Operational Issues In Radioactive Waste Management and Nuclear Decommissioning

Disposal of Radioactive Waste

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Content of the lecture

CRITERIA AND PRINCIPLES FOR DISPOSAL

NEAR SURFACE REPOSITORIES

GEOLOGICAL REPOSITORIES

SITE SELECTION AND PUBLIC ACCEPTANCE
Radioactive Waste

Radioactive waste arises from the generation of electricity in nuclear power plants, from nuclear fuel cycle operations and from other activities in which nuclear technology is applied (agriculture, education, medicine, industry).

The final step of the management is the disposal, which means the emplacement of the material, adequately conditioned, in a repository providing long term isolation of the radionuclides from the biosphere.

Disposal means that there is no intention to retrieve the waste, although such a possibility it is in principle not excluded.
Radioactive Waste Classification

There are several operational classifications of RW worldwide: for the purpose of the disposal they can be classified in two broad categories: short and long lived waste.

A nuclear waste is classified as short lived when it contains generally low concentrations of predominantly short-lived radioactive isotopes. Short-lived radioisotopes are considered those with half-lives up to around thirty years - typically Sr-90 and Cs-137. This implies that they decay to a very low level in few centuries.

(Their radioactivity level after three hundred years, ten times the half-life of the predominant radionuclides they contain decreases about one thousand times.)
Short lived waste

If the initial concentration of radionuclides in the waste is below a given limit (usually the limit of activity concentration set up for **LLW**), a thousand fold decrease leads to an insignificant level of radioactivity (close or even below to natural background).

This is why a time period of **some centuries** is the timescale over which low-level waste emplaced in a disposal system has to be segregated from the environment (three hundred year period has been historically considered the reference timescale).

Short lived waste accounts for about 95% of the radioactive material produced in the applications of nuclear energy in terms of volume or weight. Most of them are produced in the NPPs operation.
Long lived waste

Waste with significant content of long-lived and/or highly concentrated short-lived radionuclides, is a long lived waste.

For this waste a significant decrease of the initial radioactivity or its reduction to a radiation level in the waste considered not harmful in case of exposure is achieved after a period of time going from thousands to hundreds of thousands or even milions of years.

Long lived waste contain about 98% of the radioactivity generated by the applications of the nuclear technology
Type of Short Lived Waste

Short lived radioactive waste typically includes:

- Some waste from reprocessing spent fuel (LLW and ILW-SL);
- Waste from operating and decommissioning of NPP facilities (LLW and ILW-SL);
- Waste from front–end fuel cycle facilities (LLW);
- Waste from nuclear applications in medicine, research and industry.
Type of Long lived waste

Long lived radioactive waste typically includes:

• *spent fuel (SF) from nuclear reactors when not considered for fissile recovery;*

• *waste from reprocessing spent fuel (HLW);*

• *some waste from operating and decommissioning of NPP and fuel cycle facilities (LL-ILW);*

• *waste from production of MOX fuel and nuclear weapons;*

• *some of the radiation sources used in medicine, research and industry.*
Radioactive Waste Disposal

The disposal of radioactive waste and its safety are governed by two basic principles:

• The waste package (the conditioned waste and its outer container) must be placed in a repository in such way that the harmful materials they contain are prevented from coming into direct or indirect contact with the biosphere (isolation and confinement principle);

• This contact has to be prevented as long as the waste continues to be harmful and thus having health impact to individuals if released (period of the isolation principle).
Isolation and confinement

Isolation from the biosphere is provided by placing several containment barriers between the waste materials and the external environment, arranged in succession so that each barrier reinforces the preceding one (multibarrier system).

The barriers function it is basically to avoid release of radioactive isotopes from the repository in any foreseeable circumstance, both normal or accidental.

Since the only common natural agent able to mobilise and transport radioactive material is water, by a dissolution process or by physical entrainment, the actual function of the barriers is to prevent water flow in the waste material, or anyhow to avoid any release from a repository should this contact take place.
The Multi-Barrier System

The first of the barriers is the waste package itself, comprising the conditioning solid waste matrix and the container, whose materials are selected just for ensuring the segregation and immobilization of the radioactive material and the required degree of containment.

The other containment barriers are provided by the repository structures and they can be either man-made (along with waste form they are the so called Engineered Barrier System, EBS) or natural, or a combination of both.

The safety of the disposal is based on the effectiveness and durability of these additional barriers, whose nature depends on how long the containment has to be ensured and effective.
The first barrier: the Waste Package

cemented radioactive waste  LILW

vitrified canister of HLW
The other barriers

The barriers provided by the repository are directed to insure isolation and confinement for the time period needed to allow the radioactive waste to decrease to a level considered not harmful in case of release to the biosphere.

This requirement makes the fundamental difference between the kind of barriers needed for repositories for short and for long lived waste.

In the two cases in fact the barriers have to fulfil a very different duty.
Barriers and time scale

- For short lived waste the isolation must be assured at most for a few centuries (three hundred years lead to a thousand fold reduction of radiation level). This is a period of time during which it is possible to ensure performance and duration of engineered barriers (man made), provided they are adequately designed and constructed.

- For disposal of long lived waste barriers must be provided to ensure isolation for a much longer period of time, tens or hundreds of thousands of years and more. Since no engineered system can be guaranteed for such period, the containment function must be finally provided by a natural barrier, stable over geological time periods.
Disposal of Short Lived Waste (LLW)

A repository for short lived waste must fully rely upon a real EBS for isolation and containment of the radioactive materials for the required period.

Concrete is usually used for this type of barriers, either as construction material and as backfilling. Its mechanical, hydraulic and chemical properties make it an ideal material to guarantee defence not only against massive water infiltration but also against mechanical impacts or penetration.

The longevity of its properties – in this case for at least three hundred years – is obviously the main requirement that both the material and the whole structure have to meet.
Barriers in a LLW repository (conceptual)
Barriers selection and qualification

The main effort in devising, designing and constructing a repository for low radioactive waste is a technical activity that goes under the name of *qualification* of the barriers.

*Qualification* is the whole activity dealing the study of their properties, including evaluation of their long term behaviour and experimental tests carried out to confirm them.

All these actions are aimed at ensuring that barriers will actually be capable of maintaining the waste completely isolated during the required period (in this case for at least three hundred years) and limiting any releases thereafter.
Concrete selection and evaluation

Fundamental activity within the barrier qualification is the mix design of the concrete, which include the selection of cement, inert fillers and chemical additives able to insure the required performance, particularly in terms of durability over the long term.

Estimation of the long-term behaviour is made by severe evaluation of the material properties, and also by accelerated tests, whereby the material undergoes extreme tests directed to simulate the effect of stress that, in reality, are spread over considerable periods of time (for instance, immersion of specimens in water for months simulates the effect of century-long infiltration of rain water or humidity from soil).
Near Surface Repositories

A reliable and safe containment and isolation of short lived radioactive waste from the biosphere, for the required time, can thus be achieved without major technical problems.

Disposal of this waste is carried out in many countries. The repository structures in most cases are built on the surface. They are generally called near-surface repositories, and can be built below or above ground.

Underground repositories (in artificial caverns) have been constructed in Sweden and Finland. They belong to the same kind, i.e. isolation essentially relies on man-made barriers.

In its more general configuration, a near-surface repository is a succession of reinforced concrete cells in which waste packages are emplaced and immobilized.
Centre de l’Aube, France
Centre de l’Aube, France
Repository of El Cabril, Spain
Emplacement of containers at El Cabril
Artistic view of swedish repository (Forsmark)
Aerial photo of the Forsmark site
Handling of packages at Forsmark repository
Special case: Simplified Disposal of VLLW

A simplified barriers system made of various backfillings and natural occurring materials can be used for waste containing very low concentration of short lived radionuclides (the Very Low Level Waste, VLLW).

This waste requires a moderate level of containment and isolation and is suitable for disposal in a landfill type facility with limited regulatory control.

Typical waste in VLLW class includes soil and rubble with low activity concentration levels. It consists mainly of demolished material (such as concrete, plaster, bricks, metal, valves, piping etc) produced during rehabilitation or dismantling operations on nuclear industrial sites.
Disposal of VLLW

Disposal of VLLW in USA

The VLLW Disposal in France
Phases of activity of a SLW Repository

The lifetime of a repository for short lived waste involves:

- **An operational period.** Includes waste packages handling and emplacement in the disposal units. Ends when all the disposal units are filled.

- **Closure.** The operation of closing, backfilling, and sealing the disposal units as devised by the EBS design. The closure phase is usually covered by a purposely granted license.

- **Post-closure-Institutional control.** After closure the site is subject to institutional controls. It involves limiting access to the site and some level of maintenance. The duration of this institutional period depends on the established time for waste isolation and decay. At the end of this period the site can be released from control.
A closed repository for LLW

Centre de La Manche
(France)
Long lived Waste Disposal Concept

In the final repository for long-lived waste the waste material must be isolated from the biosphere for tens or hundreds of thousands of years.

This function can only be provided in a geological disposal system, in which the depth of formation and the characteristics of the host geological environment provide for isolation of the waste for the required period of time.

The safety of the repository is again based on the multibARRIER concept, whereby barriers are interposed between the waste material and the environment.
Barriers in a geological disposal

There are two principal components in the multibarrier system:

• the engineered barriers, which comprises repository structures, various containers and possible backfills used to immobilize the waste inside the repository excavations;

• the natural barrier, which is basically the host bedrock and the associated groundwater system isolating the repository and the EBS from the biosphere.

The fundamental containment function for the long period is provided by the natural barrier, which has to be stable over geological time periods.
Geological Bedrock Barriers

The requirements for long time containment are met by some sedimentary bedrocks, in particular salt formations and clay basins. Also suitable are particular types of crystalline rocks, such as non-fractured granites.

Salt formations are an almost ideal medium: formed during long evaporation process of former oceans, their very existence testifies that water has kept away from them for geological ages. Besides, the salt rock is rather plastic, therefore capable of accommodate possible tectonic stresses at its boundaries without mechanical failure.

Clay formations have equally favourable characteristics; in addition, they behave as a geochemical barrier, so providing a sorbing medium to radionuclides.
EBS in a Geological Repository

If bedrock is the key barrier to rely on for permanent isolation of waste, the EBS system also plays an important protective role, in particular to contain the highly concentrated short lived radionuclides (and the associated heat output) until they decay to insignificant level. This period is in the order of a thousand years.

In some geological systems greater emphasis is given on the role of the EBS even for more extended periods of time (Sweden and Finland, Yucca Mountain). This depends essentially on the actual geological environment, and on the national waste inventory.

Some metallic materials under consideration as outer containers for spent fuel (titanium alloys, copper), would be able of ensuring containment function for up to several hundred thousands years.
Barriers characterization in a geological repository

A key issue for demonstrating safety in a geological disposal is the characterization of the candidate host rock.

This characterization is carried out by means of underground research laboratories (URL) mined in the investigated bedrock formation. The focus of research activities includes:

- interface EBS/host rock and long term stability;
- transport and retention of radionuclides in real configuration;
- groundwater flow regime;
- techniques for tunnel excavation and waste emplacement;
- verification of data and models used in safety analysis.

Several URL are being operated worldwide.
The experimental salt mine of Gorleben (Germany)
Underground laboratory of Grimsel (CH)  
*granite formation*
Underground laboratory in clay formation (Mol, Belgium)
The tunnel of Yucca Mountain, USA
Scheme of the underground laboratory in Finland

granite formation

1. Access tunnel
2. Ventilation raise
3. Characterisation tunnel
   Main investigation level (~400 m)
4. Lower investigation level (~500 m)
Access tunnel to the underground laboratory in Finland
The unique geological repository in operation

There is only one operating (non commercial) geological repository in the world: **The Waste Isolation Pilot Plat (WIPP)**, operated by the USDOE.

WIPP is located in New Mexico. Is in a 600-meter deep salt basin, formed during the Permian Era, approximately 250 million years ago, by evaporation of an ancient sea once covering the area. It is now covered by a 300 meters impermeable layer of soil and rock.

WIPP is licensed to permanently dispose of so called TRU (Transuranic Waste), i.e. radioactive waste coming from the military productions of the Department of Energy of USA. They are long lived waste, being rather rich in plutonium.
WIPP: an operating geological repository
Safety Assessment of a Disposal System

Before construction of any repository, a comprehensive and systematic assessment aimed at demonstrating the safety of the planned facility throughout its operating lifetime and in the required post-closure period (either centuries or geological times) must be carried out.

The safety assessment study (the safety case) is prepared by the operator and reviewed by the regulatory body.

Operation of the repository is authorized only if the operator demonstrates with reasonable assurance that the safety criteria and technical requirements (established by the regulatory body) have been met.
Safety Assessment of a Disposal System (cont’d)

The safety case includes:

- an evaluation of system performance for all the situation selected;
- an overall assessment of compliance with safety requirements;
- analysis of the associates uncertainties.

In the safety case the evaluation of how the repository evolves and the performance of its components are carried out using the technique of Performance Assessment (PA).
The Performance Assessment (PA)

In the PA experimental data and scientific information are compiled and processed to construct models able to describe the possible future behaviour of the disposal system, handling time-dependant processes. This allows to quantitatively analyse, basically in terms of mass transfer processes, the performance of the barrier system or of the whole system.

A key issue in this process is the identification and characterization of scenarios describing situations from different possible features, events and processes (FEPs) that can affect the safety of the repository. In all the selected scenarios the performance of the system (by calculation of dose or risk) is analysed and checked against the criteria set up by the regulatory framework.
1. Assessment context
2. Describe system
3. Develop and justify scenarios
4. Formulate and implement models
5. Run analyses
6. Interpret results
7. Compare against assessment criteria
8. Adequate safety case
9. Effective to modify assessment components
10. Review and modification

Acceptance

YES

NO

NO

Rejection
Safety Assessment for a Near Surface repository

For a near surface repository (for short lived RW) the identification and characterization of scenarios describing the evolution of the repository is extended to a time period of centuries. This makes the calculation approach more reliable.

For instance, the behaviour and performance of the concrete barriers can be evaluated on the basis of the selected material. Expected degradation (the *design degradation*) of the confinement capacity over the time period in which the isolation has to be assured (the so called *institutional period of control*) has to be taken into consideration in the calculation model.

The safety assessment has to be extended after the end of the period of control only for some scenarios (for example, intrusion scenario)
Geological disposal safety and timescales

For geological disposal, a key issue is the setting of post-closure timescales for which the performance of a repository must be analysed and compliance with regulatory criteria demonstrated.

In various geological disposal national programs requirements for compliance period of 10,000 years have been used or recommended (Canada, Germany, Finland). Others require consideration of times beyond that period (USA) or have no limits specified (Switzerland).

PA methods, models and timescales adopted may differ among national programs, since they are site- and design-specific. Their common feature is however the need to achieve an acceptable level of confidence in the reliability of the used models and data for long term evaluation. A key role on this aspect is played by the underground research laboratories.
Some long lived radionuclides of importance for disposal

**Fission and activation products**

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<tr>
<th>Radionuclide</th>
<th>Half-life (yr)</th>
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<tbody>
<tr>
<td>Carbon 14</td>
<td>5700</td>
</tr>
<tr>
<td>Chlorine-36</td>
<td>300,000</td>
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<tr>
<td>Nickel-59</td>
<td>75,000</td>
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<tr>
<td>Technetium-99</td>
<td>210,000</td>
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<tr>
<td>Iodine-129</td>
<td>16 Mio</td>
</tr>
<tr>
<td>Caesium-135</td>
<td>2.3 Mio</td>
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</table>

**Actinide and U-Th radionuclides:**

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Half-life (yr)</th>
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<tbody>
<tr>
<td>Radium 226</td>
<td>1600</td>
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<tr>
<td>Thorium-232</td>
<td>14 Mld</td>
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<tr>
<td>Uranium-235</td>
<td>700 Mio</td>
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<tr>
<td>Uranium-238</td>
<td>4.5 Mld</td>
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<tr>
<td>Plutonium-238</td>
<td>24,000</td>
</tr>
<tr>
<td>Americium-241</td>
<td>430</td>
</tr>
</tbody>
</table>
Relative radioactivity of typical SF (a Swedish BWR) after discharge

(After Hedin, 1997)
Site selection of nuclear waste disposal facilities

It is well known that the site selection of a radioactive waste repository is today one the most challenging issues in the use of nuclear power.

The public perceptions of risk and safety associated with the siting of disposal systems have led to the development of the so-called NIMBY (Not in My Back Yard) syndrome, usually resulting in a strong negative reaction in local communities.

To overcome this problem, the centralized method first employed, a Decide-Announce-Defend approach – generally referred as DAD – was abandoned in favour of a decision-making process implying various levels of public involvement, associated with financial or structural benefits linked to the repository localisation.
Decision-making process for site selection

Methodologies based in various degrees of volunteerism play nowadays an important role in ensuring the success of the selection procedure.

Methodologies encompass pure volunteerism (a community asks to be considered for the siting process: a rather rare occurrence), and the most common case of so-called assisted volunteerism, where a form of volunteer engagement is sought after a systematic technical screening identified a list of possible sites.

A technical screening of the national territory directed to identify the suitable areas is usually the propaedeutical stage of a selection process. Exclusion criteria are commonly set up to handle the initial screening process.
The exclusion criteria procedure

The procedure is directed to identify preminarily features that are considered clearly **unsuitable** for a potential site. These can be either physical (nature of the rock or soil, geology and hydrogeology, presence of natural resources, etc) or anthropic (land uses, population density, etc).

The methodology of exclusion criteria has been applied with different details in national disposal programmes. They generally implement the safety criteria set up for a disposal site by the national Regulator, and apply the safety requirements and site-selection guidelines established by the IAEA and other international Organizations.
Site selection for a Near Surface repository

Site-selection criteria for **LLW near-surface repository** principally cover the following aspects:

- Morphology and natural environment of the territory;
- Groundwater and geochemical characteristics of soil;
- Tectonic and seismicity;
- Surface processes;
- Metereology and climate;
- Impact of human activities
Site-selection criteria for a geological repository

Site-selection criteria for **geological disposal** usually cover:

- Geological stability and hydrogeology;
- Chemical and geochemical properties of the host rock;
- Mechanical and thermal properties;
- Depth and size of the host formation;
- Presence of natural resources;
- Potential future human activities.

A general agreed principle (for any repository) is nowadays that the purpose of the selection criteria is not necessarily to select the “best” site but one that complies with established safety and environmental requirements.
I thank you for your attention
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