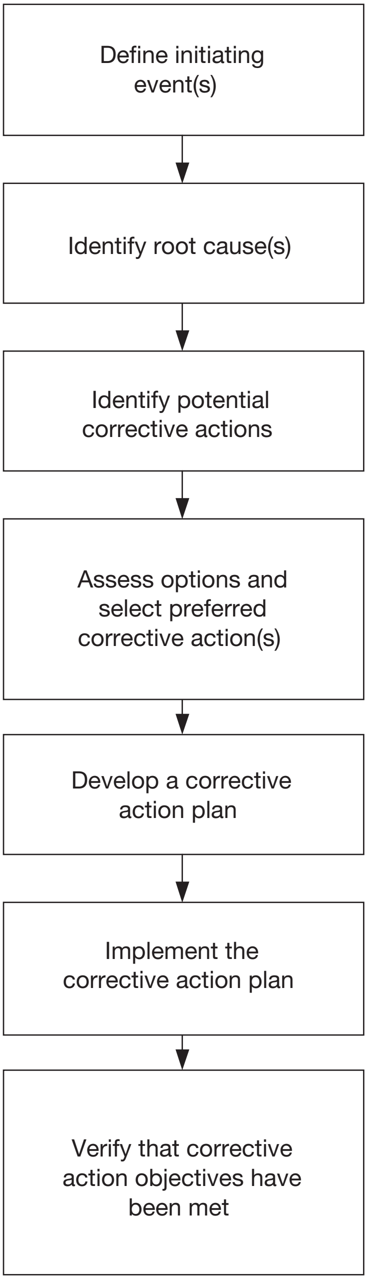

FIG. 2. The corrective action process.

— Repository specific requirements, e.g. new WAC, waste form or packaging requirements, a new source of waste requiring disposal;
— Fundamental changes in government policy leading to regulatory changes.

(b) Releases or operational exposures predicted to exceed safety standards:
— Safety system failures;
— Revised/new safety assessment results;
— Environmental monitoring results;
— Results of physical inspections;
— Premature degradation or failure of engineered barriers;
— Incomplete or inaccurate waste inventory information;
— Inadequate waste form and/or container specifications;
— Changes in physical environment;
— Independent audit or peer review findings.

(c) Actual releases or operational exposures that exceed safety standards:
— Environmental monitoring results;
— Radiation exposure;
— Accidents and incidents.

(d) Non-compliance with existing requirements:
— Operational practices;
— Post-closure care;
— Inadequate QMSs;
— Generator waste packaging or characterization.

(e) Stakeholder concerns.

3.2. EXAMPLES OF INITIATING EVENTS

Examples are provided below to illustrate the nature of various initiating events. The appendix gives examples of corrective actions carried out at specific repositories in response to various initiating events.

3.2.1. Change in regulatory standards and requirements

3.2.1.1. New QMS requirements

Many existing disposal facilities were initially designed, constructed and operated without an extensive QMS plan or procedures. The application of enhanced quality assurance processes to historical repository design, operations and safety assessments may indicate deficiencies requiring corrective action. Deficiencies may also extend to waste generator practices affecting repository performance (e.g. waste characterization, waste form or containers). Concerns may also arise regarding waste inventory documentation and other record keeping needs [47, 65].

3.2.1.2. New WAC for disposal

A key issue in the management of a disposal facility is the placing of limitations on the waste materials and contaminants accepted for disposal. This issue is addressed in most countries by the specification of WAC, identifying the types and classifications of waste that may be accepted. The WAC may reflect both generic requirements, such as those discussed in the IAEA guidance [28], and site specific restrictions. The WAC may also include specific concentration limits for certain radionuclides, with higher limits applying if the waste is conditioned and/or placed in engineered packages offering long term containment.

The WAC are generally based on current disposal regulations. If these change, the WAC may also change and may exclude wastes previously permitted for disposal, such as long lived and/or high activity LILW. This may require improved containment or removal of wastes disposed of in the past that do not comply with the new WAC. In most countries, for example, the regulations relating to chemical and radiological risks have evolved over the past 20 or so years. Hazardous chemicals previously permitted for disposal in the same facilities with radioactive waste have now been banned from disposal in certain repositories.

3.2.1.3. Fundamental change in government policy

Changes in government policy in certain Member States may cause a major change in waste management philosophy. For example, retrievable storage instead of disposal may be required. Alternatively, mined cavern or borehole disposal may replace near surface or above ground disposal. In such cases, significant changes to existing repositories or closure may be required by regulation.

3.2.2. Releases predicted to exceed, or exceeding, standards

3.2.2.1. Contaminants outside the containment barriers

If routine monitoring has detected elevated or unexpected levels of waste derived contaminants outside the containment barriers, this may indicate the failure of a safety system component.

3.2.2.2. Deterioration of waste packages

If unacceptable deterioration of the waste packages is observed, the impacts of package degradation may need to be assessed in relation to future safety performance of the facility and future waste form and packaging requirements. Potential impacts include disposal cap subsidence, increased infiltration of liquids if the deteriorating packages contain substantial void space, and accelerated release and transport of waste constituents. Waste package retrievability may also be adversely affected.

3.2.2.3. Gas generation and release

In a near surface disposal facility, there is a potential for generation of gases, particularly hydrogen, carbon dioxide and methane. Waste contained in impermeable barriers may need to include provisions for the venting of gases to ensure that gas pressurization within the isolation barriers does not occur [66–69]. Such venting may occur through natural processes (e.g. diffusion through a soil cover). If such venting does not take place or the proper design features have not been incorporated, the safety systems may be compromised.

3.2.2.4. Defective leachate collection system

Some near surface facilities include a leachate management system to remove leachate. The collected leachate may then be treated and discharged, or conditioned on-site. Defects such as clogged collection pipes or drainage layers, or inoperative leachate removal pumps, may compromise the safety system.

3.2.2.5. Clogging of drains

Drains may be employed in disposal facilities to ensure free drainage of infiltrating moisture away from the zone of waste emplacement. At such facilities, the clogging of drains may lead to retention of large volumes of water in contact with disposed wastes.

3.2.2.6. Changes to geomorphological or hydrological features

Physical changes to geomorphological or hydrological features of the site, such as changes in the direction or volume of water flow, erosion or seismic activity, could produce an immediate release of contaminants or affect long term performance and safety. This is particularly true where geomorphological or hydrogeological changes may have affected the groundwater pathway.

If there is reason to believe that the geomorphological or hydrological features have changed, e.g. by large scale construction work near the disposal facility, a revised safety assessment may be necessary to determine whether the modified groundwater pathway will affect disposal facility performance.

3.2.2.7. Accidents and incidents

Accidents and incidents, such as the dispersion of radioactive materials from a dropped or crushed waste package, failure to detect a damaged waste package prior to emplacement, a fire created by the disposal of unsegregated wastes or other unplanned events, may necessitate both immediate response and changes in ongoing practices to prevent their re-occurrence.

3.2.2.8. Safety assessment results

Improved safety assessment methodologies can be an initiating event for corrective actions if the results indicate that the repository will not meet current safety standards.

3.2.3. Stakeholder concerns

Expressions of public concern, based on perceived inadequacies in repository performance, the nature or extent of local benefits, or lack of confidence in the regulatory process, may initiate actions to increase community acceptance.

3.3. IDENTIFICATION OF ROOT CAUSES

A primary requirement for determining how to manage a problem is to focus on the underlying or root cause, and not only on the symptoms of the problem. Actions that do not address root causes may well lead to ineffective corrective actions. The root cause analysis may be relatively straightforward, especially early in the operational phase when most of the disposal system components are accessible and substantial amounts of waste have not yet been emplaced. For other problems, including those noticed during monitoring of previously capped disposal units or during the post-closure phase, extensive investigations may be needed to determine the root cause. Root cause analysis may be a useful tool for this exercise.

The first step in the identification of root causes is the assembly and analysis of existing information and the identification of information gaps that need to be addressed. The following example illustrates the importance of determining root causes. Elevated concentrations of radionuclides in groundwater may be caused by a variety of factors, e.g. inadequate waste form or packaging, excessive ingress of water into the repository, receipt of wastes that exceed the WAC limits, the presence of chelating agents, failure of a drainage or leachate collection system, or a combination of factors. Until the root cause or causes are specifically identified, the selection of corrective actions is unlikely to be optimized.

Safety assessment methodologies provide important tools for identifying root causes. If a safety analysis already exists, it can be used to compare predicted results against actual experience to help determine if a given repository component, e.g. waste package integrity or cover system effectiveness, is the source of the problem. If a safety assessment is not prepared, internationally accepted safety assessment methodologies [6, 48, 49] are available to guide this process.

There may be situations where existing information is insufficient to determine the root cause and further investigations are necessary. These information gaps may include:

(a) Baseline site characterization;
(b) Changes over time to site or repository conditions;
(c) Records on the types and amounts of waste emplaced in the repository;
(d) Knowledge of the physical and chemical characteristics of waste forms and related degradation;
(e) Knowledge of the performance of engineered barriers utilized in the repository;
(f) Extent of water ingress and egress;
(g) Environmental monitoring data for all relevant media.

If the issue is non-technical in nature, determining the underlying cause of the problem can be particularly challenging. For example, local community concern regarding the repository may be evident, but the specific reason or reasons for this concern may not be well understood. It is also possible that different stakeholders may have different concerns and opinions, requiring different solutions.

3.4. IDENTIFICATION OF POTENTIAL CORRECTIVE ACTIONS

A wide range of corrective actions may apply to a specific initiating event. It is advisable not to prematurely select an option or rule out potentially viable alternatives before an analysis of their advantages and disadvantages has been carried out. A list of corrective action alternatives can be identified by taking advantage of the substantial body of knowledge available through the published literature and the experience of other Member States, in combination with information specific to the repository requiring corrective action. Examples of corrective actions are listed in Table 1 and are described further in the appendix.

3.5. ASSESSMENT OF OPTIONS AND SELECTION OF CORRECTIVE ACTIONS

The assessment and selection of a preferred corrective action is a complex process involving diverse inputs and factors. These inputs may be grouped under four broad categories: safety/regulatory, cost, practicability, and technology.

Once a list of potential corrective action alternatives is developed, a preliminary screening step may be used to eliminate clearly unacceptable options. There are many ways of assessing and selecting a preferred corrective action or actions. Most methods use some form of decision making process, as illustrated in a general way in Fig. 3. Such methods typically consider both qualitative and quantitative factors.

A number of disciplines are typically involved in the evaluation process, including, but not limited to: engineering; health physics; earth, physical and biological sciences; cost estimation; cost–benefit analysis; public policy; regulatory and legal affairs.

Whatever process is chosen, it is advisable that the process be thoroughly documented as a means of establishing a decision record and explaining the approach taken to regulatory bodies and interested stakeholders. This record can also be preserved to assist in future corrective action evaluations.

Corrective actions capable of being implemented using existing and proven technologies offer advantages. If these do not exist, a research and development programme may be required.

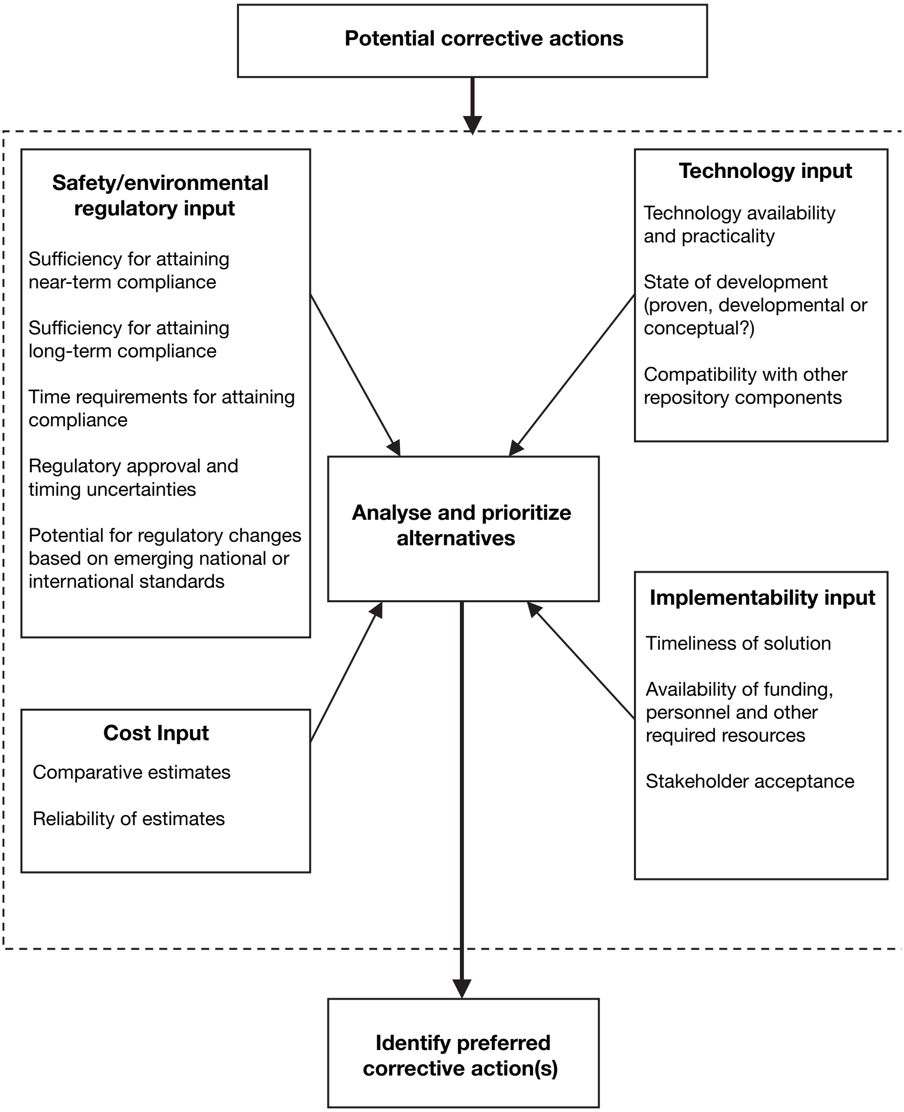

FIG. 3. Example of a decision making approach for selection of corrective actions.

The evaluation of corrective action alternatives generally includes risk estimates for both workers and members of the public. Trade-offs may apply between public risks and worker risks. In addition, risks to the environment may also need to be factored into the assessment of corrective action alternatives [8, 64].

TABLE 1. EXAMPLES OF CORRECTIVE ACTIONS

Corrective action objective	Initiating event	Possible corrective actions
Achieve compliance with changed regulatory requirements	New repository performance standards issued	Determine compliance based on updated safety assessments
		Implement any measures required to achieve compliance
	New or revised WAC adopted by repository	Impose revised waste classification requirements on generators and update waste inspection procedures at the repository
	National regulations issued requiring a new repository concept	Implement an action plan to phase out old technology and initiate a process to develop a replacement repository
Rectify an existing unsafe condition	Worker doses exceed safety standards due to inadequate procedures for handling high activity shipments	Immediately institute shielding and improve waste handling procedures to reduce doses to acceptable levels
	Groundwater contamination detected in local drinking water supply in excess of regulatory standards	Immediately protect the public by providing alternative drinking water
		Institute root cause analysis
	Incompatible wastes cause a fire, releasing unsafe levels of airborne contaminants	Revise operating procedures to ensure separation of incompatible wastes during receipt and emplacement
Prevent an unsafe condition from occurring in the future	Worker doses trending higher than normal ranges	Audit operational activities and change procedures as appropriate to reduce doses to normal ranges

TABLE 1. EXAMPLES OF CORRECTIVE ACTIONS (cont.)

Corrective action objective	Initiating event	Possible corrective actions
	Groundwater contamination detected at levels higher than predicted by the safety assessment	Determine significance based on updated safety assessment and evaluate monitoring programme sufficiency
		Design and install a clay or grout cut-off wall to control migration and enhance attenuation of contaminants
		Improve WAC and related QMS to exclude free liquids or uncontained chelating agents
		Install a cap to limit infiltration, and expand buffer zone to allow for monitored natural attenuation
	Failure of drainage system surrounding buried waste	Remove, redesign and replace drainage system
	Saturation of buried waste due to water table rise	Lower water table, and/or introduce engineering measures to prevent re-occurrence
	Saturation of buried waste due to cover failure from waste subsidence	Improve structural integrity of waste form and/or packaging
		Inject supplementary grout backfill to fill voids
		Repair covers
	Saturation of buried waste during the operational phase	Employ smaller trenches or progressively develop trenches to minimize exposure of waste to precipitation prior to cap installation

TABLE 1. EXAMPLES OF CORRECTIVE ACTIONS (cont.)

Corrective action objective	Initiating event	Possible corrective actions
	Lack of WAC has led to the disposal of unsegregated long lived, short lived, high activity, hazardous chemical, chelating agents, and other waste types	Determine significance based on safety assessment Establish suitable WAC or emplacement procedures Retrieve and segregate wastes, reprocess and repackage if necessary
	Gases resulting from decomposition or corrosion lead to airborne radionuclide releases at levels higher than predicted by the safety assessment	Determine significance based on updated safety assessment Revise WAC to exclude gas generating materials, e.g. wood, organic material, untreated biological waste
Respond to societal demands	Repository developed in the path of projected population growth	Retrieve waste and relocate the repository
	High degree of public concern that the repository is unsafe	Enhance communications programme to provide information and obtain input from the public on a regular basis
Achieve compliance with changed regulatory requirements	New repository performance standards issued	Determine compliance based on updated safety assessments Implement any measures required to achieve compliance
	New or revised WAC adopted by repository	Impose revised waste classification requirements on generators and update waste inspection procedures at the repository
	National regulations issued requiring a new repository concept	Implement an action plan to phase out old technology and initiate a process to develop a replacement repository

TABLE 1. EXAMPLES OF CORRECTIVE ACTIONS (cont.)

Corrective action objective	Initiating event	Possible corrective actions
Rectify an existing unsafe condition	Worker doses exceed safety standards due to inadequate procedures for handling high activity shipments	Immediately institute shielding and improve waste handling procedures to reduce doses to acceptable levels
	Groundwater contamination detected in local drinking water supply in excess of regulatory standards	Immediately protect the public by providing alternative drinking water Institute root cause analysis
	Incompatible wastes cause a fire, releasing unsafe levels of airborne contaminants	Revise operating procedures to ensure separation of incompatible wastes during receipt and emplacement
Prevent an unsafe condition from occurring in the future	Worker doses trending higher than normal ranges	Audit operational activities and change procedures as appropriate to reduce doses to normal ranges
	Groundwater contamination detected at levels higher than predicted by the safety assessment	Determine significance based on updated safety assessment and evaluate monitoring programme sufficiency Design and install a clay or grout cut-off wall to control migration and enhance attenuation of contaminants Improve WAC and related QMS to exclude free liquids or uncontained chelating agents Install a cap to limit infiltration, and expand buffer zone to allow for monitored natural attenuation

The potential effect of a particular approach on the performance of other aspects of the disposal system is an important consideration. As an example, additional barriers may influence the performance of the entire repository, resulting in changes to licence conditions, including possible changes (increases or decreases) in waste acceptance limits. Also, the addition of barriers may reduce disposal space. Conversely, waste retrieval may increase available capacity.

The availability of resources, as well as the funds to secure the resources, should also be taken into account in prioritizing the various corrective action options. This may be a major challenge in undertaking the upgrading of repositories. In certain cases, there may be insufficient funds, trained personnel or available technologies to undertake a preferred alternative. Determining resource availability at an early stage in the decision making process can avoid unnecessary delays and redirection of efforts. A cost–benefit analysis may be undertaken to identify the lifetime cost of the improvements, including purchase, maintenance and decommissioning (where relevant).

In general, when deciding on a corrective action strategy, the practical limitations associated with its implementation should be considered. This could lead to other, less complex or less costly but effective, options being chosen. For example, the relatively high cost of robotics technology and the difficulties of deploying qualified personnel have limited their use in repositories.

In many Member States the operation of a repository involves consultation with the public, with emphasis on the local community. Existing facilities might not be acceptable to particular stakeholders for a variety of reasons. In this case, corrective actions may involve efforts to better communicate with the public [70]. Public input on alternatives may be useful in the prioritization and selection of corrective actions.

Following careful analysis of initiating events and impact significances, no further action may be indicated. These analyses may indicate that reliance on natural processes, including sorption, retardation (physical, chemical and biological) and radioactive decay, will adequately mitigate the impacts. With this option, increased monitoring and ongoing safety assessments may be necessary to demonstrate that natural processes are indeed reducing contaminant concentrations to acceptable levels.

3.6. DEVELOPMENT OF A CORRECTIVE ACTION PLAN

Once the preferred option has been selected, a planning process is generally established to implement the selected option. The plan needs to define the sequence of processes and procedures to be followed while implementing the corrective actions. This includes:

(a) Comprehensive identification of all the activities needed to develop the plan;
(b) The schedule and critical path for delivery;
(c) Identification of critical interfaces (task linkages);
(d) Resources required(personnel, skills, infrastructure and materials);
(e) Financial requirements and sources of funding;
(f) Risks to the plan delivery and risk mitigation plan;
(g) A communications plan for relevant stakeholders;
(h) A QMS.

A detailed schedule of activities is needed to ensure that the timing for completion of the various activities is well defined, realistic and understood by all participants in the process, and that timing sequences and critical path task linkages are identified and managed.

The resources required need to be identified and a plan implemented for securing them on a timely basis. The cost of all activities needs to be thoroughly evaluated to develop a realistic budget to ensure timely delivery within funding limits.

Risks to delivering the plan need to be analysed and documented, with mitigation and contingency plans to control the risks. A major risk for many projects at disposal waste facilities is the ability to secure timely regulatory approvals.

A communications plan may be prepared that identifies the stakeholders of the process and the communications plan for each stakeholder group. A continuous process may minimize disruptions of the process and delays caused by a lack of timely and relevant information.

3.7. IMPLEMENTATION OF THE CORRECTIVE ACTION PLAN

Corrective action activities are generally carried out in accordance with an adopted corrective action plan and in coordination with the regulatory organizations responsible for oversight of the facility. Milestones with scheduled completion and inspection dates established during the planning phase may be used to track progress.

Specific instructions provided to those performing the work help ensure proper quality control. These instructions typically address responsibilities, work scope and procedures to be followed in performing each task.

3.8. VERIFICATION OF CORRECTIVE ACTION OBJECTIVES HAVING BEEN MET

Indicators and performance measures are typically defined to provide a basis for confirming that the corrective actions have accomplished their intended purpose. Repository conditions, prior to implementation of corrective actions, provide a basis against which the effectiveness of the corrective actions can be assessed.

Typical indicators of the effectiveness of corrective actions may include reduced radionuclide concentrations, less leachate production, stable trench cover contours (indicating reduced or stabilized subsidence), reduced site boundary radiation levels, or reduced radiological contamination in the surrounding environment.

In certain cases, confirmation of the effectiveness of corrective action may be difficult to determine in the short term. For example, the effectiveness of reductions in the disposed inventories of long lived radionuclides is best evaluated based on monitoring and safety assessments conducted over time, rather than immediate physical measurements. In the case of public involvement programmes, the measures of effectiveness will generally be subjective in nature.

A formalized QMS provides a basis for developing and documenting reliable information for regulators and stakeholders. This generally includes controls on organization, design, procurement, procedures and processes, documentation, inventory, inspections, tests, equipment calibration, improvements in the event of non-conformance, audits, and continuous improvement. A communications plan established during the planning phase can provide a useful tool for the transfer of information on the progress of work to the various stakeholders.

Documentation is recommended to provide a complete record of the effectiveness of the corrective actions. The verification activities may be confirmed by independent organizations, audits and peer review in addition to the required regulatory body reviews and approvals.

4. EXAMPLES OF CORRECTIVE ACTION APPROACHES AND TECHNOLOGIES

The lessons learned from managing the initiating events described above, as well as others, have led to the adoption of improved approaches and technologies for the near surface disposal of radioactive waste. General information on this topic is provided in this section. Case studies from Member States are given in the appendix.

4.1. ENGINEERED FEATURES

Engineered features are human made elements incorporated in the design of the disposal system to more effectively isolate the waste from the biosphere. This broad definition includes waste form, containers, backfill and buffer material, concrete boreholes or vaults, trench liners, covers, drainage control, leachate collection and removal, gas venting and any other feature designed to improve repository performance. The actual need for engineered features is dictated by site and waste specific considerations and safety assessment results. Some examples of the use of engineered features are provided below.

4.1.1. Repair or installation of surface caps

In some older disposal facilities, disposal units were not adequately covered, resulting in excessive infiltration and accumulation of rainwater. This water accumulation caused degradation of waste containers and subsequent leaching of radionuclides out of the repository. Erosion and burrowing animal intrusion at some repository sites have also occurred as a result of inadequate covering of the emplaced waste. This also has the potential for mobilizing radionuclides into the surrounding environment.

Surface caps, comprising both natural (e.g. clay) and human made materials (e.g. concrete, geomembranes, bitumen), have been emplaced to control water infiltration, provide intrusion resistance, control gaseous emissions, reduce erosion and attenuate radiation. Multilayered caps may contain an upper vegetative layer, a protective layer to prevent erosion and animal burrowing, a drainage layer, and a hydraulic barrier or low permeability layer. The layer thickness, permeability and materials chosen reflect specific site requirements and performance objectives. The design of surface caps generally takes settling and weathering effects into account [45].

If the corrective action requires reduced infiltration, the installation of asphalt and concrete caps may prove effective in the short term. In the longer term they are subject to degradation by weathering, cracking and subsidence.

Cap repairs, sometimes extensive in nature, have also been necessary at sites experiencing both operational and post-closure subsidence due to degradation of waste packages and localized subsurface settlement.

Re-vegetation intended to minimize infiltration and control erosion may require periodic reseeding and/or watering to establish the intended plant cover.

4.1.2. Installation of vertical hydraulic conduits

Disposal systems located in earth materials containing layers of variable permeability may experience saturation, even if located above the water table and protected by effective covers. This has been observed following periods of heavy rainfall, when temporary saturation occurs in the layers overlying less permeable strata. Horizontally migrating water in temporary, perched aquifers or saturated zones may be intercepted by vertical, high permeability drainage systems that divert horizontally flowing water away from the repository. Hydraulic conduits of this type, generally consisting of gravel or crushed rock, may be constructed during operations or after closure of the disposal units. Installations of this type, undertaken after waste has been emplaced, may raise worker exposure if there is a potential to unearth buried high activity, high dose wastes.

4.1.3. Installation of cut-off walls

Cut-off walls may be used to direct upgradient groundwater away from the repository or to channel contaminated water from the repository to a collection and treatment system. The intended effect in both cases is reduction in the potential for radionuclide migration from the repository.

Cut-off walls may be effective for restricting the lateral migration of contaminants and controlling water infiltration into the repository. The potential effectiveness of cut-off walls depends on a number of factors, including the physical characteristics of the soil (e.g. homogeneity, permeability and porosity) and the depths of both waste and groundwater. A cut-off wall may comprise a trench backfilled with clay, or a cement based grout curtain formed by pressure injection through pipes or augers. As with vertical conduits, the excavations required to install a cut-off wall may unearth buried wastes.

4.2. WASTE RETRIEVAL

In some Member States, radioactive wastes have been retrieved from storage/disposal facilities and repackaged for subsequent disposal either at the same site or at another repository. Retrieval may also be a corrective action option in cases where sealed sources containing long lived or high activity radionuclides or other problematic wastes have been placed in shallow boreholes or other near surface repositories. Actual experience with waste retrieval is limited.

Conceptually, retrieval techniques may be straightforward if the waste remains well containerized and has not been immobilized in a medium such as concrete. If the waste is not packaged or existing packages have substantially degraded, removal and subsequent repackaging may be difficult. If the waste has been immobilized in bitumen or cement or has been grouted in situ, cutting and drilling equipment may be required. In cases of high dose rate package removal, the use of shielding and mirrors may be used in combination with remote handling and/or overpacking to limit worker dose.

Improved waste characterization and/or volume reduction may be included as part of the retrieval process prior to redisposal. Upgrades to the disposal facility, e.g. repair or placement of additional engineered barriers or drainage systems, may also be carried out following waste retrieval. Volume reduction, e.g. compaction or incineration, provides a denser waste form and therefore allows more efficient use of available disposal space at the time of redisposal. Improved waste characterization may provide valuable information for updating safety assessments to improve predictions of future facility performance, or detect the presence of free liquids or other wastes within the containers that would not meet the acceptance criteria for final disposal.

In the case of borehole facilities, removal of the entire borehole unit, including the waste, has been accomplished by Ontario Power Generation in Canada by:

(a) Overcoring the borehole;
(b) Removing the soil in the annular space between the borehole and the overcoring drill (with shielding to afford worker protection);
(c) Filling the annular space with concrete;
(d) Removing the borehole intact to another storage facility.

4.3. IN SITU WASTE STABILIZATION

A repository may be upgraded by provision of additional containment or direct in situ stabilization of the waste if the waste packages no longer provide satisfactory containment.

In situ stabilization of the waste may be undertaken in relatively old disposal facilities in which waste containers have corroded or collapsed, or where the waste has been disposed of without appropriate consolidation or packaging. In situ injection of grouts may be used to encase the wastes in a monolithic structure, thereby minimizing contact between water and waste.

Consideration needs to be given to possible adverse interaction between waste constituents and grouting material. For example, the presence of certain organic compounds could inhibit the setting of cement based grout.

4.4. WATER COLLECTION AND EXTRACTION

Certain disposal practices have resulted in migration of radionuclides from the disposal facility to the nearby biosphere. Typical mechanisms include transport of radionuclides by groundwater with subsequent contamination of surface waters.

In such cases, conventional pumping systems may be used to extract contaminated groundwater. A good understanding of regional hydrogeological flow patterns will help ensure effective design and placement of the extraction well or wells and optimization of the pumping rates required for effective control.

Other water collection and extraction technologies include buried conduit pipes equipped with pumping equipment and below grade trenches to direct and collect contaminated shallow groundwater by gravity flow. These technologies can be used as means to intercept up-gradient infiltrating water or a plume of contamination.

5. CONCLUSIONS

This report considers a variety of circumstances that may require corrective actions to be assessed or implemented at near surface disposal facilities. In the context of this publication, the term ‘disposal facility’ includes those facilities defined as such in the IAEA Glossary, and those facilities that may have been designated as storage facilities but where the intent for future retrieval of the waste was not clearly established or specified. The circumstances leading to the corrective actions, or the corrective actions themselves, may be of either a technical or a non-technical nature.

A systematic process is advisable to help ensure effective planning and implementation of the corrective actions. The steps involved in the process reflect such standard project management practices as identification of activities, scheduling, controlling interfaces, managing resources, financial and progress tracking, risk identification and management, and communications.

A wide variety of corrective actions is available to remedy near surface disposal facility performance deficiencies and related regulatory and stakeholder issues. Extensive literature and international experience exists to provide guidance in the selection of corrective actions appropriate for application to specific repositories.

(a) Corrective actions may be undertaken to achieve one or more of the following objectives:
- Comply with new regulatory requirements;
- Rectify an existing unsafe condition;
- Prevent an unsafe condition from occurring in the future;
- Respond to societal demands.

(b) The corrective action process involves the following sequential steps:
- Defining the initiating event(s);
- Identifying root causes;
- Identifying potential corrective actions;
- Assessing options and selecting the preferred corrective actions(s);
- Developing the corrective action plan;
- Implementing the plan;
- Verifying the effectiveness of the corrective action(s) implemented.

(c) Initiating events may be grouped into the following general categories:
- Changes in regulatory standards and requirements;
- Releases or operational exposures predicted to exceed standards;
- Actual releases or operational exposures exceeding standards;
- Non-compliance with existing requirements;
- Stakeholder concerns.

(d) The assessment and selection of a preferred corrective action is a complex process involving diverse inputs, considerations and factors. These inputs may be grouped under four broad categories:
- Safety/regulatory;
- Technological;
- Practicability;
- Cost considerations.

Appendix

EXAMPLES OF CORRECTIVE ACTIONS IMPLEMENTED AT REPOSITORIES IN MEMBER STATES

This appendix provides examples of near surface repositories which at some time have been considered to require corrective actions, and the specific approaches used to implement such actions. The list is not exhaustive but should allow an appreciation of the types of actions that were implemented and of their efficacy.

A.1. BELARUS: ACHIEVING SAFETY AT RADON TYPE WASTE DISPOSAL FACILITIES

A.1.1. Introduction

The major issues that Belarus has confronted over the past decade in the area of radioactive waste management are addressed in Articles 12 and 28 of Ref. [3] and linked to upgrading of the Ekores National Radioactive Waste Disposal Facility, which belongs to the class of 'RADON' type facilities. (These facilities are so named because their designs were based on the same concept as that of the two central facilities near Moscow and St. Petersburg operated by the Scientific and Industrial Association RADON).

Safety issues which may raise stakeholders' anxiety for most of the existing RADON type facilities are related to their three key features:

(1) The facilities contain extra fractions of α emitters and long lived β and γ emitters in the near surface repositories;
(2) The question of safety of their borehole repositories for spent sealed sources has not been settled;
(3) They are located in the vicinity of densely populated localities.

Owing to the above considerations and especially to the long lived nature of the waste disposed of at the facilities, any measures taken to upgrade their operational conditions and safety may, in terms of public reaction, result in effects opposite to what is expected. Belarus met with such a situation when implementing its national project for upgrading and rebuilding the Ekores facility. After four years of relatively successful activities, work under the project was stopped because of great public pressure. Following many discussions at different levels, an advanced reconstruction strategy for Ekores has been developed and further efforts have been made to modernize the relevant technical solutions.

A.1.2. Reasons and objectives of the Ekores facility rehabilitation

A.1.2.1. Background

The Ekores facility was commissioned in 1964 in the vicinity of Minsk, a city with a population of approximately 2 million, and was intended for LILW storage/disposal [71]. The site comprises 2 older (historic) concrete lined trenches, each 4 m deep, and 2 subsurface reinforced second generation concrete vaults, each 3 m deep, filled with solid waste. The 2 trenches and one of the vaults are closed and the operating vault is 75% full.

There are also four ‘old’ borehole repositories (so-called ‘wells’) with S shaped loading channels for ‘free’ disposal of spent sealed radioactive sources (SSRSs). In July 2003, free SSRSs in wells were immobilized into a lead matrix in situ, using the technology developed by RADON [72].

A.1.2.2. Initial approach

Taking into account that the Ekores site had no waste segregation or waste processing procedure, no equipment for unloading containers with SSRS, no premises or facilities for the decontamination of vehicles and equipment, and no monitoring boreholes, the necessity of upgrading the facility was recognized immediately after a new regulatory regime had been established in Belarus. A national project for reconstruction of the Ekores facility was launched in late 1997. The project covered:

(a) Upgrading of the existing structures (garage, decontamination unit, fence);
(b) Construction of 3 new structures (building for SSRS disposal, building for waste predisposal treatment, vault for solid waste disposal);
(c) Introduction of more advanced technologies for the safe handling of solid and liquid LILW.

The IAEA supported this national activity by providing Ekores staff with the relevant training, expertise and equipment support.

A.1.2.3. Reasons for the development of an advanced approach

IAEA support contributed greatly to the evaluation of potential hazards posed by the Ekores site [73]. In particular, safety assessment showed that:

(a) First generation trenches contain waste with a concentration of transuranic radionuclides in excess of 4000 Bq/g;
(b) Irradiated fuel from the research reactor containing about 2 kg of uranium (in 10 stainless steel containers) is buried in the now closed vault;
(c) Owing to the migration of the radioisotopes ^{14}C, ^{36}Cl, ^{60}Co, ^{3}H, ^{239}Pu, ^{226}Ra, ^{90}Sr and ^{238}U from the Ekores repositories, contamination of groundwater within the supervised area may exceed the limits for drinking water;
(d) The long lived radionuclides ^{239}Pu, ^{226}Ra, ^{241}Am and ^{232}Th represent a danger for future generations, as long lived radiotoxic isotopes may contaminate the aquifers in the near future.

In view of public concern over the disposal of long lived radionuclides near Minsk, funding for the reconstruction was suspended and an advanced strategy for its reconstruction has been developed. This strategy is described below.

A.1.3. Advanced strategy for reconstruction of the Ekores facility

The advanced strategy states that the Ekores facility is regarded as a facility for long term storage of waste, not for disposal. All the wastes at the Ekores vaults should be identified, conditioned, packaged and labelled to ensure that the conditioned waste packages meet the most stringent safety requirements for waste storage and transport. There is a requirement to revise the national project by amending it with a section concerning the development of procedures for retrieval of waste from the existing repositories. The retrieved waste is to be sorted and conditioned into acceptable transport packages using the same technologies as those to be implemented for new incoming wastes. The main goal of the reconstruction is to provide a flexible means of relocating long lived waste to a new disposal/storage repository. The programme for siting a new repository is now in the approval process by the Government.

As for fissile material which is present in the vault, it is proposed that in the course of retrieval operations the intact containers should be removed and transported to an approved storage facility for fissile material. It is anticipated that international experts will conduct the safety assessment and develop a detailed plan for the appropriate management of this type of waste.

A.1.3.1. Strategy for managing solid radioactive waste

Solid radioactive waste requiring treatment and storage at the Ekores site will come from 2 sources, new incoming wastes and those arising from waste retrieval operations. Compactable wastes will be placed into mild steel 200 L drums and each drum will be compacted, capped with grout and consigned for storage. Non-compactable wastes will be placed in a 200 L drum during sorting, then directly grouted in situ and consigned for storage.

The packages of treated SRW will be temporarily placed in approved surface storage until the vault is empty. The surface of the vault should then be decontaminated, monitored and subjected to a structural survey. After being repaired it should be used for continued storage of the drums of cemented waste.

A new building for waste sorting, treatment and packaging is planned. A site drainage system, decontamination centre, laboratory and administrative block are also included in the project.

A.1.3.2. Strategy for management of spent radioactive sources

To enhance the safety and security of SSRSs in the old wells and to facilitate their subsequent retrieval, sources have been immobilized in metal matrices in situ. The strategy for new incoming sources is to separate them into different types, then store them in a retrievable manner. A new building for the long term storage of SSRSs has been constructed. It has been equipped with 11 modernized borehole repositories.

A.1.3.3. Modernized borehole repositories for retrievable SSRS storage

The typical design of RADON borehole SSRS repositories has been modernized in order to provide the possibility of retrieving SSRSs according to the new reconstruction concept. To achieving this, a loading channel of the well consists of 3 sections [74]. The bottom and top parts of the sections are designed to ensure their butt joint connection when in operation, and disconnection in case retrieval is required. The weight of each section is about 3 t and the weight of the underground reservoir fully filled with conditioned SSRSs is 2.2 t, meaning that a standard crane mechanism can be used for reservoir retrieval. The modernized borehole repositories are intended for storage of short lived SSRS which have half-lives of less than 30 years, mainly ^{60}Co, ^{137}Cs and $^{90}Sr/^{90}Y$.

Sources with very long half-lives of much more than 30 years are mainly 'smoke detector' type sources containing ^{239}Pu and ^{241}Am, both of which can be handled without β/γ shielding. Special borehole facilities have been constructed for storage of these SSRSs [74].

A.1.4. Conclusions

The experience of Belarus outlined above has demonstrated that ìmaking all reasonable improvements in order to upgrade the safety of existing RADON type facilities may prove to be a much more complex task than had initially been expected. Some issues remain, mainly concerning:

(a) Public perception of the presence of extra fractions of α emitters in near surface repositories;
(b) Technical procedures for safe retrieval and sorting of waste;
(c) Selection of a solution to achieve safety of the existing RADON design wells;
(d) Long term safety considerations — the existing facilities should be upgraded so as not to create future problems.

A.2. BULGARIA: NOVI HAN RADIOACTIVE WASTE REPOSITORY

A.2.1. Novi Han radioactive waste repository

A.2.1.1. Background

The Novi Han radioactive waste repository is the only national radioactive waste disposal site in Bulgaria. It is located in Losen mountain, 6.5 km from the small village of Novi Han and 35 km from the capital, Sofia. The repository accepts radioactive waste generated from nuclear applications in industry, medicine, research and education. The facility was constructed according to a modified Soviet design (type TP-4891). Its construction licence was issued in 1959 and that for commissioning in 1964. The repository was specially built for the needs of the IRT-2000 research reactor operated by the Institute of Physics and other academic and medical facilities.

In 1959, the Government appointed the Physical Institute of the Bulgarian Academy of Sciences, whose successor is now the Institute for

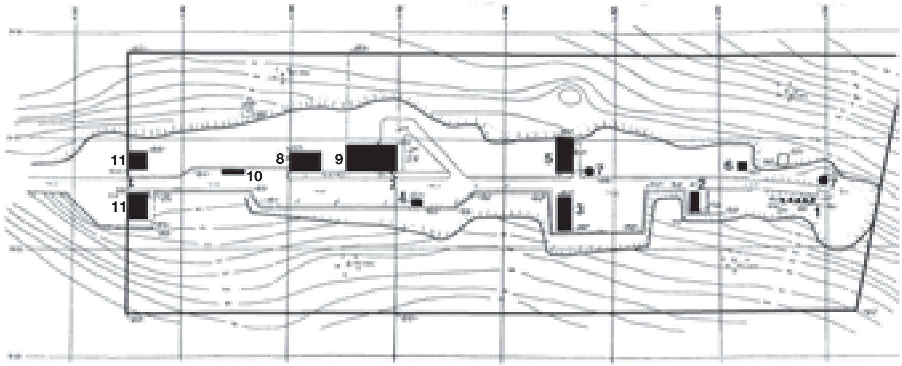

FIG. 4. Layout of facilities and buildings at Novi Han (1: trench for solid wastes, 2: vault for biological wastes, 3: vault for solid wastes, 4: storage, 5: sump water storage tanks, 6: vault for sealed sources, 7: monitoring boreholes, 8: garage, 9: radiochemical laboratories, 10: auto channel, 11: administration builidings).

Nuclear Research and Nuclear Energy (INRNE), as the central authority for the collection and disposal of radioactive waste from nuclear applications.

A.2.1.2. Description of the Novi Han repository

The Novi Han repository site covers an area of 4.25 ha.[1] The site is divided into two areas separated by a fence (Fig. 4). One area contains the administrative buildings, garage and maintenance shops. The other contains the disposal facilities, radiochemical laboratory and decontamination station.

The repository consists of several different disposal vaults:

(1) A concrete vault for low and intermediate level solid wastes, which consists of 3 separate cells with a total volume of 237 m^3;
(2) A concrete vault for biological wastes with a volume of 80 m^3;
(3) Four steel tanks for storage of low level liquid wastes with a total volume of 48 m^3;
(4) A special 1 m^3 concrete vault for spent sealed sources;
(5) A concrete trench for solid waste, which consists of 7 separate units with a total volume of 200 m^3.

[1] 1 ha = 1.00×10^4 m^2.

The disposal vault for spent sealed sources is full. The vaults for solid and biological waste and the engineered trench still have capacity for disposal of additional waste.

All the disposal vaults are engineered disposal structures (Figs 5–7) constructed of reinforced concrete with stainless steel linings, additional brick walls and asphalt insulation. The vaults are underground, with only the roofs above ground level. The four steel tanks for liquid waste are located in a reinforced concrete underground room. The reinforced concrete vault for spent sources is a cylinder 5.5 m below the surface. Heavy concrete and 5 lead slabs placed between the disposal vault and the ground surface provide protection from radiation. The engineered trench was constructed in 1984 and is the only facility with a drainage system.

All disposal units are near surface engineered multibarrier disposal facilities. Barriers to retard migration of radionuclides from the disposal vaults to the environment include reinforced concrete, stainless steel lining, hydro insulation and the site's natural geological barrier (clayey phyllite schists). The waste form itself is not considered to be a barrier.

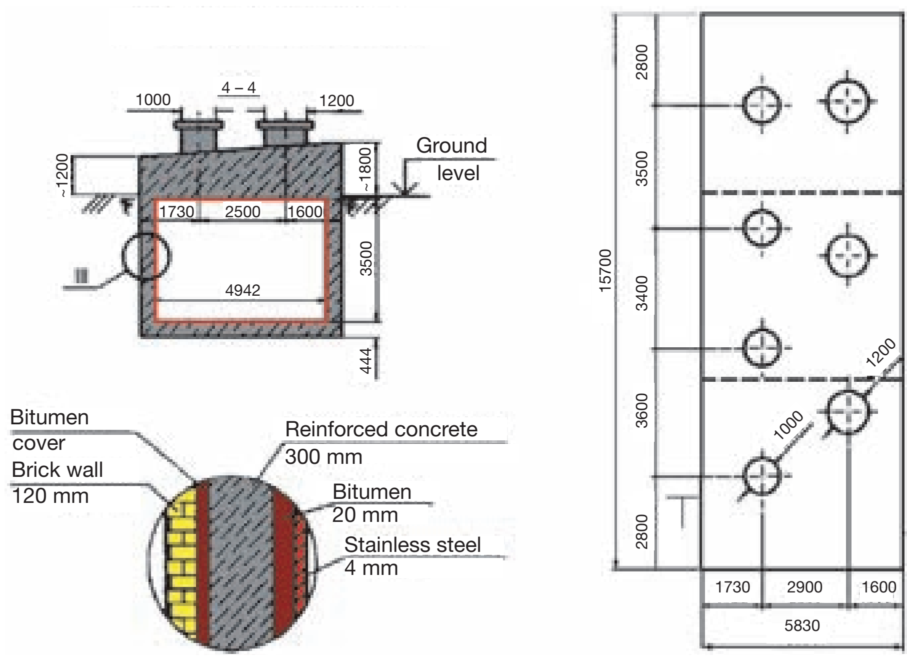

FIG. 5. Disposal vault for solid waste.

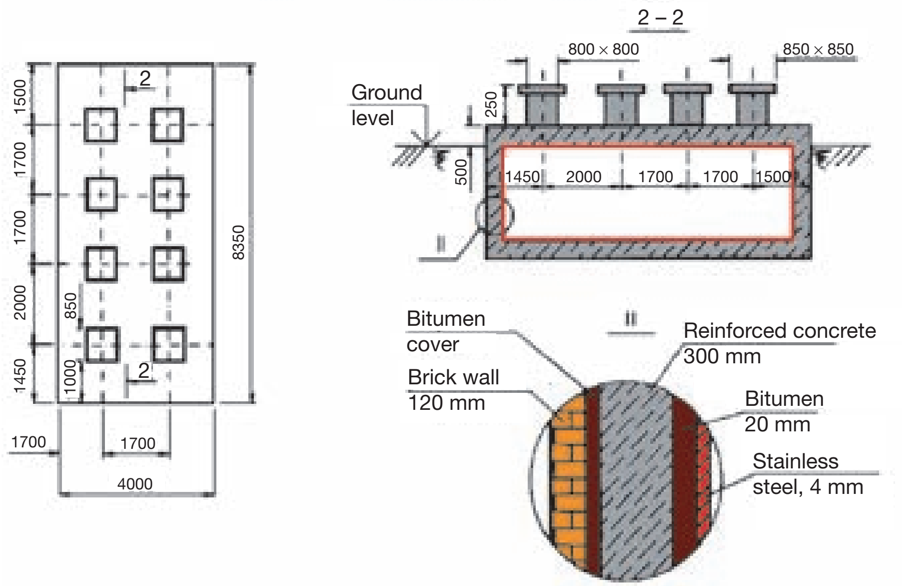

FIG. 6. Disposal vault for biological waste.

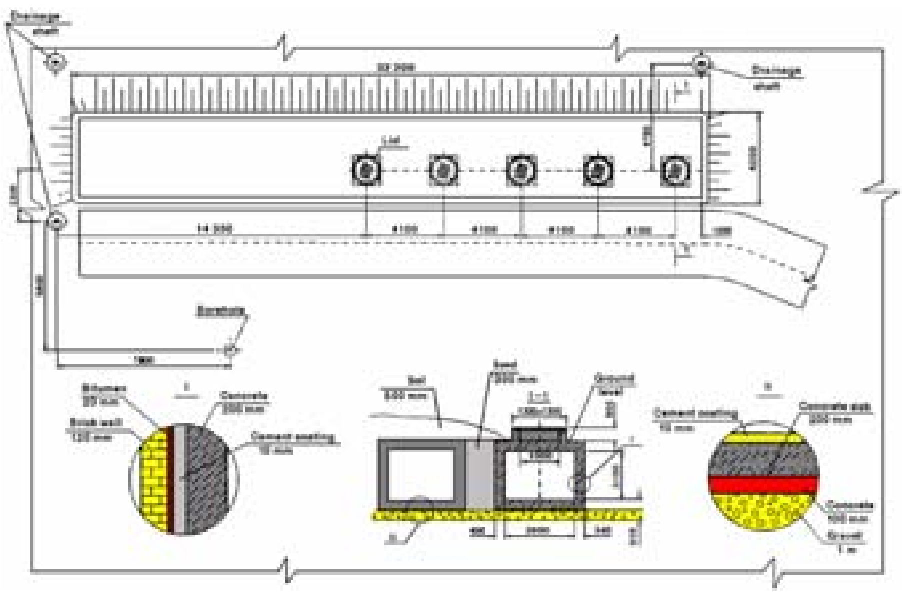

FIG. 7. Engineered trench for solid waste.

A.2.1.3. Waste acceptance criteria

Waste acceptance criteria follow Regulation No. 7 of the Committee on the Use of Atomic Energy for Peaceful Purposes on Collection, Treatment, Storage, Transport and Disposal of Radioactive Wastes in the Territory of the Republic of Bulgaria [75]. They take into account:

(a) Origin:
  (i) Radioactive wastes from industry, medicine, agriculture and scientific research are accepted for storage or disposal.
  (ii) Wastes from the uranium mining and milling industry and the Kozloduy nuclear power plant are not accepted.

(b) Activity and radionuclide inventory:
  (i) Solid and solidified low and intermediate level short lived wastes are accepted.
  (ii) Limits for α activity for solid and solidified LILWs:
    — Average activity of α radionuclides for a facility is limited to 370 MBq/t;
    — Specific α activity is limited to 3.7 GBq/t;
    — Specific α activity for single package solid waste is limited to 0.19 GBq/t;
    — Limits of ^{226}Ra and ^{232}Th for single package solidified waste: up to 3.7 MBq/t ^{226}Ra and 1.1 MBq/t ^{232}Th;
    — Limits of ^{226}Ra and ^{232}Th for single package solid waste: up to 3.7 MBq/t for ^{226}Ra and ^{232}Th;
    — In special cases the disposal of radioactive waste with specific α activity from 3.7 to 18.6 GBq/t might be permitted.
  (iii) Very short lived wastes (half-life <15 d) are not accepted for disposal.

(c) Waste form restrictions:
  (i) Waste that contains free liquids is prohibited.
  (ii) Flammable and explosive wastes are prohibited.
  (iii) Liquid wastes are accepted after neutralization to pH7.0.
  (iv) Biological waste must be treated with formalyne and solidified with gypsum in plastic waste packages.

A.2.1.4. Current situation at the Novi Han repository

The Novi Han repository has been in operation for more than 30 years without an accident or release of radioactivity to the environment, but also without investment for upgrading. As a consequence the Committee on the

Use of Atomic Energy for Peaceful Purposes temporarily suspended repository operation in 1994. In 1995 INRNE initiated a programme to upgrade the repository and developed an implementation plan. Activities are supported by the Bulgarian Academy of Sciences, the Committee on the Use of Atomic Energy for Peaceful Purposes, the International Atomic Energy Agency with Model Technical Project BUL/4/005 on Increasing Safety of Novi Han Repository from 1997–2000, and the Bulgarian Government with financing from the State budget in 1998 and subsequently from the State fund for Safety and Storage of Radioactive Waste.

A.2.2. Activities for increasing the safety of the Novi Han repository

IAEA Model Technical Project BUL/4/005 on Increasing Safety of the Novi Han Repository contributed the following to the upgrading process:

(a) Expert missions to evaluate possible options for improvement, safety assessment, evaluation of monitoring systems, evaluation of radiological monitoring and control practices, management of high level spent sealed sources, selection of treatment and conditioning processes, above ground storage facility design and quality assurance;
(b) Training of personnel and a scientific visit to a RADON facility in the Russian Federation;
(c) Procurement of Super Low Lever Liquid Scintillation Analyzer software for safety assessment, technical documentation on low and intermediate level waste processing and storage, processing and storage of spent sources, and data management.

The main tasks and achievements of the programme are discussed below.

A.2.2.1. Identification of radionuclide inventory

The radionuclide inventory of the Novi Han repository, as well as the inventory of each separate disposal vault, were identified based on existing documentation for the period of operation of the repository. Spent sealed sources, listed in Table 2, represent the majority of the waste disposed of [76].

The activity disposed of in the vaults for solid waste, biological waste and the trench is low compared with the disposal vault for sealed sources, because the waste is low level and generated mainly from scientific research. Contaminated materials from some incidents, including soil with a low contamination level, are disposed of in the trench. The activities are 8.05×10^{12} Bq, 1.91×10^{11} Bq and 1.28×10^{12} Bq, respectively. The activities of the vaults for solid

TABLE 2. RADIONUCLIDE INVENTORY OF THE DISPOSAL VAULT FOR SEALED SOURCES

Radionuclide	Activity (Bq)	Radionuclide	Activity (Bq)
^{192}Ir	1.37×10^5	^{144}Ce	2.12×10^2
^{60}Co	1.46×10^{13}	^{106}Ru	7.84×10^5
^{137}Cs	1.12×10^{13}	^{55}Fe	1.14×10^9
^{90}Sr	1.13×10^{11}	^{85}Kr	7.67×10^{10}
^{226}Ra	5.96×10^{11}	^{75}Se	2.11×10^2
^{170}Tm	5.48×10^5	^{147}Pm	1.01×10^8
^{204}Tl	2.39×10^9	^{239}Pu	1.82×10^{11}
^{65}Zn	2.79×10^4	^{241}Am	2.41×10^{10}
^{109}Cd	1.80×10^7	^{3}H	1.70×10^{10}

waste, biological waste, and the trench are shown in Figs 8–10. The inventory of the trench is made up mainly of radionuclides of Cs, Co and Sr. In addition, a total of 3 mg of ^{239}Pu from contaminated equipment, protective material, etc., is disposed of in the trench.

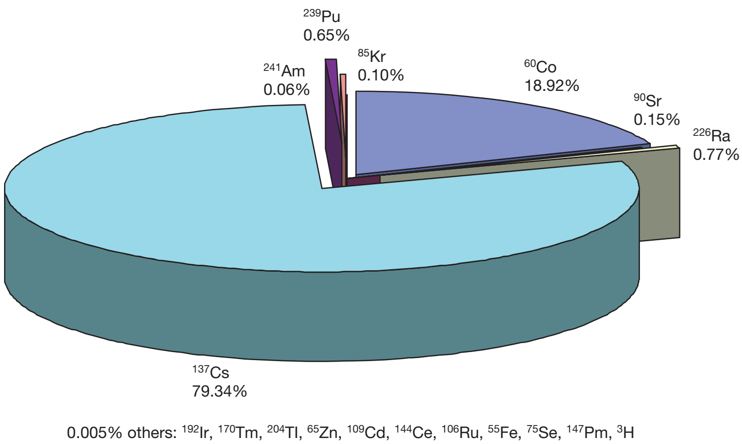

FIG. 8. Radionuclide inventory of the disposal vault for spent sealed sources.

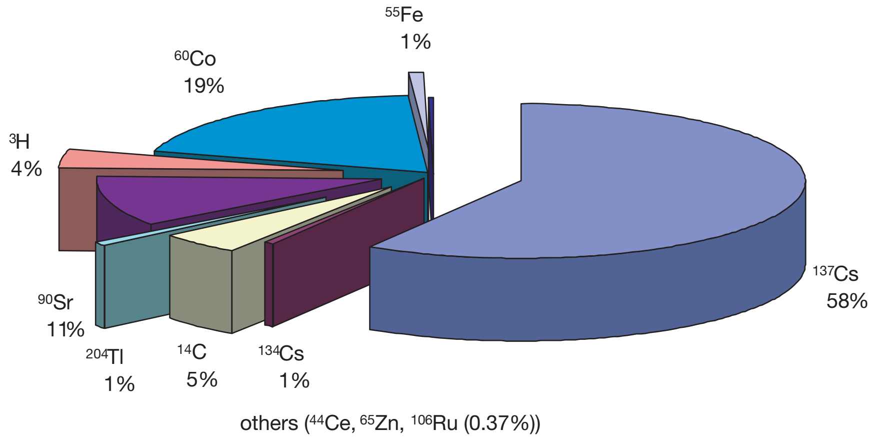

FIG. 9. Radionuclide inventory of the disposal vault for solid waste.

A.2.2.2. Characterization of the disposal vaults

According to the original design, the lifetime of the repository was to be ten years. Only the disposal vault for spent sealed sources is now full. The underground parts of the disposal facilities are in good condition and no water is to be found in the disposal structures. Leakage of radionuclides outside the vaults has not been observed.

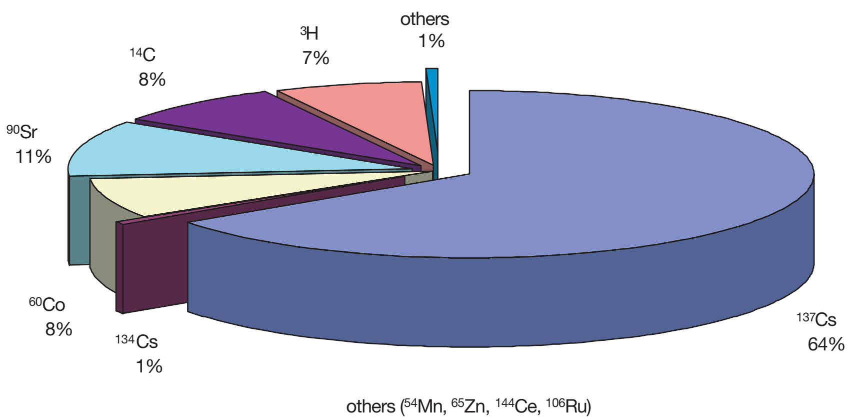

FIG. 10. Radionuclide inventory of the disposal vault for biological waste.

A.2.2.3. Characterization of the site

The Geological Institute of the Bulgarian Academy of Sciences has conducted a geological survey of the Novi Han repository site [77]. The study included characterization of the meteorology; the lithostratigraphy of Losen mountain (based on existing records of deep drillings); lithology of the site (based on 3 shallow investigation boreholes from 15–25 m deep, 2 of them in the second area of the Novi Han repository and the third outside the fence, as well as on existing data from 6 boreholes, 250–790 m deep, drilled in an area 80–1500 m from the site); tectonics and hydrogeological settings. A phyllite schist formation with a variable thickness from 300–500 to more than 800 m underlies the repository site.

According to the geological investigations, there is no evidence of hazardous atmospheric phenomena. The repository site is not endangered by flooding and gully erosion. There are no landslides or rock falls in the repository region which could be an eventual hazard and there are no conditions for the evolution of such processes. The geological strata have safe bearing capacity, providing a safe suitable foundation base for the disposal vaults. The repository area is not endangered by subsidence or significant settlement of the soil base. Additional investigations are necessary to clarify the complex tectonic structure.

The rocks in the region represent a low water bearing and low permeable formation. Their permeability is higher in the tectonic, strongly fissured and faulted zones and in the upper weathered layer. The groundwater is recharged by precipitation only. A regular aquifer has not been formed. An unstable water table of shallow groundwater at a varying depth from the surface (from 6–7 m to 15–17 m) is found in the shallow boreholes in the repository area. The hydrogeological conditions are complex. Additional site investigations are planned to provide data for the safety assessment and for the construction of new facilities on the site.

A.2.2.4. Safety assessment

The radiological consequences were determined for relevant scenarios, selected from a comprehensive list of features, events and processes developed for the Novi Han repository [78, 79]. The main scenarios are leaching, Pu capsules and intrusion (construction and residence). Results were obtained for the entire repository and for individual disposal vaults. The peak doses are lower than 1 mSv/a. Based on the results for the intrusion scenario, an institutional control period of 300 years is proposed.

A.2.2.5. Upgrading of the monitoring and control system

The activities include 2 interconnected tasks, environmental monitoring and radiation protection. The environmental monitoring programme covers the repository site, the restricted area and the supervised area. Based on monitoring during 30 years of operation and site characterization, the monitoring programme was expanded [80]. Monitoring of water sources (monitoring boreholes, permanent and seasonal springs, surface running waters, drinking, household and irrigation water), soil, sediments, vegetation, food and air is undertaken. Analytical methods include gross β, gross α, γ spectrometry, liquid scintillation measurement of ^{14}C and ^{3}H, ^{90}Sr and ^{239}Pu, direct measurement of the dose rate, permanent measurement of γ background, automatic measurement of γ background in situ and γ spectrometry. The radiation protection programme covers the dosimetric checking of personnel and control over the technological processes in order to ensure the safety of operators.

A.2.2.6. Option study and conceptual design of a new waste processing and storage facility

The option study is aimed at selecting treatment and conditioning processes based on existing information on the different waste streams, regulatory requirements and best practices [81]. The conceptual design for a new waste processing and storage facility is taken into account in the feasibility study for reconstruction and modernization of the repository [82].

A.2.2.7. Direct measures for improvement of safety

The improvement measures were financed by the 1998 State budget and subsequently by the State Fund for Safety and Storage of Radioactive Waste. They cover the recommendations of the regulators as well as technological needs, and a new organizational structure that includes quality assurance. Some of the important activities are the following:

(a) Repair and improvement of the existing disposal vaults (Fig. 11). This includes repair of the concrete in the above-ground parts of the disposal facilities, new hydro-insulation and new lids, as shown in Fig. 12 for the disposal vault for biological waste. A new heavy protective cover was installed over the disposal vault for spent sealed sources (Fig. 13).

FIG. 11. Disposal vault for solid waste before improvement.

FIG. 12. Disposal vault for solid waste after improvement.

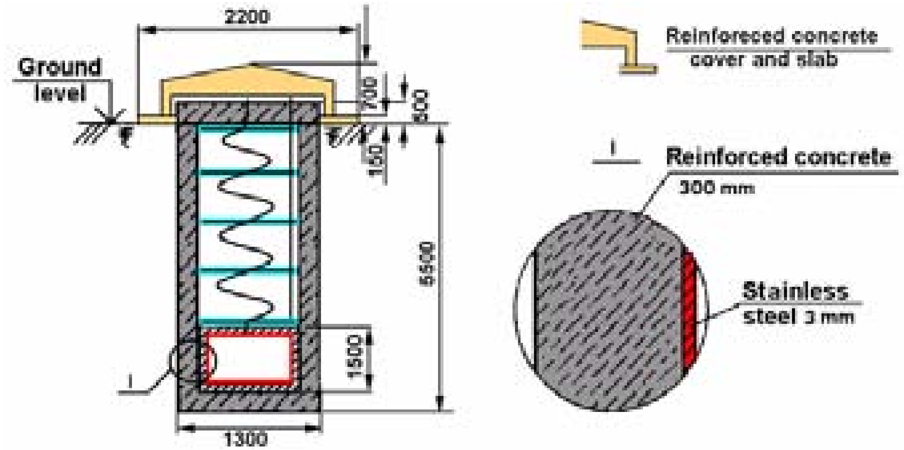

FIG. 13. Improvement of the disposal facility for spent sealed sources.

(b) Repair and improvement of the building; complete refurbishment of the electricity supply and reserve electricity supply, water supply, specialized sump water collection and ventilation systems, and decontamination facility.
(c) Improvement of the fire fighting system and the physical protection.
(d) Improvement of security and personnel access control.
(e) Improvement of the infrastructure.
(f) Reliable communications and new transport vehicles for personnel and for the transport of radioactive waste.

A.2.3. Conclusions and further developments

Work accomplished to date has improved safety and allowed transport of radioactive waste stored in Sofia to Novi Han for temporary above ground storage. INRNE will apply for relicensing of the facility. Further development will cover construction of a facility for long term storage/disposal of spent sealed sources. An above ground storage facility might be constructed. A possible long term solution is the utilization of a deep shaft at Gabra for construction of a monitored disposal facility.

A.3. CANADA: CORRECTIVE ACTIONS IMPLEMENTED IN THE WASTE MANAGEMENT AREAS OF CHALK RIVER LABORATORIES

A.3.1. Background

This section discusses some of the corrective actions that have been carried out in response to initiating events associated with the waste management areas at the Chalk River Laboratories.

Chalk River Laboratories was established in 1944 by Atomic Energy of Canada Ltd (AECL), a corporation that is owned by the Government of Canada. Operations at the Chalk River Laboratories site began in the autumn of 1944. Over the years, Chalk River Laboratories has served the needs of basic research, radioisotope production, and research and development in support of AECL's CANDU heavy water reactor. The facilities provide storage for radioactive wastes arising from the operation of research and development facilities at Chalk River Laboratories, isotope processing operations, prototype CANDU reactors, hospitals, universities and industries across Canada.

Chalk River Laboratories are located in the Province of Ontario on the southern shore of the Ottawa River, 160 km northwest of Ottawa. The Chalk River Laboratories site is typical of its immediate surroundings — a mixture of exposed bedrock, glacial till, fluvial gravel and sand, small lakes and marshes. Elevations vary from the level of the Ottawa River to 120 m above the river level. The Ottawa River is the dominant drainage feature in the area.

Canada does not currently have any near surface disposal facilities for radioactive waste. However, some storage practices were undertaken decades ago at AECL sites with no specific plans to retrieve the waste at a future time, and thus the associated facilities meet the definition of a repository as given in Section 1.3. As a result, some of the experience gained in addressing issues surrounding the waste management areas is relevant to the topic of the application of corrective actions to near surface disposal facilities for radioactive waste.

A.3.2. Waste management areas

The information below provides a brief description of the larger waste management areas with radiological inventories on the Chalk River Laboratories site that have been subjected to corrective actions.

A.3.2.1. *Waste management area A*

The first emplacement of radioactive waste at Chalk River Laboratories took place in 1946 into what is now referred to as waste management area A. These emplacements took the form of direct disposal of solids and liquids to excavated trenches in the sand overburden.

A.3.2.2. *Liquid dispersal area*

This area contains seepage pits that went into operation in 1953 to receive active liquids from various laboratories and facilities associated with reactor operations. The seepage pits (reactor pits 1 and 2, and the chemical pit) are located on a small dune, in an area bounded on the east and south by wetlands and by waste management area A to the west.

A.3.2.3. *Waste management area B*

This was established in 1953 to succeed waste management area A as the site for solid waste management. It contains a wide variety of waste burial structures, such as unlined sand trenches, concrete monoliths containing solidified liquid wastes, asphalt lined trenches, concrete bunkers and tile holes for high level wastes. Tile holes are below grade concrete pipes set vertically on a poured concrete base; some of the tile holes have a steel lining.

A.3.2.4. *Waste tank farm*

This was built to store high and intermediate level liquid wastes in tanks that are housed in stainless steel lined concrete bunkers. Water level sensors in the concrete bunkers, which are tested periodically, are wired to alarms at response centres in the inner area.

A.3.2.5. *Waste management area C*

This is a sand trench facility that went into service in 1963 to receive LLWs with hazardous lifetimes of less than 150 years and wastes that cannot be confirmed to be uncontaminated. Some of the older trenches at waste management area C have been covered with an impermeable membrane of high density polyethylene (HDPE).

A.3.2.6. Waste management area F

This was established in 1976 to store contaminated soils and slags containing low levels of ^{226}Ra, uranium and arsenic. Emplacement was completed in 1979 and the site is now considered closed.

A.3.3. Initiating events

In the past at Chalk River Laboratories, radioactive waste was placed in non-engineered waste management facilities, as well as in asphalt trenches and concrete bunkers. Unique 'one of a kind' wastes were encased in such materials as concrete and bitumen. In the intervening years, environmental monitoring and inspections have revealed the presence of radioactive contamination in both groundwater and surface water, and the fact that the structures used to contain radioactive wastes were either failing or had not been built as indicated in records.

A.3.4. Corrective actions

A.3.4.1. Installation of surface caps

Waste management area C: In those cases where wastes were placed in unlined sand trenches, the groundwater contamination is due in large part to advection. To mitigate this situation, some of the older trenches have been covered with an impermeable membrane of HDPE. Subsequent monitoring of both groundwater and surface water has shown an appreciable reduction in the levels of such radionuclides as tritium as a consequence of this corrective action.

A.3.4.2. Monitoring and assessment

Waste management area F: In the establishment of waste management area F a layer of clay was placed as a cap to reduce the flow of water through the waste body. However, subsequent drying of the clay layer has led to cracking, and the efficacy of this cap has been markedly reduced. The corrective actions to date have largely served to enhance our knowledge of the behaviour of the contaminants in order to determine if and when additional corrective actions will be required. The results of monitoring and assessment have shown that the contaminants are migrating very slowly and that it will be decades before local surface water is affected.

A.3.4.3. Installation of cut-off walls and sorbing barriers

As noted above, the use of non-engineered waste management facilities at Chalk River Laboratories has led to several plumes of contamination. In one case, wall and curtain technology has been employed to direct the plume of contaminated groundwater through a bed of clinoptilolite to adsorb such species as ^{90}Sr. The concentration of ^{90}Sr prior to passing through the adsorbing medium is approximately 85 Bq/L, and after passing through the clinoptilolite bed the ^{90}Sr levels are approximately 0.6 Bq/L, which is well below the Canadian drinking water standard of 5 Bq/L. Given the sorptive capacity of the clinoptilolite bed, together with the half-life of ^{90}Sr, there may be no requirement to retrieve the sorbing medium. Similar technology is being considered for application to other plumes.

A.3.4.4. Water collection and extraction

In the case of other plumes it has been possible to install a series of interception wells to pump the contaminated water to the surface for subsequent treatment. In these cases the levels of ^{90}Sr in the untreated water can be as high as several thousands of Bq/L, while after treatment they are at a level of approximately 10–20 Bq/L.

A.3.4.5. Retrieval

To an increasing extent, problematic wastes are being retrieved for processing and/or repackaging. The retrieval and processing technology varies considerably based on the nature and condition of the waste.

In the case of stored liquid wastes, efforts are now under way to retrieve liquid wastes from more than 20 tanks, some of which are more than 40 years old. The intention is to retrieve the liquid wastes, blend and consolidate them in new tanks and then ultimately solidify the liquids using, for example, vitrification technology.

A project to retrieve old research reactor fuels that have undergone significant corrosion due to exposure to water has also recently been initiated. The current plan is to retrieve those fuels which are predominantly made of uranium metal, dry and repackage the fuel, and then store it in a dry state until a disposal facility for used nuclear fuel is available.

Other retrieval operations have been and are focusing on a diverse range of wastes including:

(a) Hazardous chemicals encased in large concrete monoliths;

(b) Drums of contaminated solvents;
(c) Abandoned research equipment;
(d) Reactor parts that were buried following a reactor accident in 1952;
(e) Contents of some early unlined trenches;
(f) Localized areas of contaminated soils.

A.4. CZECH REPUBLIC: UPGRADING OF EXISTING REPOSITORIES

A.4.1. Introduction

In the Czech Republic, radioactive wastes are disposed of in the repositories at Dukovany, Richard near Litoměřice and Bratrství in Jáchymov. The repositories had previously been operated by private operating organizations. Dukovany was commissioned in 1985 and operated by ČEZ plc (the nuclear power plant operating organization); Richard was commissioned in 1964 and its last operating organization was ARAO; Bratrství was commissioned in 1975 and its last operating organization was also ARAO. Their management has been transferred to a State organization, the Radioactive Waste Repository Authority (RAWRA), which is now responsible for the safe operation of all repositories. WAC have been changed several times during repository operation as the safety requirements have been improved.

A.4.2. Dukovany

A.4.2.1. Description of the facility

The Dukovany repository was designed for the management of low and medium level radioactive waste generated by nuclear power plants. It is the largest and most modern of all the repositories in the Czech Republic, situated within the area of the Dukovany nuclear power plant in the community of Rouchovany.

Four rows of 28 concrete vaults (each 17.3 × 5.3 × 5.4 m) have been built to dispose of all operational and decommissioning wastes from the Dukovany and Temelín nuclear power plants. The vault walls and floor are made of monolithic 10 cm reinforced concrete integrated with another waterproof layer of 10 cm asphalt-polypropylene concrete. Eight vaults have been filled with bituminized, cemented or supercompacted wastes. Wastes are accepted in 200 L drums. These drums are filled in 6 layers, so approximately 1600 drums can be disposed of in 1 vault. Empty space in the vault is filled with concrete and the vault is protected against rainwater.

The total volume of the disposal rooms is 55 000 m^3 (about 180 000 drums). This is sufficient for disposal of all the low and medium level waste from both power plants, even in the case of a prolongation of their planned lifetimes by 10 years (to 40 years).

A.4.2.2. Planned activity

Because of its modern design, no upgrade is planned. Some improvements have been made in the arrangement of the drums, increasing the capacity of the vault from 1200 to 1600 drums.

Current plans are divided into 2 main fields:

(1) Protection of the vault construction against wear (concrete carbonization, etc.);
(2) Improvement of the waste packages.

A.4.2.3. Protection of the vault construction against negative effects of weather

Maintenance of the repository construction must continue throughout its operating period, which will extend decades into the future. The vault construction is exposed to the negative influence of rainwater leakage through improper seals in the ceiling panels. Carbonization of the concrete can also cause long term damage to the construction.

To deal with these problems the possibility of protecting 2 rows of the vault by construction of a temporary roof is being studied. To protect the rest of the vaults the use of modern conservation materials for water insulation is being studied.

A.4.2.4. Waste packages

The supercompaction campaign did not achieve sufficient volume reduction. The waste producer is looking for a new overpack to enable more efficient use of the disposal volume. A possible option may be the adaptation of the entire vault to accept raw supercompacted drums.

A.4.3. Richard

A.4.3.1. Description of the facility

The Richard repository is situated in a former limestone mine. The disposal chambers and corridors are situated in a thin (3–5 m) layer of limestone. This layer is insulated from the top and bottom by a wide (30–60 m) stratum of water impermeable claystone. During the conversion of the mine for disposal purposes it was necessary to reinforce some parts with concrete and to construct a drainage system for any water which might eventually reach the mine.

The total volume of the repository exceeds 16 000 m^3, and its disposal capacity is approximately half that volume. The temperature in the repository is a practically constant 10°C. Since 1964 radioactive waste from non-power applications (institutional waste) has been disposed of there.

The WAC have been changed several times since 1964. Very simple waste packages were used in the early years of the repository's operation. The first waste was disposed of in 60 L zinc coated drums, which were placed in 100 L drums, which in turn were inserted into 200 L drums. The space between the drums was filled with concrete, forming a 5 cm thick concrete barrier. The wall of the outer drum was coated on both sides with zinc and the outside was covered with a thin layer of bitumen paint. The drums were placed in prepared chambers which had been formed during the mining phase.

A.4.3.2. Maintenance and refurbishment

The repository has been operating for almost 40 years without accident, but requires systematic maintenance in the following main areas:

(a) Mine construction (conservation of underlying constructions, mine entry, ventilation, lights, electricity supply);
(b) Surface area reconstruction and upgrading (new administration building, new fences, reconstruction of hot cells and operations building, road surface, etc.);
(c) Radiation protection (measurement devices, monitoring programme, control area regime (including hygienic room));
(d) Records (records of disposed waste).

A.4.3.3. Records

The changes in the requirements for records of disposed wastes are similar to changes in WAC. Previously only dose rate and main radionuclide listings were sufficient, but the level of detail has been increased. The current system collects detailed radionuclide composition, waste tracking data, and waste conditioning, chemical and other important data. Since 2000 the exact unit location has been recorded for each disposed waste unit. The new system is Internet based but hardcopies of the records are kept as well. Research into historical records is being carried out to combine historical and contemporary data.

A.4.3.4. Upgrading filled chambers

The closure of one or more disposal chambers filled, in particular, with historical waste would enable RAWRA to:

(a) Improve the safety of the disposal of LLW packages in the Richard repository;
(b) Improve radiation protection of personnel;
(c) Demonstrate the feasibility of a safe closure of the repository;
(d) Establish a programme to evaluate the long term behaviour of the backfill material and sealed waste packages for verification and validation of the data necessary for the safety assessment of the repository.

Realization of the project is expected to reduce the operational expenses connected with management of the historical waste.

After evaluation of the results of the project, RAWRA plans to adopt the concept of step by step closure of the disposal chambers at the Richard repository after they have been filled with the waste packages. This concept has been adopted and approved by the State Office of Nuclear Safety at the Dukovany repository, where the waste in the disposal vault is sealed with concrete grout immediately after the vault has been filled with waste packages.

The experience from sealing one or more disposal chambers at the Richard repository will allow RAWRA to realize a similar project at the Bratrství repository, where the situation is similar and corrective actions also need to be taken.

A.4.3.5. Consideration of new chambers

Waste producers are not satisfied with the current waste package model (a 100 L drum inside a 200 L drum) because of the increasing need for fragmentation of some kinds of solid waste. The repository contains 2 empty unprepared chambers. These chambers can be adapted for disposal use, including creation of a special space for large dimensioned waste.

A.4.4. Bratrství

A.4.4.1. Description of the facility

The Bratrství repository is designed for the disposal of waste containing natural radionuclides. It was constructed by adaptation of a mining shaft, during which 5 disposal chambers were created. The facility was put into operation in 1974.

A.4.4.2. Consideration of safety improvements

The situation at the Bratrství repository is similar to that at Richard. Detailed research on historical waste records has identified great uncertainties in nuclide inventory. Safety analyses have shown that the repository is and will continue to be safe but that one chamber of the repository should be checked with the aim of decreasing uncertainty. This may include picking out disposed historical waste, analysing it and disposing of it again.

A.5. EUROPEAN UNION: EUROPEAN COMMISSION STUDIES AND PROJECTS IN THE FIELD OF LOW LEVEL RADIOACTIVE WASTE IN COUNTRIES OF CENTRAL AND EASTERN EUROPE

A.5.1. Background

European Commission studies and projects in the nuclear sector are financed through a number of programmes and managed by various technical services in the Commission. These programmes can be broadly categorized as follows:

(a) The PHARE Nuclear Safety Programme, managed by the Task Force for Nuclear Issues (TFNI) of DG-Enlargement, which provided funding for projects in the then EU candidate countries in Central and Eastern Europe. Following the accession of the majority of these countries in 2004 there are no further PHARE annual programmes and the TFNI, along with DG-Enlargement, will be disbanded.
(b) The Euratom Research Framework Programme, managed by DG-Research (Unit JO4).
(c) The so-called 'B7 budget line Co-operation Programme' that until the end of 2000 was managed by the Nuclear Safety Policy Unit (then in DG-Environment). Following the reorganization of the Commission's

services, this budget line has been discontinued. The management of the ongoing projects was transferred to TFNI (with technical support from DG-Energy and Transport).

(d) The Tacis Programme, managed by the Europe Aid Co-operation Office and DG-External Relations, which provides funding for projects in the Russian Federation and the countries of the former Soviet Union.

Although these programmes are not concerned exclusively with radioactive waste, several individual projects involve waste issues, specifically low level waste.

A.5.2. PHARE programme

Over the course of the PHARE Nuclear Safety Programme there were numerous projects in the field of LLW within the EU candidate countries. A useful, though not definitive, guide to projects before 1999 can be found in Commission Report EUR 19154. This report can be downloaded from DG-Energy and Transport's 'Nuclear Issues' web site:
http://europa.eu.int/comm/energy/nuclear/pdf/radwaste_in_ceec.pdf.

As a result of the reorganization of the Commission's services in 2000, no PHARE multicountry programme was launched that year. TFNI restarted the nuclear safety programming activities in 2001. However, one project of interest was implemented during the intervening period: Project -006-RO/PHARE-SCR/A6-01, Preparatory Measures for the Long-term Safety Assessment of the Low Level Radioactive Waste Repository Baita Bihor, Romania (completed in September 2001).

In the past, availability of PHARE reports depended on the express agreement of the beneficiary. Although this condition is now more relaxed, reports are not routinely made available on-line in the same way as the 'B7 budget line' reports.

A.5.2.1. 2001 programme

Starting with the 2001 programme, a decentralized approach to PHARE tendering and implementation has been applied. This brings the nuclear safety programme in line with other PHARE assistance programmes, but has resulted in additional delays in the tendering and contracting procedures. The financial disbursements for these projects were only authorized until the end of 2004. Projects that have relevance in the LLW field are:

(a) Solution for closure of a chamber in the Richard repository, Czech Republic (CZ 01.14.03);
(b) Reconstruction of the hot cell at the Richard repository, Czech Republic (CZ 01.14.04);
(c) Evaluation of waste retrieval and disposal options at the Püspökszilágy radioactive waste treatment and disposal facility, Hungary (HU 01.11.02);
(d) Design of an additional waste disposal vault and integral storage facility for long lived waste at Baldone, Latvia (LE 01.09.01);
(e) Improvement of storage conditions and closure of the national radioactive waste repository at Rózan, Poland (PL 01.13.01).

A.5.2.2. 2002 programme

The following projects from the 2002 programme are relevant in the LLW field. In view of the delays imposed by the tendering and contracting procedures, these projects were not expected to get under way much before the end of 2003.

(a) Supply of equipment for characterization of institutional radioactive waste and development of the technical design for a waste processing and storage facility, Bulgaria (632.01.01);
(b) Realization of closure of a chamber in the Richard repository as input for establishing a safety case, Czech Republic (632.02.04);
(c) Providing free storage/disposal space in the Püspökszilágy repository, Hungary (632.04.03);
(d) Safety Assessment and upgrading of the Maisiagala repository, Lithuania (632.06.01);
(e) Preliminary safety analysis report for the Baita Bihor low level radioactive waste repository, Romania (632.08.01) (follow-up to Project -006-RO/PHARE-SCR/A6-01);
(f) Characterization of institutional low and intermediate level radioactive waste currently stored in a central facility, Slovenia (632.10.03).

A.5.2.3. 2003 programme

The 2003 programme was the last annual programme, although in the case of Bulgaria and Romania, further funding was to have been made available in the years 2004–2006.

A.5.3. Euratom framework programme

Most of the research projects in the field of radioactive waste launched under the fifth Euratom Framework Programme were concerned with high level waste issues. Nonetheless, the work that has been undertaken within this programme could be of general interest to anyone in the radwaste sector. A complete summary of the various research conducted within the programme can be found in the following publication: Nuclear Fission and Radiation Protection Projects Selected for Funding 1999–2001, which is available at http://www.cordis.lu/fp5-euratom/src/lib_docs.htm.

The first call for proposals for the sixth Framework Programme has now taken place. However, the emphasis of this programme is on issues essentially related to geological disposal.

A.5.4. 'B7-line' Co-operation Programme

Most of the information related to this programme can be obtained by consulting the reports page of DG-Energy and Transport's 'Nuclear Issues' web site: http://europa.eu.int/comm/energy/nuclear/index_en.html/.

From this page, final reports can be accessed and downloaded. Most of the relevant studies are to be found under the heading 'Radioactive Waste'. The principal reports in the LLW field are:

- EUR 19260: Use of the Existing Buildings of the Püspökszilágy Radioactive Waste Treatment and Disposal Facility for Temporary Storage of Radioactive Wastes (Hungary);
- EUR 19842: Management of Spent Sealed Radioactive Sources in Central and Eastern Europe (Czech Republic, Estonia, Hungary, Poland and Slovenia);
- EUE 20052: Feasibility Study of a Waste Assay System and the Possibility of Volume Reduction at the Püspökszilágy RWTDF (Hungary);
- EUR 20053: Assessment of the Proposed Design of a New Spent Sealed Radioactive Sources Storage Facility at Novi Han (Bulgaria);
- EUR 20054: Long-term Safety Analysis of Baldone Radioactive Waste Repository and Updating of Waste Acceptance Criteria (Latvia);
- EUR 20654: Management of Spent Sealed Radioactive Sources in Bulgaria, Latvia, Lithuania, Romania and Slovakia;
- EUR 20655: Detailed description of a New Management System for Solid, Short-lived Low and Intermediate Level Radioactive Waste at Ignalina NPP (Lithuania).

- EUR 20656: Assessment and Upgrading of the Novi Han Monitoring System (Bulgaria).

Finally, the fifth Situation Report on radioactive waste management in the enlarged European Union can also now be downloaded from the above web page — the report reference is EUR 20653. For the first time, the Situation Report also covers the then candidate countries in Central and Eastern Europe.

A.5.5. Tacis programme

Limited details of the Tacis programme can be accessed via the following web page on the Commission's Europa web site:
http://europa.eu.int/comm/external_relations/nuclear_safety/intro.

However, there are probably very few projects that are relevant to the LLW sector. The only one perhaps of any significance is described briefly on the following web page:
http://europa.eu.int/comm/external_relations/north_dim/examples.htm.

A.6. FRANCE: IMPROVEMENTT MEASURES IMPLEMENTED AT THE CENTRE DE LA MANCHE

A.6.1. Background

The French repository at the Centre de la Manche was the first near surface disposal facility in France. It underwent numerous operational improvements from 1969 to 1994. For example, WAC were introduced in 1979 after a number of operating years with a very simple set of proper disposal requirements and conditions. The long term safety objectives have been continuously improved and were formalized in 1985 in the 'Basic Safety Rules'.

A.6.2. Disposal units

Initially and according to its activity, the waste, packaged in various forms (drums, etc.), was either buried in earth trenches (Fig. 14) or placed in concrete lined trenches. Rainwater was collected in a sump located at the downgradient end of each trench, and then conveyed to a retention tank. Depending on its activity it was either sent to the adjacent Sainte-Hélène River or discharged into the sea via the COGEMA installations. After one year of operation, the earth trenches were deemed unsafe and abandoned. Use of the concrete lined

FIG. 14. Creation of the Centre de la Manche, 1969 (courtesy of ANDRA).

trenches continued for some years. These trenches were subdivided into cells by concrete walls, and waste packages placed within the cells were stabilized with sand (Fig. 14).

Following the discovery of water in one of these trenches in 1972, the sand was replaced with cement to guarantee better sealing of the structure. The construction of the structures, which represented the second barrier, was also improved. The earth trenches, which had been built in 1969, were dismantled, with the exception of one still in existence today. Waste packages were retrieved, reconditioned and disposed of in other disposal units.

Starting in 1975, 'platforms', consisting of levelled soil covered with a layer of local materials and a bitumen emulsion, were put into operation, replacing the earth trenches. These platforms were equipped with water drains. The waste packages were stacked on the platforms to form mounds. To guarantee overall mechanical strength and to facilitate disposal operations, the framework of the structures was built of concrete blocks containing the waste arranged stepwise on the edge of the structure. Metallic drums were placed in these compartments, covered with a plastic sheet and a layer of soil. The voids between the packages were filled with gravel. The plastic sheet was later eliminated in favour of a layer of gravel and soil.

After 1981, platforms made of reinforced concrete became the standard disposal structures for the repository. These structures were built to withstand earthquakes and to collect rainwater efficiently. The concrete trenches were also abandoned and replaced with monoliths comprised of prefabricated reinforced concrete bins or concrete blocks. Monoliths were used to enhance protection of the waste package, depending on the type and activity of the radionuclides contained in it. They were built on the platform structures. A lower level consisted of a raft foundation on which the monoliths containing the waste packages were positioned. A second raft was placed on the monoliths supporting the waste packages arranged in a mound. Following a tritium leak, the water collection system was modified in 1980 to separately collect rainwater and infiltration water percolating around the waste packages. To prevent line breakage and to avoid the use of pumps, which were liable to failure, a separate underground gravity water recovery system was built in 1982.

A.6.3. Tritium leak incident

In 1976, the Sainte-Hélène River adjacent to the centre was contaminated with tritium at values that were high yet below the regulatory limit. The structures responsible for the tritium leak were old concreted trenches in which the waste was immobilized by sand. These concreted trenches were opened, modified and rebuilt. The faulty packages were retrieved and reconditioned in special drums. Acceptance thresholds for tritiated waste were then reduced and inspections conducted at the generators. This incident nevertheless left a trace that persists today in the groundwater table and in the streams around the Centre.

A.6.4. Long term safety and waste inventory

During the 25 years of operation of the Centre de la Manche, from 1969 to 1994, long term safety principles have been gradually developed and consolidated.

These principles were reflected in the Fundamental Safety Rules that serve as reference for the WAC. Safety relies not only on the containment barriers but also on the control over the activity of the waste packages received and over the radiological inventory of the disposal facility. The information is provided through delivery documentation that has been registered in a computerized system since 1985. For the previous period, the inventory of disposed packages and the assessment of their activity were derived from the entire set of operational data or, in the absence of sufficient information, from

FIG. 15. Tumulus of concrete packages and metal boxes at the Centre de la Manche (courtesy of ANDRA).

simulations based on delivery analogies as well as on the knowledge of waste contamination and activation processes (Fig. 15).

This safety objective necessitates the required package information to be submitted systematically upon delivery. These statements have benefited from the technological advances and were input into a computerized system in 1985, allowing for the waste packages to be followed from their production site up to their final placement in the disposal modules. This mechanism has been applied at the Centre de l'Aube since that facility was commissioned in 1992.

The Centre de la Manche's operating organization has created significant documentation by keeping the whole set of waste delivery slips, and has archived delivery registers as well as disposal registers and maps. All of these data have been used to develop the facility's inventory, showing both the exact location of the packages in the different disposal modules and their activity (Fig. 15). Data collected between 1969 and 1985 have been consolidated with those entered electronically after 1985.

However, the activity of some packages had to be determined by simulation, a method that could only be used with a good knowledge of the waste generating facilities. Coupled with the computerized tool, competence of the disposal facility's operating organization in activity assessment techniques and knowledge of the contamination or activation processes are necessary for good inventory control. A commission appointed in 1996 by the French Government to assess the situation at the Centre de la Manche audited this approach and encouraged the Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA) in its efforts, ruling that the reconstituted inventory should meet the needs of the safety analysis.

By the end of its operation in 1994 the repository, with an area of 600 × 300 m, had accepted 527 214 m^3 of waste. When disposal activities were terminated, a final cover was constructed to divert rainwater away from the repository. The cover consists of a bitumen membrane and several layers of earth and sand (Fig. 16).

A.6.5. Corrective actions taken before the institutional control period

ANDRA filed an application in September 1994 for authorization to begin the institutional control period. The safety authority felt that the application for authorization to begin the institutional control period should be submitted to a public inquiry. The inquiry commission gathered the opinions of the population and, in February 1996, issued a favourable opinion on the

FIG. 16. Aerial view of the Centre de la Manche. A multilayer, non-permeable cover protects the waste disposal area (courtesy of ANDRA).

transition of the Centre de la Manche to the institutional control period. Subsequently, after complaints were filed against ANDRA by ecological associations for water pollution and violations of the regulations, the Government decided to set up a second evaluation commission in February 1996. This commission was asked to evaluate the situation concerning the Centre de la Manche disposal facility and to express its opinion on the impact of the Centre on the environment.

The Government made the commission's conclusions public in July 1996. The commission concluded that if the intended measures were taken, the Centre would present no significant health risk for the local population. The commission also found the absence of a local and department-wide 'cancer register' to be anomalous from the health standpoint. Its conclusions were primarily as follows:

(a) The cap adds a vital safety factor to the installation.
(b) The waste inventory compiled by ANDRA is satisfactory.
(c) Heterogeneities associated with long lived alpha emitters exist in the facility and predate Basic Safety Rule RFS I.2. The work necessary to eliminate these hot spots results in greater risks to the public and the environment than maintaining the status quo.
(d) The site cannot be opened to unrestricted use for 300 years. The heterogeneity of the structures would lead to subsidence. Areas containing long lived radioelements would lead to exposures in cases of intrusion and destruction of packages. Radon would be released due to the presence of radium bearing waste, and this radionuclide could accumulate in the subsoil of a dwelling built on the site. A chemical risk exists due to heavy metals present in the facility, particularly lead.
(e) It is unwise to make any assumptions about the evolution of society and its structure after a few centuries. In recognition of this, transfer of the facility to future generations who might or might not be willing to assume responsibility for it is discouraged. As a consequence, the commission proposed the formation of a new organization for the institutional control period, with responsibilities that include developing a design for a final cap to guarantee passive safety of the facility. The Centre will also have to be placed in a state such that it would not present any significant risk to the environment and the population if it were to be suddenly abandoned.
(f) The memory of the site should be preserved and necessary measures should be taken to limit the types of structures or equipment that can be installed. It was also felt that the site should remain subject to the requirements of public interest.

(g) The public had expressed the feeling that they had been disregarded in the decisions concerning the Centre. For example, the installation of the cap was carried out without consulting the public. The commission proposed the formation of an information and monitoring commission, composed of ANDRA and representatives of the local population and the ministries concerned, which would be informed and would express its opinion on the decisions to be taken.

In June 1998, ANDRA sent the supervisory ministries a new preliminary report concerning the transition to the institutional control period, taking into account the commission's conclusions. The review of the inventory was completed, including the provision of information about missing data, and the impact of the facility was reassessed from the radiological and chemical standpoint.

ANDRA then proposed phased institutional control divided into three periods. The first period would be one of highly active surveillance, to last about five years and intended in particular to assess the satisfactory operation of the present cap. The second period would be one of active surveillance, to last between 50 and 100 years. ANDRA's presence at the centre is necessary to ensure monitoring and maintenance, as well as environmental surveillance. Its task will be to confirm that any change in the cap does not compromise its ability to meet the requirements and to investigate any further arrangements needed for its strengthening. The third period will be one of passive surveillance, during which ANDRA will conduct reduced surveillance of the Centre and its environment, but during which time the complete abandonment of the Centre will not result in any unacceptable consequences to the environment. The transition to this period can only take place after verification of certain assumptions about the behaviour of the Centre over the long term, and the possible implementation of technical improvements aimed at supplementing the present containment systems with other passive systems which are simple and reliable over the very long term.

In December 1998, in response to the questions of ecological associations, the safety authority examined the preliminary report for transition to the institutional control period, which takes into account the conclusions of the report of the evaluation commission named by the Government [83]. The application for a permit to create a new basic nuclear installation was submitted to a new public inquiry in 1999. It was intended that the Centre would enter the highly active institutional control period in the year 2003.

The results of the studies to investigate the behaviour of the facility and particularly its cap will be communicated periodically to the information and monitoring commission. This commission was formed in December 1996 and includes representatives of the administration, the operating organization, elected representatives, ecological associations and the agricultural sector. Armed with full knowledge of the facts, it can make a statement about the operation of the centre and its future, and will also be able to advise the local population with complete openness.

A.7. HUNGARY: CONSIDERATIONS IN THE PLANNING OF CORRECTIVE ACTIONS AT THE PÜSPÖKSZILÁGY NEAR SURFACE REPOSITORY

A.7.1. Introduction

The Püspökszilágy repository was sited in 1971 and designed and commissioned in 1976 according to the international guidelines in effect at that time. In 1983 the site was licensed to dispose of solid low level radioactive waste from the Paks nuclear power plant until the expected opening of the power station's own disposal facility. Shipments from the nuclear power plant continued until 1996.

The site was expanded in the late 1980s. The Hungarian Geological Survey, one of the authorities participating in the licensing procedure, has not consented to issue a permanent licence for the new vaults. The new vault expansion has, however, been granted a limited operating licence.

To date, approximately 4970 m^3 of solid and solidified waste have been emplaced. 1580 m^3 came from the Paks nuclear power plant, which took up some 2500 m^3 of repository volume in the disposal site. About 3000 m^3 have been sealed and temporarily covered. More than 80% of the disposed waste is classified as LLW. The current total activity emplaced is approximately 1000 TBq. More than 80 isotopes are accepted by the facility for storage or disposal. The main radioisotopes in the waste disposed of are ^{3}H, ^{14}C, ^{60}Co, ^{90}Sr, ^{99}Tc, ^{137}Cs, ^{192}Ir, ^{226}Ra, $^{238/239}Pu$ and ^{241}Am.

The currently remaining unused capacity at the site has been reduced to 30 m^3. This is sufficient to accommodate the annual volume of waste shipped from non-power generation activities in the next years.

A number of safety assessments of the site have recently been undertaken. The main conclusions of these assessments were that with appropriate management action and reductions in performance uncertainties it is likely that a future post-closure safety case can be developed, demonstrating compliance with relevant regulatory requirements. To achieve this, certain improvements were recommended. Based on the findings of the safety assessment, consideration is given to possibly retrieving certain waste types from the vaults and putting them into interim storage pending final disposal in a geological repository. Allowing for disposal of further wastes by providing free capacity within the existing facility is also under evaluation. A key objective of the planned corrective actions is to ensure that the facility provides appropriate long term performance.

A.7.2. Description of the facility

The disposal site is located on the ridge of a hill near Püspökszilágy village. The facility is a typical shallow land, near surface engineered type repository with concrete vaults and shallow (6 m) boreholes (Fig. 17).

The disposal units are categorized into 4 classes, referred to as A, B, C and D. The A type disposal system consists of the original 48 vaults, with 70 m^3 capacity each and the expansion, 6 vaults with 140 m^3 capacity each, plus 12 vaults with a capacity of 70 m^3 (Fig. 18).

The solid waste is presently packaged into drums (formerly into plastic bags). Spent sealed sources are received from waste generators in shielding containers. Solid waste placed in plastic bags by the producers is repackaged into drums at the facility. Liquid waste is absorbed in siliceous marl or solidified with cement. At the beginning of the site operation, both unconditioned and

FIG. 17. The disposal site at Püspökszilágy.

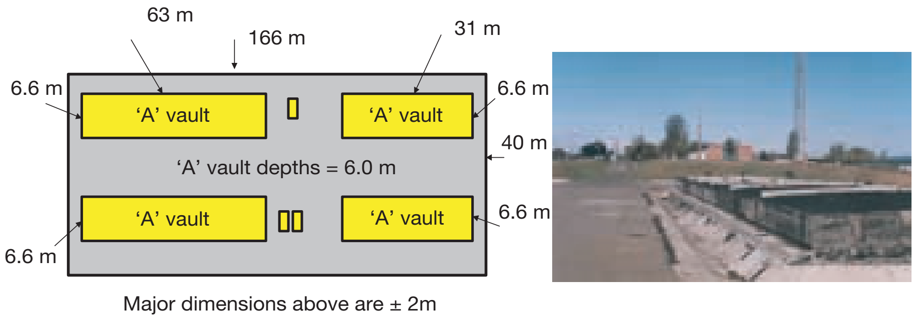

FIG. 18. Layout and dimensions of A type vaults.

conditioned wastes, packaged in plastic bags or metal drums, were placed in layers in the vault and each vault was backfilled with concrete. A few years ago the authorities required retrievability of the newly emplaced waste pending confirmation of post-closure safety. Since that time no backfilling has taken place. Two vaults have already been sealed and temporarily covered.

The B type disposal system consists of 16 boreholes, each with a diameter of 40 mm, and 16 boreholes with a diameter of 100 mm each. The boreholes are stainless steel lined and 6 m long, located inside a concrete monolith structure. The boreholes are used for the disposal of high activity wastes. These wastes come from isotope users and are regarded as high level waste based on the Hungarian National Standard for the Classification of Radioactive Wastes (HLW: dose rate at the surface is higher than 10 mGy/h). High activity gamma sources are usually put into a special disposal container (because of its shape it is called a 'disposal torpedo') and sealed. Gamma sources having no surface contamination are not packaged for disposal, but lead containers are used for transport. Alpha and beta sources have to be packaged into polyethylene casings. Gamma sources are not conditioned prior to disposal in boreholes. Usually twice a year the boreholes are partially filled with cement grout up to the level of the sources.

The D type disposal system consists of four 6 m long boreholes, each with a diameter of 200 mm. These disposal boreholes are designed to accomodate the longer lived (half-life longer than 5 years) spent sealed sources.

A particular feature of the site is that neither the original licence nor licensing of the expansion specified WAC. For this reason, high activity sources and sources consisting of long half-life and alpha emitting materials have also been disposed of in vaults. There were two important exemptions. Radium sources (needles, capsules, etc.) from medical applications had been collected and stored at the National Institute of Oncology. In the early years, the

Püspökszilágy repository accepted ^{238}Pu and ^{239}Pu sources for disposal. These two cases were found to comply with safeguards requirements, but these practices have been terminated and Pu sources are now collected and stored at the Institute of Isotopes.

The lack of defined WAC means that, other than external dose rate, an acceptable standard is not established against which the type of waste received can be judged to be in conformance or not. The operating organization of the facility recognized that inconsistencies existed in the records of waste historically stored at the site.

A.7.3. Refurbishment activities

The repository has operated for more than 30 years without accident or significant release of radioactivity to the environment, but also without any investment for upgrading. As a consequence, the equipment has become obsolete and the physical conditions of the operating systems have degraded. In 1998 the new operating organization of the facility, the Public Agency for Radioactive Waste Management (PURAM), started a systematic programme to upgrade repository safety. During 1998 and 2002 the safety re-evaluation of the repository was the primary focus, in parallel with some basic modernization and refurbishment measures.

One of PURAM's objectives has been to upgrade the physical state of the facility and to provide better conditions for future operation. The main areas of upgrading are the following:

(a) Physical protection (new fence system, new access control, new security equipment);
(b) Radiation protection (replacement of obsolete measuring devices, enhancement of environmental monitoring);
(c) Data acquisition (new data recording system, waste characterization capability, new meteorological station);
(d) Transport (new transport vehicles and containers).

Repair and improvement of buildings, refurbishment of the electrical system and water supply, specialized sump water collection and ventilation systems, a decontamination facility and improvement of the fire fighting system will complete the modernization activities. The other main upgrading objective is to prepare for conversion of the existing treatment building into a centralized interim storage facility for institutional radioactive waste not suitable for near surface disposal. The building was designed in the 1970s to treat and condition raw low and intermediate level radioactive wastes (liquid and solid) from isotope applications, but remained unused. The centralized interim storage facility can also be used for storage should a need arise to receive a large amount of waste at the repository site pending later disposal.

A.7.4. Safety concerns

The safety of the facility has not previously been the subject of a comprehensive assessment. When the temporary licence of the expanded part of the repository expired in December 2000, the regulatory body required a comprehensive safety assessment as a condition for issuing the permanent licence.

Two safety analyses were completed to answer the questions of whether the site would remain safe in the future, or if corrective measures were needed [84, 85]. To support the safety assessment, an uncemented and a cemented compartment in an A type vault were opened in March 2000 to check on the condition of the engineered barriers and the disposed wastes. The basic objective of the investigations was to evaluate the condition of the disposed wastes as well as the condition of the more than 20 year old concrete and metal structures. The vaults were found to be dry, and the vaults, cap and wastes were found to be in good condition with little apparent degradation of either concrete or waste packaging. When the investigations were finished, the compartments were closed and sealed.

Based on the safety assessments it was concluded that the same level of operational and environmental safety can be expected up to the end of passive institutional control of the site. The facility as a whole is suitable for safe disposal of low and intermediate level short lived waste. Beyond the passive institutional control period, mostly because of the significant amount of long lived components still disposed of (^{14}C, ^{226}Ra, ^{232}Th, $^{234,235,238}U$, ^{239}Pu and ^{241}Am), inadvertent human intrusion (or any other scenario resulting in exposure to waste after deterioration of concrete barriers) could exceed both the dose constraint and the dose limit. Consequently, the Püspökszilágy repository is considered to be unsuitable for some of the long lived waste formerly emplaced there [86].

Key recommendations relating to the future management of the site were as follows:

(a) Certain long lived waste and high activity spent sources should be removed from the facility.
(b) The repository cap should be redesigned.
(c) Long term settlement within the vaults should be minimized. At an appropriate time, the vaults should be completely backfilled.

(d) Steps should be taken to minimize the chances of future human intrusion by recording information about the facility and by providing an extensive period of administrative control following repository closure.

International recommendations (e.g. from the ICRP and IAEA) addressing exposure from past practices call for obligatory intervention above 100 mSv/a, and a more optimized (efforts and compliance) intervention when doses of between 10 and 100 mSv/a are observed. The basis for optimization is the real dose associated with intervention activities versus reduction of the potential dose in the future. Such an optimization has not yet been performed in Hungary.

A.7.5. Identification and evaluation of the the corrective action options

Further work is planned to resolve these issues with the objective of providing full assurance of post-closure safety. This further work is likely to involve changes in the characteristics of the facility, updating of closure plans and enhancements of the methods used to evaluate potential post-closure radiological impacts [87].

In covering this work, Hungary has been relying on external assistance and collaboration. Aside from Hungary's part and IAEA support (Technical Cooperation Project HUN/4/015), the third 'pillar' of the technical cooperation in the safety enhancement programme is the European Union's PHARE project. The aim of the PHARE project is to decide on the most appropriate method of safety upgrading [88].

Due to the large number of parameters involved, an optimized intervention programme will be established on the basis of a feasibility study. Decisions reached concerning the favoured waste management option will be of interest to a number of stakeholders. Implementation will involve the commitment of substantial sums of money. It is therefore important that the decision is well developed. This will be enhanced by application of a formal multi-attribute analysis approach (Fig. 19).

As well as conducting an options assessment to identify the preferred waste management options, it is necessary to determine how to implement the preferred option. The implementation plan will be used as the basis for defining an equipment list and a timeline.

The feasibility work will provide a view as to the best option for the management of all of the waste currently disposed of at the site (except the spent sources in the B and D boreholes, which will be dealt with separately).

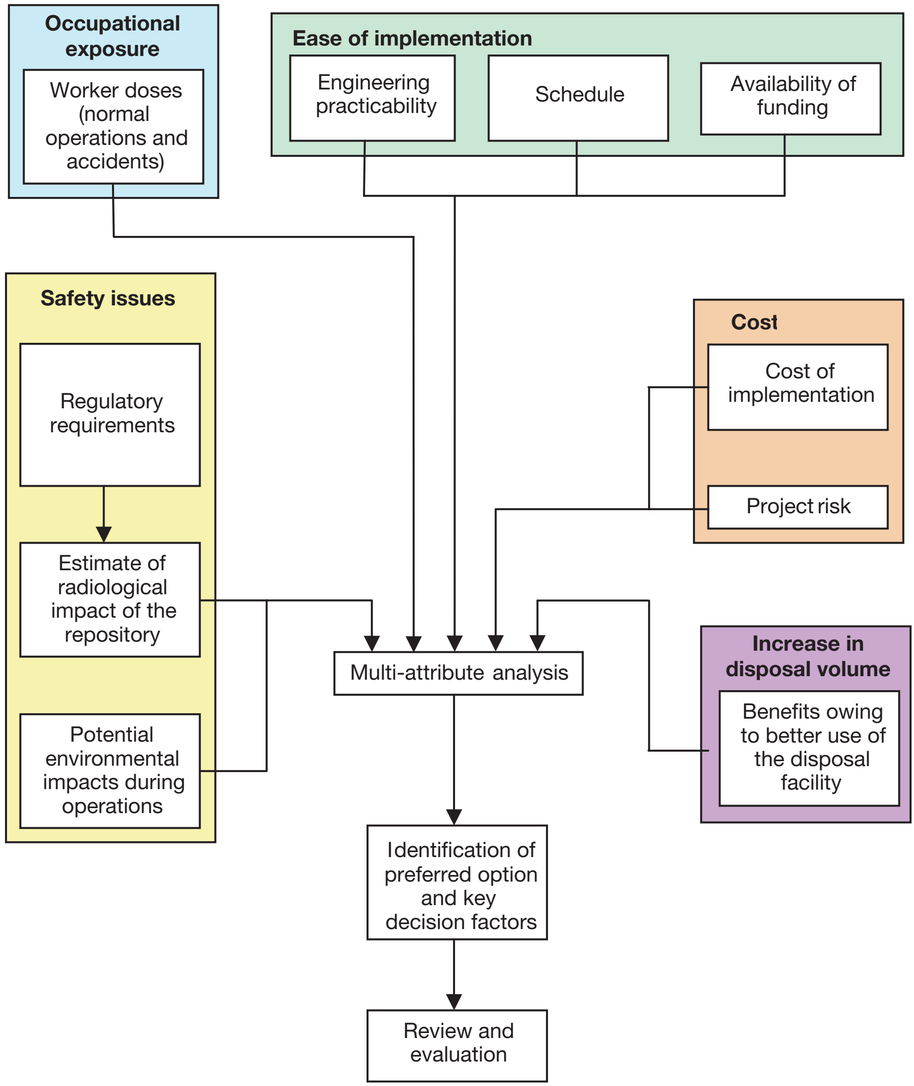

FIG. 19. The multi-attribute analysis approach.

Recommendations on the most effective options will be accompanied by explanations of why the specific option is preferred. This will provide a good basis for discussion with interested stakeholders. The overall outcome will be a plan that provides for improvement of the long term safety of the facility, while at the same time allowing for more effective use of the available disposal volume.

The following corrective action options are being considered in the feasibility study:

(a) Minimum intervention: This option will most probably result in little or no change in the environmental impact to the surrounding area in the short term. The vaults and wastes have been found to be dry and in good condition, indicating that there is little or no leaching of water into the vaults. In the longer term it is expected that the cement and bitumen sealing the vaults will degrade, allowing water to leach in and subsequently allowing contaminants to leach into surrounding soils. Local communities will be affected by an increased traffic burden on the roads, noise and dust generation. When the repository reaches capacity (in about two years) it will be necessary to construct a new repository to cope with subsequent waste generation.

(b) Administrative approaches: This option is similar to option (a) except that there will be an increase in administrative control to help ensure that the site is recorded as unsuitable for building after closure and reduce the likelihood of future intrusion into the waste. When the vaults reach capacity a new repository will need to be constructed to cope with any subsequent waste.

(c) Better isolation through construction of an improved cap: This option, along with options (a) and (b), will allow wastes to be deposited at Püspökszilágy for another two years but will involve further development work, including grouting voids within vaults and construction of a multi-layered cap with clay and impermeable membranes. Construction of a clay cap has many positive environmental impacts. These include minimization of exposure on the surface of the waste facility; prevention of vertical infiltration of water into wastes that would create contaminated leachate; control of gas emissions from underlying waste and creation of a land surface that can support vegetation and/or be used for other purposes. However, the cap will consume large quantities of clay (depleting national resources), increase traffic levels, fuel consumption and air pollution, and may destroy flora and fauna in the clay source area. This option will also result in the need for a new repository when the current facility reaches capacity.

(d) Better isolation through construction of an improved cap and installation of curtain walls: This option will entail similar environmental impacts to option (c), but will have additional impacts due to the construction of a curtain wall. The curtain wall will be a major construction project using substantial amounts of energy, cement and other construction materials.

(e) Removal of readily identifiable sources from easily accessible vaults: The environmental impacts of this option will be mainly radiological. However, the removed sources will likely be contaminated with non-radioactive contaminants and will require appropriate disposal elsewhere, generating additional environmental impacts.

(f) Removal of readily identifiable sources from all vaults: The radioactive sources will likely be contaminated with non-radioactive contaminants and will require alternative disposal. It is also expected that small quantities of cement rubble will be generated from source recovery. The cement rubble, however, will be deposited back into the vaults as it is likely to be contaminated. The voids will then be filled using large quantities of cement.

(g) Removal of all sources from easily accessible vaults: This option will have a largely radiological impact. However, the removed sources will likely be contaminated with non-radioactive contaminants which will also require disposal at another facility. This option will involve an increase in cement use during the grouting phase.

(h) Removal of all sources from all vaults: This option involves greater intrusive work into the vaults, and therefore it is expected that there will be larger volumes of cement waste, dust produced and resources used (energy, water, etc.).

The options for management of waste generated in the future are:

(1) Storage at source pending disposal at Püspökszilágy: The environmental impacts of this option are comparable to options (e)–(h) above with regard to the removal of sources, waste sorting, grouting and capping. However, additional environmental impacts may be incurred by the temporary storage of wastes. A temporary storage facility will have to be constructed which will ensure that waste containers are stored safely. This will reduce the likelihood of accidents through human activities, e.g. dropped loads, spills. A monitoring and measuring regime will also be necessary until the wastes are disposed of. Wastes already stored in the Püspökszilágy vaults will be sorted, creating space for additional wastes and extending the lifetime of the repository.

(2) Storage at source pending disposal elsewhere: This option will involve construction of a temporary storage facility at the source of generation to house wastes awaiting disposal at a new repository. The temporary facility will have to ensure that the waste containers are stored safely,

TABLE 3. CRITERIA AND SCORING

		Favourable	Intermediate	Unfavourable	Unacceptable
		10 9 8	7 6 5 4	3 2 1	0
1	Dose to operators	Not credible for any worker to approach regulatory limits Individual dose < 0.1 mSv/a	One or more workers might approach regulatory limits, but action to constrain doses unlikely to be necessary 0.1 mSv/a < individual dose <5 mSv/a	One or more workers likely to approach legal limits – action to constrain worker doses probably necessary 5 mSv/a < individual dose <20 mSv/a	Breach of regulatory limit likely for one or more workers Individual dose >20 mSv/a
2	Dose to the public	Not credible for any member of the public to approach regulatory limits under any circumstances		Dose to members of the public may approach regulatory limits under accident conditions	Dose to members of the public may exceed regulatory limit under accident conditions
3	Non-radiological risks to workers	Risks to workers comparable to those in low risk industry and easily managed	Risks to workers comparable to those in low risk industry, but significant effort required to ensure that this is the case	Risks equivalent to those in moderate risk industry but requiring significant management effort to ensure they are controlled	Risks comparable to high risk industry: $>10^{-3}$/a
4	Non-radiological risks to the public	Risks to the public similar to those generated by low risk industry and easily controlled	Risks to the public similar to those generated by low risk industry, but significant effort required to ensure that this is the case	Risks to the public similar to those generated by moderate risk industry but requiring significant management effort to ensure they are controlled	Risks to the public significant

TABLE 3. CRITERIA AND SCORING (cont.)

		Favourable	Intermediate	Unfavourable	Unacceptable
		10 9 8	7 6 5 4	3 2 1	0
5	Post-closure performance	Substantial improvement on current performance	Some improvement on current performance	Performance similar to current performance	Performance worse than current performance
6	Non-radiological environmental impact	No significant impact	Moderate impact	Significant impact	Impact unlikely to be acceptable under Hungarian conditions
7	Engineering feasibility and project risk	Uses established technology and methods with minimal chance of the project failing to meet objectives	Some development work or use of novel technologies and some chance of project failing to meet objectives	Significant development work required on technology and significant chance that the project will fail to meet objectives	High chance of failure
8	Timescale for implementation	No credible uncertainty could lead to option being implemented later than required	Some chance of time overrun, but probability and consequences minor	Chance of time overrun — probability and/or consequence significant	Cannot be implemented on the required timescale
9	Socio-political acceptability	No significant socio-political concerns expected	Some concerns possible	Concerns likely	

TABLE 3. CRITERIA AND SCORING (cont.)

		Favourable	Intermediate	Unfavourable	Unacceptable
		10 9 8	7 6 5 4	3 2 1	0
10	Potential for allowing disposals at Püspökszilágy beyond those allowed by current arrangements	Can accommodate at least 25 years of arisings at current projections; no significant risks to this outcome	Some risks of failing to accommodate at least 25 years of arisings at current projections	Only a fraction of the next 25 years of arisings can be accommodated within Puspokszilágy	No further arisings can be accommodated

reducing the likelihood of accidents through human activities. Storage of waste containers in this manner will require monitoring and measuring regimes.

(3) Storage at Püspökszilágy pending disposal at Püspökszilágy: This option involves the temporary storage of waste at Püspökszilágy before it is deposited in the vaults. The original waste in the vaults will be subject to retrieval and better packaging methods, creating more space for additional wastes and extending the lifetime of the repository. However, the sources that have been removed from the vaults will have to be disposed of in a more suitable facility elsewhere and will require the construction of a new repository.

(4) Storage at Püspökszilágy pending disposal elsewhere: This option would involve the transport of wastes from the source to the Püspökszilágy site, temporary storage there, and then transport to a new repository. The criteria against which options are scored will allow options to be distinguished in a useful way and help ensure that all important issues are addressed (Table 3). Options are assigned a score between 0 and 10, with increasing numerical scores representing increased favourability. An option scoring zero against any criterion is regarded as unacceptable either because:

— It is unacceptable in principle (for example because it breaches Hungarian law);
— It is unacceptable as a stand-alone option (for example, it may not allow further disposal at Püspökszilágy, but might offer desirable improvements in post-closure performance if coupled with another alternative).

One of the options was to have been selected by the end of 2004.

A.7.6. Summary

The Püspökszilágy repository is considered to be unsuitable for certain wastes formerly emplaced in it. Based on recent safety assessments, a judgement has been made that long term safety of the Püspökszilágy repository may be ensured, but only with some technical and administrative modifications to the facility.

In 1998, Hungary started a systematic programme to upgrade the repository. During 1998 and 2002, the safety re-evaluation was the primary focus, together with some basic modernization and refurbishment measures (replacement of the obsolete equipment, supplementary site investigations, re-inventory, near-field and far-field studies).

In 2003, a project was launched to select the most appropriate methods for enhancing safety and to prepare for corrective actions. Important elements of this phase include construction of the central interim storage facility, inventory re-evaluation, a feasibility study, a detailed work programme, licence preparations and application for international assistance. The final step is the implementation of safety upgrading measures based on the selected option.

For any proposed intervention, the benefits (in terms of risk or dose averted) should be balanced against cost. In addition to the work on safety reassessments, it is necessary to develop short term and long term plans for providing disposal and storage capacity for all the waste types currently disposed of at the site.

According to PURAM's plan, the repository will be operational for an additional 40–50 years, receiving radioactive wastes from non-nuclear power plant waste producers. By the end of this period a deep geologic repository should be available to receive those long lived wastes temporarily stored in the Püspökszilágy facility that are not amenable to disposal in a near surface repository. Bearing this approach in mind, measures will first be taken to provide additional disposal capacity at the site.

A.8. LATVIA: RADIOACTIVE WASTE PROGRAMME

A.8.1. History and magnitude of the use of radiation sources in Latvia

In 1959 it was decided to build a radioactive waste repository in Baldone. Operation of the repository began in 1962, applying the technology and standards which were valid all over the Soviet Union at that time.

Since then, the Baldone repository has been regularly adapted and modernized, the last major development being construction of a new treatment/storage facility, which has been in operation since 1995.

In Latvia, radioactive wastes are produced by four main groups of activities: research, medical practice, various industrial activities and, until 1994, activities of the Soviet armed forces.

The former Soviet Union made extensive use of lighthouses powered with radionuclide thermoelectric generators (RTGs). These are still used by the Russian Federation. Such generators contain thermocouples that convert the decay heat from a radioactive substance to electricity. Nine such lighthouses, containing of the order of a few hundred PBq of ^{90}Sr, were located in Latvian waters. These lighthouses are no longer a problem as the RTGs have all been returned to the Russian Federation.

As a military centre in the Baltic region under the former Soviet Union, Latvia received relatively large quantities of radioactive waste for disposal at Baldone, compared to the amounts of waste delivered to the corresponding facilities in Estonia and Lithuania.

In many cases it is difficult to clearly separate defence related applications of radioactive materials from civilian applications in Latvia.

A.8.2. Radioactive waste management

Since the beginning of the 1960s the radioactive waste produced in Latvia has been collected and transported to the central storage/treatment/disposal facility at Baldone, which is operated by the State enterprise RADONs. Decommissioning waste from the Salaspils research reactor is being temporarily stored on-site.

A new radioactive waste management agency, RAPA, has been established. Its main tasks are radioactive waste management and decommissioning of the Salaspils research reactor.

A.8.2.1. Basic design of the Baldone repository

The storage and disposal vaults of the Baldone repository are constructed close to the top of a small hill with soft slopes, about 60 m above the level of the Baltic Sea. The waste handling part of the facility is shown in Fig. 20 and includes 7 vaults. They are concrete underground vaults with capacities from 40 to 200 m^3 (Table 4). As the vaults for solid waste have been filled up, a new 1200 m^3 vault has been built and has been in use since the end of 1995.

Figure 20 shows the entire fenced-in area. Waste is received from the main road shown at the lower left corner. Near this entrance is the administration building. There is a long access road to the place where the waste is stored.

Solid waste was generally placed in the concrete vaults without conditioning. When a waste layer reached a thickness of about 1.5 m, the voids were filled with mortar using ordinary construction cement. A similar process was used for biological waste. In these cases, however, the waste was first sterilized and then embedded in gypsum before final disposal. Spent sealed sources were disposed of in their industrial shielding containers. It was not until the mid-1980s that such sources were conditioned, that is, removed from their original containers and placed together with several other sources in lead containers,

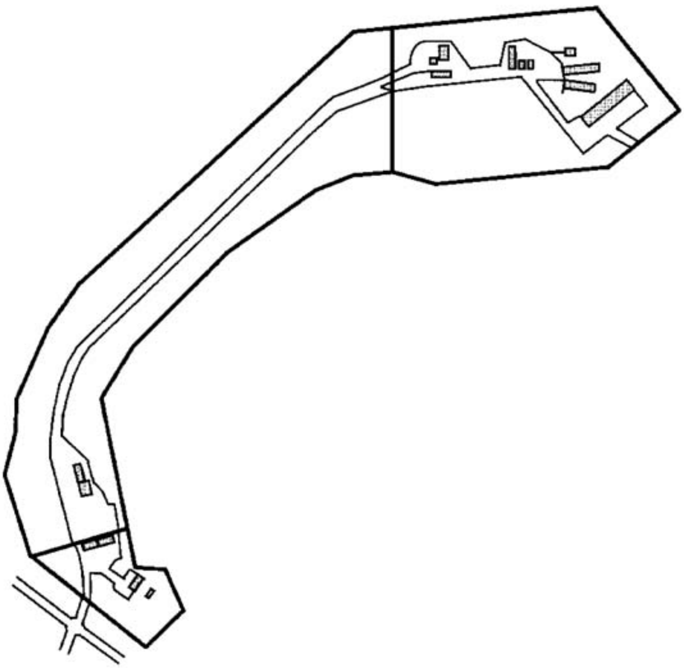

FIG. 20. Storage/disposal vaults at the Baldone repository.

TABLE 4. BASIC INFORMATION ON VAULTS AT THE BALDONE REPOSITORY

Vault number	Volume (m^3)	Proportion of radwaste (%)	Operating period
1	200	50	October 1962 – June 1973
3	200	80	May 1973 – November 1986
4	40	90	May 1974 – November 1988
5	40	85	June 1987 – September 1991
6	200	70	May 1988 – August 1996
7	1200	In use	December 1996 to present

any voids being filled with molten lead. In this way the volume of the waste was reduced and long term storage safety was improved.

Before 1985 all liquid waste was stored in the stainless steel tank. Thereafter, according to new regulations all of the stored liquid waste was treated and the secondary solid waste disposed of. Treatment of the liquid waste took place in 1988 and the processing equipment was later removed. Since then, liquid waste has been stored only through the winter. In all other seasons such waste is solidified with cement and used as mortar for the conditioning of solid waste.

Since 1995, when vault 7 was put into operation, and until recently, all radioactive waste has been conditioned and stored in transportable containers, either steel drums or concrete containers. Now a new concept, based on IAEA recommendations for near surface waste disposal, has been adopted and is in use.

A.8.2.2. Radioactive waste inventory

It is clear that a number of questions exist about the completeness and correctness of the data, especially for the older shipments of waste for which only partial information is available. The quality of the data is also much better for the sealed sources, which represent most of the radioactivity, than for other waste.

In 1991 a computer based database was developed and all the data previously accumulated in handwritten records were progressively transferred into it. The database is still being completed and developed. However, the inventory of each disposal/storage unit can be seen at any time. Information on the individual vaults is given in Table 4. The radioactivity of the main long lived isotopes in the disposal vaults, of which more than 10^{10} Bq were present in the vaults in January 2003, is given in Table 5.

A.8.3. EU DG-Environment project — Consortium d’assistance opérationnelle aux pays d’Europe de l’Est (CASSIOPEE) activities in Latvia

The main objective of the project was to provide advice to the Latvian authorities on the safety enhancements needed and WAC for the near surface radioactive waste disposal facilities of the Baldone repository. The project included the following activities:

(a) Examination of the current status of the management of radioactive waste in Latvia in general, and at the Baldone repository in particular.

TABLE 5. RADIOACTIVITY OF LONG LIVED ISOTOPES IN THE DISPOSAL VAULTS OF THE BALDONE REPOSITORY

Radionuclide	Total activity (Bq)
Vault 1	
^{3}H	5.7E10
^{232}U	5.9E10
^{228}Th	6E10
^{238}Pu	1.3E11
^{60}Co	2.5E11
^{14}C	4.6E11
^{210}Pb	5.9E11
^{226}Ra	8E11
^{90}Sr	1.6E12
^{137}Cs	2.3E12
Total	6.23E12
Vault 3	
^{147}Pm	1.1E10
^{239}Pu	1.5E10
^{22}Na	4.6E10
^{226}Ra	8.3E10
^{40}K	8.8E10
^{55}Fe	9.3E10
^{241}Am	2.3E11
^{60}Co	3.6E11
^{14}C	1E12
^{210}Pb	1E12
^{26}Al	1.1E12
^{238}Pu	1.3E12
^{3}H	1.8E13
^{63}Ni	9.5E12
^{137}Cs	5.8E13
Total	9.31E13

TABLE 5. RADIOACTIVITY OF LONG LIVED ISOTOPES IN THE DISPOSAL VAULTS OF THE BALDONE REPOSITORY (cont.)

Radionuclide	Total activity (Bq)
Vault 4	
^{63}Ni	1.91E10
^{3}H	3.62E10
^{238}Pu	9.96E10
^{241}Am	1.15E11
^{60}Co	2.10E11
^{14}C	3.13E11
^{90}Sr	4.91E11
^{137}Cs	1.95E12
Total	3.26E12
Vault 5	
^{63}Ni	1.91E10
^{3}H	2.65E10
^{14}C	3.31E10
Total	1.05E11
Vault 6	
^{239}Pu	1.06E10
^{152}Eu	2.44E10
^{147}Pm	2.53E10
^{14}C	3.57E10
^{204}Tl	5.86E10
^{210}Pb	6.32E10
^{226}Ra	2.62E11
^{241}Am	3.16E11
^{238}Pu	1.14E12
^{90}Sr	1.15E12
^{3}H	6.32E12
^{60}Co	6.98E12
Total	3.61E13

(b) Long term safety analysis of the Baldone repository, including:
   - Radiological safety in the operational phase, including the planned increase of capacity for disposal and long term storage;
   - Radiological analysis for the post-closure period;
   - Environmental impact assessment (non-radiological components).
(c) Recommendations for future updating of radioactive WAC.
(d) Recommendations for safety upgrades, if necessary, to the facility.

According to the conclusions of the CASSIOPEE analysis:

(1) The status of the Baldone repository in the short term is not a matter of concern;
(2) Radioactive waste is stored under sufficiently safe conditions, and in normal circumstances no significant migration of radioactive substances into the environment or impact on nearby residents is foreseen;
(3) In general, radioactive waste management creates a comparatively low level of risk;
(4) Regarding forthcoming needs for intervention, the resulting doses after 30 years without a cover justify the installation of a cover for the closure period of the repository.

Current plans for building new waste storage facilities in Latvia have been elaborated on the basis of recommendations derived from a long term safety analysis of the Baldone repository performed by CASSIOPEE from 2001 to 2002:

(i) Building of a dedicated long term storage facility for spent sealed sources and long lived waste;
(ii) Modification of the design of the disposal vaults to correspond to the best practices in other countries.

The anticipated radiological impact of the new vault no. 8 has been evaluated in the framework of the CASSIOPEE analysis, including applicable criteria, conditions, input data, hypotheses and recommended corrective measures.

A.9. LITHUANIA: THE MAIŠIAGALA REPOSITORY

A.9.1. Background

The radioactive research, medical and institutional wastes from Lithuania and the Kaliningrad district were disposed in the Maišiagala repository, which is currently managed by the Radioactive Waste Management Agency (RATA).

The repository is situated about 35 km northwest of Vilnius, the capital of Lithuania. It was built in 1963 on the top of a sandy hill. The eastern side of the hill ends in a swampy lowland by a steep slope. The volume of the repository is about 200 m^3. The dimensions of the monolithic reinforced concrete vault are 5 m × 15 m × 3 m, the thickness of the sidewalls is about 0.25 m and the thickness of the bottom is about 0.2 m. The side walls are covered from inside and outside with 0.02 m thick cement and sodium aluminate coatings. From outside they were also covered with 2 layers of hot bitumen. The bottom of the vault is covered with bitumen and 2 layers of rubberoid. Figure 21 gives a schematic view of the repository.

The repository consists of 6 sections which were filled sequentially during its operating period. Each section was filled with radioactive waste in a random manner. Disused radioactive sources embedded in biological shielding were buried together with their shielding. Sources without shielding were buried in 2 stainless steel containers. As of 20 November 2001 the total activity of buried radioactive nuclides was 3.42×10^{15} Bq.

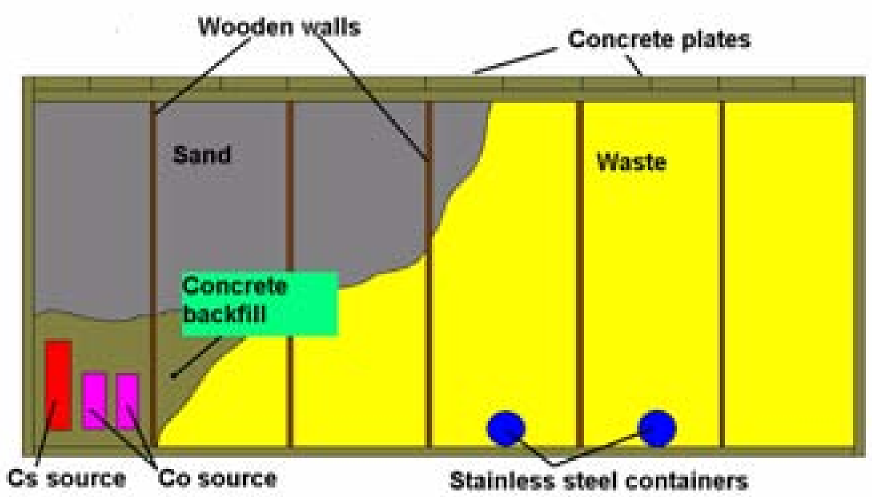

FIG. 21. Schematic view of the Maišiagala repository.

In the course of burial, radioactive waste was constantly interlaid with concrete. When the disposal facility was closed in 1988, only three-fifths of its volume were filled. The empty two-fifths of the vault were filled with concrete, then sand, then with concrete (0.01 m), hot bitumen and 0.05 m asphalt layers. Monolithic concrete that was covered with bitumen and a 0.05 m thick layer of asphalt closed the vault. A sand layer, the thickness of which was not less than 1.2 m, formed the cap.

A.9.2. Radiological situation

Four wells near the repository (one on each side of the repository) and 4 wells in the anticipated direction of groundwater flow were drilled to control the radiological situation around the repository. The groundwater level was about 5 m below the repository bottom and had never been higher than 1.5 m below the repository bottom. The content of radionuclides and their activities were investigated in the water probes from each well once per quarter.

More than 95% of the total activity in the repository is from tritium. The maximum volume of ^{3}H activity in the wells in the year of measurement is shown in Fig. 22.

The total β activity was measured, showing the presence of the natural radionuclides ^{214}Pb, ^{214}Bi and ^{40}K in the groundwater. A very low activity of the

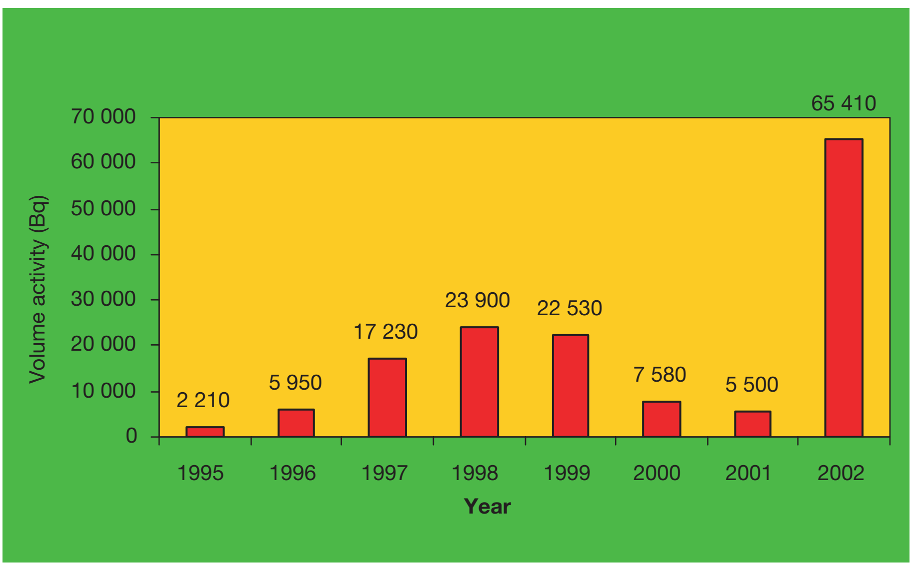

FIG. 22. Maximum volume of ^{3}H activity in the control wells at Maišiagala.

^{137}Cs isotope was found at times. Measurements performed after the radiochemical analysis of samples showed that $^{239,240}Pu$ and ^{90}Sr activities in all the samples were almost the same and did not exceed the background values.

A.9.3. Safety assessment

In 1998, SKB performed an assessment of the long term safety of the Maišiagala repository [89]. The conclusions and recommendations of the SKB study are as follows:

(a) Storage building: Assessment of the existing structure has shown that it is not designed and constructed to a defined set of criteria to serve as a long term disposal facility. Documentation of the materials used is insufficient to evaluate the facility. The current status of the facility is not known.
(b) Safety of long and short term storage: There are no indications that the waste currently stored presents an environmental problem in the short term. The doses from reference wells will exceed 1 mSv/a during the first several hundred years. Doses will result mainly from release of ^{3}H and ^{137}Cs.
(c) Recommendation: Both waste retrieval and construction of additional surface barrier concepts were examined. It was concluded that both concepts are possible, but the retrieval option is more cost effective.

A.9.4. Upgrading of the repository

A.9.4.1. Background

RATA is in charge of the Maišiagala site and has initiated a new project for upgrading the repository.

The initiating event for upgrading of the repository had already been mentioned in the conclusions of the SKB study and changes in the regulatory requirements. A new classification of radioactive waste was established and requirements for disposal of LILW in a near surface repository were issued [90].

A.9.4.2. Objectives

The objectives of the project on safety assessment and upgrading of the Maišiagala repository are:

(a) Drafting of the safety analysis report;
(b) Specification of the physical protection and environmental monitoring systems;
(c) Definition of an upgrading programme for the Maišiagala repository;
(d) Implementation of the upgrading programme.

The main results to be achieved during the project are:

(1) Establishment of the waste inventory and specification of special software and hardware for recording radioactive waste;
(2) Preparation of a detailed territorial plan of the Maišiagala repository;
(3) Ground drilling for groundwater sampling;
(4) Investigation of the repository's status with regard to the presence of free water inside it;
(5) Analysis of the geological, hydrological and radiological situation of the repository's surroundings;
(6) Analysis of the status of the repository's structure and its durability;
(7) Definition of an environmental monitoring programme, specification of equipment and training of RATA staff in performing environmental monitoring;
(8) Preparation of a safety analysis report for the existing facility;
(9) Preparation of a conceptual and detailed design for upgrading of the repository;
(10) Preparation of a safety analysis report for the upgraded facility;
(11) Preparation of an implementation plan and cost estimate for the implementation phase;
(12) Preparation of licensing documents;
(13) Preparation of licensing documents for work related to safety improvements;
(14) Specification of upgrading of the physical protection system.

The project should be implemented in the coming years.

A.10. NORWAY: RETRIEVAL FROM A NEAR SURFACE REPOSITORY AT THE INSTITUTE FOR ENERGY TECHNOLOGY, KJELLER

A.10.1. Background

The Norwegian nuclear programme was initiated in 1948 with the establishment of the Institute for Atomic Energy (known from 1980 as the Institute for Energy Technology, IFE). IFE is an industrial foundation funded from the national budget and from commercial research programmes [91].

Waste management was initiated by the construction of a treatment plant in 1959 and a storage facility at the Kjeller site in 1965. The drums are standard 210 L steel drums with variable shielding and inner drums, depending on the type and activity of the waste.

In 1970 IFE received permission from the National Institute for Radiation Hygiene (now Norwegian Radiation Protection Authority, NRPA) to bury approximately 1000 drums of radioactive waste and 19 other waste items in a near surface repository at Kjeller.

A.10.2. Description of the near surface repository

In 1970, all LILW generated before then was disposed of in a shallow land disposal at the Kjeller site.

The waste drums were embedded in clay and stacked in 2 horizontal layers. The area covered 11.5 m × 23 m. The upper layer of drums was covered by 1.5–2 m of clay and soil with no engineered barriers. The area where the repository was established has a slightly inclined surface. A drainage sump was established at the lowest level at one end of the repository. The amount of water and content of radionuclides has been measured on a regular basis [92].

A.10.3. Decision making and political process

The process of selecting a site for the disposal of low and intermediate level radioactive waste in Norway started in 1989, when a steering committee was appointed by the Government to investigate possible solutions for the final disposal of all Norwegian LILW [93].

In April 1994 the Norwegian Parliament decided to build a new combined disposal and storage facility for LILW in Himdalen, approximately 30 km from Kjeller. Decisions were also made that the waste in the old near surface repository should be retrieved, that the waste drums containing plutonium should be placed in storage at the new facility, and that the rest of the waste should be disposed of at the new facility.

The decision was not based on technical problems with the old repository, but on findings that shallow land disposal for this kind of waste was not up to date in 1994. Another reason was that, with the construction of the new facility, it was better to have all Norwegian LILW at 1 site (and in a modern facility) than split between 2 sites.

A.10.4. Regulatory requirements

The NRPA is the official regulatory body in Norway for nuclear safety, security and radiation protection. In 1994, the NRPA requested that IFE propose a method of retrieving the waste packages from the repository. They were requested to address the condition of the drums; the method to be used for their removal, repackaging and transport; radiation protection measures and estimated doses to the workers; how the soil close to the drums, in the repository and surrounding area would be handled if contaminated; and identification of the 'plutonium drums'.

IFE's proposal was accepted by the NRPA, and in the second half of 1994 investigations were carried out. Soil samples were obtained from core drillings in the disposal area (i.e. above the drums) and in the surrounding area. Ten drums were retrieved to test the proposed retrieval method (lifting by hooking a chain to the drums with the use of an excavator) and to examine the condition of the drums. Previously, in 1993, 10 drums had been removed, so this was a second check.

It was agreed that retrieval should not start until the new facility was in operation and enough waste packages had been moved from the storage facility at the Kjeller site to the Himdalen disposal facility. This approach allowed indoor storage capacity for the retrieved and repackaged waste. Also, the old repository was functioning well and there was no need for immediate action.

A.10.5. Information and reporting

In June 2001, IFE organized an information meeting for the municipality, neighbouring companies and residents, and interested groups. The NRPA also participated. The planned work was described. During the retrieval work, IFE sent weekly reports to the NRPA. Information was also given on IFE's intranet and on their web site. The work was covered in the local press (newspaper and television). A planned information meeting at the finalization of the work was cancelled because of lack of interest.

A.10.6. Retrieval work

Fencing, barriers and monitoring and surveillance systems had to be established for the retrieval work. The area around the repository was fenced off and classified as a controlled area. The dose rate was not meant to exceed 7.5 μSv/h at the fence. The fence had 2 openings, one for transport of drums out of the area and one personnel entrance. To avoid contamination, the area where the waste drums were handled was covered with a tarpaulin.

The soil was removed from above the drums in phases. Only a part of the repository was uncovered at one time to protect the drums from unnecessary exposure to air and rain. The drums were lifted up from the ground one by one with the excavator and moved to a cleaning and checking position where soil on the drum was removed (Fig. 23). This soil was collected and treated as radioactive waste. The total amount of waste generated was two drums. The drums were then moved into slightly larger drums. The void between the 2 drums was filled with cement. The drums were closed with a lid, moved to a checking and monitoring place and then to the storage facility.

A.10.7. Personnel and environmental monitoring

People working inside the controlled area were classified as occupationally exposed. Individual doses were monitored using personal dosimeters supplemented by electronic personal dosimeters (EPDs). Doses on the EPDs were recorded every working day in the controlled area. During the retrieval period, 14 August to 22 October, doses to 21 persons were recorded. The maximum total dose recorded on the EPDs was 2.06 mSv.

During the excavation work, airborne releases of radioactive dust particles were monitored. Activity was measured daily to calculate the level of

FIG. 23. Retrieval of waste drums at the Kjeller site.

contamination. The filter was changed once a week. An instrument located at the other end of the IFE's premises approximately 400 m away monitored background levels. Workers frequently in contact with drums and clay were equipped with personal air samplers. Filters in these instruments were changed weekly and the activity on the filters was measured. Based on these daily and weekly measurements, breathing masks were not required during the work. The maximum committed dose from inhalation of contaminated dust was calculated. The maximum committed dose from ^{137}Cs was below 5×10^{-4} μSv, from ^{238}Pu below 5×10^{-3} μSv and from $^{239,240}Pu$ less than 0.2 μSv.

Once weekly, dose rates were measured at the entrance to the controlled area, at the IFE's border fence, at the tent, along the walls and at a door to the nearby building.

The dose rate in the environment depended on the number of drums stored above ground in the controlled area. The mean dose rate was 0.19 μSv/h [94].

A.10.8. Contamination and clearance

Retrieval of radioactive waste took place from 14 August to 22 October 2001. Clearance of the area was finished in early November 2002. Clearance levels for contaminated clay specified by the NRPA specifically for this work were 100 Bq/g dry weight for ^{137}Cs, and for ^{239}Pu + ^{240}Pu + ^{241}Am, 10 Bq/g dry weight.

After a section of drums in the repository had been removed, samples of clay from the clay bed were taken and analysed by gamma spectroscopy. All samples were below the clearance levels and the former repository could therefore be closed without removal of contaminated clay. The NRPA gave permission for closure on 26 October 2001.

A.10.9. Repackaging and transport to the Himdalen facility

Major portions of the drums were difficult to identify because labels were either difficult to read or had been destroyed by corrosion (Fig. 24). However, the drums containing plutonium were successfully identified and 166 drums containing plutonium were recovered, given new identification numbers, and repacked in new 330 L stainless steel drums. The other 831 drums were repacked in 300 L steel drums and given new identification numbers. To give the old drums a protective layer and fix the contamination on the surfaces, concrete was pumped into the spaces between the old drums and the new outer drums. The drums were then cleared for transport and brought to a nearby storage building to await transport to Himdalen.

The new combined disposal and storage facility for LILW in Himdalen began operation in March 1999. The main purpose of the facility is direct disposal of conditioned waste packages, and 25% of the facility’s present capacity is for storage. Waste packages placed there are in a ‘disposal ready form’ and will either be encased in concrete in the repository portion of the facility or disposed of at another site [95].

FIG. 24. Drums recovered from the Kjeller site.

A.10.10. Conclusions

As a result of discussions preceding construction of the Himdalen facility, the Norwegian Parliament decided that the contents of the shallow land repository on the premises of IFE at Kjeller should be retrieved and transferred to Himdalen. The repository contained 997 drums and 19 other items of low and intermediate level radioactive waste which had been buried in clay in 1970. Retrieval of the drums started in August 2001 and was completed after 11 weeks of work. The NRPA, as well as the local community and media, were kept informed throughout the process.

The waste drums were in remarkably good condition and their handling caused no significant problems. The original drums were cemented into slightly larger drums prior to preliminary storage at IFE and subsequent transport to Himdalen. Radiological monitoring of the remaining clay in the hole showed contamination far below clearance levels. The total dose received by the involved personnel was less than 2.1 mSv. The total cost of retrieval, repacking, internal transport and radiological and environmental control was 3.6 million NOK, not including Himdalen related transport and disposal/storage costs. Of the 997 drums, 166 are plutonium drums containing a total of 30 g of $^{239/240}Pu$. These drums will be placed in the storage hall of the Himdalen facility.

A.11. RUSSIAN FEDERATION: CORRECTIVE MEASURES FOR NEAR SURFACE DISPOSAL FACILITIES

The RADON system of specialized enterprises for collection, transport, treatment and disposal of institutional LILW in near surface repositories was established in the former Soviet Union in the early 1960s. The following sections present two aspects of the upgrading of near surface facilities, the upgrading of historical vault type repositories and matrices for borehole type repositories with disused sealed sources.

A.11.1. Historical repositories for solid radioactive waste

Two upgrading options were always considered for vault type historical repositories at MosNPO RADON:

(1) Complete or partial retrieval of the waste;
(2) Planning for and upgrading of the facility’s safety.

The first option requires complex design work, development of a variety of special techniques, a system of radioactive waste management, a specific safety assessment and possibly licensing. Only the second option is presented here as an example of RADON's present practice [96].

As part of the monitoring system, wells are drilled in the ground surrounding the repository to monitor the groundwater regime, and the chemical and radionuclide composition of the groundwater. The results of the monitoring allow the integrity of engineering barriers to be estimated and the corrosion resistance of the construction materials to be predicted. Analysis of the monitoring data indirectly showed that the system of engineered barriers does not perform to satisfaction and some corrective measures for restoration of engineered barriers are probably needed.

To find the root cause and define effective corrective measures, it is necessary to study the previously built and operated engineered barrier system.

The work that has been performed to evaluate near surface repositories constructed at the MosNPO RADON site in the 1960s includes the following:

(a) Geophysics studies to determine possible degradation in the construction and waste matrix;
(b) Exploratory boring in the waste matrix and geomedia;
(c) Sampling of the waste matrix, soil and water (if present);
(d) Gamma logging of the boreholes;
(e) Laboratory studies of the samples;
(f) Hydrogeological studies of the site and the repository.

The state of each element of the multibarrier system has been studied. Studies of one repository have shown a disturbance of the protective soil layer, with significant losses in the waterproof properties of the clay.

If the initial hydraulic conductivity of the clay layer was 10^{-8}–10^{-9} m/d, the samples studied had a value of 10^{-6} m/d. Clays of the natural barrier in the near field were found to be water saturated. The lowering of the clay permeability caused destruction of storage integrity. After removal of the soil layer a system of cracks on the surface of the repository can be observed. Studies have shown that the water flow through the cracks in the concrete may reach 10^{-7} m^3/d per 1 m^2 of the surface. Additionally, geophysics and drilling work revealed the presence of a disturbance (fracture and cavity) in the waste matrix.

To restore impermeability of the engineered barriers, a special method of restabilization was developed and tested at MosNPO RADON. The method is based upon secondary grouting of previously cemented waste with special high penetrative compositions on a cement base. Special polyfunctional additives to increase permeability, frost and water resistance, crack resistance, biological resistance, stability and strength, as well as to decrease leachability of radionuclides from the cement compound, are in use at MosNPO RADON. Pumping of the grouting compounds is done through special boreholes drilled into the repository. The method was tested in a pilot compartment of one of MosNPO RADON's units and then implemented at several repositories built in the early 1960s.

The results of secondary cementation were checked by repeated determination of the cement matrix (and waste) permeability. It was observed that, as a result of the regrouting, the permeability of the waste in the compartment decreased, reaching the permeability of host rock. The average injection rate in practice is about 0.08 m^3 of the grout solution for 1 m^3 of preliminary cemented waste form. The results obtained indicate that the method developed for repairing near surface repository engineered barriers can be recommended.

After the work has been done and the initial permeability of cement matrix and waste inside the repository has been restored to the previously low level, the next step to increase safety could be construction or reconstruction of the final cap.

Upgrades at MosNPO RADON have achieved the following:

(1) Isolation of radioactive waste from the environment with maintenance of the required operational qualities of the construction within the stated period, depending on its status or throughout the entire institutional control period;
(2) Possible fulfilment of the planned radiation monitoring activities;
(3) Possible fulfilment of the repair work required during predicted unfavourable conditions.

As a result, a multilayer construction was designed and its parameters are now being investigated in mathematical models from a hydrogeological and temperature point of view to find the optimum values for the specific conditions of the MosNPO RADON site.

A.11.2. Shallow borehole repository for disused sealed radiation sources

Sealed radiation sources are widely used in different branches of industry, medicine, agriculture and scientific research. According to the concept which was established in the early 1960s in the former Soviet Union, spent sources are collected and transported to regional specialized RADON facilities. Special transport containers with top loading and bottom discharging are used for this purpose. Spent sealed radiation sources with short lived radionuclides are disposed of in shallow ground borehole repositories at regional specialized RADON facilities, while sources with long lived radionuclides are stored in shielded containers pending a decision on their final disposal in a deep geological formation.

Spent sealed radiation sources are radioactive waste that can have an extremely high radioactivity level. For example, the specific radioactivity of a ^{60}Co source can be of the order of 10^4 Ci/kg or higher.[2] Delivered to and stored in a special facility they represent the main part of the radionuclide inventory. In a Russian special storage facility the average radionuclide composition of spent ionizing sources is ^{137}Cs (40%), ^{60}Co (25%), ^{90}Sr (22%), ^{192}Ir (8%) and ^{170}Tm (4%).

The borehole repository for spent sealed radiation sources (Fig. 25) is a stainless steel cylindrical vessel with a diameter of 200 mm and a height of 1500 mm which is placed in a steel reinforced concrete well at a depth of 4 m. The vessel's walls are 5 mm thick. The stainless steel loading channel of the repository has a curved (spiral) tube with a diameter of 108 mm × 5 mm. At the upper part of the repository there is a carbon steel conical socket which provides for safe discharge of the transport containers. This socket is closed with a carbon steel lid. The concrete wall of the repository is surrounded by a clay–cement (or clay) mixture which fills the initial construction hole in the original soil as a seal material.

Initially typical repositories were designed for the disposal of sources with the radioactivity corresponding to a radium equivalent of 50 000 g-eq. The maximum dose rate on the surface of a repository near the loading channel must not be higher than 0.82 mR/h.[3] The underground reservoir is heated radiogenically by heat generation from sources. According to the initial design the maximum allowable temperature in the reservoir is 230°C. As a rule there are a few borehole repositories for the disposal of spent sources at regional specialized RADON facilities [96].

The initial repository design was developed at the end of the 1950s on the assumption that the stainless steel underground vessel and the reinforced concrete wall would provide enough protection against the possibility of radionuclide migration into the environment. Therefore only the hazard of irradiation of personnel was calculated in the typical design of a borehole repository.

[2] 1 Ci = 3.70×10^{10} Bq.

[3] 1 rad = 1.00×10^{-2} Gy.

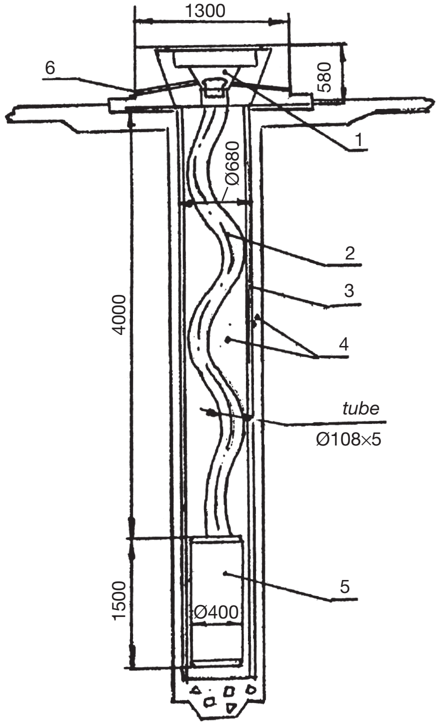

FIG. 25. Design of a typical borehole repository (dimensions in millimetres). 1: carbon steel conical socket, 2: stainless steel loading channel, 3: steel reinforced concrete well, 4: concrete, 5: stainless steel cylindrical vessel, 6: drainage channel.

Practical use of borehole repositories showed the possibility of accelerated corrosion of engineered barriers in powerful radiation fields inherent to repositories containing powerful sealed sources. Investigations of repositories showed dose rates in the underground reservoirs of up to 20 MR/h, temperatures higher than 80°C and concentrations of radiolytic hydrogen of 3.5%. Because of the non-uniform allocation of sources in the underground vessel the radiation fields are extremely high, even for a total radioactivity below the repository limit (radium equivalent of 50 000 g-eq). Due to condensation of water vapours on the cold walls of the loading channel, small amounts of water were accumulated. Although this is a slow process, only a portion of the water is accumulated in the underground reservoir during many years of operation in dumping conditions, when there is a flow of hot air from the radiogenically heated zone of the repository upward and a reverse flow of dump air. The presence of water and powerful ionizing fields significantly decreases the safety of the disposal of sources. Radiolysis of water and air causes an accumulation of hydrogen, ozone and nitrogen oxides which produce nitric acid when they come into contact with water. These processes accelerate the corrosion of source containers [97].

The contamination of water in the underground reservoir was determined to be as much as 10^6 Bq/L and there was considered to be a potential danger of penetration into the surrounding soil, although this has not been observed in practice. Radionuclide releases are also possible through the loading channel by gas–aerosol phase. Therefore, free storage of powerful sealed radiation sources in borehole repositories does not have a sufficiently high level of safety. The technology should be improved and corrective measures for already operating boreholes should be developed.

In order to ensure the long term safety of the disposal it was proposed that the sources be allocated uniformly in a matrix material in the underground reservoir. An additional barrier would be provided to protect sources against direct contact with air and water and decrease radiation and temperature fields.

Different metals were considered for the matrix material. To facilitate the encapsulation process and minimize the influence of high temperature on radionuclides, metals with low melting temperature must be used. For damaged sources this minimizes the volatilization of radionuclides in the process of their encapsulation. For powerful sources, only metals can be applied as matrices.

A technological scheme was developed, which provides immobilization of sources in a metal matrix after they have been loaded into the borehole repository. It was proposed that this scheme be used at regional specialized RADON facilities to correct deficiencies in the initial design concept with open storage of sources in underground reservoirs of borehole repositories. Use of this new technology was then extended to the conditioning of sources in interim or long term storage. The considerable reduction of radiation and temperature fields in repositories due to the application of metal matrices permitted an approximately six-fold increase in their capacity.

This technology includes methods of examining the repository, which allow determination of the following parameters:

(a) Activity of the spent sources disposed of;
(b) Number of spent ionizing sources in the repository;
(c) Proportion of backfill material;
(d) Dose rate on the cap of the repository and in the bottom of the reservoir;
(e) Temperature in the reservoir;

(f) Presence and level of water in the reservoir;
(g) Nuclide content in the water;
(h) Contamination of the receiving component;
(i) Presence of hydrogen in the gaseous phase;
(j) General dimensions of the reservoir.

To realize this technology, MosNPO RADON developed a mobile system which encompasses all the operations needed to encapsulate the spent ionizing sources in a metallic matrix (Fig. 26). This mobile plant can be used for different types of repositories [98]. Sources are encapsulated into metal matrices directly in the underground reservoirs of repositories, allowing the initial plan of disposal of spent sources with subsequent encapsulation to be followed, as well as the implemention of improved technology from the very beginning. Encapsulation technology has been in use at MosNPO RADON since 1986. At regional specialized RADON facilities (e.g. Volgograd, Nizhniy Novgorod, Sverdlovsk) disused sealed radiation sources with total activities of more then 1 million Ci previously loaded into boreholes have been immobilized since 1991 using this movable plant. Only lead and lead based alloys are used for this purpose, lead being the most reliable matrix material.

The results of calculations for different scenarios show that the maximum annual dose to a member of the population does not exceed (5.5–7.5) × 10^{-5} Sv/a, even in the case of complete destruction of the engineered barriers and complete flooding of the repository. This shows the high degree of safety of borehole repositories when sources are immobilized by encapsulation in a metal (lead) matrix. The predicted data comply with the annual dose limit (10^{-4} Sv/a) used at the present time in the Russian Federation and confirm that the operation of these repositories is safe for the population [98].

Due to its chemical properties, lead forms only insoluble compounds with groundwater anions. The heterogeneous character of the exchange reaction fixes the lead corrosion products near the repository. In addition, according to the proposed approach the metallic matrix with sources would be removed and melted for reuse after a period of time (500–1000 years). This is an additional means of protecting the environment from potential contamination by lead corrosion products (Fig. 26).

A.11.3. Conclusions

Over a long storage period, destruction of the integrity of the multi-barrier system may occur in the subsurface repository, which may increase the potential for radionuclide migration into the environment. For historical

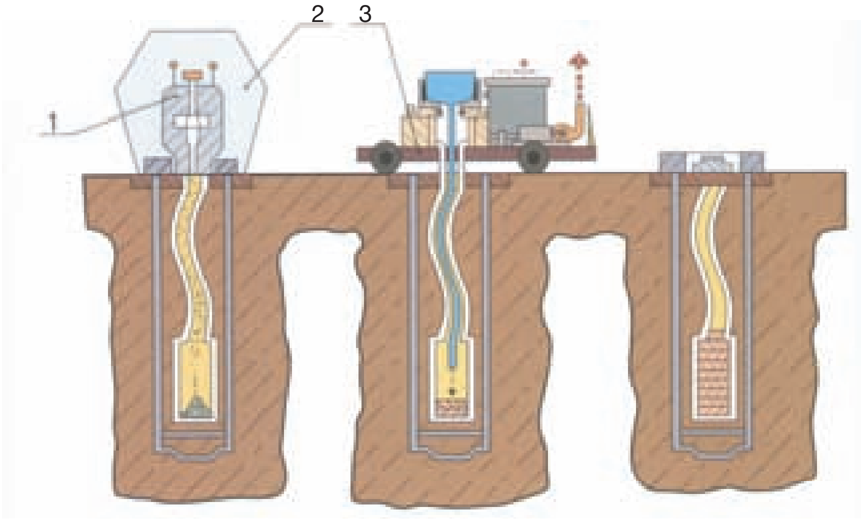

FIG. 26. Immobilization technology of the MOSKIT mobile plant (1: container, 2: uploading unit, 3: MOSKIT mobile plant).

RADON type repositories the most often used corrective measure to reduce possible contact of the waste with water is stabilization of the waste in a matrix [99]. Special cement based grout is used for vault type repositories with solid and solidified waste, and for borehole repositories with spent sealed radiation sources a metal matrix is usually installed.

A.12. SOUTH AFRICA: INVESTIGATIONS AT THE VAALPUTS NEAR SURFACE REPOSITORY

A.12.1. Background

Vaalputs is the national repository for the disposal of LILW in South Africa. The Koeberg nuclear power plant is currently the dominant waste generator disposing of waste at Vaalputs. Near surface trenches are utilized for the disposal of LILW (short lived). The LLW (short lived), contained in metal drums, and the intermediate level waste (short lived), contained in concrete containers, is disposed of in separate trenches.

The disposal trenches at Vaalputs were designed and constructed to be 100 m long, 7.7 m deep and 20 m wide at the bottom and with sides that slope

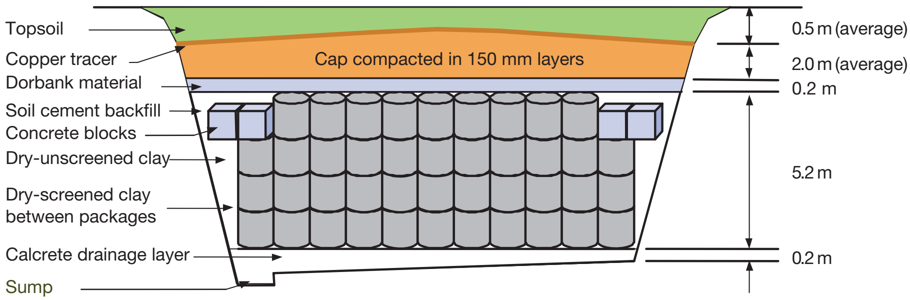

FIG. 27. Intermediate level waste concrete drum trench.

upwards at an angle of 80º. The waste emplacement, backfilling and capping methods were as indicated in the vertical cross-sectional diagram shown in Fig. 27.

The relatively big trenches described above were designed in anticipation that the LILW from 4 nuclear power plants the size of Koeberg would eventually be disposed of at Vaalputs. Because the nuclear energy programme in South Africa did not develop as originally anticipated, together with strategic changes at Koeberg (e.g. waste minimization techniques), the volumes of radioactive waste shipped to Vaalputs for disposal did not materialize as initially expected. This resulted in the trenches being left open for extended periods of time, thereby causing prolonged exposure of the waste packages to adverse weather conditions.

A.12.2. Initiating event: cracking of concrete containers and leaching of waste

In May 1997 it was observed that some of the exposed concrete containers inside the disposal trench showed minor cracks and that material containing traces of ^{60}Co and ^{137}Cs was leaching from these cracks, as shown in Fig. 28.

Although the cracking phenomenon was not reported until May 1997 it was evident that it had evolved over time, since some of the waste packages had been emplaced in the trench as early as 1986. Cracks were observed on the rims and sidewalls, and on the bases of the drums. Only some of the drums could be inspected at the time as the back of the waste stack was already backfilled and covered. It may also be possible that cracked drums that passed the receiving inspection may have been delivered to the repository during the earlier years.

The local regulatory authority was informed of a nuclear occurrence. The occurrence was communicated to the local public via the Vaalputs Communication Forum. The incident raised concerns in the local surrounding

FIG. 28. Leachate from a cracked concrete container at Vaalputs.

communities with regard to the safety of the repository, and also contributed to negative media coverage.

A.12.3. Investigations

Several investigations were launched to determine the extent of the release of activity from the disposal trenches. An IAEA expert team was invited to South Africa to review the situation regarding the cracked drums and also to communicate their findings and observations to the local community. Results of all the investigations showed that no radioactivity had leaked into the environment around the disposal trench.

Both the waste generator and the repository operating organization appointed independent experts in the field of reinforced concrete to investigate the drum cracking mechanism and to report on the findings.

A.12.4. Corrective actions

The following initial corrective actions were undertaken to rectify unsafe conditions:

(a) The leachate from the containers was sealed with a durable sealant to prevent the possibility of fixed contamination becoming available for transport to the environment;
(b) The remaining waste packages were backfilled and covered with dry clay to prevent further exposure to adverse weather conditions;
(c) The waste stack was sealed off with a double wall constructed of concrete blocks;
(d) The section of the trench containing the damaged packages was effectively isolated from the environment by capping.

From an administrative point of view, the following corrective actions were implemented:

(1) The operating organization temporarily suspended further deliveries of concrete waste packages to the repository;
(2) Working groups were established with the waste generator to review the waste management process (e.g. quality assurance measures in the waste package design, manufacturing and procurement processes, the waste conditioning and waste packaging processes, as well as the waste package preservation measures during storage);
(3) The WAC for Vaalputs were revised to include more stringent quality related requirements for waste packages;
(4) Operational procedures and systems at Vaalputs were improved to include more stringent receiving inspections and regular inspections of waste packages with the aim of proactively identifying degrading waste packages;
(5) Annual audits are undertaken on waste generators.

A.12.5. Regulatory requirements

Due to premature signs of deterioration of the exposed waste containers, the national regulatory body required that the existing trenches be backfilled and capped as soon as possible. At that stage, only 70% of the intermediate level waste trench and 30% of the LLW trench had been filled with waste packages. This requirement in itself implied a modification of licence requirements and operational procedures for the site. In addition, the national regulatory body required that in future the metal LLW drums be covered within one month and the concrete intermediate level waste containers within two months after being emplaced in the respective trenches.

A.12.6. Improvements

The premature closure of the trenches necessitated some changes to their original design. The main feature of the implemented changes entailed a concrete cut-off wall, the width of each trench, at the live end of the waste stacks. These walls effectively sealed off the waste stacks and provided a barrier against which backfill and capping could proceed. A schematic diagram of the vertical cross-section of this configuration is shown in Fig. 29.

To prevent the difficulties encountered with the partial closure of waste filled sections of trenches, it is intended to utilize smaller trenches in the future to accommodate a pre-defined number of waste packages delivered in concentrated consignments, and backfilling and capping of the trenches as soon as possible after emplacement of the waste packages (Fig. 30).

A.12.7. Root cause identification

The root causes of the occurrence were identified to be a combination of the following:

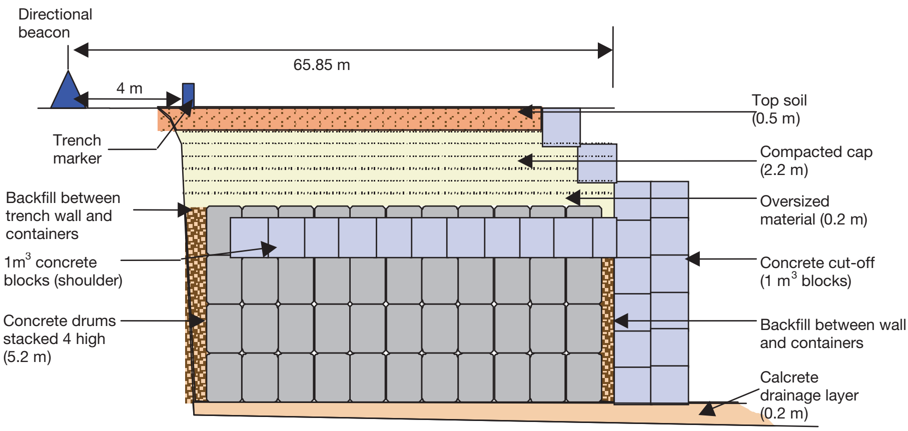

FIG. 29. Vertical cross-section through the intermediate level waste trench, indicating the position of the cut-off wall and details of the capped section.

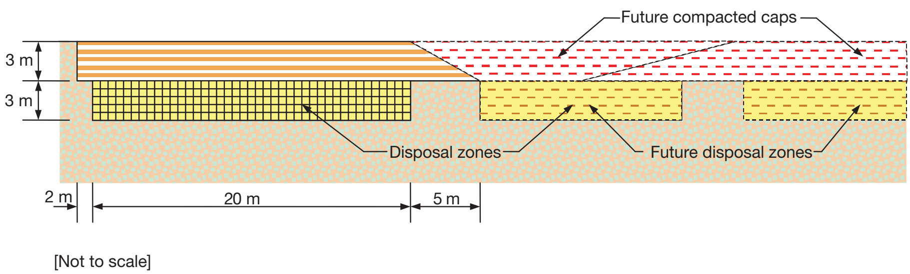

FIG. 30. Vertical cross-section through new LLW trenches, showing the existing disposal zone and future extensions.

(a) Premature degradation of the waste packages due to their prolonged exposure to the elements (e.g. thermal cycling, corrosive effects due to corrosive elements in the soils);
(b) Insufficient quality control and enforcement of the WAC by the repository's operating organization;
(c) Insufficient quality control measures implemented by the waste generator;
(d) Uncertainties and lack of understanding regarding the specifications, lifetime expectancy and performance of the concrete containers in the repository environment (the containers and container specifications were initially adopted from a French design);
(e) Anomalies in the waste segregation, conditioning and waste package filling procedures.

A.13. UNITED KINGDOM: DEVELOPMENT OF THE DRIGG LLW DISPOSAL SITE

A.13.1. Introduction

British Nuclear Fuels plc (BNFL) owns and operates the Drigg disposal site, which is the UK's principal facility for the disposal of low level radioactive waste.

This section describes the development of the Drigg site to date, in particular the upgrading of the site in the late 1980s and early 1990s, which centred around the phasing out of disposal into trenches and the introduction of a revised waste form and disposal into engineered vaults.

A.13.2. Development of the site until the late 1980s

The Drigg site (Fig. 31) is located on the West Cumbrian coast about 0.5 km inland and some 6 km to the southeast of the Sellafield site. The total area of the site is about 100 ha. The ground slopes gently towards the sea, falling from about 20 m above sea level at the northeastern boundary to about 7 m above sea level on the southwestern boundary. To the east the site is bounded by the Whitehaven to Barrow-in-Furness railway line. A small stream flows through the site, discharging into the Irish Sea to the south of the site via the River Irt.

The site was originally developed in 1939 as a Royal ordnance factory and some of the surface features date from this period. Ownership of the site subsequently passed to the Atomic Energy Authority, which was granted planning consent in 1957 for the disposal of LLW in the northern 40 ha of the site, referred to as the 'consented area'. The first certificate of authorization for disposal of LLW was granted in 1958 under the terms of the Atomic Energy

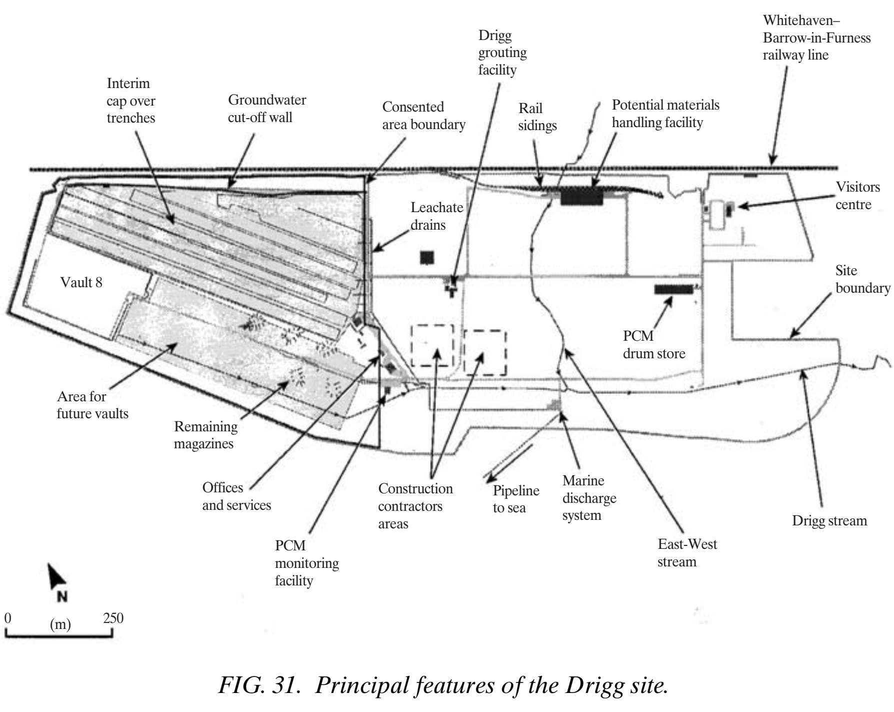

FIG. 31. Principal features of the Drigg site.

Act 1954, and disposal operations commenced in 1959. Responsibility for the site was transferred to BNFL when the company was formed in 1971.

Within the consented area there is clay, typically at a depth of about 5–8 m below ground level, which forms a low permeability base to the disposal trenches. Since the mid-1980s, in excavated areas where the clay layer dipped below the excavated trench base level, bentonite was rotavated into the base of the trench in order to provide a low permeability layer and optimize leachate drainage. The trench bases are graded and include simple drains which, in conjunction with the underlying clay, serve to direct infiltrating rain or groundwater to the southern end of the trenches for collection in a series of drains. Until 1991 the collected leachate was discharged into the Drigg stream and hence the River Irt and the Irish Sea.

Drummed, bagged and loose waste was tumble tipped into the trenches starting from the northern end and progressively covered with at least 1 m of backfill (1.5 m from 1988 onwards) to create a stable surface for continued disposal operations. Disposal was conducted solely by tipping into trenches until 1988, when vault 8 commenced operation as part of the site upgrade discussed below. Trench 7 operations continued, however, until commencement of operation of the LLW high force compaction (HFC) facility at Sellafield in 1995. The total area occupied by the trenches is about 16 ha and the total volume about 500 000 m^3. Due to the effects of self-compaction in the trenches, a total volume of loose waste of about 800 000 m^3 was actually disposed of.

On the western side of the consented area there were 10 former royal ordnance factory storage magazines. Between 1959 and 1967 some intermediate level waste, known as plutonium contaminated material (PCM), was placed in these magazines for storage. The construction of a purpose built PCM store at the southern end of the site allowed the 5 northerly magazines, which contained waste principally in 200 L drums, to be emptied between 1976 and 1986 and subsequently demolished. All the PCM wastes on the site are currently being retrieved and transported to the BNFL Sellafield site as discussed in Section A.13.3.

A.13.3. Site upgrade and current status

In 1987 a major upgrade of disposal operations at the Drigg site commenced with the principal aims of improving waste management practices and the efficiency of space utilization, and enhancing the visual impact of disposal operations. The main features of the upgrade were as follows:

(a) Installation of a groundwater cut-off wall around the north and east sides of the trenches and construction of an interim cap over the completed trenches;
(b) Refurbishment and enhancement of the leachate drainage system;
(c) Phasing out of trench disposal of loose waste in favour of orderly emplacement of compacted, containerized and grouted wastes in engineered concrete vaults.

This section first describes the interim cap and cut-off wall associated with the trenches and the upgrade to the leachate management system. The basis for the revised disposal strategy and associated waste form of high force compacted, containerized and grouted waste is then discussed, followed by a description of vault 8. The upgrading to the new waste form of some of the wastes initially placed in vault 8 and the retrieval of the PCM wastes currently stored at the Drigg site are then briefly discussed.

A.13.3.1. Interim cap and cut-off wall

The interim cap over trenches 1–6 was completed in 1989 to minimize rainwater infiltration into the trenches and hence reduce leachate volumes. During 1995 it was extended to cover trench 7 after completion of trench disposals (Fig. 31).

The cap comprises a 1:25 (vertical:horizontal) graded earth mound incorporating a low density polyethylene membrane at a depth of about 1 m. This membrane sheds rainwater to the perimeter of the cap, where it is collected in drains and discharges into the Drigg stream. The surface of the cap has been seeded with grass and mixed shrubs.

Waste degradation gas is vented passively by means of 15 cm diameter probes driven through the cap into the waste. The probes are perforated along their length within the waste and capped to prevent rainwater ingress. They also facilitate water level measurements and sampling for leachate and gas composition. Periodic monitoring of cap settlement is also carried out. This supports any maintenance required and will input to later decisions on the precise timing of final cap construction.

Prior to installation of the interim cap, a 450 m long, 1 m wide, low permeability cement/bentonite groundwater cut-off wall was installed around the northeast corner of the site running from the northern end of trench 7 to the northeast corner of vault 8. In 1995 the wall was subsequently extended alongside trench 7 prior to the capping. The base of the wall is keyed into the same clay layer that forms the base of the trenches, with a typical wall depth of about 8 m.

A.13.3.2. Leachate management

Refurbishment of the leachate drainage system was completed in early 1991 to allow improved leachate monitoring and controlled discharge directly to the Irish Sea (rather than via the Drigg stream and River Irt). Leachate from each trench and vault 8 is routed by gravity flow through interceptor drains to a common point and then to a set of holding tanks on the western edge of the site, where it is held pending automated discharge. The leachate is flow proportionally sampled and pumped directly to the Irish Sea through a marine outfall.

A.13.3.3. Waste form

In changing from trench to vault type disposals, BNFL considered the range of potential options for the revised waste form. The 2 primary functional requirements of the revised waste form in terms of post-closure safety are:

(1) Residual voidage should be low in order to minimize the potential for significant settlement of the final cap;
(2) Loads across the base area of each container stack should as far as practicable be uniformly distributed in order to ensure that ground bearing pressures are not exceeded when the final cap is emplaced and to minimize differential, uneven settlement.

Also very important in considering waste form options was the objective of the site upgrade programme to significantly increase the projected operational life of Drigg. In this regard, waste volume reduction and space utilization in terms of waste packaging are the principal issues.

The chosen waste form is that based on HFC of the waste, emplacement in 20 m^3 steel International Organization for Standardization (ISO) containers and grouting of the voidage within the ISO container to form a solid product (Fig. 32). This waste form was selected from a comprehensive analysis of the range of options available, taking into account the above requirements and also considering operational aspects. Two new BNFL facilities were introduced in the mid-1990s for production of the new waste form.

A.13.3.4. Vault 8

The first disposal vault at the Drigg site (vault 8) was introduced in 1988. Its purpose is to:

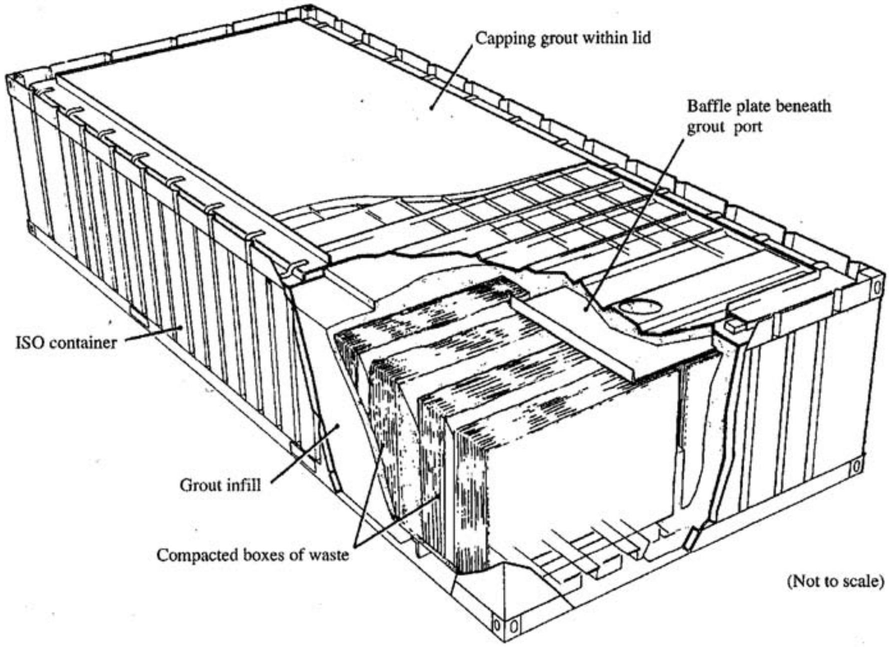

FIG. 32. Fully grouted product container.

(a) Form a platform for the orderly emplacement of containerized wastes;
(b) Maintain a free draining environment around the wastes.

Vault 8 consists of 3 bays and is approximately 175 m wide by 200–265 m long. The average depth is about 5 m and the total capacity about 200 000 m^3 of containerized waste. This depth allows 4 high stacking of nominal half-height ISO containers. The containers are handled in the vault by fork-lift truck.

The vault has surface water drains which collect rainwater from the surface of the vault base slab while an under-slab drainage blanket and perimeter drains collect groundwater from beneath and around the vault. As with the trenches, the principal means of leachate containment is the naturally occurring clay layer at about 5 m below ground level.

A.13.3.5. Upgrading of initial vault 8 disposals

From 1988, waste from non-Sellafield sites was placed in vault 8, principally in either full- or half-height ISO containers. With the progressive introduction of the revised waste form, the majority of disposals to vault 8 until

1995 were in need of further treatment to upgrade them to the new waste form standard. Such wastes are called ‘backlogged wastes’.

The design of the nearly 1000 backlogged half-height ISO containers generally did not minimize associated voidage nor provide a lid with a groutable top surface. They have been transferred to the Drigg grouting facility, where the lids were modified to allow in-fill grouting and the remaining shortcomings to be sufficiently overcome by limiting their positioning to the top of container stacks in vault 8.

The full-height containers, containing predominantly drummed waste, are being transferred to Sellafield for processing at the WAMAC facility, with the wastes being compacted and returned to the Drigg site in half-height containers. By 2003 the WAMAC facility had processed the wastes from about 600 of the nearly 1000 full-height backlogged ISO containers. Upgrading of the full-height ISO backlogged LLW is planned to be complete by about 2005, commensurate with the end of vault 8 operations.

A.13.3.6. Retrieval of PCM

The PCM wastes currently stored on the Drigg site are intermediate level waste and are not associated with the LLW disposal facilities. These wastes must be removed from the site, and it is BNFL policy to recover and transfer them as soon as practicable to Sellafield for further treatment and storage. Retrieval of crated PCM from the 5 remaining magazines commenced in 1998. Retrieval modules have been constructed with facilities for preliminary monitoring and packaging of the crated items in new, size specific overpacks. Retrieved items are then transferred to a central crate monitoring facility at the south end of the consented area so that a detailed radionuclide inventory can be compiled prior to transfer to Sellafield in purpose-built transport containers. Retrieval of PCM wastes from the PCM drum store is also in progress, having commenced in 1997, also using project specific packaging, monitoring and transport arrangements. It is planned that PCM retrieval will be fully complete by the end of 2006, with decommissioning of all the associated facilities by 2010.

A.13.4. Conclusions

This section has described the development of the Drigg site to date. Development has taken place from tipping of wastes into trenches to the orderly emplacement of compacted, containerized and grouted wastes in engineered vaults.

Future operational facilities will consist of a series of vaults within the consented area. In the longer term, engineering measures associated with the eventual closure of the site are planned to include a final cap, groundwater cut-off wall and vertical drain.

The plans for site development will be regularly updated to reflect both operational developments at the site and engineering and scientific developments and understanding. Engineering design work will be used to assess and optimize potential alternative development plans. These plans will continue to be discussed with regulatory authorities and other stakeholders.

A.14. UKRAINE: RADIOACTIVE WASTE FROM PAST PRACTICES AND SAFETY UPGRADING

Radioactive waste resulting from the utilization of ionizing radiation sources is managed by the Ukrainian State Association RADON (UkrSA "RADON"). This organization comes under the authority of the Ministry of Emergencies. Special enterprises were commissioned in 1961 and 1962 to deal with the collection, transport, storage and disposal of radioactive waste from industrial enterprises, medical and research institutes, including spent radiation sources (Tables 6, 7). Radioactive waste repositories located in Kiev and Kharkov caused contamination of groundwater by tritium.

The main cause of radionuclide contamination outside the waste repositories is non-compliance with the technical requirements of engineering structures and with the nuclear and physical requirements for radioactive waste containing tritium. Water accumulated in the repositories results from condensation or infiltration, together with fallout due to poor repository designs developed in the 1950s. Radionuclide contamination has resulted from the migration of unfiltered fallout, subsequent migration into the groundwater and diffusion of radioactive waste in underlying moist soil as the result of failures in the hydraulic isolation.

The causes of a radiation accident at the Kiev RADON location were the following:

(a) Shortcomings in the repository's design;
(b) Absence of a clay layer at the base of the repository;
(c) Lack of layered cementing, disposal of unsolidified radioactive waste containing tritium;
(d) Operation of repositories over more than 40 years with no modernization to improve operational safety.

TABLE 6. UkrSA RADON RADIOACTIVE WASTE FROM PAST PRACTICES AS OF 1 JANUARY 2003

Waste type	Location	Volume (m^3)	Mass (kg)	Activity (Bq)	Main Radionuclides
Solid low and intermediate level radioactive waste	Kiev	1 897	256 163	7.86E15	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Dnipropetrovsk	428	1 001 143	7.04E15	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Odessa	497	279 491	1.51E15	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Lvov	492	1 469 733	1.133E14	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Kharkov	1 321	2 269 291	3.51E14	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
Liquid low and intermediate level radioactive waste	Kiev	413		2.16E12	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Dnipropetrovsk	60		5.7E10	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Odessa	138		4.5E11	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
	Kharkov	28		4.37E10	^{60}Co, ^{90}Sr, ^{137}Cs, ^{226}Ra
Spent radiation sources	Kiev			1.33E15	
	Dnipropetrovsk			1.27E14	
	Odessa			8.41E15	
	Lvov			4.73E14	
	Kharkov			4.57E13	

The primary task in the upgrading of radiation safety at the RADON plants was the isolation of both currently operated and closed repositories from atmospheric fallout. In light of this, on the request of the regulatory authority at Kharkov, RADON, 2 repositories were isolated. This action focused on preventing water infiltration and stopping tritium migration into the groundwater. Predictions indicate that this should make practically impossible tritium

TABLE 7. UkrSA RADON RADIOACTIVE WASTE MANAGEMENT FACILITIES

Facility	Main purpose	Design capacity	Year of commissioning
Kiev	Storage, disposal	Solid radioactive waste: 3075 m^3 Liquid radioactive waste: 1000 m^3 Spent radioactive sources: 4.4E6 GBq	1962
Dnipropetrovsk	Storage, disposal	Solid RW: 450 m^3 Liquid RW: 200 m^3 Spent RS: 1.8E6 GBq	1961
Odessa	Disposal	Solid RW: 583 m^3 Liquid RW: 400 m^3 Spent RS: 1.8E6 GBq	1961
Lvov	Storage, disposal	Solid RW: 1140 m^3 Liquid RW: 200m^3 Spent RS: 2.9E6 GBq	1962
Kharkov	Storage, disposal	Solid RW: 2384 m^3 Liquid RW: 1000 m^3 Spent RS: 2.2E6 GBq	1962

contamination beyond the sanitary and protection zone in an amount that represents a hazard for the population and the environment. A system of monitoring wells has also been completely redeveloped to monitor performance.

To prevent the infiltration of atmospheric fallout into repositories and reduce tritium migration into groundwater, in August 1997 3 repositories at Kiev RADON were covered with special roofing.

Because Kiev RADON, like other repositories, is located within a city's boundaries, extraction of radioactive waste from the emergency repositories and its redisposal at a central disposal site is planned.

The following specific measures are planned:

(1) Preparation of a place for loading operations;
(2) Preparation of radioactive waste in the containers;
(3) Removal of the waterproof covering from the emergency repositories;
(4) Pumping of liquid radioactive waste from the emergency repositories;

(5) Extraction of solid radioactive waste;
(6) Sorting and loading of solid radioactive waste into containers;
(7) Layered cementing of solid radioactive waste;
(8) Transfer of containers to the central disposal site.

The Complex Radioactive Waste Management scientific centre (Zhovty Vody) developed a preliminary safety analysis report entitled Design for Minimization of the Radiation Accident Impact at Radioactive Waste Repositories 5, 6, 7 of Kiev RADON. This report analysed radiation conditions pertaining to the disposal of on-site short lived radioactive waste and long term storage of long lived radioactive waste to be retrieved from Kiev RADON.

The Science and Operational Issuing Enterprise STRUM has developed a mobile remote system for solid radioactive and hazardous toxic waste treatment and for radioactive waste unloading from the emergency disposal facilities and its reloading into lead lined storage and transport containers.

Mobile remote system management is carried out from a remote control installation located 500 m from the complex using a television system for observation. The mobile remote system was tested at Kiev RADON. Extraction of the waste and its delivery to the central disposal site of the exclusion zone was to have been realized after its commissioning.

Using hands-on experience with radioactive waste redisposal carried out in 2000, the Makariv military radioactive waste disposal facility can be addressed. The radioactive waste was retrieved from the disposal facility at Makariv in compliance with the design approved by the State Nuclear Regulatory Committee. Wastes with a volume of 6.6 m^3 with an activity of 2.61×10^9 Bq were removed, put into a special container and placed in the Kiev RADON for temporary storage.

Summary

The comprehensive programme on radioactive waste management approved by a resolution of the Cabinet of Ministers in 2002 includes:

(a) Development of a feasibility study and designs for temporary storage of radioactive waste in containers (2002–2005);
(b) Reconstruction of the UkrSA RADON monitoring system (2002–2005);
(c) Development of a mobile remote system for unloading solid radioactive waste from the emergency disposal facilities and its putting it into transport containers (2002–2005);
(d) Commissioning of the Vector complex facility in 2004;

(e) Extraction of radioactive waste from emergency repositories 5–7 of Kiev RADON and its redisposal at the central disposal site (2002–2005);
(f) Retrieval of radioactive waste from disposal trenches at Kiev RADON and redisposal in 2005.

A.15. UNITED STATES OF AMERICA: CORRECTIVE ACTIONS UNDERTAKEN AT NEAR SURFACE DISPOSAL FACILITIES

A.15.1. Introduction

Near surface disposal of commercial low and intermediate level radioactive waste generated by industry, nuclear power plants, and academic and medical facilities has taken place in the USA since the early 1960s. This experience includes repositories developed in both humid, high precipitation regions (Barnwell, South Carolina; West Valley, New York; Maxey Flats, Kentucky; and Sheffield, Illinois) and arid, low precipitation regions (Richland, Washington; Beatty, Nevada; and Tooele, Utah). This discussion does not address near surface disposal at US Department of Energy facilities which handle LILW from the research, development and production of nuclear weapons. The facilities in New York, Illinois and Nevada are no longer in operation. With the exception of the Utah facility, which was developed in the early 1990s for low activity wastes, each of these repositories has been upgraded in various ways. These corrective actions are summarized below for the six facilities involved.

A.15.1.1. Humid region sites

A.15.1.1.1. Barnwell, South Carolina (operating facility)

The Barnwell, South Carolina, disposal facility opened in 1971 and continues to dispose of low and intermediate level radioactive waste in a humid, coastal plain environment. The facility is located on State of South Carolina owned land leased to the private company that operates the facility. The 235 acre[4] site has sufficient capacity to continue operating for several decades at presently expected waste volumes.

The Barnwell site opened in 1982, prior to the existence of comprehensive US regulations, and has made improvements over time. Early

[4] 1 acre = 4.047×10^3 m^2.

corrective actions included requirements for solidification of liquids, improved isotopic analysis for ion exchange media, use of high integrity waste containers for intermediate level wastes, and environmental monitoring. Federal Government requirements for site suitability, waste form and classification, packaging, segregation of wastes by waste class, structural stability for class B and C level wastes, increased shipment manifest requirements, financial assurance, institutional control and other features took effect in 1982. These are set forth in the US Nuclear Regulatory Commission (NRC) 10 CFR Part 61 regulations [100]. In those regulations, class A waste, which generally decays to acceptable levels in 100 years or less, is considered to be low level waste. Class B and C wastes are considered intermediate level wastes.

In 1980, increased stability of intermediate level waste forms was required. Stability could be achieved by solidification or improved containerization. Use of high integrity containers for stabilization began in 1981. Most containers were made of HDPE. In 1989, the NRC determined that these did not meet long term structural stability requirements [101]. A cylindrical concrete disposal vault was adopted to allow continued use of HDPE containers. In 1993 a rectangular concrete vault was required for irradiated non-fuel bearing reactor components.

South Carolina regulations changed in 1996, requiring all wastes to be disposed of in engineered barriers. Two previous vault designs were continued and a third rectangular design was added, primarily for disposal of class A wastes packaged in drums and metal boxes. Large decommissioning project components such as nuclear power plant steam generators and reactor pressure vessels were determined to be structurally equivalent to the concrete disposal vaults. Additional requirements included enhanced caps on all disposal trenches and improved infiltration monitoring in class A trenches.

Three types of trenches are used at the Barnwell site. Large, open, class A trenches are used for low dose rate wastes. Narrow, steep walled class B and C trenches are used for disposal of intermediate dose rate containers. Long, narrow trenches are used for the disposal of irradiated hardware with high dose rates to minimize worker exposure.

Tritium was detected during 1978 in a monitoring well outside the disposal trenches. As a result of this observation, the site's operating organization increased the level of environmental characterization and monitoring down-gradient of the disposal trenches. In 1991, tritium was found in a groundwater monitoring well on facility property. An enhanced trench cap using a bentonite clay mat, an HDPE geomembrane liner and soil drainage layers was constructed over the older trenches. To date about 80 acres of the approximately 100 acres of disposal trenches have been capped in this manner to minimize infiltration and resulting leachate production. These measures have led to adequate control of tritium concentrations outside the disposal trenches. State controlled funds collected during operations are set aside for closure and post-closure monitoring and maintenance.

A.15.1.1.2. Maxey Flats, Kentucky (closed facility)

The Maxey Flats, Kentucky, disposal facility opened in 1964. The facility stopped accepting waste in 1978. During the operational phase, filled disposal units were capped with a layer of soil. Waste package degradation and related trench cap subsidence resulted in substantial water infiltration, inundation of wastes and transport of radionuclides out of disposal trenches. Relatively high annual precipitation and low permeability site soils contributed to a 'bathtub effect' which resulted in large amounts of contaminated water in the trenches.

After extensive studies and experimentation, a final corrective action programme was implemented by the Federal Government, which had shipped the majority of waste disposed of at the facility. The work, completed and found to be satisfactory in 2003, consisted of the following major elements:

(a) An additional 620 acres of adjacent land was purchased and fenced as a buffer zone surrounding the 280 acre disposal facility.
(b) Leachate was pumped out of the disposal trenches, mixed on-site with cement and other additives and poured into concrete bunkers at the site, where it solidified.
(c) New disposal trench caps consisting of clay and a synthetic liner were installed to minimize infiltration and leachate generation. Sumps with data loggers were installed to monitor water levels in the capped trenches.
(d) New drainage channels with auto-samplers and flow meters were installed around the perimeter of the facility.

The State of Kentucky undertakes long term institutional control activities.

A.15.1.1.3. Sheffield, Illinois (closed facility)

The Sheffield, Illinois, disposal facility opened in 1966. A separate, hazardous chemical waste site was later opened there. The radioactive waste repository stopped accepting waste in 1978. During site closure planning, tritium was detected migrating through extensive sand layers beneath the disposal trenches. Carbon-14 was also detected migrating from the disposal units at a slower rate. Trench cap subsidence, relatively high precipitation rates and shallow groundwater contributed to off-site migration of radionuclides.

Tritium and ^{14}C were found to be discharging into a small surface water body near the site. These conditions required a series of studies and subsequent corrective actions to properly close and stabilize the site. This work, completed in 2001, consisted of the following:

(a) An additional 170 acres of adjacent land was purchased and fenced as part of an expanded exclusion zone around the 20 acre site. The exclusion zone includes the small surface water body which serves to sufficiently dilute and impound the migrating tritium.
(b) A low permeability clay cap was installed over all trenches in 1989 to minimize water infiltration and related leachate production and subsurface transport of radionuclides. The cap was covered with soil and revegetated to control erosion.
(c) Groundwater monitoring wells were installed to verify the adequacy of these corrective actions.

Tritium concentrations have decreased steadily since installation of the new trench cap, and are generally now at natural background level or at the point of regulatory compliance. Post-closure monitoring and maintenance is performed by the site's operating organization under contract to the State of Illinois, which is responsible for long term institutional control. Corrective actions, primarily groundwater pumping and treatment, continue to be performed by the former operating organization at the adjacent, now closed chemical waste disposal facility.

A.15.1.1.4. West Valley, New York (closed facility)

The West Valley, New York, disposal facility opened in 1963 and stopped accepting waste in 1975. The 15 acre disposal site is located within a 200 acre area. The larger area was primarily devoted to spent fuel reprocessing and high level liquid waste storage and vitrification.

During the early 1990s, a bentonite clay and natural soil slurry wall and geomembrane enhanced cover were placed over 2 of the 14 disposal trenches to minimize infiltration and water accumulation. This geomembrane cover was extended over the entire site by 1997. These corrective actions have been successful in controlling leachate accumulation. Environmental monitoring and maintenance continues. Final closure and long term arrangements are subject to additional studies by the State of New York.

A.15.1.2. Arid region sites

A.15.1.2.1. Hanford, Washington (operating facility)

The Hanford, Washington, disposal facility opened in 1965 and continues to dispose of low and intermediate level radioactive waste in an arid, desert environment. This repository is located on State leased land within the US Department of Energy's Hanford Reservation, a restricted area formerly used by the national Government for nuclear weapons research and development, that contains 2 additional near surface repositories for weapons related wastes.

A private company operates the facility under a sublease with the State. The 100 acre commercial waste disposal site has substantial remaining capacity and is expected to operate until approximately 2056.

The Hanford repository had been accepting waste for many years when the NRC issued the 10 CFR Part 61 regulations in 1982. Chemical wastes previously disposed of along with certain radioactive materials also became subject to comprehensive but separate regulations some years after the Hanford repository opened. These new regulatory requirements, previous experience gained from operating the repository, and technological advances resulted in a series of operational enhancements over time.

The first major operational improvements were adopted in 1980 when solidification of certain ion exchange resins, segregation of wastes containing chelating agents, use of absorbents for contained liquids and environmental monitoring were required. A minimum burial depth was specified for high dose rate waste packages, transuranic waste concentrations were limited, and a site closure plan was required.

In 1982, NRC regulations introduced detailed waste form and characterization requirements, including structural stability standards for intermediate level wastes. Stability was provided through the use of NRC approved designs for high integrity containers including HDPE, reinforced concrete, steel alloy and other designs intended to maintain their physical form for 300 years.

In 1986, on-site groundwater monitoring wells were installed to supplement off-site wells. In this same timeframe, the facility stopped accepting wood containers and chemical wastes, including liquid scintillation media. Daily placement of soil backfill between waste containers was adopted to minimize trench subsidence. Packaged sealed sources and other high activity level wastes are now disposed of in modular engineered concrete barriers offering structural stability, an additional barrier to contaminant migration, and shielding to minimize worker exposure. These barriers were first introduced in 1987.

The facility stopped accepting absorbed liquids in 1999 and required all liquids to be solidified or stabilized prior to disposal. Also in the 1990s, additional groundwater monitoring wells were added and monitoring of unsaturated zone soil gases was initiated. An investigation of chemical constituents was also started. The data indicate limited unsaturated zone transport of chemical substances, but no groundwater or off-site effects. Investigation of these chemical substances is continuing to determine if corrective action is needed. The facility performance assessment was also updated using shipment manifest information for the site's early years of operation.

Construction of a multi-layer disposal cap over previously filled disposal units is expected to take place in 2005 following completion of environmental impact studies and a further updated safety assessment. This cap is intended to minimize infiltration, promote natural revegetation and enhance erosion control. Government controlled funds collected during operations are set aside to pay for final closure and post-closure monitoring and maintenance.

A.15.1.2.2. Beatty, Nevada (closed facility)

The Beatty, Nevada, disposal facility was the first commercial operation in the USA, having opened in 1962. The 36 acre low level radioactive waste facility closed at the end of 1992 for public policy reasons unrelated to facility performance. An adjacent but separate 44 acre chemical waste disposal site continues to operate on the site. The facility is located in an arid desert environment 100 miles north of Las Vegas near the proposed Yucca Mountain geological high level radioactive waste repository.

Like the Hanford and Barnwell repositories, the Beatty site opened before the existence of comprehensive US regulations and adopted improvements over time. These included requirements to solidify liquids, prohibition of scintillation media, transuranic isotope concentration limits and environmental monitoring requirements. Government mandated requirements on waste form and packaging, structural stability for intermediate level wastes, expanded environmental monitoring and other features of the comprehensive 1982 NRC regulations followed. Quality assurance requirements for waste shippers using the facility and a ban on co-disposal of chemical wastes were also instituted. Tritium migration was detected in the unsaturated zone, but the regulatory body determined that no corrective actions are required beyond continued monitoring.

The approved repository closure and stabilization activities were implemented beginning in 1993. Key features of the closure programme included the following:

(a) All structures were removed and the area was surrounded with a security fence;
(b) Drainage contours were cleared of debris and sediments and regraded;
(c) All disposal trenches were capped with an 8 ft[5] layer of on-site soils, mounded in the centre at a gentle slope and covered with a thin gravel layer to prevent erosion and promote natural revegetation.

Post-closure monitoring and maintenance is performed by the site's operating organization under contract to the State of Nevada regulatory body responsible for long term institutional control. State controlled funds collected during operations pay for these activities.

---

[5] 1 ft = 3.048×10^{-1} m.

REFERENCES

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

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Near Surface Disposal of Radioactive Waste, IAEA Safety Standards Series No. WS-R-1, IAEA, Vienna (1999).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, INFCIRC/546, IAEA, Vienna (1997).

[4] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTNERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety, IAEA Safety Standards Series No. GS-R-1, IAEA, Vienna (2000).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Siting of Near Surface Disposal Facilities, Safety Series No. 111-G-3.1, IAEA, Vienna (1994).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulatory Control of Radioactive Discharges to the Environment, IAEA Safety Standards Series No. WS-G-2.3, IAEA, Vienna (2000).

[8] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Radiation Protection Principles for the Disposal of Solid Radioactive Waste, Publication 46, Pergamon Press, Oxford (1985).

[9] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Radiological Protection Policy for the Disposal of Radioactive Waste, Publication 77, Pergamon Press, Oxford (1997).

[10] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Protection of the Public in Situations of Prolonged Radiation Exposure, Publication 82, Pergamon Press, Oxford (1999).

[11] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste, Publication 81, Pergamon Press, Oxford (2000).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Shallow Ground Disposal of Radioactive Wastes, Safety Series No. 53, IAEA, Vienna (1981).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Experience in Shallow Ground Disposal of Radioactive Wastes, Technical Reports Series No. 253, IAEA, Vienna (1985).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Investigations for Repositories for Solid Radioactive Wastes in Shallow Ground, Technical Reports Series No. 216, IAEA, Vienna (1982).
[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Radioactive Waste Management: An IAEA Source Book, IAEA, Vienna (1992).
[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of Available Options for Low Level Radioactive Waste Disposal, IAEA-TECDOC-661, IAEA, Vienna (1992).
[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Report on Radioactive Waste Disposal, Technical Reports Series No. 349, IAEA, Vienna (1993).
[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Radioactive Waste Management Practices and Issues in Developing Countries, IAEA-TECDOC-851, IAEA, Vienna (1996).
[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Requirements for the Safe Management of Radioactive Waste, IAEA-TECDOC-853, IAEA, Vienna (1995).
[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Planning and Operation of Low Level Waste Disposal Facilities (Proc. Int. Conf. Vienna, 1996), IAEA, Vienna (1997).
[21] OECD NUCLEAR ENERGY AGENCY, Low-level Radioactive Waste Repositories: An Analysis of Costs, OECD, Paris (1999).
[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Considerations in the Disposal of Disused Sealed Radioactive Sources in Borehole Facilities, IAEA-TECDOC-1368, IAEA, Vienna (2003).
[23] EUROPEAN COMMISSION, Management and Disposal of Disused Sealed Radioactive Sources in the European Union, Rep. EUR 18186 EN, EC, Luxembourg (2000).
[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Radioactive Waste Management (Proc. Int. Conf. Córdoba, Spain, 2000), IAEA, Vienna (2000).
[25] MATERIALS RESEARCH SOCIETY, Scientific Basis for Nuclear Waste Management XXIII (Proc. Symp. Boston, MA, 1999), MRS, Warrendale, PA (2000).
[26] HAN, K.W., HEINONEN, J., BONNE, A., Radioactive waste disposal: Global experience and challenges, IAEA Bull. **39** 1 (1997) 33–41.
[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Classification of Radioactive Waste, Safety Series No. 111-G-1.1, IAEA, Vienna (1994).
[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Acceptance Criteria for Disposal of Radioactive Wastes in Shallow Ground and Rock Cavities, Safety Series No. 71, IAEA, Vienna (1985).
[29] OECD NUCLEAR ENERGY AGENCY, Shallow Land Disposal of Radioactive Waste: Reference Levels for the Acceptance of Long Lived Radionuclides, OECD, Paris (1987).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Derivation of Activity Limits for the Disposal of Radioactive Waste in Near Surface Disposal Facilities, IAEA-TECDOC-1380, IAEA, Vienna (2003).
[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Scientific and Technical Basis for the Near Surface Disposal of Low and Intermediate Level Waste, Technical Reports Series No. 412, IAEA, Vienna (2002).
[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Performance of Engineered Barrier Materials in Near Surface Disposal Facilities for Radioactive Waste, IAEA-TECDOC-1255, IAEA, Vienna (2001).
[33] US DEPARTMENT OF ENERGY, Low-level Waste Management Handbook Series: Corrective Measure Technology for Shallow Land Burial, Rep. DOE/LLW-13Te, USDOE, Washington, DC (1989).
[34] ATOMIC ENERGY OF CANADA Ltd, Preliminary Safety Analysis Report (PSAR) for the Intrusion Resistant Underground Structure (IRUS), AECL-MISC-295, AECL, Mississauga, Ontario (1996).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Investigations, Design, Construction, Operation, Shutdown and Surveillance of Repositories for Low- and Intermediate-level Radioactive Wastes in Rock Cavities, Safety Series No. 62, IAEA, Vienna (1984).
[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Surveillance and Monitoring of Near Surface Disposal Facilities for Radioactive Waste, Safety Reports Series No. 35, IAEA, Vienna (2004).
[37] DAYAL, R., MOGHISSI, A.A., Cementitious materials in radioactive waste management, Waste Manage. **12** 2/3 (1992).
[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Characterization of Radioactive Waste Forms and Packages, Technical Reports Series No. 383, IAEA, Vienna (1997).
[39] INTERNATIONAL ATOMIC ENERGY AGENCY, Containers for Packaging of Solid Low and Intermediate Level Radioactive Wastes, Technical Reports Series No. 355, IAEA, Vienna (1993).
[40] INTERNATIONAL ATOMIC ENERGY AGENCY, Considerations in the Development of Near Surface Repositories for Radioactive Waste, Technical Reports Series No. 417, IAEA, Vienna (2003).
[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Requirements and Methods for Low and Intermediate Level Waste Package Acceptability, IAEA-TECDOC-864, IAEA, Vienna (1996).
[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Inspection and Verification of Waste Packages for Near Surface Disposal, IAEA-TECDOC-1129, IAEA, Vienna (1999).
[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of the Factors Affecting the Selection and Implementation of Waste Management Technologies, IAEA-TECDOC-1096, IAEA, Vienna (1999).

[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Design, Construction, Operation, Shutdown and Surveillance of Repositories for Solid Radioactive Wastes in Shallow Ground, Safety Series No. 63, IAEA, Vienna (1984).

[45] INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures and Techniques for Closure of Near Surface Disposal Facilities for Radioactive Waste, IAEA-TECDOC-1260, IAEA, Vienna (2001).

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

[47] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of Quality Assurance to Radioactive Waste Disposal Facilities, IAEA-TECDOC-895, IAEA, Vienna (1996).

[48] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment Methodologies, Coordinated Research Project on Improvement of Safety Assessment Methodologies for Near Surface Waste Disposal Facilities (ISAM), IAEA, Vienna (1997) http://www-ns.iaea.org/projects/isam.htm.

[49] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment Methodologies (ASAM), Coordinated Research Project on Application of Safety Assessment Methodologies for Near-surface Radioactive Waste Disposal Facilities (ASAM), IAEA, Vienna (2002) http://www-ns.iaea.org/projects/asam.htm.

[50] CHEM-NUCLEAR SYSTEMS, Barnwell Waste Management Facility Site Disposal Criteria, S-20-AD-010, Chem-Nuclear Systems, Columbia, SC (2001).

[51] URSULEAN, I., BALABAN, V., "Operational safety of near surface waste disposal facilities in the Republic of Moldova", Safety of Radioactive Waste Management (Proc. Conf. Córdoba, Spain, 2000), IAEA, Vienna (2000) (CD-ROM).

[52] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, Dounreay Site Restoration Plan, UKAEA, Dounreay (2001).

[53] BACKE, S., CHRISTENSEN, G.C., LUNDBY, J.E., BERGAN, T.D., "Retrieval of LLW and ILW from a 25 year old near surface repository", Planning and Operation of Low Level Waste Disposal Facilities (Proc. Symp. Vienna, 1996), IAEA, Vienna (1997) 477–484.

[54] INTERNATIONAL ATOMIC ENERGY AGENCY, Regional Workshop on Experiences in the Upgrading of Novi Han Repository (Proc. Workshop Borovetz, Bulgaria, 2000), IAEA, Vienna (2002) CD-ROM.

[55] UNITED KINGDOM ATOMIC ENERGY AUTHORITY, The Dounreay Waste Shaft, UKAEA, Dounreay (2001).

[56] CHARLESWORTH, D.H., CHAMP, D.R., "Transition from storage to disposal of LLW at Canada's Chalk River Laboratories", Planning and Operation of Low Level Waste Disposal Facilities (Proc. Symp. Vienna, 1996), IAEA, Vienna (1997) 79–87.

[57] POŠKAS, P., RAGAIŠIS, V., ADOMAITIS, J.E., "Dealing with existing radioactive waste storage/disposal facilities in Lithuania", Safety of Radioactive Waste Management (Proc. Conf. Córdoba, Spain, 2000), IAEA, Vienna (2000) (CD-ROM).
[58] ORMAI, P., BERCI, K., TAKATS, F., JUHASZ, L., "Long lived radiation sources and waste in near surface repository: Legacy of the past, challenge of today", in Proc. ICEM Conf. 2001, Bruges, Belgium, International Federation of Chemical, Energy, Mine and General Workers' Unions, Brussels (2001) (CD-ROM).
[59] BULYGA, P.V., RODZYALUSKAYA, L.F., IVANOV, V.B., "Experience in operational control of the national LLW disposal facility Ekores", Planning and Operation of Low Level Waste Disposal Facilities (Proc. Symp. Vienna, 1996), IAEA, Vienna (1997) 485–491.
[60] VOKÁL, A., NACHMILNER, L., HAWORTH, A., "Performance of the engineered barriers of the Dukovany shallow land repository", ibid., pp. 536–542.
[61] COYLE, A., GRIMWOOD, P.D., PAUL, W.J., "Waste acceptance policy and operational developments at the UK's Drigg LLW disposal site", ibid., pp. 339–349.
[62] DUTZER, M., NICOLAS, M., "Operating the Centre de l'Aube", ibid., pp. 243–247.
[63] POTIER, J.-M., "Final closure of a low level waste disposal facility", ibid., pp. 523–530.
[64] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 1990 Recommendations of the ICRP, Publication 60, Pergamon Press, Oxford (1991).
[65] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance of Records for Radioactive Waste Disposal, IAEA-TECDOC-1097, IAEA, Vienna (1999).
[66] OECD NUCLEAR ENERGY AGENCY, Gas Generation and Release from Radioactive Waste Repositories (Proc. Workshop Aix-en-Provence, 1991), OECD, Paris (1992).
[67] US NUCLEAR REGULATORY COMMISSION, Microbial Degradation of Low-Level Radioactive Waste, Rep. NUREG/CR-6341, NRC, Washington, DC (1996).
[68] ONTARIO POWER GENERATION, Impacts of Gas Generation in Ontario Power Generation's Low Level Wastes, Rep. 05386-REP-034693-0015-R00, Ontario Power Generation, Toronto (1999).
[69] EUROPEAN COMMISSION, Progress Project: Research into Gas Generation and Migration in Radioactive Waste Repository Systems. Final Report, Rep. EUR 19133 EN, EC, Luxembourg (2000).
[70] INTERNATIONAL ATOMIC ENERGY AGENCY, Socio-economic and other Non-radiological Impacts of the Near Surface Disposal of Radioactive Waste, IAEA-TECDOC-1308, IAEA, Vienna (2002).

[71] BYLUGA, P.V., ROZDYALOVSKAYA, L.F., IVANOV, V.B., “Experiences in operation control of the national LLW disposal facility Ekores”, Planning and Operation of Low Level Waste Disposal Facilities (Proc. Symp. Vienna, 1996), IAEA, Vienna (1997).
[72] ARUSTAMOV, A.E., et al., “Metal matrices for the immobilization of highly-radioactive spent Sealed radiation sources”, WM ’98 (Proc. Conf. Tucson, AZ, 1998), WM Symposia, Inc., Tucson, AZ (1996).
[73] SHIRYAEVA, N.M., SKURAT, V.V., MYSHKINA, N.K., “The state safety assessment of the low level waste repositories in the territory of Belarus”, WM ’2000 (Proc. Conf. Tucson, 2000), WM Symposia, Inc., Tucson, AZ (2000).
[74] ROZDYALOVSKAYA, L.F., IVANOV, V.B., “Management of spent sealed sources: Belarus experience”, WM ’01 (Proc. Conf. Tucson, AZ, 2001), WM Symposia, Inc., Tucson, AZ (1996).
[75] COMMITTEE ON USE OF ATOMIC ENERGY FOR PEACEFUL PURPOSES, Collecting, Storage, Processing, Keeping, Transport and Disposal of Radioactive Waste on the Territory of the Republic of Bulgaria, Regulation No. 7, CUAEPP, Sofia (1992).
[76] STEFANOVA, I.G., Characterization of Radioactive Waste and Identification of Radionuclide Inventory, Rep. BUL/4/005, Institute for Nuclear Research and Nuclear Energy, Sofia (1999).
[77] EVSTATIEV, D.P., PETROV, R., ANGELOVA, D., KARASTANEV, D., “Geoenvironmental assessment of radioactive waste repository”, in Proc. 7th IAEG Congress, Lisbon, Portugal, 1994, Balkema, Rotterdam (1994).
[78] MATEEVA, M., Preliminary Safety Assessment of the Novi Han Repository, Sofia (2001).
[79] MATEEVA, M., KOZAK, M., Safety Assessment of the Radioactive Waste Repository — Novi Han, Sofia (2001).
[80] MISHEV, P.L., CHRISTOSKOVA, M.Z., Development of a System for Environmental Monitoring and Control, Rep. BUL/4/005, Institute for Nuclear Research and Nuclear Energy, Sofia (1999).
[81] STEFANOVA, I.G., Selection of Treatment and Conditioning Processes, Rep. BUL/4/005, Institute for Nuclear Research and Nuclear Energy, Sofia (1999).
[82] STEFANOVA, I.G., Conceptual Design of LILW Processing and Storage Facility, Rep. BUL/4/005, Institute for Nuclear Research and Nuclear Energy, Sofia (1999).
[83] AGENCE NATIONALE POUR LA GESTION DES DÉCHETS RADIOACTIFS, Post-Closure Preliminary Safety Report of Centre de la Manche, Rep. SUR RP ADCS 98-002, ANDRA, Châtenay-Malabry (1998).
[84] EUROPEAN COMMISSION, Safety Analysis of the Püspökszilágy Radioactive Waste Treatment and Disposal Facility: An Assessment of Post-closure Safety, Rep. PHARE 990167, PH4.12/95(01)N2, EC, Luxembourg (2001).
[85] EDV-EROTERV, Safety Analysis of the Püspökszilágy Radioactive Waste Treatment and Disposal Facility (2002) (in Hungarian).

[86] ORMAI, P., TAKÁTS, F., “Hungarian near surface repository development: Is intervention necessary?”, Management of Radioactive Wastes from Non-power Applications — Sharing the Experience (Proc. Conf. Malta 2001), C&S Papers Series 15/CD, IAEA, Vienna (2002) (CD-ROM).
[87] ORMAI, P., “Safety upgrading of the Püspökszilágy disposal facility”, Issues and Trends in Radioactive Waste Management (Proc. Conf. Vienna, 2002), IAEA, Vienna (2003).
[88] SERCO ASSURANCE, RWE NUKEM, PHARE Project proposal (2003).
[89] SVENSK KÄRNBRÄNSLEHANTERING AB, Assessment of the Long-term Safety of the Existing Storage Facility for Radioactive Waste at Maišiagala, SKB, Stockholm (1998).
[90] BALTRUNAS, D., BUTKUS, D., KALINAUSKAS, R., PLUKIS, A., REMEIKIS, V., “Safety assessment of the RADON type radwaste facility at Maišiagala, Lithuania”, Safety of Radioactive Waste Management (Proc. Int. Conf. Córdoba, Spain, 2000), IAEA, Vienna (2000).
[91] SØRLIE, A.A., Strålevern Report 2003:15, Norwegian National Report, Joint Convention, Strålevernet, Østerås (2003) www.nrpa.no.
[92] LUNDBY, J.E., Burying of Drums with Radioactive Waste, Rep. IFA/CH-91, Institutt for Atomenergi, Kjeller, Norway (1971) (in Norwegian).
[93] KVESETH, K., Repository for Low and Intermediate Level Waste, Rep. NOU 1991:9, Olje- og Elnergidepartementet, Oslo (1991) (in Norwegian).
[94] BACKE, S., et al., “Retrieval of a near surface repository at IFE-Kjeller”, Radiological Protection in the 2000s, Theory and Practice (Proc. 13th Ordinary Mtg Turku, Finland, 2002), Nordic Society for Radiation Protection, Åbo, Finland (2002).
[95] SØRLIE, A.A., “The Combined disposal and storage facility for LLW and ILW in Himdalen in Norway”, WM ‘01 (Proc. Conf. Tucson, AZ, 2001), WM Symposia, Inc., Tucson, AZ (1995).
[96] SOBOLEV, I.A., et al., “Experience in the disposal of L/ILW in the Russian Federation”, Planning and Operation of Low Level Waste Disposal Facilities (Proc. Symp. Vienna, 1996), IAEA, Vienna (1996) 297–304.
[97] BARINOV, A.S., OJOVAN, M.I., SOBOLEV, I.A., OJOVAN, N.V., Sov. Radiochem. **32** 4 (1990) 417–421.
[98] KASCHEEV, V.A., et al., Sov. At. Energy **69** 1 (1990) 572–576.
[99] OJOVAN, M.I., et al., Nucl. Waste Manage. Environ. Remed. Prague **1** (1993) 155–157.
[100] US NUCLEAR REGULATORY COMMISSION, Title 10, Code of Federal Regulations, Part. 61, NRC, Washington, DC (1982).
[101] US DEPARTMENT OF ENERGY, Low-level Waste Management Handbook Series: Corrective Measures Technology for Shallow Land Burial, Rep. DOE/LLW-13Te, USDOE, Washington, DC (1989).

CONTRIBUTORS TO DRAFTING AND REVIEW

Beyleveld, C.	South African Nuclear Energy Corporation, South Africa
Bohdan, L.	Nuclear Power Plant Operational Support Institute, Ukraine
Champ, D.	AECL Chalk River Laboratories, Canada
Dayal, R.	International Atomic Energy Agency
Faltejsek, J.	Radioactive Waste Repository Authority, Czech Republic
Gera, F.	Consultant, Italy
Green, T.	RWE Nukem Ltd, United Kingdom
Grimwood, P.	British Nuclear Fuels Ltd, United Kingdom
Guskov, A.	SIA RADON, Russian Federation
Lange, B.	AECL, Chalk River Laboratories, Canada
Ormai, P.	Public Agency for Radioactive Waste Management, Hungary
Romano, S.	American Ecology Corporation, United States of America
Rozdyalouskaya, L.	Ministry for Emergencies, Belarus
Salmins, A.	Radiation Safety Centre, Latvia
Sörlie, A.	NRPA, Norway
Stefanova, I.	Institute for Nuclear Research and Nuclear Energy, Bulgaria
Vinskas, A.	State Nuclear Safety Inspectorate, Lithuania
Voizard, P.	Agence Nationale pour la Gestion des Déchets Radioactifs, France
Webster, S.	European Union

Consultants Meetings

Vienna, Austria: 30 September – 4 October 2002; 10–14 May 2004

Technical Committee Meeting

Budapest, Hungary: 25–29 August 2003

Providing guidance on the disposal of radioactive waste constitutes an important and integral component of the IAEA programme on radioactive waste management. Low and intermediate level waste, even though it contains a small fraction of the total activity of all radioactive waste produced globally, represents more than 90% of the total volume of radioactive waste. Most of the radioactive waste produced in many developing Member States is primarily low and intermediate level waste. A number of activities have been initiated by the IAEA to assist Member States in the disposal of low and intermediate level waste, focusing on both technology and safety aspects. Many existing disposal facilities were developed and began operations long before current regulatory requirements took effect or more recent site suitability guidance, technological advances, safety assessment methodologies and quality assurance systems became available. National laws, regulations and disposal methods have evolved and improved with time. Various Member States have ongoing programmes both to upgrade these facilities and/or develop new near surface disposal facilities. Upgrading measures are being implemented or planned at a number of disposal facilities in numerous countries, and are described in this publication.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 92–0–112704–9
ISSN 0074–1914