Geologic repositories
and
waste conditioning

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August 2019
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Note

In this document the references are coded by Q-numbers (e.g. Q6). Each reference has a unique number in this coding system, which is consistently used throughout all publications by the author. In the list at the back of the document the references are sorted by Q-number. The resulting sequence is not necessarily the same order in which the references appear in the text.

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Nuclear legacy - isolation from the human environment

A nuclear power plant irreversibly generates each year an amount of human-made radioactivity equivalent to about 1000 exploded atomic bombs of about 15 kilotons (Hiroshima bomb). Each year the civil nuclear power plants of the world add some 300000 atomic bomb equivalents to the world inventory, present in spent nuclear fuel, in construction materials and in auxiliary materials. Radioactivity cannot be destroyed nor can be made harmless.

During the disasters of Chernobyl and Fukushima jointly about 0.01% of the world civil inventory of human-made radioactivity has been released into the biosphere. This corresponds with the amount generated by one nuclear power plant during one year at full power. The irreversible and harmful consequences of these disasters are still noticeable on continental scales, affecting millions of people, costing hundreds of billions of euros, and will continue for undetermined time periods into the future. Adequate settlement of the nuclear legacy is a *conditio sine qua non* to avoid dispersion of the remaining 99.99% of the nuclear legacy into the biosphere, in order to keep vast areas on the Northern Hemisphere habitable.

To fulfill this challenge all human-made radioactive materials should be isolated from the biosphere, the human environment, in the physically best way. There is just one practicable solution to achieve such a situation, that is disposal of all human-made radioactive materials, packed in appropiate containers, in deep geologic repositories. In this way return of the radionuclides by groundwater flows into the human environment may be postponed for thousands of years.
It may be important to dispose of *all* human-made radioactive materials, because in practice it proved to be impossible to guarantee the correct contents of radionuclides in each container. Measurements, if done at all, are incomplete because not all kinds of radioanuclides can be detected. How reliable are inspections? Financial and political arguments may play an important role.

What is 'nuclear waste'?

Spent nuclear fuel at a burnup of 33 GWday/Mg, contains about 94 mass% of unused uranium -238, 0.8% of unfissioned U-235, about 0.9% plutonium, 0.07% minor actinides (neptunium, americium and curium), in addition to about 3.5% fission products. Usually the nuclear industry regards spent fuel as a resource of future energy production, due to its content of uranium and fissile nuclides. The vast majority of the human-made radioactivity is confined in spent fuel. Reuse of the uranium and plutonium for energy generation requires that the fission products and minor actinides are removed from the spent fuel. This requires a complicated chemical separation sequence, called reprocessing. A thermodynamic analysis points to two prohibitive features of the proposed advanced closed-cycle reactor systems needed to generate useful energy from spent fuel:

- the proposed concepts cannot operate as designed,
- even if they would be feasible the closed-cycle reactor concepts would have a negative energy balance.

Development of a nuclear energy generating capacity equalling the present world capacity would imply a timescale of one or two centuries, ignoring the above mentoned prohibitive features. Moreover the risks of proliferation and nuclear terrorism would greatly increase when such systems would come into operation.

There is another good reason not to reprocess spent nuclear fuel: it would massively increase the acute nuclear entropy generation by releasing substantial fractions of the latent and delayed entropy, notions that are explained in report m38 *Nuclear power and the Second Law*. As is explained in report m20 *Reprocessing of spent fuel* reprocessing means a redistribution of the human-made radioactivity over large volumes of materials, in addition to substantial discharges into the biosphere. As a result health hazards and risks of large nuclear accidents greatly increase, and may become uncontrollable.

Conclusion: based on the above mentioned arguments this study regards all radioactive materials generated and/or mobilised by the nuclear energy system as unusable and consequently as nuclear waste.

Radioactive waste management

With respect to nuclear power the primary challenge of the present time and in the future is to curb the latent nuclear entropy, in order to limit the hazards posed by the human-made radioactivity. The challenge has different aspects. During its operational lifetime a nuclear power plant produces spent fuel elements, containing more than 95% of human-made radioactivity in a relatively small mass (some 700 Mg) and volume (several hundred m^3).
Simultaneously the nuclear system produces other radioactive wastes (mill tailings, operating waste and dismantling waste) containing smaller amounts of radioactivity (some 5% of the total human-made radioactivity), but distributed over enormous volumes and masses of materials, to be measured in millions of cubic meters and millions of tons.
The chances that radioactive materials get dispersed into the environment from spent fuel may seem low, but the released quantities of radioactivity will be huge, in case of an accident. Millions of people may be affected by radioactive contamination as a result of a large-scale release, as the disasters at Chernobyl and Fukushima proved.
The chances that radioactive materials from other wastes get dispersed into the environment as a result of leakages, deteriorating containment and/or accidents, are practically unity: dispersion is certainly occurring, year after year. The quantities of the escaped radioactivity in case of a particular event may be relatively low, but exposure to the radioactive materials may last forever. Moreover there are routine discharges of radioactive materials. Due to these low-rate dispersion hundreds of millions of people are exposed to a continuously growing level of radioactivity in air, food and water.

The health hazards posed by radioactive materials dispersed in the human environment are disproportionally greater than might be expected based on the extent of dispersion of materials alone at a given moment, due to the long term effects: chronic exposure by permanently contaminated drinking water, radionuclides entering the food chain and redispersion of radioactive materials as dust and aerosols by wildfires and human activities. A number of radionuclides tend to accumulate in specific organs. Some radionuclides are extremely radiotoxic, so deleterious health effects are possible at very low concentrations of those radionuclides. Underexposed in the radiological models of the nuclear industry are the biological behaviour and destructive effects of radionuclides within living tissue.
Radioactivity cannot be made harmless to living organisms. Chemical elimination or extraction from soil, water and air is impossible because of the large number of different radionuclides, each having its specific physical and chemical properties, that are identical to non-radioactive isotopes.

There is just one option to limit the radioactive contamination of the biosphere caused by nuclear power: by isolating the radioactive materials effectively from the biosphere as soon as possible after their generation. There is just one way to achieve isolation from the biosphere, that is by disposal of the radioactive waste in deep geological repositories excavated in geologically stable formations.

Construction of a geologic repository for spent fuel

Which geologic formations are best suited to accomodate a geologic repository? Each country seems to answer this question on its own way, dependent on the geologic options present and the political situation of the moment. For example, in the Netherlands and in Germany salt domes are discussed, in Belgium and France old clay formations, in the USA a volcanic formation (cancelled, without naming a new option) and granitic formations in Sweden, Finland and Switzerland.

How large and how many?

Each canister in the KBS-3 concept holds 2 Mg of spent fuel. Assume a modern nuclear power plant of 1 GWe power generates 25 Mg spent fuel a year, and the distance between the holes for the canisters in the gallery floor is 6 m, then 75 m of gallery plus holes have to be constructed per reactor per year.
The current global generation of spent fuel is some 10 500 Mg/a. A large geologic repository at a capacity of 40 000 Mg of spent fuel would comprise 120 km of galleries, excluding the access tunnels. This would imply that every four years a repository with 120 km of galleries more than 400 m below surface in a geologically stable formation has to be opened to dispose of the global generation of spent fuel at the current rate. To dispose of the existing backlog of 60 years civil nuclear power, more than 310 000 Mg in 2016, eight of such large repositories would be needed.

Uncertainties

Corrosion proceeds fast at elevated temperatures in the presence of water and nuclear radiation. It is not known how long the containers (usually coined 'canisters' in nuclear jargon) will last before they go leaking. The bentonite is for retarding the migration of radionuclides leaking from corroded containers. There are doubts regarding the sealing and retarding function of bentonite under the extreme conditions to be expected in the repository.
Also the depth of the repository below surface might be a matter of concern: during an ice age the permafrost may penetrate as deep as 400 m below surface, so the depth of geologic repositories has to be greater than 400 m.
During the required storage period of a million years or more several ice ages may occur. The glaciers could scrape away thick layers of soil and rocks, exposing the radioactive waste.

Ingression of water might cause great hazards during operation of a repository. During this period, that may take years or even decades, the containers with spent fuel and with other radioactive materials are placed in their definitive positions. In contact with water and the presence of nuclear radiation the materials of the containers rapidly corrode, and the radioactive contents will react with water. Large amounts of radionuclides may enter the groundwater and may so reenter the human environment.
Such events may also happen after definitive closure of a repository, caused by geologic properties and activities of the formation, resulting in cracks and fissures, or by human intrusion.

During the 1970s Germany has disposed of 126 000 containers of plutonium-bearing waste in an old salt mine in Asse, assuming that this would be a safe geologic repository. But now, the walls of the Asse mine are collapsing and cracks forming, thanks to pressure from surrounding rocks. Despite hurried backfilling of much of the mine, the degradation continues. Brine seeps in at a rate of around 12 m^3/day So the race is on to dig it all up before radioactive residues are flushed to the surface. It could take decades and cost billions of euros to resolve the problem. In the meantime, excavations needed to extract the drums could cause new collapses and make the problem worse [Pearce 2016].

At this moment some 12 million nuclear bomb equivalents of radioactivity in temporary and often vulnerable storage facilities are still awaiting definitive isolation from the biosphere.

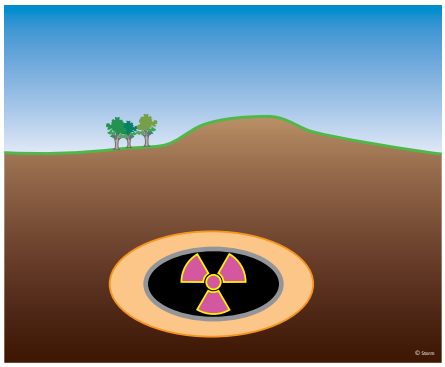

Figure 1
Symbolic presentation used in this study of a deep geologic repository. The purpose is to isolate the radionuclides from the biosphere for geologically long periods.

Other heat-generating waste

When spent fuel is reprocessed the newly formed plutonium and the remaining uranium are separated from the other contents. The other actinides and fission products end up in liquid and solid heat-generating wastes. A part of the waste can be vitrified and the containers with the resulting glass are to be handled similar to spent fuel canisters. A number of radionuclides cannot be fixed in a glass matrix due to their physical and chemical properties, for example noble gases, iodine, technetium and cesium. A significant part of these nuclides are released into the environment.
It is not clear what will happen with the remaining highly radioactive reprocessing waste that is not vitrifiable, It has to be packed in durable containers, and has to be stored in a similar repository as for spent fuel because it is heat generating and/or contains high concentrations of long-lived radionuclides.

Spent nuclear fuel management options

Concerning spent fuel and other radioactive wastes this study starts form the viewpoint: keep the volumes containing radioactive materials as small as possible, so keep spent fuel elements intact, and isolate the spent fuel elements and other radioactive wastes from the biosphere as soon as possible.
These recommendations are not self-evident to the nuclear industry, on the contrary. In its publications the nuclear industry in Europe and the USA discusses various options:

- direct disposal (the least popular option within the nuclear industry)
- retrievable storage, for future generations
- reprocessing, to make possible the following options:
  - vitrification of fission products and actinides,
  - reuse of plutonium in MOX (Mixed OXide) fuel in light-water reactors (LWRs)
  - reuse of reprocessed uranium in LWRs
  - closed-cycle systems (breeders)
  - partitioning and transmutation
- use as ‘fuel’ in thorium waste burners, integrated fast reactor (IFR) or in other concepts.

During the fission process in an advanced light-water reactor (LWR) not more than 0.5% of the uranium

nuclei (U-235 + U-238 via Pu-239) can be fissioned. A small part of the uranium-238 nuclei are transformed into plutonium by neutron capture. In a reprocessing plant spent fuel is separated into three fractions: fission products, plutonium and uranium.
According to the nuclear industry the recovered plutonium and uranium could be used for more energy generation per kg uranium than possible in LWRs by means of MOX fuel in LWRs or in breeder reactors. MixedOXide fuel consists of uranium oxide and plutonium oxide, see m15 *Plutonium recycling in light-water reactors by MOX fuel* and m01 *Uranium-plutonium breeder systems*. Use of MOX fuel in LWRs has a negative energy balance: more energy is cconsumed to reprocess spent fuel and to fabricatie MOX fuel elements than can be extracted from those elements. Breeder reactors use plutonium as fissile material and depleted or natural uranium as fertile material, from which new plutonium can be formed by neutron capture. Plutonium to start up the first generation breeders would have to be extracted from spent fuel of the present generation nuclear reactors by reprocessing

Reprocessing would also make waste reduction possible by vitrification of only high-level radioactive component of spent fuel and/or by partitioning and transmutation. According to a popular view within the nuclear industry the latter two technical concepts, vitrification and P&T, could reduce the high-level waste problem to a routine job, see for example [MacKay 2009] Q399. Both concepts may seem plausible at a first glance. However, on closer examination these concepts prove to be unfeasible as radioactive waste reduction strategies. Contrary to assertions of the nuclear industry vitrification of high-level radioactive waste greatly increases the volumes of radioactive waste to be sequestered in gelologic repositories. Partitioning and transmutation of long-lived radionuclides from spent fuel is an infeasible concept, due to phenomena governed by the Second Law of thermodynamics. see reports m16 *Partitioning and transmutation*, m20 *Reprocessing of spent fuel* and m30 *Vitrification of nuclear waste*.

Permanent storage in a geologic repository without any treatment other than packing in durable containers, is the least dangerous way to isolate the radioactivity in spent fuel from the human environment. The next section addresses technical aspects of direct disposal of spent fuel. By leaving the spent fuel elements intact, the volume containing the radioactivity remains minimal and the contents are in least unstable form. Consequently the chances of dispersion of the radioactivity into the environment remain minimal. The less activities involving the spent fuel, the less chances of accidents and releases of the radioactive contents. In addition, the energy consumption of this option is less than of the other options, all involving reprocessing of spent fuel. The other options depend on succesful reprocessing of spent fuel, a highly energy-intensive and polluting process, see report m20 *Reprocessing of spent fuel*.

Direct disposal of spent fuel

After shutdown of the reactor spent fuel generates enough heat to melt it within a short time. The residual heat generation slowly decays.
Therefore the first step in the management of spent fuel is a cooling period of 30-60 years, to let decay the residual heat, in water-filled cooling pools or dry casks. This so-called interim storage period poses serious public health risks, because the facilities are vulnerable to natural disasters and terrorism.
The second step to an effective isolation is an appropriate packing of the spent fuel. The containers should be resistant to water for long periods, a demanding task, for most materials rapidly deteriorate in the presence of water and strong radiation fields and at elevated temperatures.
The third step is the construction of a facility for permanent disposal of the waste containers. In view of the geologic timescales (millions of years) the waste has to be isolated, the final storage facility should be embedded in a geologically very stable formation, a deep geologic repository.
The last step is placing the waste containers into the geologic repository and back filling the remaining galleries definitively.

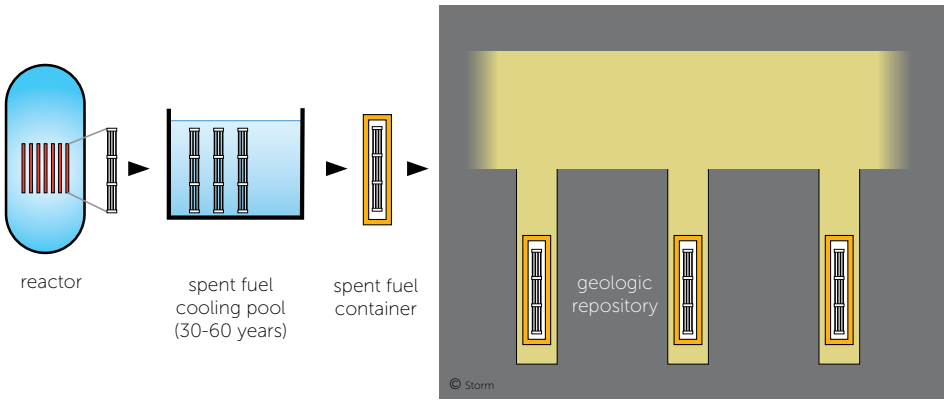

Figure 2

Outline of the sequence of activities needed for direct disposal of spent fuel. After removal from the reactor the spent fuel elements are stored in cooling basins for extended periods, before packed in a special container. These containers are placed in a deep geologic repository by means of robotic equipment. If a deep geologic repository is available, it may take 30-60 years before a spent fuel element is effectively isolated from the biosphere, as far as possible.

The spent fuel in the containers generates heat for long periods after removal from the reactor, so each spent fuel container has to be placed into a separate hole in the floor of a gallery to avoid melting and cracking. Each hole is filled up with bentonite. When the holes of one gallery are occupied, which has to be done with robotic equipment because of the high radiation fields, the gallery itself is back filled with a bentonite-sand mixture and permanently closed. Bentonite is a clay mineral that swells by absorption of water and so forms a poorly permeable mass. Ion migration through bentonite is very slow.
This scenario is based on the Swedish KBS-3 concept, one of the farthest developed designs of a geologic repository for direct disposal of spent fuel, see next section.
Several other concepts have been published for final disposal of high-level waste and spent fuel, such as deep boreholes, see for example [IAEA NW 2018] Q843, [CTECH 2003] Q824, [Smith 2008] Q823, [SKB P-10-47 2010] Q836. This study bases its assessment on the KBS-3 design.

Residual heat generation of spent fuel

At removal from the reactor the spent fuel generates so much heat, due to the radioactive decay of the fission products and actinides, that the fuel elements will melt within a short time if not effectively cooled. The residual heat generation falls to less than 1% during the first year and slowly decays thereafter. A 100 years after removal from the reactor it is still 100-200 watt/Mg. For that reason spent fuel has to be cooled actively during many years after removal from the reactor, before it can be processed for final storage or other purposes. Even then the residual heat generation requires precautionary measures.

Figure 3 represents the residual heat generation of spent fuel of relatively low burnup as function of the time; both axes, the vertical as well the horizontal, have logarithmic scales. To people who are not used to read diagrams with logarithmic scales may easily give a wrong impression of the quantities on the axes. If the horizontal axis of Figure 3 would be stretched to a linear time scale, the diagram would be 1 km long. A linear vertical scale would be 10 km high.

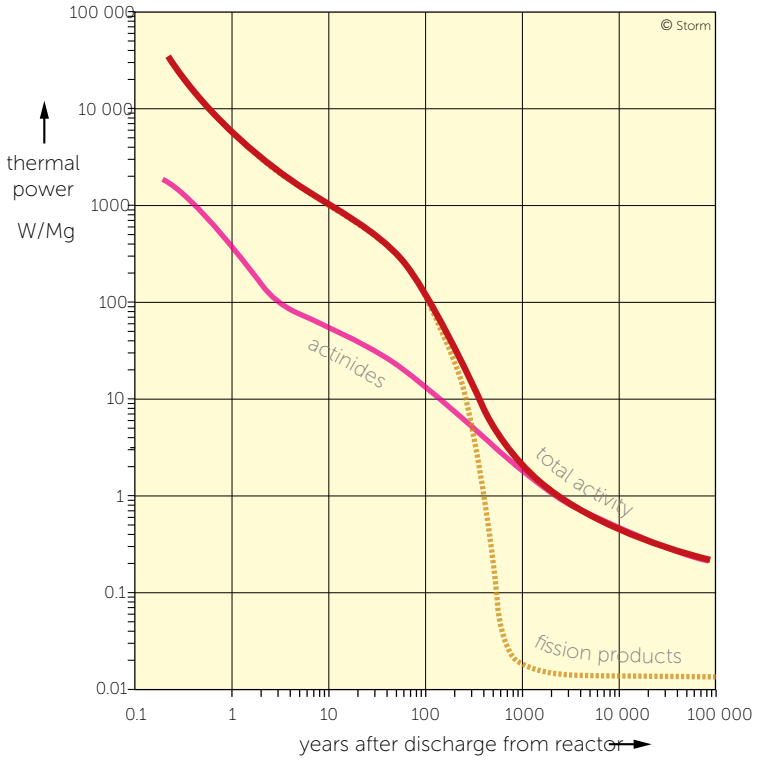

Figure 3

Residual heat generation of spent fuel (burnup B = 33 GWday/Mg) as function of cooling time. At the moment of reactor shutdown, the fuel radiates about 8 MW/Mg (not shown in the diagram). Source: [Hollocher 1975] Q262.

Note that both axes of this diagram have logarithmic scales: each division marks a factor of ten. Spent fuel from current nuclear power plants generally has a higher fuel burnup (B = 40-50 GWday/Mg) and has a about two times higher residual heat generation [IPFM 2011] Q513.

Interim storage of spent fuel

By far the largest part of the anthropogenic radioactivity is confined within the spent fuel elements, containing dozens of different kinds of radionuclides. The highly dangerous spent nuclear fuel is unloaded from the reactor and transported to cooling ponds, usually on the site of the nuclear power plant.

The decay of the radioactivity to levels comparable with an uncontaminated environment takes millions of years. Because of its heat generation and extremely high radioactivity spent fuel has to be cooled for decades, in order to avoid melting, hydrogen explosions and consequently release of its contents, as happened in Fukushima. At high temperatures the zirconium cladding of the fuel elements reacts with water, forming hydrogen. Radiolysis of water due to the nuclear radiation from the spent fuel, also forming hydrogen, compels continuous purification of the cooling water.

Cooling pools

After removal from the reactor spent fuel elements are stored in water-filled cooling pools for a long period, this is called interim storage. After some 30 years interim storage in cooling ponds the heat production has decayed sufficiently to handle the fuel elements for further processing.

Interim storage may become a source of inadvertent emission of radioactivity.

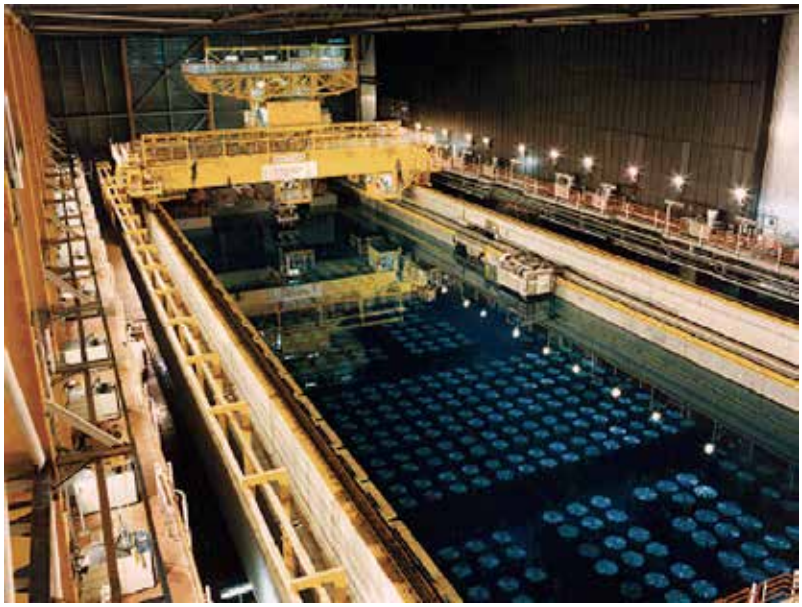

Figure 4
Storage basin for spent fuel in the British reprocessing plant THORP at Sellafield. The basin is filled with demineralised water, which has to be cooled and purified continuously. The spent fuel elements give off a blue glare caused by the interaction of the nuclear radiation with water (Cherenkov effect). The cooling pools of nuclear power plants have a similar construction, but are smaller. Photo World Nuclear Association.

Operation and maintenance of the interim storage facilities are expensive. The water in the pools has to be actively cooled and decontaminated during a period of at least 30 years. The spent fuel of the new generation of reactors with a higher fuel burnup, such as the EPR, may have to be cooled for much longer periods. The basins deteriorate and may go leaking, as happened at several occasions in the past, and have to be replaced. In addition the integrity of the fuel elements deteriorates inevitably over time, due to a number of natural mechanisms, exacerbated by the intense nuclear radiation, so unplanned releases of radioactivity into the environment increase over time as well as the risks of the occurrence of large accidents. If the cooling water is drained during a certain period, as a result of an accident or terroristic action, the fuel elements will melt. The zirconium cladding reacts vigorously with water at high temperatures, producing hydrogen. Explosions may be unavoidable and thousands of nuclear bomb equivalents of radioactivity will escape into the environment. This scenario actually happened at Fukushima in 2011.

Activities related to interim storage of spent fuel do not generate financial profits for the company which operated the nuclear power plant during its productive life. Does that company still exist 30-60 years after closedown of the plant?

Dry casks

In the United States during the late 1970s and early 1980s, the need for alternative storage of spent fuel began to grow when pools at many nuclear reactors began to fill up with stored spent fuel. Utilities began looking at options such as dry cask storage for increasing spent fuel storage capacity. Designers of nuclear power plants anticipated that the spent fuel would be reprocessed, with usable portions of the fuel to be recycled and the rest to be disposed as waste. However, commercial reprocessing was never successfully developed in the United States, and a permanent waste repository has not yet been developed. As a result, many of the spent fuel pools at commercial nuclear power plants are approaching their full capacity [NRC 2012a] and [NRC 2012b]. A similar development may be expected also in other countries.

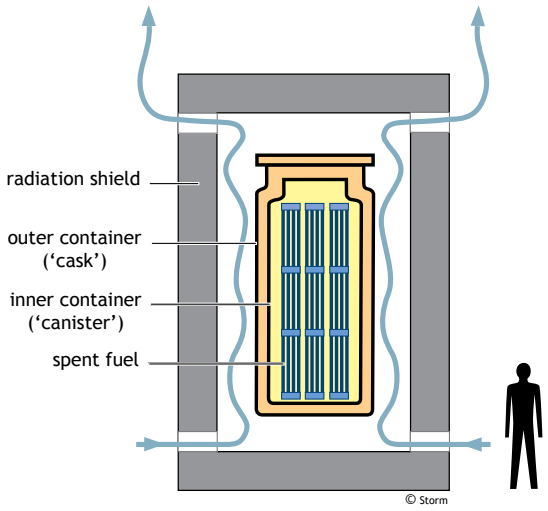

Figure 5

Principle of dry storage of spent fuel elements. The residual heat has to be removed by convection and natural circulation of air around the steel vessels containing the heat generating spent fuel.

Spent fuel is typically cooled at least 5 years in the spent fuel pool at the reactor site before transfer to cask; the industry norm is 10 years. In the United States nearly 63000 Mg (metric tons) of commercial spent fuel accumulated at the end of 2009. Of that total nearly 49000 Mg (~78%) were in pools and nearly 14000 Mg were stored in dry casks. The total amount of spent fuel increases by 2000-2400 Mg annually in the US.
U.S. nuclear utilities are operating dry-storage facilities for used fuel that are licensed for operating periods of up to 60 years. The fuel in these facilities and the used fuel that will be discharged in the foreseeable future may need to remain in storage for much longer periods. Some have suggested that this period could extend to as long as 300 years [NRC 2012c].
In Europe the reprocessing option is still kept open and large numbers of spent fuel elements are stored in pools at the reactor sites and in sizeable pools at the sites of the reprocessing plants, La Hague in France and Sellafield in the UK.

Figure 6

Dry storage casks of spent fuel. Source: US Nuclear Regulatory Commission (photo retrieved from wikipedia).

In case of dry cask storage the spent fuel elements are placed in double-walled steel containers after an

initial cooling period in a cooling pool. These containers, casks in nuclear jargon, are placed vertically inside concrete cilinders as radiation shields, or horizontally into a concrete bunker.
During dry storage the spent fuel elements have to be cooled by natural air circulation. This implies that the steel containers inside the concrete outer cask (radiation shielding) has to stay constantly in direct contact with air and that the heat transfer from the fuel elements to the outside air has to be sufficient to keep the temperature of the fuel elements at a safe level.
The integrity of the casks containing the fuel elements deteriorates inevitably over time, due to a number of natural degrading mechanisms and by reaction with gases and pollutants in the cooling air, exacerbated by the intense nuclear radiation, so unplanned releases of radioactivity into the environment increase over time. It would be very risky to transport a cask with a leaking fuel container. Terrorsitic attacks might cause severe accidents with dispersion of large amounts of radioactivity.

Uncertainties

Corrosion proceeds fast at elevated temperatures in the presence of water and nuclear radiation. It is not known how long the containers (usually coined 'canisters' in nuclear jargon) will last before they go leaking. The bentonite is for retarding the migration of radionuclides leaking from corroded containers. There are doubts regarding the sealing and retarding function of bentonite under the extreme conditions to be expected in the repository.
Also the depth of the repository below surface might be a matter of concern: during an ice age the permafrost may penetrate as deep as 400 m below surface, so the depth of geologic repositories has to be greater than 400 m.
During the required storage period of a million years or more several ice ages may occur. The glaciers could scrape away thick layers of soil and rocks, exposing the radioactive waste.

Isolation of operating waste and dismantling waste

In addition to spent fuel the nuclear system generates large volumes of radioactive waste containing the balance of the human-generated radioactivity: operating waste and dismantling waste. Due to the large volumes of these wastes the concentrations of the radionuclides are relatively low, and as a result these wastes are not heat generating. Because the chance that dangerous radionuclides get dispersed from these materials is very high, they have also to be isolated from the biosphere in repositories deep in geologically stable strata.

Examples are the VLJ repository at Olkiluoto in Finland and and the SFR repository at Forsmark in Sweden. The nuclear power plants at Olkiluoto and Loviisa have both a repository for operating waste. The Olkiluoto repository is commissioned in 1992 and consists of two rock silos at a depth of 60-100 meters inside the bedrock in the Ulkopää peninsula of Olkiluoto Island.
Low level and intermediate-level operating waste generated at the Loviisa NPP is finally disposed of in the facilities built in the bedrock of Hästholmen Island. The Loviisa disposal facility consists of a 1170 m long access tunnel and hall facilities built at a depth of about 110 m; it was put to disposal use in 1999. Both facilities are being expanded.
Waste will be packed in 200 liter drums, compacted to about 100 liter, and these are packed in steel crates (1.3 or 1.4 m^3 each) and concrete crates (5.2 m^3 or 3.9 m^3).
The repositories in Sweden and Finland are destined for definitive storage of operating waste from the nuclear power plants in Sweden (10 or 12) respectively Finland (2). The Finnish repositories are also meant to receive decommissioning waste.

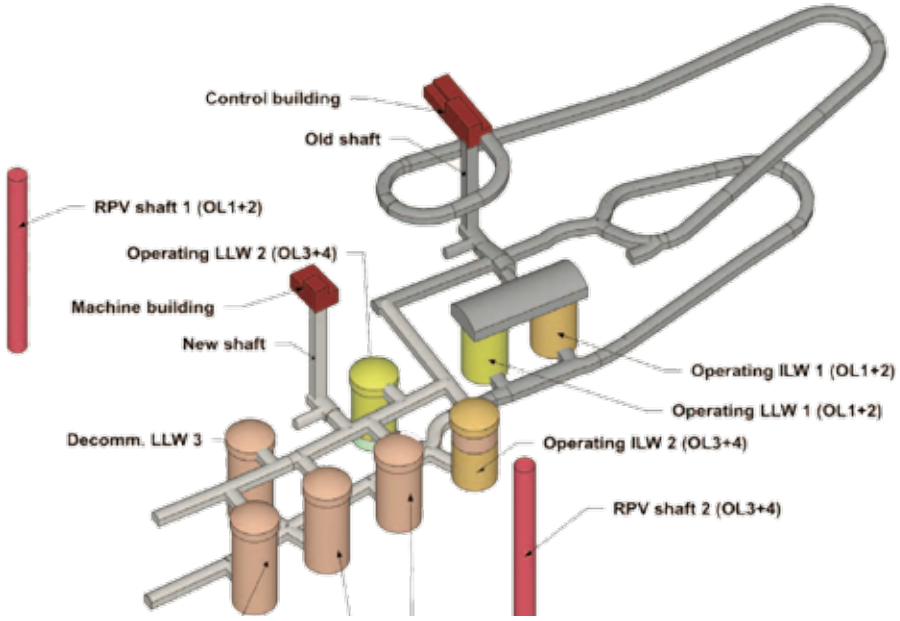

Figure 7

Design of the Olkiluoto VLJ repository of Low-Level Waste (LLW), Intermediate-Level Waste (ILW) and decommissioning waste. The waste containers are disposed of in silos at a depth of 60-100 meters below level. Source: [Posiva 2013] and [Henglanti 2011].

WIPP

The world's only operating deep geological repository is the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico in the USA. The WIPP is run by the US Department of Energy (DOE) and is used to dispose of laboratory equipment, clothing and residues from the nation's nuclear-defence programme. In the past 15 years around 91 000 m^3 of such transuranic waste, containing plutonium, americium and curium, has been placed there . The storage rooms are carved out of a 250 million year old saltbed, 600 meters below ground level.

On 14 February 20214 a drum exploded as a result of an unexpected chemical reaction within the drum, releasing plutonium and americium (about 3.7 GBq). Airborne radioactive material reached the surface through the ventilation system and spread some 900 meters from the exhaust shaft.

In the long term risks are described of contamination of aquifers and surface as a result of water intrusion in the repository, for example by inadvertently drilling a borehole through the WIPP [Tracy *et al.* 2016].

French concepts

The French National Radioactive Waste Management Agency ANDRA (Agence Nationale pour la Gestion des Déchets Radioactifs) developed two concepts for disposal of radioactive wastes: a shallow burial facility and a deep geologic repository [ANDRA-*solutions* 2014]. The French waste classification system makes a distinction between short-lived and long-lived radioactive waste. For short-lived waste shallow burial disposal is proposed. For some classes of waste a storage period is foreseen to let decay some radionuclides, before final disposal in a surface facility.

For high level waste, such as spent fuel and vitrificated reprocessing waste, ANDRA proposes a geologic repository, about 500 meters deep in a clay formation.

Figure 8
Concept of ANDRA of a surface disposal facility for 'low- and intermediate-level short-lived' waste. Source: [ANDRA-*solutions* 2014].

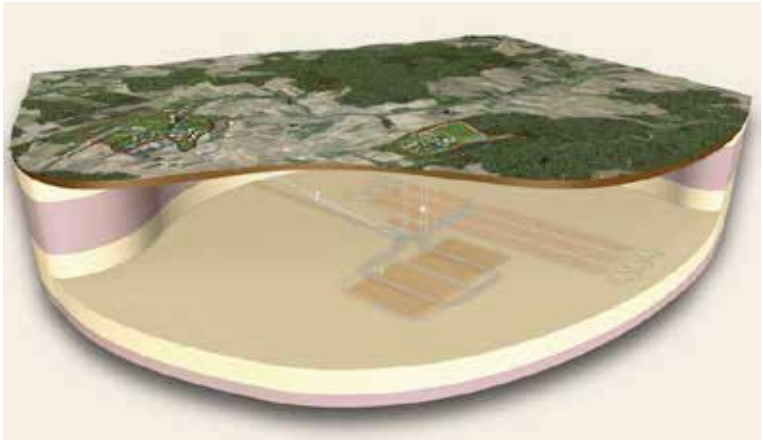

Figure 9
Concept of ANDRA of a geological repository for 'HLW and ILW/LL' waste, otherwise known as the Industrial Centre for Geological Disposal (Centre industriel de stockage géologique – Cigéo). Source: [ANDRA-*solutions* 2014].

Progress in Europe

The start of operation of geologic repositories in Europe varies widely in time, as the following quote from [EU-*9329* 2017] proves:

> For the disposal of intermediate level waste, high level waste and spent fuel, the concepts for disposal as per Article 12(1)d of the Directive (e.g. site selection, development of design) are not concrete in most of the Member States, often due to the need for policy decisions to be made or sites to be selected.22 Of the Member States that are planning to develop geological disposal facilities in the coming decades, only Finland, France and Sweden have so far selected sites, demonstrating the challenges of moving from the planning stage to practical implementation. Globally, Finland is the first country where the construction of a deep geological facility has begun and is expected to be in operation by 2022, with France and Sweden expected to start operation by 2030 (see Figure 2). Another 12 Member States have plans for a deep geological repository and are at different stages of implementation. The majority of Member States without nuclear programmes cover activities up to interim storage and repatriation of spent fuel (if relevant) to the supplier in their national programme and have not yet defined a policy or a route for the disposal of radioactive waste.

Figure 10

Planned start of operation of deep geological facilities. This is Figure 2 from [EU-9329 2017].

KBS-3 concept

A good example of a geologic repository for permanent storage of spent fuel is the KBS-3 concept designed by SKB (Svensk Kärnbränslehantering AB, Swedish Nuclear Fuel and Waste Management Co). The KBS-3 project started in 1973 and has been presented in 1983 by the Nuclear Fuel Safety project KBS: [Papp 1998a] Q37, [Papp 1998b] Q38, [IAEA-349 1993] Q43, [Thegerstrom 2010] Q453, [SKB 2006a] Q176a, [SKB 2006b] Q176b, [SKB P-10-47 2010] Q836. The design, which is similar to the design described in [INFCE-7 1980] Q277, has been approved in 1984 by the Swedish government. SKB expects to start disposal of spent fuel canisters by 2020, 47 years after the start of the KBS project.
The KBS-3 concept is envisioned as a system of galleries in a granitic formation or in very stable rock strata some 500-700 meters below the surface. The spent fuel elements would be packed in containers of cast iron, clad by a thick layer of very pure copper. The designers assume this combination of materials to be resistant to (sea)water for thousands of years. However this assumption may be optimistic in view of the elevated temperatures and the presence of nuclear radiation. There are also doubts regarding the sealing function of bentonite under the extreme conditions to be expected in the repository.
The spent fuel canisters are placed in boreholes in the floor of the galleries by remotely piloted vehicles. The holes are to be filled up with bentonite, and the gallery itself would be backfilled with a bentonite-sand mixture after filling the holes. The canister must remain subcritical with sufficient certainty in the rare cases where highly enriched uranium or uranium mixed with plutonium has to be placed in an individual canister, even if water penetrates into the canister.

There are three main phases of spent fuel final disposal:
- construction of the repository
- operation of the facility during 60 years, sequestering the spent fuel canisters
- definitive closure of the repository.

The complete sequence of these activities may take a period of a century.

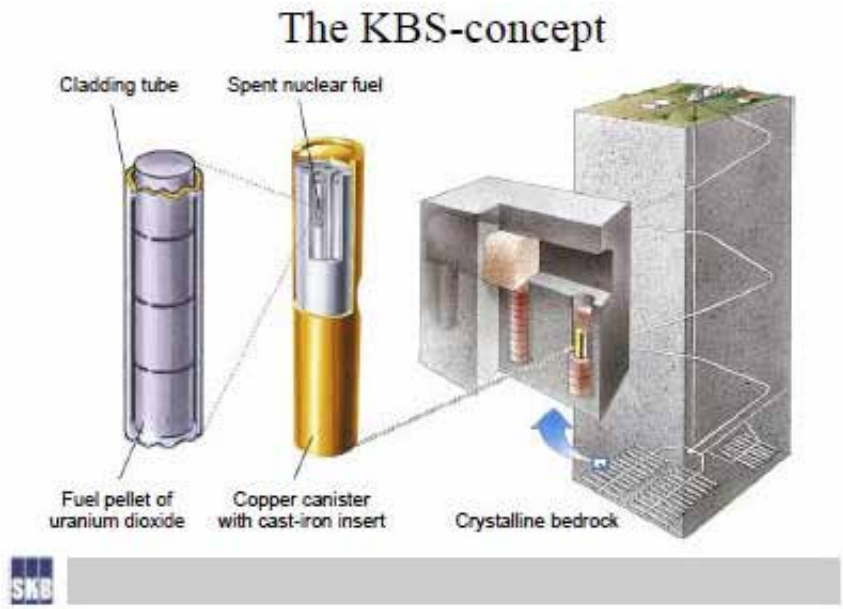

Figure 11
Swedish KBS-3 concept for deep geological disposal of spent nuclear fuel. Source: [SKB 2006a] Q176a.

Each canister in the KBS-3 concept holds 2 Mg of spent fuel. Assume a modern nuclear power plant of 1 GWe power generates 20 Mg spent fuel a year, and assuming the distance between the holes for the canisters in the gallery floor to be 6 m, then 60 m of gallery plus holes have to be constructed per reactor per year.

If the current global nuclear capacity of some 470 GWe would consist of reference reactors, then 9400 Mg of spent fuel would be generated per year. The actual production of spent fuel is much higher because the majority of the currently operating reactors discharge spent fuel at lower burnup than the reference reactor, so we assume for convenience that 10000 Mg of spent fuel is generated annually worldwide.
A large geologic repository at a capacity of 40000 Mg of spent fuel would comprise 100 km of galleries, excluding the access tunnels. This would imply that every four years a repository with 100 km of galleries 500 m below level in a geologically stable formation has to be opened to dispose of the global generation of spent fuel at the current rate. To dispose of the existing backlog of more than 60 years civil nuclear power, some 368 000 Mg at the end of 2013 [IAEA-NW 2018] Q843, more than nine of such large repositories would be needed.

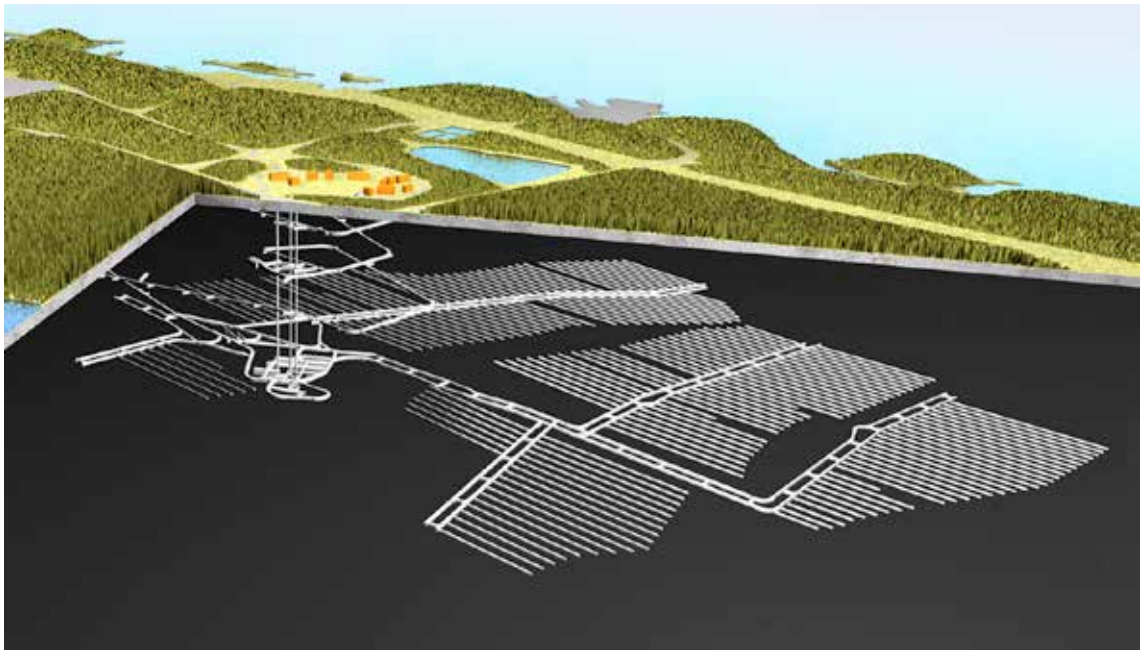

Figure 12
Spent nuclear fuel disposal repository at Olkiluoto (Finland), according the KBS-3 concept. The current plans involve excavation of the repository on one level at about 400-450 m depth. Deposition holes will be bored in the floors of the deposition tunnels for inserting the canisters. The canisters will be completely surrounded by bentonite blocks that will swell considerably when becoming wet. The repository will be expanded as the disposal operations progress by excavation of more disposal and central tunnels. Source: Posiva Oy [www.posiva.fi/en/media/image_gallery].

Construction

Numerical data on the energy investment and CO_2 emission of the final disposal of spent fuel are not found in the open literature. To store 7000 Mg spent fuel, about $5.8*10^6$ m^3 granite has to be mined, according to [Papp-1 1998] Q37.
The specific repository volume per Mg spent fuel then would be:

V_{rep} = 829 m^3/Mg

This figure is adopted in this study to estimate the energy investment and CO_2 emission of the onstruction of the repository.

Papp cites a total cost of DM 8.5bn in 1998 for a repository with a capacity of 7000 Mg spent fuel, one of the very few cost estimates found. This estimate, concerning only the construction phase, is based on a mining cost of 600 DM/m^3 and a bentonite cost of 3000 DM/m^3.
From these figures follows per Mg spent fuel:

c = 1.21*10^6 DM/Mg in 1998 = 0.63*10^6 $(2000)/Mg spent fuel

In practice the construction cost and corresponding energy investment might be higher. At the time of writing (2019) not one deep geologic repository in the world is operational, so no empirical data exist. Large cost escalations are intrinsic to new technology projects according to [RAND 1981] Q126:

"Severe underestimation of capital costs is the norm for *all* advanced technologies."

According to [RAND 1979] Q127 escalations in cost estimates of energy process plants with factors 2-5 are not uncommon. The nuclear industry provides ample examples of this rule and of the observations mentioned in both publications.

The specific energy investment of the construction of the KBS-3 repository in granite can be estimated by comparison with underground mining in hard rock. A specific figure of mining hard rock can be found based on a process analysis of the Ranger mine in Australia (see report **m26** *Uranium mining and milling*):

$J_{mining} = J_e + J_{th}$ = 1.1 GJ/m^3 rock J_{th}/J_e = 8

The thermal energy input then is:

J_{th} = 0.98 GJ/m^3 rock

Both bentonite and sand, needed to backfill the access tunnels and galleries, have to be mined, prepared and transported to the repository. From a process analysis (see report **m41** *Uranium mine rehabilitation*) follows a specific thermal energy consumption of:

J_{th} = 20 GJ/Mg bentonite

At an average density of d =2.15 Mg/m^3 this corresponds with:

J_{th} = 43 GJ/m^3 bentonite

Assumed that the backfill consists of a 1:1 mixture of sand and bentonite and neglecting the energy consumption of the sand component, the energy input of the backfill per m^3 of repository would be:

$J_{bentonite} = J_{th}$ = 21.5 GJ/m^3 repository

The total thermal energy investment of the construction of the repository, including the supply of bentonite, but excluding the backfilling operation would be:

J_{th} = 22.5 GJ/m^3 repository

Per Mg spent fuel:

J_{th} = 22.5*829 = 18.6 TJ/Mg spent fuel

The electric component is small compared with the thermal component (less than 1%) and is neglected in this assessment.

Sequestering and backfilling

Estimation of the cost and energy investments of the operational phase (handling and sequestering the spent fuel canisters) and closure (backfilling the galleries and access tunnels) during the 60+ years following the construction may be compared with operation + maintenance of a nuclear power plant. Many operations have to be remotely piloted, due to high radiation levels.in the disposal galleries.
From the figures given by [Papp-1 1998] Q37 follows per Mg spent fuel:

c = 1.21*10^6 DM/Mg in 1998 = 0.63*10^6 $(2000)/Mg spent fuel

From the cost figure the specific energy investment and CO_2 emission can be estimated using the energy/cost ratio of new construction, e = 12.34 MJ/$ (2000):

$J_{rep.sf} = c*e$ = 0.63*10^6*12.34 = 7.8 TJ/Mg spent fuel J_{th}/J_e = 4.8

This study assumes an annual operation cost of 2% of the construction cost, or:

$c = 0.02*0.63*8.5*10^9$ $(2000)/a = $ $107*10^6$ $(2000)/a$

Assumed that phase 2 (sequestering) will takes 60 years and phase 3 (final closure) 10 years, the total cost would become:

$c = 70*107*10^6 = 7.49*10^9$ $(2000)

Per Mg spent fuel :

$J_e + J_{th} = c*e = 1.07*10^6*12.34$ =13.2 TJ/Mg spent fuel $\quad J_{th}/J_e = 4.8$

The thermal component is:

J_{th} = 10.9 TJ/Mg spent fuel

Complete sequence

The specific energy investment per Mg spent fuel is:

$J_e + J_{th}$ = 18.6 + 13.2 = 31.8 TJ/Mg spent fuel

The specific thermal energy investment per Mg spent fuel is:

J_{th} = 18.6 + 10.9 = 29.5 TJ/Mg spent fuel

The specific CO_2 emission is:

$m = 29.5*10^6 *75$ = 2200 Mg CO_2/Mg spent fuel

The lifetime figures of the reference reactor are:

Energy investment:

$E_e + E_{th}$ = 31.8*583.4 = 18552 TJ

E_{th} = 29.5*583.4 = 17 210 TJ

CO_2 emission:

m = 2200*583.4 = 1283000 Mg CO_2

The corresponding specific CO_2 emission is:

$\gamma = 1283*10^9/219*10^9$ = 5.86 gCO_2/kWh

Table 1

Energy investment and CO_2 emission of the final disposal of spent fuel from the advanced reference reactor and the EPR design.

reactor	mass spent fuel Mg	number of V5 canisters	$E_e + E_{th}$ input TJ	E_{th} input TJ	mCO_2 Gg	specific emission gCO_2/kWh
advanced reference reactor	583.4	292	18 552	17 210	1283	5.86
EPR design	1506	753	47 891	44 427	3313	4.24

Deep geologic repository for other radioactive waste

This study starts from the idea that all radioactive waste produced by the nuclear system has to be disposed of in a geologic repository. Radioactive waste other than spent fuel has to be stored in a separate repository. Temporary or even permanent storage in above-ground facilities or shallow burial sites, such as in France [ANDRA-*solutions* 2014] Q757 is no option in the long run, in the view of this study.
Estimates of energy investments and CO2 emission are based on the Swedish SFR concept [IAEA-349 1993] Q43, [Vattenfall 2005] Q152, [Sjöland 2014] Q704, for the same reasons as for the spent fuel repository concept. The waste containers are stored in large caverns, mined in a stable rock formation. In the extended SFR concept 6.33 m^3 rock has to be mined for each m^3 of packaged waste [SKB 2018] Q839, or 17.5 Mg rock per m^3 waste (density of granite d = 2.76 Mg/m^3).

Figure 13
The existing SFR on the right, the new section on the left. After extension the SFR will be three times bigger than it is today. Source: [SKB 2018] Q839.

Assumed that the specific energy investment of the construction of the SFR repository, including the supply of bentonite, but excluding the backfilling operation would be the same as for the KBS-3 spent fuel repository:

J_{th} = 22.5 GJ/m^3 repository

Sequestering + closure per m^3 repository:

J_{th} = 13.2 TJ/Mg spent fuel/829 m^3 repository/Mg spent fuel = 15.9 GJ/m^3 repository J_{th}/J_e = 4.8 (see SKB-3)

The total energy investment of construction + sequestering + closure:

$J_e + J_{th}$ = 22.5 + 15.9 = 38.4 GJ/m^3 repository

The thermal component of sequestering + closure is:

J_{th} = 13.2 GJ/m^3 repository

The total thermal energy investment of construction + sequestering + closure then becomes:

J_{th} = 22.5 + 13.3 = 35.7 GJ/m^3 repository

Per m^3 waste:

J_{th} = 35.7*6.33 = 226 GJ/m^3 waste

The corresponding specific CO_2 emission is:

γ = 226*75 = 16950 kgCO_2/m^3 waste

The total energy investment and CO_2 emission of the final disposal of the radioactive wastes from the nuclear process chain of the advanced reference reactor, except spent fuel and uranium mine tailings, would become:

thermal component:

J_{th} = 61,3 PJ

electric component:

J_e = 4,63 PJ

and the specific CO_2 emission per kilowatthour:

γ = 4594*10^9/219*10^9 = 21.0 gCO_2/kWh

Table 2

Data on the final disposal of the radioactive waste other than spent fuel from the upstream processes of the advanced reference reactor

process advanced refer. reactor	number of containers	displaced volume m^3	repository volume m^3	E_e+E_{th} input TJ	E_{th} input TJ	mCO_2 Gg	specific CO_2 g/kWh
refining + conversion	5400	5400	34 182	1313	1220	91.53	0.42
enrichment	3100	3100	19 623	754	701	52.55	0.24
reconv. + fuel fabr.	7500	7500	47 475	1823	1695	127	0.58
reactor OMR	100 000	100 000	633 000	24307	22 600	1695	7.74
sum 1 upstream	116 000	116 000	734 280	28 196	26 216	1966	9.0
reconverted depleted U	26668	26668	168808	6482	6027	452	2.06
decomm. + dismantling	46700	137 300	869109	33 374	31 030	2327	10.63
sum 2 downstream	73368	163 968	1 037 917	39 856	37 057	2779	12.7
total sum 1 + sum 2	189 368	279 968	1 772 197	68 052	63 273	4745	21.7

Table 3

Data on the final disposal of the radioactive waste other than spent fuel from the upstream processes of the EPR design

process advanced refer. reactor	number of containers	displaced volume m^3	repository volume m^3	E_e+E_{th} input TJ	E_{th} input TJ	mCO_2 Gg	specific CO_2 g/kWh
refining + conversion	11880	11880	75 200	2888	2685	201	0.26
enrichment	10120	10120	64 060	2460	2287	172	0.22
reconv. + fuel fabr.	16500	16500	104 445	4011	3729	280	0.36
reactor OMR	220 000	220 000	1 392600	53576	49720	3729	4.77
sum 1 upstream	258 500	258 500	1 636305	62834	58521	4382	5.6
reconverted depleted U	84608	84608	535 569	20566	19121	1434	1.84
decomm. + dismantling	54873	159 978	1 012661	38886	36155	2712	3.47
sum 2 downstream	139481	244 586	1 548229	59452	55276	4146	5.3
total sum 1 + sum 2	397 981	503 086	3 184534	122286	113 697	8527	10.9

Except for the uranium mill tailings, all radioactive wastes of the nuclear energy system are to be packed in appropiate containers and isolated from the biosphere in geologic repositories.
This study assumes the use of five types of standard containers, V1 through V5, depending on the type of waste. These container concepts are among the frequently quoted types in the nuclear literature, e.g. [IAEA-349 1993] Q43 and [IAEA-355 1993] Q62.
Container V1 is not much more than a common steel drum, not suitable for permamanent disposal of radioactive waste. For that reason the V1 container is not included in the assessment of this study. In the past huge amounts of radioactive waste have been packed in this kind of drums and dumped into the sea or into shallow burial pits.
The V3 container corresponds with the German Type II container; the V4 container is not suitable for waste containing apha emitters. The four types of waste containers for permanent disposal, V2 - V5, are shown in Figure 4. The dimensions, materials, masses and specific applications of these containers are addressed in the following tables.

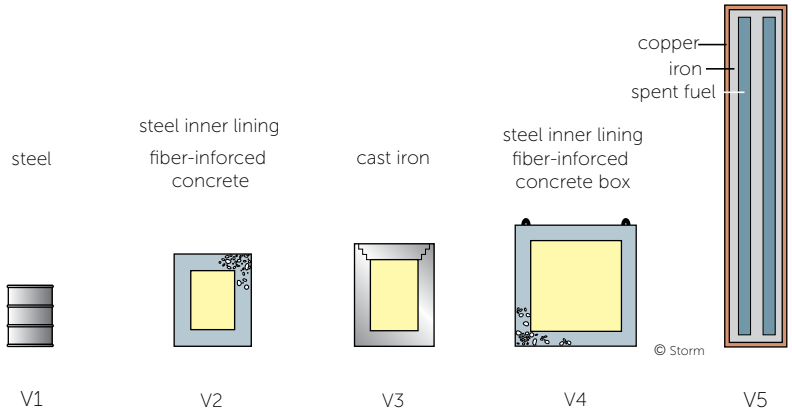

Figure 14
Containers for all categories of radioactive waste, used as reference in this study. Container V1, not much more than a common oil barrel, is not suitable for permamanent disposal and is not included in the assessment of this study. Container V5 is specificly designed for permanent storage of spent fuel elements.

Table 4
Dimensions and masses of the containers for packaging of radioactive waste.

type	outer diameter m	height m	wall thick m	external volume m^3	capacity m^3	concr. Mg	steel/ iron Mg	copper Mg	mass empty Mg
V2	1.02	1.22	0.20	1.00	0.25	1.80	0.04	-	1.84
V3	1.05	1.36	0.21	1.18	0.29	–	6.46	-	6.46
V4	1.60 x 1.60	1.60	0.20	4.10	1.73	5.68	0.137	-	5.82
V5	1.05	4.84	0.05	4.19	-	-	16.4	7.5	23.9

The average energy requirements J_v for construction, loading, handling and transport of the waste containers are assumed to be similar to the energy investments of 'new construction'. This quantity is found by means of an input/output analysis and is used to calculate the energy investments of construction and operation, maintenance + refurbishments (OMR) of the nuclear power plant (report **m39** *Construction and OMR of*

nuclear power plants) and of decommissioning + dismantling. The I/O method is explained in report m06 *Energy analysis, the method*.

$J_v = J(e) + J(th) = 80$ GJ/Mg $J(th)/J(e) = 4.8$

$J(th) = 66.2$ GJ/Mg

Assumed an average specific CO_2 emission of $\gamma = 75$ g/MJ_{th}, the specific CO_2 emission, excluding CO_2 from chemical reactions (e.g. steel and concrete production), would be:

$\gamma_v = 75 \bullet J(th) = 5.0$ Mg CO_2/Mg

Table 5

Construction data of the waste containers used as reference in this study for the packaging of radioactive wastes.

type	waste type	displaced volume m³	capacity m³	mass empty Mg	mass loaded Mg	Je + Jth constr. GJ/V#	Je GJ/V#	Jth GJ/V#	mCO2 Mg/V#
V2	LLW + ILWα	1.00	0.25	1.84	2.4 *	147	25	122	9.2
V3	HLW + α	1.18	0.29	6.46	8.0 *	517	89	428	32.3
V4	LLW + ILW	4.10	1.73	5.82	14.7 *	466	80	386	29,1
V5	spent fuel	4.19	-	23.90	26.8	4000	690	3310	248

* Assumed the content of V2 containers has an average density of $d = 2.4$ Mg/m³ (concrete) and that the V3 and V4 containers are half filled with steel scrap and the remaining volume is filled with concrete. The density of cast iron is $d = 7.3$ Mg/m³ and that of steel and stainless steel $d = 7.9$ Mg/m³.

V5 canister

Canister V5 is in accordance with the Swedish SKB-3 concept [SKB-*TR-10-14* 2010] Q675. Wall thickness of the canaister is 5.0 cm ultrapure copper, steel (iron) insert, containing 4 nuclear fuel elements, from a PWR: $m = 2.000$ Mg HM + 0.900 Mg zircalloy + control rods

Based on cost estimates by [Konings&Dodd 1999] Q57 the specific energy input for construction, handling and transport can be estimated by means of an input/output analysis - $c = 0.13$ M$(2000)/Mg, energy intensity factor $e = 12.34$ MJ/$(2000) - at:

$J(V5) = J_e + J_{th} = c*e = 1.60$ TJ/Mg HM

Assumed $J_{th}/J_e = 4.8$

A loaded V5 canister contains about 2 Mg of heavy metal (HM) - spent fuel plus control rods, excluding cladding - so the energy input for production per V5 canister is would be:

$J(V5) = 2*1.60 = 3.2$ TJ/V5

Cost estimates of new technologies are usually underestimated and spent fuel packaging and handling for definitive disposal is a new technology.

Another estimate is possible starting from the assumption that the energy intensity of the V5 production equals that of 'new construction', calculated by another version of I/O analysis, see previous section:

$J(V5) = J_e + J_{th} = 80$ GJ/Mg $J_{th}/J_e = 4.8$

Packaging the highly radioactive fuel elements and sealing off the canisters has to be done under remote control. The canisters must be fabricated from very pure materials, only exceedingly pure electrolytic copper can be uesed for the outer wall. Fabrication and handling are to be done under strict quality control with high quality specifications [SKB-*TR-10-14* 2010] Q675. In view of these considerations the specific energy requirements for construction and handling of a V5 canister may be estimated at:

$J(V5) = J_e + J_{th} = 4.0$ TJ/V5 $J_{th}/J_e = 4.8$

This value, slightly higher than found according to the first estimate, is used throughout this study. Probably the value of 4.0 TJ/V5 is a low estimate. In practice the cost and energy input might become significantly higher. In view of the electrolytic materials and remote control the energy input of production and handling of V5 canisters might show a high electric component, corresponding with a lower ratio of *J*(th)/*J*(e) than assumed in this study.

CO_2 emission from construction materials

To the energy consumption and the coupled CO_2 emission of construction of the waste containers, as listed in Table 2, should be added the energy consumption and CO_2 emission of the production of the used construction materials concrete, steel and electrolytic copper (embodied energy and CO_2 emission).

Table 6

Specific embodied energy and CO_2 emission of the construction materials of the waste containers. Sources for steel and concrete: [IAEA-TecDoc-753 1994] Q148, [IPCC 2006] Q215, [NRMCA 2012] Q216, for copper: [White 1998] Q299.

quantity	unit	steel/iron	concrete	electrotytic copper
specific thermal energy input	MJ/kg	29.54	1.83	1612
specific CO_2 emission	Mg CO_2/Mg	2.41	0.139 *	97.15

* Portland cement: 927 kgCO_2/Mg cement. Assumed the high quality concrete used for nuclear applications contains 15% cement, then the chemical specific emission of concrete would be 139 kgCO_2/Mg concrete.

Table 7

Embodied CO_2 emission of the construction materials concrete, steel/iron and copper of the waste containers.

type	mass concrete Mg	mass steel Mg	mass copper Mg	mass total Mg	*m*CO2 concrete Mg	*m*CO2 steel Mg	*m*CO2 copper Mg	*m*CO2 materials Mg
V2	1.80	0.04	–	1.84	0.250	0.006	–	0.26
V3	–	6.46	–	6.46	-	15.6	–	15.6
V4	5.68	0.137	–	5.82	0.790	0.019	–	0.81
V5	–	16.40	7.50	23.90	–	39.52	728.6	768.1

Table 8

Specific energy investment and CO_2 emission of the waste containers.

type	*J*e + *J*th construction GJ/V#	*J*e + *J*th materials GJ/V#	*J*e + *J*th total GJ/V#	*J*th total GJ/V#	*m*CO2 construction Mg/V#	*m*CO2 materials Mg/V#	*m*CO2 total Mg/V#
V2	147	4.47	152	127	9.2	0.26	9.46
V3	517	191	708	619	32.3	15.6	47.9
V4	466	14.4	480	400	29,1	0.81	29.9
V5	4000	12600	16600	3792	248	768	1016

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