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Technical assessment of nuclear energy with respect to the ‘do no significant harm’ criteria of Regulation (EU) 2020/852 (‘Taxonomy Regulation’)

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Contents

Abstract

Tackling climate change is an urgent challenge. It calls for the EU to step up its action to show global leadership by becoming climate-neutral by 2050 in all sectors of the economy. This requires compensating, by 2050, not only any remaining CO_2 but also any other remaining greenhouse gas emissions, as set out in the Communication 'A Clean Planet for all – A European strategic long-term vision for a prosperous, modern, competitive and climate-neutral economy' and as confirmed by the 'European Green Deal' Communication.

To complement the existing policy framework, several European Green Deal Initiatives have been adopted and other initiatives are under preparation. Among the adopted initiatives is the Regulation (EU) 2020/852 (the 'Taxonomy Regulation') on the establishment of a framework to facilitate sustainable investment which provides appropriate definitions to companies and investors on which economic activities can be considered environmentally sustainable.

Inclusion or exclusion of nuclear energy in the EU taxonomy was a debated subject throughout the negotiations on the Taxonomy Regulation. While there are indirect references in the regulation to the issue of nuclear energy (including on radioactive waste), co-legislators ultimately left the assessment of nuclear energy to the Commission as part of its work on the delegated acts establishing the technical screening criteria.

The Technical Expert Group on Sustainable Finance (TEG), which was tasked with advising the Commission on the technical screening criteria for the climate change mitigation and adaptation objectives, did not provide a conclusive recommendation on nuclear energy and indicated that a further assessment of the 'do no significant harm' aspects of nuclear energy was necessary.

As the in-house science and knowledge service of the Commission with extensive technical expertise on nuclear energy and technology, the JRC was invited to carry out such analysis and to draft a technical assessment report on the 'do no significant harm' (DNSH) aspects of nuclear energy including aspects related to the long-term management of high-level radioactive waste and spent nuclear fuel, consistent with the specifications of Articles 17 and 19 of the Taxonomy Regulation.

This report is the result of that JRC analysis.

Executive summary

To reach the objectives of the European Green Deal, it is fundamental to direct investments towards sustainable projects and activities with clear assessment of their co-benefits and risks for human health and the environment. The Taxonomy Regulation (Regulation (EU) 2020/852), on the establishment of a framework to facilitate sustainable investments, sets out the conditions, including environmental objectives, that an economic activity has to meet in order to qualify as environmentally sustainable. It also sets the framework for the development of an EU classification system ("EU Taxonomy") of environmentally sustainable economic activities for investment purposes.

The European Commission established a Technical Expert Group (TEG) on sustainable finance in July 2018 to develop recommendations for technical screening criteria for economic activities that can make a substantial contribution to the climate change mitigation or adaptation objectives, while avoiding significant harm to the four other environmental objectives of the Regulation:

— sustainable use and protection of water and marine resources;

— transition to a circular economy;

— pollution prevention control; and

— protection and restoration of biodiversity and ecosystems.

In June 2019, the TEG provided preliminary recommendations for a first set of economic activities, together with the associated technical screening criteria, that should deliver a substantial contribution to climate change mitigation and adaptation, while not significantly harming any of the other environmental objectives.

In its assessment of nuclear energy as part of its review on energy generation activities, the TEG concluded that nuclear energy has near to zero greenhouse gas emissions in the energy generation phase and can be a contributor to climate mitigation objectives. While consideration of nuclear energy from a climate mitigation perspective was therefore warranted, the TEG could not reach a definite conclusion on potential significant harm to other environmental objectives, in particular considering the lack of operational permanent experience of high-level waste disposal sites. Therefore, nuclear energy was not included at this stage in the EU Taxonomy. Instead, the TEG recommended that more extensive technical work be undertaken on the "do no significant harm" (DNSH) aspects of nuclear energy.

During the summer of 2020, in agreement with the Directorate-Generals for Energy (DG ENER), for Environment (DG ENV), for Research and Innovation (DG RTD), for Climate Action (DG CLIMA) and the Secretariat-General of the European Commission, the Directorate-General for Financial Stability, Financial Services and Capital Markets Union (DG FISMA) requested JRC to carry out this "more extensive technical work on the DNSH aspects of nuclear energy" as recommended by the TEG.

The JRC conducted a review to assess nuclear energy generation under the "do no significant harm" (DNSH) criteria, considering the effects of the whole nuclear energy life-cycle in terms of existing and potential environmental impacts across all objectives, with emphasis on the management of the generated nuclear and radioactive waste. This report presents the result of this extensive review.

For practical and editorial reasons, the report is divided into two distinct parts (Part A and B), supplemented by several annexes.

Part A is titled "Review of the state-of-the-art to assess nuclear energy generation under the "do no significant harm" (DNSH) criterion" and deals with the review of the environmental impacts corresponding to the various lifecycle phases of nuclear energy and comparison with the environmental impacts of other electricity generation technologies, such as coal, oil, gas, and renewables (including hydropower).

Part B is titled "Specific assessment on the current status and perspectives of long-term management and disposal of radioactive waste" and deals with the state-of-the-art and DNSH aspects of radioactive waste management, focusing on the final disposal of high-level radioactive waste and spent nuclear fuel.

During the preparation of this report, the need for a detailed overview of the relevant legal and regulatory framework became evident. This has been included as an annex entitled "Legal and regulatory background of nuclear energy" (Annex 1). It is a common background document for parts A & B of the report, outlining the main elements of the associated nuclear and environmental legal and regulatory frameworks.

This report will be reviewed by Member States' national experts on radiation protection and waste management appointed by the Scientific and Technical Committee under Article 31 of the Euratom Treaty, as well as by experts on environmental impacts from the Scientific Committee on Health, Environmental and Emerging Risks (SCHEER).

Policy context

To reach the objectives of the European Green Deal and to meet the EU's climate change mitigation and energy-mix targets for 2030, it is fundamental to direct investments towards sustainable projects and activities with clear assessment of their co-benefits and risks for human health and the environment. To achieve this, a common language and a clear definition of what is 'sustainable' is needed. This is why the action plan on financing sustainable growth called for the creation of a common classification system for sustainable economic activities, or an "EU taxonomy".

The EU Taxonomy is a classification system, establishing a list of environmentally sustainable economic activities. This EU-wide classification system will mean that the EU has a uniform and harmonised way of determining what economic activities can be regarded as sustainable. This is essential in order for the EU to become the first climate-neutral continent by 2050, as well as to mitigate biodiversity loss and other increasingly urgent environmental challenges. This system is being developed through delegated acts and will be published in two batches: one on the climate-related objectives and one on the other four environmental objectives mentioned above.

The Taxonomy Regulation (REGULATION (EU) 2020/852) empowers the Commission to adopt delegated and implementing acts in order to establish the actual list of environmentally sustainable activities along with the associated technical screening criteria for each environmental objective. Although nuclear energy has been recognised by the TEG as "climate-neutral energy", the compliance with the "do no significant harm" criteria of the nuclear energy life-cycle, and in particular the disposal of radioactive waste, requires further considerations.

Key conclusions

- The protection of people and the environment in countries with nuclear installations relies on the existence of a solid regulatory framework that oversees the safety and environmental impacts of these installations. The achievement and maintenance of a high level of safety during the lifetime of nuclear facilities and the duration of related activities requires a sound governmental, legal and regulatory framework, which includes regular safety reviews and strict monitoring and reporting.
- The EU and its Member States have developed and established a comprehensive regulatory framework to ensure the safety of nuclear installations, in line with international requirements and recommendations for enhancing regulatory systems for the control of nuclear installations throughout their lifetime. As contracting parties to the Convention on Nuclear Safety and to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, the EU and its Member States commit to a set of obligations and safety on a global scale, including those relating to their legislative and regulatory framework and regulatory bodies.
- The detailed assessment of the impacts of nuclear energy in its various lifecycle phases shows that all non-radiological effects and potential impact indicators are dominated by the mining & milling phase, except the greenhouse gas (GHG) emissions, where Nuclear Power Plant (NPP) operation gives the largest contribution (see Figure 3.3.1-12 of Part A and Tables A.2-1 and A.2-2 in Annex 2).
- The analyses did not reveal any science-based evidence that nuclear energy does more harm to human health or to the environment than other electricity production technologies already included in the Taxonomy as activities supporting climate change mitigation.
- The comparison of impacts of various electricity generation technologies (e.g. oil, gas, renewables and nuclear energy) on human health and the environment, based on recent Life Cycle Analyses (LCA) presented in Chapter 3.2 of Part A, shows that the impacts of nuclear energy are mostly comparable with hydropower and the renewables, with regard to non-radiological effects.
- For nuclear energy, its impact on water consumption and potential thermal pollution of water bodies must be appropriately addressed during the site selection, facility design and plant operation phases.
- With regard to potential radiological impacts on the environment and human health, the dominant lifecycle phases of nuclear energy significantly contributing to potential radiological impacts on the

environment and human health are: uranium mining and milling (ore processing); NPP operation (production of electricity by means of nuclear fission reactors); and reprocessing of spent nuclear fuel.

- Related analyses demonstrate that appropriate measures to prevent the occurrence of the potentially harmful impacts or mitigate their consequences can be implemented using existing technology at reasonable costs.
- Management of radioactive waste and its safe and secure disposal is a necessary step in the lifecycle of all applications of nuclear science and technology (nuclear energy, research, industry, education, medical, and other). Radioactive waste is therefore generated in practically every country, the largest contribution coming from the nuclear energy lifecycle in countries operating nuclear power plants. Presently, there is broad scientific and technical consensus that disposal of high-level, long-lived radioactive waste in deep geologic formations is, at the state of today's knowledge, considered as an appropriate and safe means of isolating it from the biosphere for very long time scales.
- Similarly, carbon capture and sequestration (CCS) technology is based on the long-term disposal of waste in geological facilities and it has been included in the taxonomy and received a positive assessment. The Taxonomy Expert Group therefore considers that the challenges of safe long-term disposal of CO_2 in geological facilities, which are similar to the challenges facing disposal of high-level radioactive waste, can be adequately managed. There is already an advanced regulatory framework in place in the communities for both carbon dioxide storage and radioactive waste management (see Annex 1). In terms of practical implementation, there is currently no operational geological disposal for carbon dioxide or for radioactive waste.
- Most of the LCA consulted are comprehensive, and include in their results the contribution of the disposal phase to the overall environmental impacts from both radiological and non-radiological aspects.
- From a non-radiological aspect, the disposal phase contributes only slightly to the overall greenhouse gas emissions, use of land, and generation of technological waste. It does not contribute (the results are zero or negligible) to those indicators representative of the impacts to the Taxonomy Regulation objectives of sustainable use and protection of water and marine resources, pollution prevention and control, and protection and restoration of biodiversity and ecosystems.
- With regard to the transition to a circular economy, the raw materials used to build the multiple engineered barriers of the disposal facilities (e.g. copper) cannot be recovered. The amounts needed are small, in particular when compared with the world production and the long timeframes of the disposal. Some materials resulting from the construction of facilities, e.g. part of the rock excavated to construct the tunnels of a crystalline rock repository, can be commercialized.
- Measures to ensure that radioactive waste does not harm the public and the environment include a combination of technical solutions and an appropriate administrative, legal and regulatory framework. Although there remain contrasting views, it is generally acknowledged, that the necessary technologies for geological disposal are now available and can be deployed when public and political conditions are favourable. No long-term operational experience is presently available as technologies and solutions are still in demonstration and testing phase moving towards the first stage of operational implementation. Finland, Sweden and France are in an advanced stage of implementation of their national deep geological disposal facilities, which are expected to start operation within the present decade.
- The radiological impact of nuclear energy lifecycle activities, including radioactive waste management and disposal, is regulated by law in the Member States, setting the maximum allowed releases and radioactivity exposure to the professionally exposed groups, to the public and to the environment. Respecting these limits, establishing the boundaries below which no significant harm is caused to human life and to the environment, is a precondition for any nuclear lifecycle activity to be authorized and is subsequently monitored by independent authorities.
- Provided that all specific industrial activities in the whole nuclear fuel cycle (e.g. uranium mining, nuclear fuel fabrication, etc.) comply with the nuclear and environmental regulatory frameworks and related Technical Screening Criteria, measures to control and prevent potentially harmful impacts on human health and the environment are in place to ensure a very low impact of the use of nuclear energy.
- An important outcome from the report is the demonstration of the development of appropriate Technical Screening Criteria (TSC) for nuclear energy-based electricity generation according to the approach practised by the TEG in their work. The TSC published here are preliminary proposals, illustrating that adequate criteria can be compiled to ensure that the application of nuclear energy does no significant

harm to human health and the environment. The process for developing the relevant TSC tables is outlined in Chapter 5 of Part A and some illustrative Technical Screening Criteria (TSC) for selected lifecycle phases of nuclear energy are given in Annex 4.

Main findings

The comparison of environmental impacts of various electricity generation technologies on human health and the environment, leads to the following main findings:

- Average lifecycle GHG emissions determined for electricity production from nuclear energy are comparable to the values characteristic to hydropower and wind (see Figure 3.2-6 of Part A);
- Nuclear energy has very low NO_x (nitrous oxides), SO_2 (sulphur dioxide), PM (particulate matter) and NMVOC (non-methane volatile organic compounds) emissions. The values are comparable to or better than the corresponding emissions from the solar PV and wind energy chains (see Figure 3.2-8 and -18 of Part A);
- With regard to acidification and eutrophication potentials, nuclear energy is also comparable to or better than solar PV and wind (see Figure 3.2-9 and -10 of Part A);
- The same is true for freshwater and marine eco-toxicity (see Figure 3.2-11 of Part A); ozone depletion and POCP (photochemical oxidant creation potential, see Figure 3.2-19 of Part A);
- Land occupation of nuclear energy generation is about the same as for an equivalent capacity gas-fired power plant, but significantly smaller than wind or solar PV (see Figure 3.2-15 of Part A).

Some areas where utilization of nuclear energy needs special attention were also identified:

- Potential thermal pollution of freshwater bodies: Large inland nuclear power plants utilizing once-through cooling systems withdraw a large amount of water from the river or lake used as ultimate heat sink for normal plant operation. When the heated-up cooling water is returned to the water body, it represents a significant thermal pollution potential that must be handled adequately. In order to avoid harmful thermal pollution effects, the maximum discharge temperature of the condenser cooling water, as well as the maximum temperature of the freshwater body after mixing have to be strictly controlled. Water withdrawal options and the avoidance of excessive thermal pollution must be carefully analysed during the site selection process.
- Water consumption: A general feature of power plants utilizing a specific thermal cycle to convert heat to mechanical energy (energy of the turbine) is the need for continuous cooling. While water consumption is very low for once-through cooling, technologies using recirculation cooling, evaporative cooling towers or pond cooling usually consume a significant amount of water to compensate for losses due to evaporation. Water consumption characterizing these cooling technologies remains comparable to concentrating solar power and coal, for both recirculation and pond cooling (see Figure 3.2-7 of Part A). During site selection, the available water resources and the potential environmental effects of excessive water consumption must be carefully analysed and an optimal solution must be implemented.

In addition to the analysis of state-of-the-art lifecycle assessment results, the impact of ionizing radiation on human health and the environment (see Chapter 3.4) and the potential impact of severe accidents (see Chapter 3.5 of Part A) have been discussed extensively. The corresponding main findings are as follows:

- The average annual exposure to a member of the public, due to effects attributable to nuclear energy-based electricity production is about 0.2 microsievert, which is ten thousand times less than the average annual dose due to the natural background radiation (see Figure 3.4-1 of Part A).
- According to the LCIA (Life Cycle Impact Analysis) studies analysed in Chapter 3.4 of Part A, the total impact on human health of both the radiological and non-radiological emissions from the nuclear energy chain are comparable with the human health impact from offshore wind energy.
- Potentially harmful effects of ionizing radiation to professionally exposed personnel are prevented by strict radioprotection measures, monitoring and limiting occupational doses. The ALARA (as low and reasonably achievable) principle is applied also to optimize plant maintenance works for minimizing worker's radiation doses.
- With regard to public exposure in case of accidents, severe accident fatality rates and maximum consequences (fatalities) are compared in Figure 3.5-1 of Part A. The current Western Gen II NPPs have a

very low fatality rate (≈5·10-7 fatalities/GWh). This value is much smaller than that characterizing any form of fossil fuel-based electricity production technology and comparable with hydropower in OECD countries and wind power (only solar power has significantly lower fatality rate).

- Severe accidents with core melt did happen in nuclear power plants and the public is well aware of the consequences of the three major accidents, namely Three Mile Island (1979, USA), Chernobyl (1986, Soviet Union) and Fukushima (2011, Japan). The NPPs involved in these accidents were of various types (PWR, RBMK and BWR) and the circumstances leading to these events were also very different. Severe accidents are events with extremely low probability but with potentially serious consequences and they cannot be ruled out with 100% certainty.
- After the Chernobyl accident, international and national efforts focused on developing Gen III nuclear power plants designed according to enhanced requirements related to severe accident prevention and mitigation. The deployment of various Gen III plant designs started in the last 15 years worldwide and now practically only Gen III reactors are constructed and commissioned. These latest technology developments are reflected in the very low fatality rate for the Gen III EPR design (≈8·10-10 fatalities/GWh, see Figure 3.5-1 of Part A). The fatality rates characterizing state-of-the art Gen III NPPs are the lowest of all the electricity generation technologies.
- The consequences of a severe accident at a nuclear power plant can be significant both for human health and the environment. Very conservative estimates of the maximum consequences of a hypothetical severe nuclear accident, in terms of the number of human fatalities, are presented in Chapter 3.5 of Part A and are compared with the maximum consequences of severe accidents for other electricity supply technologies.
- While the number of human fatalities is an obvious indicator for characterising the maximum severity of accident consequences, nuclear accidents can lead to other serious direct and indirect impacts that might be more difficult to assess. Whereas the public is well aware of the devastating consequences on property and infrastructure, as well as on the natural environment, from historical cases of anthropogenic catastrophes, the disaster and risk aversion might be perceived somehow differently for nuclear related events. Evaluating the effects of such impacts is not in the scope of the present JRC report, although they are important for understanding the broader health implications of an accident.
- The analyses outlined in Chapter 3 of Part A revealed some potentially harmful impacts of nuclear energy on human health and the environment. The implementation of specific measures, such as careful site selection, appropriate facility design and construction, as well as rigorous operation and waste management practices, as required by the applicable regulatory and legislative provisions, ensure that these potential impacts remain within established limits. Some of the impacts belonging to the three "dominant" lifecycle phases (mining & milling, NPP operation and reprocessing) need particular attention and management (see details in section 4.4 of Part A).

On the current status and perspectives of long-term management and disposal of radioactive waste and spent fuel, it can be stated that:

- Radioactive waste is generated during all stages of the nuclear energy lifecycle. A basic ethical requirement is the principle that the activities of today shall not cause negative impacts and shall not impose undue burdens on future generations. Radioactive waste management and in particular waste disposal aims at meeting this principle.
- The impact associated with the construction and operation of radioactive waste handling, transportation, storage and disposal facilities is essentially of conventional, non-radiological nature, and different studies estimate it as a small share of the overall impact of the entire fuel cycle.
- Although the geological disposal concepts can vary, the environmental impacts are dominated by the activities related to excavating the tunnels and building the multiple engineered barriers. The environmental impact analysis of the disposal facilities includes a description of the measures implemented to mitigate specific effects. Mitigation measures are considered also in the mining of raw materials needed to construct a repository (e.g. metals and bentonite for the engineered barriers) to limit the environmental impact of the disposal phase.
- The long-term potential impacts of radioactive waste relevant to the "do no significant harm" criteria, are of a radiological nature. Due to its potential to cause harm, radioactive waste and spent fuel must be managed aiming at radionuclide containment and isolation from the accessible biosphere for as long as

the waste remains hazardous. The maximum radioactive dose limits to humans and to the environment due to waste management activities and disposal facilities are set by the relevant regulations.

- In terms of volume, the largest fraction of the radioactive waste comes from the operation and decommissioning of nuclear power plants and associated nuclear fuel cycle activities. This is generally very low or low level waste.
- A significant portion of the potential radioactive waste is in fact non-radioactive or very slightly radioactive (primarily originating from decommissioning activities). If allowed by the national legal and regulatory framework, materials with radioactivity levels below clearance thresholds can be removed from regulatory control through a clearance process, i.e. it is no longer considered as radioactive waste and can be reused, recycled, or further managed as conventional waste. Some materials or equipment that cannot be removed from regulatory control can anyhow be authorised to be reused or recycled maintaining the regulatory control.
- Uranium mining and milling also produces large amounts of very low-level waste due to formation of waste rock dumps and/or tailings. These dumps and tailings are located close to the uranium mines and the related ore processing plants and their environmentally safe management can be ensured by the application of standard tailings and waste rock handling measures.
- In terms of radioactivity, the main contributors are spent fuel and high-level waste. These materials contain long-lived radionuclides which remain radioactive over a very long time – up to a hundred thousand years or more, encompassing many generations.
- The radioactive waste is collected and characterised to determine its physical, chemical and radiological properties, and then sorted and segregated depending on the management route, which depends on the properties of the waste and national strategy. Radioactive waste is treated and conditioned in preparation for disposal. Storage is a necessary step to allow for the decay of short-lived radionuclides, and to collect and accumulate a sufficient amount of radioactive waste for treatment, conditioning or disposal. Storage also ensures the safety of radioactive waste until the disposal facility starts its operation.
- The safety of radioactive waste and spent fuel during storage before disposal is ensured by adequate passive safety features (containment, shielding, etc.), but also relies upon active monitoring and control by the operators of the facilities.
- Very low and low level waste, as well as certain intermediate level waste are disposed of in surface or near surface disposal facilities that isolate the waste with engineered and natural barriers for a period of typically 300 years, after which the radioactivity has decayed to harmless levels. On such a timescale, the behaviour of the engineered barriers is well known and predictable, and they are considered sufficiently reliable. As part of the licensing process, the safety demonstration must prove that during the first 300 years, the doses to the public caused by any foreseeable circumstance (including extreme natural events and human intrusion) are kept below the limits established by the regulatory authority.
- Disposal of very low and low level waste in surface and near surface facilities is an industrial reality, and facilities have been constructed and operated in many countries. Some of them have completed their operation and have entered the institutional control phase. The mechanisms and processes put in place are robust, allow for the identification of non-safe situations and provide for the improvement of the safety of the disposal.
- Intermediate level waste that cannot be disposed of in surface or near surface facilities shall be disposed of at greater depths, in geological disposal facilities.
- For high-level radioactive waste and spent fuel, there is a broad consensus amongst the scientific, technological and regulatory communities that final disposal in deep geological repositories is the most effective and safest feasible solution which can ensure that no significant harm is caused to human life and the environment for the required timespan. The final disposal of spent fuel and radioactive waste in a repository foresees its emplacement in a multi-barrier (engineered and natural) system in a stable geologic formation several hundred metres below ground level. The specific configuration of the repository depends on the characteristics and radioactivity content of the waste. The multi-barrier configuration of the repository prevents radioactive species from reaching the biosphere over the time span required. In the absence of releases of radioactive species to the accessible biosphere, there is neither radiological pollution nor degradation of healthy ecosystems, including water and marine environments.

- The safety of deep geological repositories during operation includes active monitoring and control. The long-term safety of radioactive waste in the geological repository, especially after its closure, must not depend on any institutional control and must be based on inherent passive features. Passive features include engineered and natural barriers that do not require continuous supplies to active systems (e.g. electricity), periodic maintenance, replacement of parts, or permanent surveillance. In the case of a deep geological repository for final disposal of spent fuel and high-level waste, the structures of the facility and the natural media must perform their containment functions without external interventions for as long as necessary.
- The implementation of a deep geological repository to ensure that radioactive waste does not harm the public and the environment is a stepwise process, which includes a combination of technical solutions and a strong administrative, legal and regulatory framework. Each step is taken based on a documented decision-making process, in which relevant scientific and technical state of the art, operational experience, social aspects and updates in the legal and regulatory framework are incorporated. Compliance must be ensured and demonstrated for all the steps subjected to active monitoring by the operators and also for the very long-term duration associated with the final disposal of long-lived and high-level waste and spent fuel (post-closure phase). This process allows making decisions that are flexible, and allows deciding among different options for the way forward.
- With the partial exception of the so-called natural analogues (i.e. sites where natural nuclear reactors occurred billions of years ago), there is no empirical evidence generated by a radioactive waste disposal facility that has gone through the pre-operational, operational, and post-closure stages for the entire timeframe foreseen (up to a hundred thousand years or more for a deep geological repository). For this reason the safety of the disposal during the post-closure phase is demonstrated by a robust and reliable process which confirms that dose or risk to the public are kept below the established limits under all circumstances during the time scales of interest and in the absence of direct human monitoring and control.
- The safety demonstration includes calculations and models of the behaviour of the engineered barriers under different circumstances, of the release and transport of the radioisotopes through the barriers, of the effects of climate events, including extreme hydrogeological, seismic and other phenomena, and of the impacts on the human life and/or the environment of potential releases of radionuclides from the waste. The models and calculations represent the state of the art of the knowledge generated by several decades of study and research on all relevant properties and mechanisms that affect the entire disposal system. The analysis is underpinned by the application of the natural laws that govern the long-term behaviour of the geological bedrock and the evolution of the relevant external factors (e.g. the climate). The safety demonstration is thoroughly reviewed independently and critically by the regulatory authority, and the authorisation procedure includes the involvement of the local communities in the decision making process.
- The safety demonstration involves scenario analysis, model representation and developing an understanding of how likely, and under what circumstances, radionuclides might be released from a repository, and what would be the consequences of such releases for humans and the environment. A challenging feature of these studies is the very long timeframe and the complexity of the phenomena that govern the safety functions, as well as the treatment of uncertainties in the scenarios, in the models, and in the data. The safety demonstration provides quantitative indicators that are compared to the requirements of the regulations. The results can be expressed in terms of dose to humans as a function of time covering the reference case, which must yield values well below regulatory limits as illustrated in Figure 5.2.4-4 of Part B, and including *what-if* scenarios that consider very unlikely extreme circumstances, which might yield higher doses.
- The research, development and demonstration (RD&D) carried out in support of safe radioactive waste management, including disposal, is a key component of each National and International Programme. Given the long timescales and socio-political dimension, RD&D provides primarily the scientific basis for implementing safe radioactive waste management solutions, whilst also contributing to building stakeholder trust, public acceptance, and training for the next generations of experts.
- A significant research effort has been devoted to maximising the fraction of spent nuclear fuel that can be recycled in nuclear reactors and reducing the long-term radiotoxicity of HLW to be disposed of in the geological repository. Both aims are relevant to the environmental objective "Transition to a circular economy, waste prevention and recycling". Due to the fact that fast reactors allow multiple (re)cycling of the fractions of fuel/waste not consumed/burned, the final result of iterating this process would be an

almost complete use of the fuel and an increasingly reduced fraction of long-lived species (mostly in terms of the minor actinides content) in the irradiated fuel. Although essentially all steps of this process, also known as partitioning and transmutation, have been demonstrated at laboratory scale, the Technology Readiness Level is not yet corresponding to industrial maturity.

— A variety of tools and approaches is used to provide scientific evidence in support to safe disposal of radioactive waste. Representative waste forms, including real spent fuel and vitrified high-level waste, are studied in hot laboratory facilities to determine the relevant properties and behaviour of the waste exposed to combinations of simulated environmental features. Tailor-made analogues are used to investigate single effects and reactions. The study of natural analogues can yield very valuable information, for example, on the migration of radionuclides across a geological formation. Experiments carried out in underground research laboratories allow acquiring knowledge and data on the properties of the host rock and their impact in the migration of radionuclides. All the experimental data and knowledge are used to develop and validate models using state of the art codes. Modelling is extensively used to understand behaviours and trends observed experimentally and to obtain prediction capabilities for complex systems.

Quick guide

Part A describes relevant aspects to assess nuclear energy generation under the "do no significant harm" (DNSH) criteria and deals with the review of impacts corresponding to the various lifecycle phases of nuclear energy.

The structure of Part A of the report is the following:

— Chapter 1 contains the introductory part, outlining the motivation and objectives of the JRC report. It also describes the report's structure and the approach for its development.

— Chapter 2 introduces the basic processes, advantages and limits of lifecycle analysis. The purpose of this chapter is to provide information on the methodology, applicability, merits and limitations of the currently used LCA procedures, in order to highlight what can be expected from an LCA and what is beyond its scope.

— Chapter 3 constitutes the main body of Part A. First, it provides a concise comparison of the impacts of various electricity generation technologies: coal, oil, gas, hydropower, nuclear and renewables on the six environmental objectives of the Taxonomy (see subchapter 1.3.2) with the aim of illustrating the magnitude of the impacts of nuclear energy in comparison with the other electricity generation methods.

  The next section of Chapter 3 is devoted to the assessment of the environmental and human health impacts characterizing the individual lifecycle phases of nuclear energy. The following LC phases are discussed:

  - Uranium mining and uranium ore processing;
  - Conversion to uranium hexafluoride (UF_6) gas;
  - Enrichment of uranium;
  - Fabrication of UO_2 nuclear fuel;
  - Reprocessing of spent nuclear fuel;
  - Production of MOX fuel;
  - Nuclear power plant operations (this includes construction, electricity generation and long-term operation of NPPs, as well as NPP decommissioning and site remediation);
  - Management and disposal of radioactive and technological waste (in Part A only the related lifecycle analysis results are discussed).

  This impact assessment uses results from adequate lifecycle emission analyses (LCAs) carried out for electricity generation by means of various nuclear reactor types. The assessment discusses the "open" and "closed" fuel cycles, as well. The applied impact indicators are described in subchapter 1.3.2.

Subchapter 3.4 (Impact of ionizing radiation on human health) provides a brief overview of possible effects of ionizing radiation on human health, in order to put into perspective the anticipated effects of radioactive releases from various nuclear facilities.

Subchapter 3.5 is devoted to the assessment of the impacts resulting from potential severe accidents, also containing a comparison with other electricity generation technologies.

- Using the conclusions of the analyses outlined in Chapter 3, Chapter 4 provides a concise overview of the impact assessment results and formulates recommendations on the compatibility of nuclear energy with the basic principles and objectives of the Taxonomy. This section also uses some results of the analysis performed in Part B, dealing with the assessment of the impacts of radioactive waste management and disposal.
- Chapter 5 provides illustrative – preliminary – TSC tables for some lifecycle phases. Here only those LC phases were selected which provide dominant contribution to at least one of the impact categories used. The DNSH sections in these TSC tables were completed using the data and recommendations outlined in Chapter 4. The following lifecycle phases are covered in this section:
  - Uranium mining and ore processing;
  - NPP operation (electricity production);
  - Reprocessing of spent nuclear fuel;
  - Storage and disposal of radioactive waste (including interim storage and disposal of spent nuclear fuel).

The following annexes are relevant to Part A:

- Annex 1 – Description of legal and regulatory framework of nuclear energy.
- Annex 2 – Summary of LCA results for all lifecycle phases of nuclear energy.
- Annex 3 – NACE codes corresponding to main LC phases of nuclear energy.
- Annex 4 – Illustrative TSC tables.
- Annex 5 – Ionising radiation: definitions, units, biological effects and radiation protection.

Part B describes relevant aspects of the management of radioactive waste, with particular attention on the long-term management of spent fuel and high-level waste, along the lines envisaged by the Terms of Reference of the present Report.

The structure of Part B of the report is the following:

- Chapter 1 presents the objectives, main principles and a summary of the legal framework of the management of radioactive waste and spent fuel.
- Chapter 2 highlights the typologies and the classification of radioactive waste generated during the various steps of the nuclear fuel cycle described in part A, and summarizes the current global and EU radioactive waste and spent fuel inventories.
- Chapter 3 presents the strategies and technologies available for the management of radioactive waste, focusing especially on the processes rather than in the details of the technologies.
- Chapter 4 presents the different aspects of interim storage of radioactive waste and spent fuel as a necessary step prior to disposal.
- Chapter 5 is dedicated to the final disposal of radioactive waste and spent fuel. It addresses the surface and near-surface disposal of low-level short-lived radioactive waste and provides a schematic description of the main geological disposal concepts for HLW and spent fuel in Europe. The rationale and conceptual approach, the tools and criteria informing the validation and the implementation of deep geological repositories are described, together with specific safety criteria, and features associated with the safety case and long-term performance assessment.
- Chapter 6 describes the strong contribution of R&D to the development and the implementation of the long-term solutions for the management of radioactive waste, including a historical perspective, the main

scope of current research efforts, how research is organised in the EU, main actors, tools, trends, and future perspectives.

The following annexes are relevant to Part B:

- Annex 1 – Description of legal and regulatory framework of nuclear energy.
- Annex 6 – Long-term radioactivity and radiotoxicity of radioactive waste

PART A

Review of the state-of-the-art to assess nuclear energy generation under the “do no significant harm” (DNSH) criterion

1 Introduction, motivation, approach and structure

1.1 Introduction

According to the Final Report of the Technical Expert Group (TEG) on Sustainable Finance (March 2020, see Ref. [1-1]):

"*The EU's Action Plan on Financing Sustainable Growth (March 2018) called for the creation of a <u>classification system for sustainable activities or Taxonomy</u>. In May 2018, the European Commission issued a proposal for a regulation which sets out the obligations for investors and the overarching framework for the Taxonomy (Proposal for a regulation on the establishment of a framework to facilitate sustainable investment – hereafter, Taxonomy Regulation (TR)). This will be supplemented by delegated acts containing the technical screening criteria.*

The TEG was asked to <u>develop recommendations for technical screening criteria</u> which respond to the framework set out in the TR. The TEG mandate has been to focus on economic activities that can make a <u>substantial contribution to climate change mitigation or adaptation</u>, <u>while avoiding significant harm to the other environmental objectives.</u>

In December 2019, the co-legislators reached political agreement on the overarching Regulation."

Note that the Taxonomy Regulation has been officially adopted in June 2020, see Ref. [1-2].

1.1.1 Deliberations of the Taxonomy Expert Group on nuclear energy

Nuclear energy was not included in the EU Sustainable Finance Taxonomy [1-1] for various reasons, but in the Technical Annex [1-3], the TEG outlined also positive considerations on nuclear energy, acknowledging that it can certainly contribute to climate change mitigation. As an explanation for <u>not</u> including nuclear energy into the Taxonomy, the section <u>TEG deliberations on nuclear energy</u> of [1-3] states the following:

"The TEG assessed nuclear energy as part of its review on energy generation activities. Nuclear energy generation has <u>near to zero greenhouse gas emissions in the energy generation phase</u> and can be <u>a contributor to climate mitigation objectives</u>. Consideration of nuclear energy by the TEG from a climate mitigation perspective was therefore <u>warranted</u>.

The proposed Taxonomy regulation and thus TEG's methodology for including activities in the Taxonomy explicitly includes two equally important aspects, Substantial Contribution to one environmental objective and Do No Significant Harm (DNSH) to the other environmental objectives.

...

Evidence on the potential substantial contribution of nuclear energy to climate mitigation objectives was extensive and clear. <u>The potential role of nuclear energy in low carbon energy supply is well documented</u>.

On potential significant harm to other environmental objectives, including circular economy and waste management, biodiversity, water systems and pollution, the evidence about nuclear energy is <u>complex</u> and more difficult to evaluate in a taxonomy context. Evidence often addresses different aspects of the risks and management practices associated with nuclear energy. Scientific, peer-reviewed evidence of the risk of significant harm to pollution and biodiversity objectives arising from the nuclear value chain was received and considered by the TEG. Evidence regarding advanced risk management procedures and regulations to limit harm to environmental objectives was also received. <u>This included evidence of multiple engineered safeguards, designed to reduce the risks</u>. Despite this evidence, there are still empirical data gaps on key DNSH issues.

For example, regarding the long-term management of High-Level Waste (HLW), there is an international consensus that a safe, long-term technical solution is needed to solve the present unsustainable situation. A combination of temporary storage plus permanent disposal in geological formation is the most promising, with some countries are leading the way in implementing those solutions. <u>Yet nowhere in the world has a viable, safe and long-term underground repository been established</u>. It was therefore <u>infeasible</u> for the TEG to undertake a robust DNSH assessment as <u>no permanent, operational disposal site for HLW exists yet from which long-term empirical, in-situ data and evidence to inform such an evaluation for nuclear energy.</u>

Given these limitations, it was not possible for TEG, nor its members, to conclude that the nuclear energy value chain does not cause significant harm to other environmental objectives on the time scales in question. The TEG has therefore not recommended the inclusion of nuclear energy in the Taxonomy at this stage. Further, the TEG recommends that more extensive technical work is undertaken on the DNSH aspects of nuclear energy in future and by a group with in-depth technical expertise on nuclear life cycle technologies and the existing and potential environmental impacts across all objectives."

During the summer of 2020 – after compiling an appropriate Terms of Reference document – DG FISMA of the European Commission (in agreement with DGs ENER, ENV, RTD, CLIMA and the Secretariat-General) requested JRC to carry out this "more extensive technical work on the DNSH aspects of nuclear energy" as recommended by the TEG.

1.2 Main tasks defined in the Terms of Reference document

The Terms of Reference (ToR) document defines the following main tasks to be implemented (see Ref. [1-4] for details):

"*Conduct a review of the state-of-the-art to assess nuclear energy generation under the "do no significant harm" (DNSH) criterion."*

"The assessment should consider the effects of the whole nuclear life cycle on the existing and potential environmental impacts across all objectives. As per the TEG recommendations, special attention should be given to impacts on the objectives relating to circular economy, pollution and biodiversity criteria; but ensuring the protection of water and marine resources is also very important and should be considered."

"For this task it is deemed relevant to consider the process followed by the TEG to determine the technical screening criteria."

"After establishing that a given activity could make a substantial contribution to the climate objectives, the TEG screened activities that could risk doing significant harm to one of the four (non-climate) environmental objectives. It followed a full life-cycle approach, to avoid errors such as considering an activity sustainable with a negative effect during a given stage (upstream or downstream)."

"...the Final Report of the TEG [1-1] includes comments on the impact that other energy sources (i.e. solar PV, wind power, hydropower) have on the four environmental objectives, which should be used as a minimum basis for the nuclear energy assessment:

- Protection of water and marine resources (water deterioration, changes to hydrological regimes)

- Transition to a circular economy (production and end of life management of materials and components)

- Pollution prevention and control (high emissions to air, water and land compared to thresholds included in current regulation)

- Protection and restoration of biodiversity and ecosystems (impacts on areas with high biodiversity values, disturbance or collision of animals)"

"...The technical assessment should gather and present evidence that helps evaluating the existing problems the pros and cons of existing and proposed solutions with a specific focus on the risks and nature of potential environmental impacts over the timescales[1] commensurate with long term nuclear waste management, treatment and storage."

The structure of the report and the approach selected by the JRC to carry out the analyses envisaged in the Terms of Reference are outlined in subchapter 1.3.

1.3 Structure and approach

The ToR prescribed that "the JRC should draw on its broad range of technical experts to produce **one in-depth report** assessing nuclear energy under the "do no significant harm" criterion". For practical and editorial reasons during the development of the JRC report it was decided to deliver the report in two separate parts (Part A and B), supplemented by several annexes, among them a common annex describing the legal

[1] Safe long-term management of radioactive waste must ensure that potential environmental impacts over the decades, centuries and even millennia following the closure of a deep-geological repository are acceptable.

and regulatory framework of nuclear energy, including the long-term management of spent fuel and high-level radioactive waste.

Part A is titled Review of the state-of-the-art to assess nuclear energy generation under the “do no significant harm” (DNSH) criterion and it deals with the review of impacts corresponding to the various lifecycle phases of nuclear energy.

Part B is titled Specific assessment on the current status and perspectives of long-term management and disposal of radioactive waste and it deals with the state-of-the-art and DNSH aspects of radioactive waste management, focusing on the final disposal of high level radioactive waste and spent nuclear fuel.

During the preparatory work the need for a third – legal – part became obvious for the authors. Entitled Legal and regulatory background of nuclear energy, it is a common background document for the two parts dealing with the technical issues.

It outlines the main elements in the associated legal and regulatory frameworks, with focus on regulating nuclear safety, the associated environmental impacts, nuclear safeguards and security in the EU. Its main purpose is to recall that the EU has established the necessary legal and regulatory framework to ensure the safe and secure operation of nuclear facilities, and the appropriate limitation of environmental and other impacts of nuclear energy.

It is attached to the present document as Annex 1.

1.3.1 Structure of Part A

The structure of Part A of the report is the following:

— Chapter 1 contains the introductory part, outlining the motivation and objectives of the JRC report. It also describes the report’s structure and the approach for its development.

— Chapter 2 introduces the basic processes, advantages and limits of lifecycle analysis. The purpose of this chapter is to provide information on the methodology, applicability, merits and limitations of the currently used LCA procedures, in order to highlight what can be expected from an LCA and what is beyond its scope.

— Chapter 3 constitutes the main body of Part A. First, it provides a concise comparison of the impacts of various electricity generation technologies: coal, oil, gas, hydropower, nuclear and renewables on the six environmental objectives of the Taxonomy (see subchapter 1.3.2) with the aim of illustrating the magnitude of the impacts of nuclear energy in comparison with the other electricity generation methods.

— The next section of Chapter 3 is devoted to the assessment of the environmental and human health impacts characterizing the individual lifecycle phases of nuclear energy. The following LC phases are discussed:

  - uranium mining and uranium ore processing;
  - conversion to uranium hexafluoride (UF_6) gas;
  - enrichment of uranium;
  - fabrication of UO_2 nuclear fuel;
  - reprocessing of spent nuclear fuel;
  - production of MOX fuel;
  - nuclear power plant operations (this includes construction, electricity generation and long-term operation of NPPs, as well as NPP decommissioning and site remediation);
  - management and disposal of radioactive and technological waste (in Part A only the related lifecycle analysis results are discussed).

  This impact assessment uses results from adequate lifecycle emission analyses (LCAs) carried out for electricity generation by means of various nuclear reactor types. The assessment discusses the “open” and “closed” fuel cycles, as well. The applied impact indicators are described in subchapter 1.3.2.

- Subchapter 3.4 (Impact of ionizing radiation on human health) provides a brief overview of possible effects of ionizing radiation on human health, in order to put into perspective the anticipated effects of radioactive releases from various nuclear facilities.
- Subchapter 3.5 is devoted to the assessment of the impacts resulting from potential severe accidents, also containing a comparison with other electricity generation technologies.
- Using the conclusions of the analyses outlined in Chapter 3, Chapter 4 (Summary DNSH assessment for nuclear energy and recommendations) provides a concise overview of the impact assessment results and formulates recommendations on the compatibility of nuclear energy with the basic principles and objectives of the Taxonomy. This section also uses some results of the analysis performed in Part B, dealing with the assessment of the impacts of radioactive waste management and disposal.
- Chapter 5 (Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy) provides illustrative – preliminary – TSC tables for some lifecycle phases. Here only those LC phases were selected which provide dominant contribution to at least one of the impact categories used. The DNSH sections in these TSC tables were completed by using the data and recommendations outlined in Chapter 4. The following lifecycle phases are covered in this section:
    - uranium mining and ore processing;
    - NPP operation (electricity production);
    - reprocessing of spent nuclear fuel;
    - storage and disposal of radioactive waste (including interim storage and disposal of spent nuclear fuel).

The following annexes are relevant to Part A:

- Annex 1 describes legal and regulatory background of nuclear energy.
- Further Annexes contain supporting materials, numerical tables, etc., as follows:
    - Annex 2 – Summary of LCA results for all lifecycle phases of nuclear energy
    - Annex 3 – NACE[2] codes corresponding to main LC phases of nuclear energy
    - Annex 4 – Illustrative TSC tables
    - Annex 5 – Ionising radiation: definitions, units, biological effects and radiation protection.

1.3.2 Details of the approach selected

1.3.2.1 The specificities of nuclear energy

Industrialisation has undoubtedly brought great benefits to mankind. Among them, the access to reliable sources of electricity has resulted in very high living standards and increased life expectancy. However, all our industrial activities have an environmental footprint, from the greenhouse gases emitted in the production of concrete, steel and other materials required for construction, to the diesel emissions from the trucks used to transport materials, to the chemical emissions from industrial processes and the destruction of natural habitats to make way for industry, to name but a few examples. In fact, all human activities have an environmental footprint, including those linked to basic survival needs, such as farming. In many cases the environmental impact has generally been tolerated, or not identified as a priority requiring immediate action, on the basis that the benefits are considered to outweigh the disadvantages. However, it has now become evident, especially in relation to the potential damages caused by climate change, that some industrial activities cannot continue as they are and that we need to start doing things in a more sustainable way.

All electricity generation technologies, like other industrial activities, interact with our environment. They do so in different ways (for example by emitting different pollutants or by using different natural resources) and to different extents, some much more than others. Nuclear energy is no exception. In Chapter 3.2 of this report, nuclear electricity generation is compared with some other electricity generation technologies with regard to different environmental impact categories.

[2] NACE = Statistical classification of economic activities in the EC

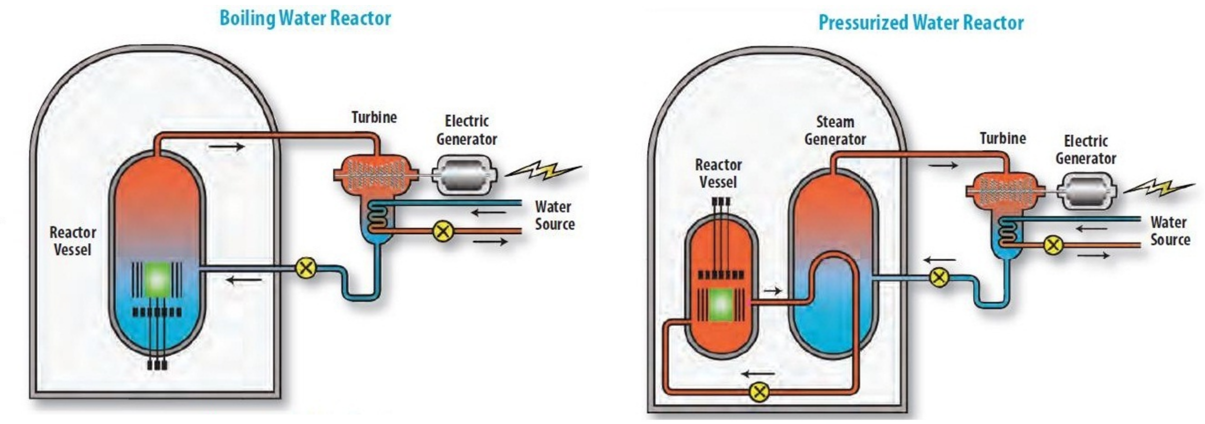
What sets nuclear energy apart from other electricity generation technologies is its association with ionising radiation[3] and radioactive substances[4], an association which attracts considerable public attention.

A nuclear power plant is an electricity production facility utilizing the nuclear fission process that generates heat from the nuclear fuel. The heat is then transferred to the coolant medium and converted to electricity through appropriate technological processes (usually by using a steam turbine driving an electric generator, see Figure 1-1).

A simplified scheme of an NPP can be depicted as a conventional power plant, where the "boiler" part applied for combusting gas, oil, coal, biomass, etc. has been replaced by a nuclear reactor, accommodated in specially constructed reinforced buildings forming the so called nuclear island of the NPP. Outside of the nuclear island the applied equipment and the characteristic technological processes do not essentially differ from those in conventional power plants, i.e. the main steam system, the turbine with its auxiliaries, the condenser, the cooling water inlet and discharge works, the generator, the transformers, the electric switchyards and the power transmission lines are the same in both cases.

Fundamental differences exist between the fuel extraction/production and waste treatment in a conventional power plant and an NPP, therefore no analogies can be used to develop appropriate Technical Screening Criteria (TSC) for these NPP lifecycle phases.

Figure 1-1. Operating scheme of the two most widely used reactor types



Source: Nuclear Energy Institute (NEI), https://www.nei.org/home

The front-end of the nuclear fuel cycle (uranium mining and milling, conversion, enrichment and nuclear fuel manufacturing) is an entirely nuclear-specific activity, which must be handled separately and must have a unique TSC set. The same is true for the back-end of the nuclear fuel cycle, where the so called "closed" and "open" cycles must be distinguished and separately handled. In the closed cycle reprocessing of spent nuclear fuel (SF) is performed, with or without fabrication of MOX[5] fuel. In the open cycle no reprocessing takes place and after a temporary storage period the SF is to be disposed at a final disposal facility.

The major difference between nuclear and conventional power plants is the presence of radioactive materials in the NPP during its operation and decommissioning phases. The irradiated nuclear fuel is highly radioactive, and during reactor operations waste containing radioactive nuclei is also generated. Radioactive nuclei are primarily created in the nuclear fuel as fission products from the fission process, but structural materials of the reactor may also become radioactive through neutron activation, induced by neutrons escaping the fuel. Radioactive nuclei may emit alpha, beta or gamma radiation, or even neutrons, depending on the type of radioactive decay involved. All these radiation types have harmful effect on humans and the biota, although to a different extent and the nature and severity of harm depends on the intensity of ionizing radiation.

[3] Any radiation capable of displacing electrons from atoms or molecules, thereby producing ions. Some examples are alpha, beta, gamma, x-rays, neutrons, and ultraviolet light. High doses of ionizing radiation may produce severe skin or tissue damage (http://www-naweb.iaea.org/nafa/aph/resources/nuclearglossary-APH.pdf).

[4] Material designated in national law or by a regulatory body as being subject to regulatory control because of its radioactivity (IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Edition, https://www.iaea.org/publications/11098/iaea-safety-glossary-2018-edition)

[5] In contrast to the UOX (uranium oxide) fuel – which is currently the most widely used nuclear fuel type – the MOX (mixed oxide) fuel also contains plutonium oxide, which is mixed with uranium oxide (see Chapter 3.3.6 for details).

Radiation levels above certain – scientifically established – thresholds are definitely harmful and therefore in an NPP appropriate measures are taken to protect the operating personnel, the public and the environment from the harmful effects of radioactive materials. The appropriate protection is ensured by the design of the facility, by operation and maintenance rules, strict measures for controlling the discharge of radioactive gases and effluents, as well as legal instruments and regulations, including overarching regulatory supervision during all lifecycle phases of the NPP. Radiation protection and discharge control generally relies on the application of the ALARA (as low as reasonably achievable) or ALARP (as low as reasonably practicable) principle, which is an internationally acknowledged method to minimize the radiation effects of NPP operation[6].

The amount and impact of ionising radiation from the nuclear power lifecycle will be discussed further in several chapters of this report. Chapter 3.4 describes the impact of radiation on human health and the environment; Chapter 2.4 of Part B and Annex 6 describes radioactivity and radiotoxicity, as well as the main natural radionuclides and those present in radioactive waste; Annex 5 illustrates ionising radiation definitions, units, biological effects and basic principles of radiation protection..

1.3.2.2 The environmental objectives of TEG

The Taxonomy Expert Group applies the following six environmental objectives, which correspond to those in the Taxonomy Regulation (see [1-3]):

- Climate change mitigation;
- Climate change adaptation;
- Sustainable use and protection of water and marine resources;
- Transition to a circular economy, waste prevention and recycling;
- Pollution prevention and control;
- Protection of healthy ecosystems (protection and restoration of biodiversity and ecosystems).

1.3.2.3 DNSH assessment of economic activities

The "Do No Significant Harm" (DNSH) analysis is an integral part of any sustainability analysis, since one cannot declare an activity "compliant" if it supports one of the objectives but undermines other objectives.

The DNSH analysis is to ensure that the technical screening criteria (TSC) and the Taxonomy itself do not include economic activities undermining any of the environmental objectives. The approach applied by the TEG focused on identifying practices and criteria through which potential harm to environmental objectives can be mitigated. In cases where the TEG could not identify practices or criteria to mitigate an identified potential harm, then the activity was not included in the Taxonomy (see Ref. [1-3] for more details). In addition, Ref. [1-6] provides descriptions of the Taxonomy usage and also contains some examples for DNSH analyses.

In our understanding the Taxonomy is not a tool for assessing the safety of the related industrial facilities or to provide an in-depth analysis of their predicted environmental impacts. These issues must be appropriately covered by the safety analysis report and the Environmental Impact Assessment (EIA) of the facility. The main function of the DNSH analysis in the frame of the Taxonomy is to define the conditions under which economic activities are considered not to be detrimental to the achievement of the various environmental objectives of the Taxonomy. The criteria applied in the DNSH assessment must be based on an adequate and thorough analysis of the potential environmental impacts of the economic activity under investigation, in order to ensure that the conditions for its acceptance/rejection will be defined appropriately.

The TEG used the following approach to perform the DNSH and define TSC associated with specific industrial activities. First the corresponding lifecycle impact assessment has to be reviewed and the potentially significant environmental impacts occurring during the whole life cycle have to be identified. After having identified the potentially harmful effects thorough the whole lifecycle, it has to be decided whether these impacts can be successfully prevented or mitigated or not. If not, then the activity cannot be part of the Taxonomy, it has to be eliminated.

If there are viable and well-proven practices or criteria which are applicable to mitigate the impacts then the activity can be included in the Taxonomy, provided that the realized installation applies the mitigating

6 See e.g. https://www.radiation-dosimetry.org/what-is-alara-and-alarp-principle-definition/ for explanations

practices or fulfils the criteria. According to the nomenclature of the Taxonomy, these conditions are formulated as Technical Screening Criteria (see [1-3] for many examples).

The applicable mitigation practices are known to the experts working in the specific industry, while the criteria can be derived from the relevant EU Directives, standards, the BAT (Best Available Technologies) Reference Documents or other acknowledged reference documents (see e.g. Refs. [1-7] and [1-8] as examples).

1.3.2.4 Development of technical screening criteria

Our approach to define TSCs for the individual nuclear energy lifecycle phases basically followed the process taken by the TEG for developing the TSCs (see Ref. [1-3]). Potentially harmful impacts of nuclear energy based electricity generation were identified by using results from relevant LCAs (lifecycle analyses) and by analysing the underlying technological processes. The selected analyses covered all lifecycle phases of nuclear energy and treated both open and closed fuel cycles. In order to characterize environmental and human health impacts, internationally acknowledged and widely used impact indicators were applied.

The following internationally accepted impact indicators were used to characterize the non-radioactive impacts of nuclear energy:

- green-house-gases emissions (GHG);
- atmospheric pollution (SO_x and NO_x);
- water pollution;
- land use;
- water consumption and withdrawal;
- production of technological waste;

Moreover, the following impact indicators were used to take into account the nuclear-specific impacts of nuclear energy (these are the so-called "radiological impacts"):

- gaseous radioactive releases;
- liquid radioactive releases;
- solid radioactive waste production.

Additional, internationally applied impact indicators were also used, such as acidification and eutrophication potentials, photochemical ozone formation potential, eco-toxicity and human toxicity, resource use) to facilitate the comparison of results published in various studies. When available, particulate matter emissions are taken into account, because these can also contribute to radioactive contamination e.g. by dusting. The analysis also reviewed relevant legal aspects and regulations, focusing on EU Directives and industry-specific standards.

The above described analyses were documented in Chapter 3.3 of Part A, constituting the "Do No Significant Harm" (DNSH) analysis section of our study.

By using the results and conclusions of the above analyses, one can derive and synthesize data and other information (e.g. applicable standards or relevant best available techniques) required to fill in the corresponding DNSH sections in the TSC tables defined for the various lifecycle phases of nuclear energy. The following TEG environmental objectives are addressed in the TSC tables:

(2) Adaptation = climate change adaptation;

(3) Water = protection of water and marine resources;

(4) Circular Economy = transition to a circular economy;

(5) Pollution = pollution prevention and control;

(6) Ecosystems = protection and restoration of biodiversity and ecosystems.

The fulfilment of the first environmental objective ((1) = Climate change mitigation) is determined from the magnitude of the associated GHG emissions and the Taxonomy uses it to decide whether a specific electricity generation technology can be included into the Taxonomy or not. The final TEG report [1-1] states that "*Any electricity generation technology can be included in the Taxonomy if it can be demonstrated, using an ISO*

14067 or a GHG Protocol Product Lifecycle Standard compliant Product Carbon Footprint (PCF) assessment, that the life cycle impacts for producing 1 kWh of electricity are below the declining threshold". (The threshold is currently set to 100g CO_{2e}/kWh). Note that the ISO 14067 standard is focusing on the determination of the carbon footprint of a product, and it is fully consistent with the ISO 14040 and ISO 14044 international standards on life cycle assessment (LCA).

During the development of the TSC the relevant non-nuclear criteria were complemented by criteria accounting for the radiation protection and radioactive emission control aspects of nuclear energy.

The relevant EU directives and regulations – together with the national laws and regulations in effect – are considered as legal obligations to be compulsory satisfied in the EU and their fulfilment is a minimum condition.

1.4 References for Chapter 1

[1-1] Financing a Sustainable European Economy, Technical Report, Taxonomy: Final report of the Technical Expert Group on Sustainable Finance, March 2020

[1-2] Taxonomy Regulation: Regulation (EU) 2020/852 of the European Parliament and of the Council of 18 June 2020 on the establishment of a framework to facilitate sustainable investment, and amending Regulation (EU) 2019/2088

[1-3] Taxonomy Report, Technical Annex, Updated methodology & Updated Technical Screening Criteria, March 2020

[1-4] Terms of reference for a technical assessment implemented by the JRC on Nuclear energy under the "Do no significant harm" criterion, EC document ARES(2020)3473004, 2 July 2020

[1-5] Radiation effects and sources, United Nations Environment Programme, ISBN 978-92-807-3517-8, 2016.

[1-6] Using the Taxonomy, Supplementary Report 2019 by the Technical Expert Group on Sustainable Finance, 2019

[1-7] T. Lecomte et al.: Best Available Techniques (BAT) Reference Document for Large Combustion Plants; Report EUR 28836 EN, JRC Science for Policy Report, 2017

[1-8] Directive (EU) 2015/2193 on the limitation of emissions of certain pollutants into the air from medium combustion plants

2 Lifecycle assessment: methods, benefits and limitations

This chapter introduces the basic processes, advantages and limits of life cycle analysis (LCA). The purpose of this chapter is to provide information on the methodology, applicability, merits and limitations of the currently used LCA methods, in order to highlight what can be expected from LCA and what is beyond its scope.

The information outlined here helps the Reader to understand the main steps of the LCA process and facilitates the proper interpretation of the details and conclusions of the technical assessments outlined in Chapter 3.

2.1 Brief overview of LCA

2.1.1 Short history of LCA

The idea of life cycle analysis (LCA) emerged in the 1960s from concerns about the environmental impacts of alternative products. In the late 1960s, an internal study for a well-known beverage company in the USA, comparing the impacts of its packaging products, laid the foundations for current methods. The study considered not only the use but also production, transportation and disposal of the product [2-1]. Although the analysis primarily focused on a single-use beverage package, the life cycle inventory (LCI) approach began to gain importance across the USA and Europe.

During the 1970s, many companies worldwide developed similar methods of LCI comparisons aiming at energy analysis, environmental resources requirements, emissions as well as waste generation. The focus shifted to concerns on limitations of energy resources and materials in a broader sense. However, the increased interest required a common theoretical framework. Between 1970 and 1975, in the USA, the Environmental Protection Agency (EPA) developed a protocol for quantifying releases to the environment and for characterizing the use of resources of products, standardized as Resource and Environmental Profile Analysis (REPA)[7]. In parallel, the environmental dimension and life cycle measures became an integrated part of all areas of European Commission Policies following the establishment of the Directorate-General for the Environment in 1973[8].

Over time, the life cycle concept has proven to be a suitable tool for environmental comparison of product value chains around the world. However, its assumptions and techniques have evolved in a non-harmonized manner. Therefore, in the 1990s, the International Standards Organization (ISO) formally created the 14000 series of standards that cover life cycle assessment methods. In 2002, to support life cycle thinking more practically, the United Nations Environment Programme (UNEP) together with the Society of Environmental Toxicology and Chemistry (SETAC) established the Life Cycle Initiative. The Initiative provides a global platform for the tools, data and indictors supporting the development of scientific consensus and exchange of best practices[9]. Consequently, in 2005, the European Platform on Life Cycle Assessment[10] was established. The initiative implements the International Life Cycle Data (ILCD) system to promote the availability, exchange and use of quality-assured life cycle data, methods and studies.

Today, the LCA is a widely accepted method supporting decision makers in capturing the overall environmental impacts associated with any given activity, from raw material acquisition, through the production and use phases, to the final disposal of all residuals back to the earth.

2.1.2 Scope and main steps of LCA

The goal of LCA is to quantify the potential environmental impacts of a given product (or activity) during its entire lifespan. To ensure consistency of such assessments, ISO 14040:2006 (Environmental management — Life cycle assessment — Principles and framework) lays down a systematic standardized approach consisting of four basic steps, as described in Table 2-1.

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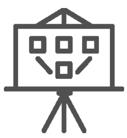
7 For example: Resource and Environmental Profile Analysis of Nine Beverage Container Alternatives: Final Report, EPA/530/SW-91C/1974

8 For example: Directive 85/339/EEC of 27 June 1985 on containers of liquids for human consumption, which provided measures related to the production, use, recycling, refilling and disposal of liquid food containers

9 See: https://www.unenvironment.org/explore-topics/resource-efficiency/what-we-do/life-cycle-initiative

10 See: https://eplca.jrc.ec.europa.eu/

Table 2-1. Framework for life cycle assessment

	**Goal Definition and Scoping**: Sets the frame of the analysis and defines all the detailed aspects, such as: — Purpose and method; — System boundaries; — Data requirements for all inputs and outputs across all stages of product life cycle; — Organization of results.
	**Life Cycle Inventory**: Collects all relevant data of the process flows, such as: — Materials; — Energy; — Emissions; — Waste; and assesses how these flows affect the environment.
	**Life Cycle Impact Assessment:** Calculates the potential impact on human health and environment, as well as addressing resource depletion. The phase consist of 4 steps: — Selection and classification of the relevant impacts according to the impact categories; — Characterization of the potential impact using science-based conversion factors; — Normalization of the potential impacts in a manner that allows comparison; — Weighting according to the most important potential impacts.
	**Life Cycle Interpretation:** Analyses and interprets the results of the life cycle assessment in order to answer questions outlined in the goal definition and provides comprehensive conclusions or recommendations. Other elements to be considered in the analysis include: — Assumptions and data, including engineering estimates; — Sensitivity analysis associated with each alternative and its relative magnitude; — Consistency check; — Limitations and constraints of the analysis.

Source: elaborated from [2-1] and [2-2].

2.1.3 Benefits and limitations of LCA

LCA allows decision makers to compare and to select the product or process that result in the least impact to the environment and human health, when deciding between two or more options. It provides a holistic view on the environmental impacts through all life cycle stages and thus identifies hotspots that point to possible improvements in the process to achieve environmental benefits.

The method is widely recognized and the framework for conducting the assessment builds on internationally accepted standards. However, the scope and the implemented impact assessment method can vary between studies and hence the comparability of the resulting data is often limited. The scope defines which activities or processes actually relate to the system being analysed and guides the data collection effort. In fact, the data collection is the most time and resource consuming phase of the LCA as it requires a large amount of data. If not enough data are available, assumptions, engineering estimates, and decisions need to be made based on the stakeholders values. Frequently, the information gathered is based on empirical experience following the use of the products. Some products have been thoroughly studied, while others less so. For example, performing the LCA on new technology systems that are still in the research and development phase can be challenging [2-3].

Several methods for the quantification of the impacts have been developed focusing on different impact indicators (see Chapter 2.2.1). All assumptions and scenarios must be clearly reported along with the results. In addition, it is important to realize that not all environmentally relevant information can be quantified. In this case, the LCA represents benefits or drawbacks of each alternative. The final interpretation of the results is essential for a better understanding of the environmental and health impacts associated with each alternative. It should be noted, that it does not determine which alternative is better. Rather, the results reveal which alternative performs better on certain impacts.

Further special attention needs to be paid to the allocation of recycling as part of the life cycle approach [2-4]. Another noteworthy issue is that LCA only considers impacts related to the normal and abnormal operation of processes and products. Hence, the assessment does not cover impacts from accidents[11] or spills. Finally, depending on the system boundaries, it usually excludes social and other workplace related aspects, such as workplace-exposure and indoor-emissions [2-2].

2.2 Life Cycle Impact Assessment (LCIA)

2.2.1 The most common LCIA methodologies

As presented in Table 2-1, the Life Cycle Impact Assessment (LCIA) determines the relative impact of the potential to cause harm to humans and environment. There are a number of scientifically based methods for calculating the impacts, which usually consist of four steps [2-5]. The first step selects the impacts on human health, the natural environment, and the availability of natural resources that will be considered as part of the overall LCA. Impacts are divided into impact categories. The most common impact categories are climate change, ozone depletion, photochemical ozone formation, respiratory inorganics, ionising radiation, acidification, eutrophication, human toxicity, ecotoxicity, land use and resource depletion. The purpose of the categories is to classify identified inventory items. Table 2-2 provides examples of different inventory items and their linkage to the impact categories.

In the second step, the impact of each emission or resource consumption is quantitatively modelled. The framework uses characterization factors to convert the inventory results into representative indicators determining impact scores. This generally provides two different types of indicator, so-called mid-point and end-point impact indicators. Mid-point indicators characterise contributions to the different environmental issues at some intermediate point in the cause-effect chain, whereas end-point approaches go a step further with the aim of assessing the actual damage resulting from these contributions.

ISO 14040:2006 states that these first two steps are mandatory for each LCIA. The next step is optional and is called normalisation. Normalisation associates impact scores with a common reference. This facilitates comparison between impact categories. The last step, which is also optional, is weighting, which assigns relative weights to the different impact categories and ranks them according to their perceived importance or relevance. This step may be necessary when comparing between different alternatives to evaluate trade-off situations.

Since the early 1990s many LCIA methods have been developed and their scope has evolved over time. The assessments focused primarily on the burden associated with emissions to the environment and resources. Later the cost assessment was included, considering the complete supply chain, and today the assessment may be supplemented by impact categories focusing on social aspects. Another difference among LCIA approaches is the different geographical scope or different fields of applications. The International Reference Life Cycle Data System (ILCD) reviews the existing methods in its Handbooks and develops a set of recommendations for their use. With regard to the ILCD analysis, Table 2-2 presents the most suitable method for each category in the European context and the following paragraphs briefly describe the purpose of the different methods. Relevant information and references to each method can be found in [2-9] along with recommendations on the use of the methods for each category in [2-8].

11 Usually leakages, spills and other types of releases potentially caused by accidents are not included as part of the normal life cycle inventory since they are fundamentally different in nature from the production or operation related normal and abnormal operating conditions that LCA relates to. Work on Life Cycle Accident Assessment is still under development (see Ref. [2.1-2] for more details).

Table 2-2. Commonly used LCIA categories, examples of inventories linked to each category and recommended methods for quantification of the impacts with their respective characterisation factor

Impact category	**Examples of inventories**	**Recommended method by ILCD**
Climate change	Carbon Dioxide (CO_2) Nitrogen Dioxide (NO_2) Methane (CH_4) Chlorofluorocarbons (CFCs)	Baseline model of 100 years of the IPCC (Global Warming Potential)
Stratospheric Ozone Depletion	Chlorofluorocarbons (CFCs) Hydrochlorofluorocarbons (HCFCs) Halons Methyl Bromide (CH_3Br)	EDIP99 (Ozone Depleting Potential)
Acidification	Sulfur Oxides (SO_x) Nitrogen Oxides (NO_x) Hydrochloric Acid (HCL) Hydrofluoric Acid (HF) Ammonia (NH_4)	Accumulated Exceedance (Acidification Potential)
Eutrophication	Phosphate (PO_4) Nitrogen Oxide (NO) Nitrogen Dioxide (NO_2) Ammonia (NH_4)	Accumulated Exceedance (Eutrophication Potential)
Photochemical Smog	Non-methane hydrocarbon (NMHC)	ReCiPe (Photochemical Oxidant Creation Potential)
Terrestrial Toxicity	Toxic chemicals with a reported lethal concentration to rodents	USEtox (Comparative Toxic Unit for ecosystems)
Aquatic Toxicity	Toxic chemicals with a reported lethal concentration to fish	USEtox (Comparative Toxic Unit for ecosystems)
Human Toxicity	Total releases to air, water, and soil	USEtox, (Comparative Toxic Unit for humans)
Resource Depletion	Quantity of minerals used Quantity of fossil fuels used	CML 2002 (Resource Depletion Potential/Scarcity)
Land Use	Quantity disposed of in a landfill or other land modifications	SOM (Soil quality indictor)
Water Use	Water used or consumed	Ecopoints 200 (Water Shortage Potential)
Particulate Matter/ Respiratory inorganics	Sulphur Dioxides (SO_2) Nitrogen Oxides (NO_x) Solid and liquid particulates Non-methane volatile organic compounds (NMVOC)	RiskPoll (Intake fraction for fine particles)
Ionising radiation, human health	Routine atmospheric and liquid releases in the nuclear fuel cycle	Dreicer et al. 1995 [2-6] (Human exposure efficiency relative to ^{235}U)
Ionising radiation, ecosystems	Radioactive releases to freshwater and its sediments	Garnier-Laplace et al. 2006 [2-7] (Comparative toxic unit for ecosystems)

Source: elaborated from [2-1] and [2-8].

2.2.1.1 Methods covering different impacts

The method that marked a milestone in the development of LCIA in Europe is based on the CML 1992 LCA Guide & Backgrounds, developed by the Leiden University's Centre of Environmental Science (CML) in the Netherlands. This method is known as CML2002 and provides best practices for indicators within the ISO14040 series of standards. The method includes approximately 800 substances, often with characterisation factors for more than one impact category. The database contains global normalisation factors as a baseline but without the weighting method. The Eco-indicator 99 method has further advanced the approach to simplify the interpretation and weighting of results. The method proposes single-point eco-indicator scores that can be used in decision making. Subsequently, the ReCiPe method integrates and harmonises both of the above approaches in a consistent framework. Although this method has not yet been published as a single document, most impact categories have been described in peer-reviewed journals. Likewise, IMPACT 2002+ combines previously used approaches, and links all types of life cycle inventories via 14 midpoint categories to four damage categories: human health, ecosystem quality, climate change, and resources. In this way, the method ensures a comparative scope of LCIA. IMPACT 2002+ today provides characterisation factors for almost 1500 different LCI results.

Looking at the impact coverage, the EDIP97 method is unique in a sense that it implements the classical emission-related impact categories and resources, as well as the working environment. The method covers seven categories: Monotonous repetitive work, noise, accidents, cancer, reprotoxic damage, allergy and neurotoxic damage due to occupational exposure to chemicals. Another method introducing unique features is EPS, developed in Sweden in 1990. It was the first method that used monetisation. It produces category indicators expressed in monetary terms, such as the Willingness to Pay (WTP). In addition, this method integrates Monte Carlo analysis and thus covers the uncertainties of the modelling results.

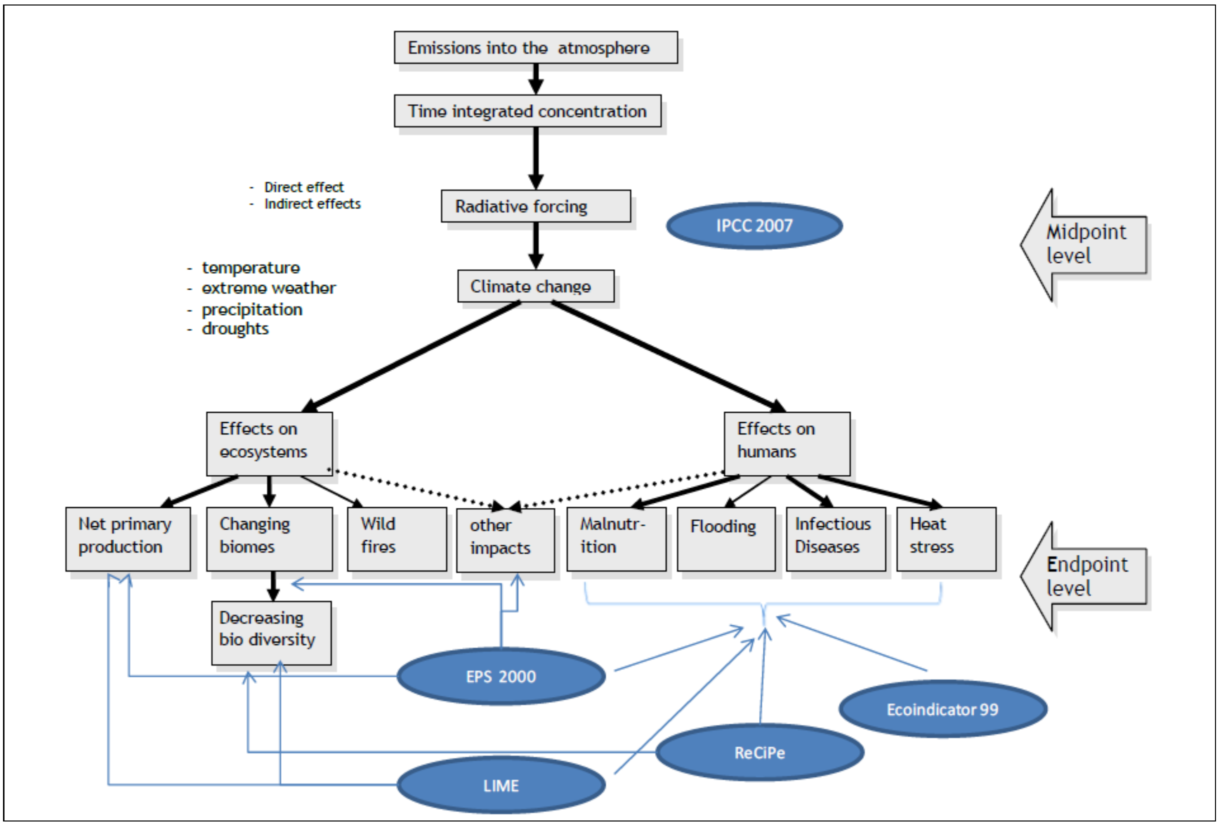
2.2.1.2 Methods focusing on specific impacts

There are approaches scrutinizing a specific category of impacts (see examples in Table 2-2). For example, all LCIA methodologies have a Climate Change impact category, and they all use the Global Warming Potentials (GWPs) developed by the Intergovernmental Panel on Climate Change (IPPC). The GWP is the ability to absorb additional heat in the atmosphere over time caused by greenhouse gases. The increase of greenhouse gases in the atmosphere is expressed in terms of CO_2 equivalents. There is broad consensus on the use of the IPCC's GWPs over 100 years to characterize the category of climate change. However, it can be considered a midpoint of the cause-effect chain. Figure 2-1 shows the use of complementary methods to define the endpoint impacts caused by the release of emissions into the atmosphere.

USEtox is a model that specifically focuses on characterisation factors for human toxicity and freshwater ecotoxicity in LCIA. Its development evolved through a scientific consensus among various developers and LCA practitioners with the support of the UNEP-SETAC Life Cycle Initiative. Another example is EcoSense, which supports the assessment of the impacts and damages of airborne pollutants from single point sources in Europe (SO_2, NO_x, primary particulates, NMVOC, NH_3 and a selection of toxic metals). It covers the impacts on photochemical ozone formation, acidification, eutrophication, and respiratory inorganics. Furthermore, the RiskPoll model has been developed to simplify the understanding of the assessment of the impacts and costs of damage due to primary and secondary particulate matter (PM) emissions. The model is based on a detailed and thorough review of epidemiological evidence and is applicable on all continents.

The method of Accumulated Exceedance includes a spatially differentiated approach providing European country-dependent characterisation factors for acidification and terrestrial eutrophication. This approach allows comparison of values within the impact category. Another specified method is Soil Organic Matter (SOM), which defines a framework for assessing land use impacts in LCA. The impacts are defined as an indicator of soil quality and, site-specific data are needed for its determination. An alternative resource-oriented model is ecological footprint (EF). The EF analysis considers biologically productive land and water area to produce all consumed products and to absorb generated waste by fossil fuels and nuclear fuel consumption.

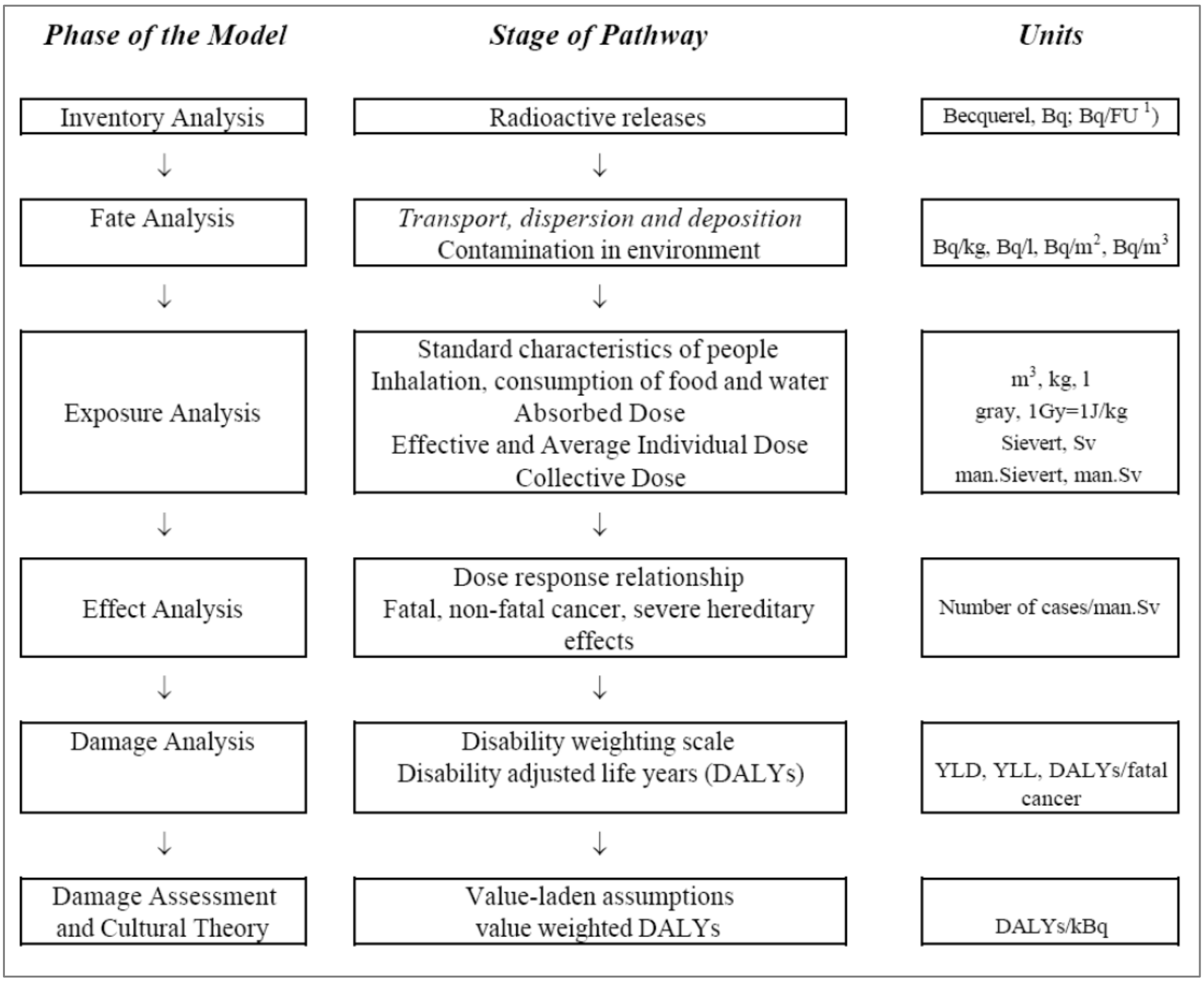
Figure 2-1. The impact pathway of emissions into the atmosphere



Source: [2-8]

For the category of ionising radiation, ILCD distinguishes between damage to human health and the ecosystem. Figure 2-2 shows the approach for quantification of the impact on human health. The method is described by Frischknecht et al. 2000 [2-10] and analyses Disability Adjusted Life Years (DALY) caused by routine releases of radioactive material into the environment. The method is compatible with the human toxicity category and used in Ecoindicator 99, IMPACT 2002+, ReCiPe and Ecopoints 2006. The framework enables the provision of separate fate and exposure intermediary results based on work carried out by Dreicer et al. 1995 [2-6]. This is based on the assessment of 14 routine atmospheric and liquid discharges in the French nuclear fuel cycle. The data have been generalised for site-independent assessment and are therefore valid on a global scale. Regarding ecosystem damages, ILCD recommends the approach developed by Garnier-Laplace et al. in 2006 [2-7]. The model converts the radiological doses to the corresponding concentration in the corresponding medium. It only addresses the effects caused by the release of radiation into freshwater and its sediments. However, the method is fully comparable and consistent with methods used for ecotoxicity, such as the USEtox framework.

Figure 2-2. Framework for the quantification of the ionising radiation impact on human health

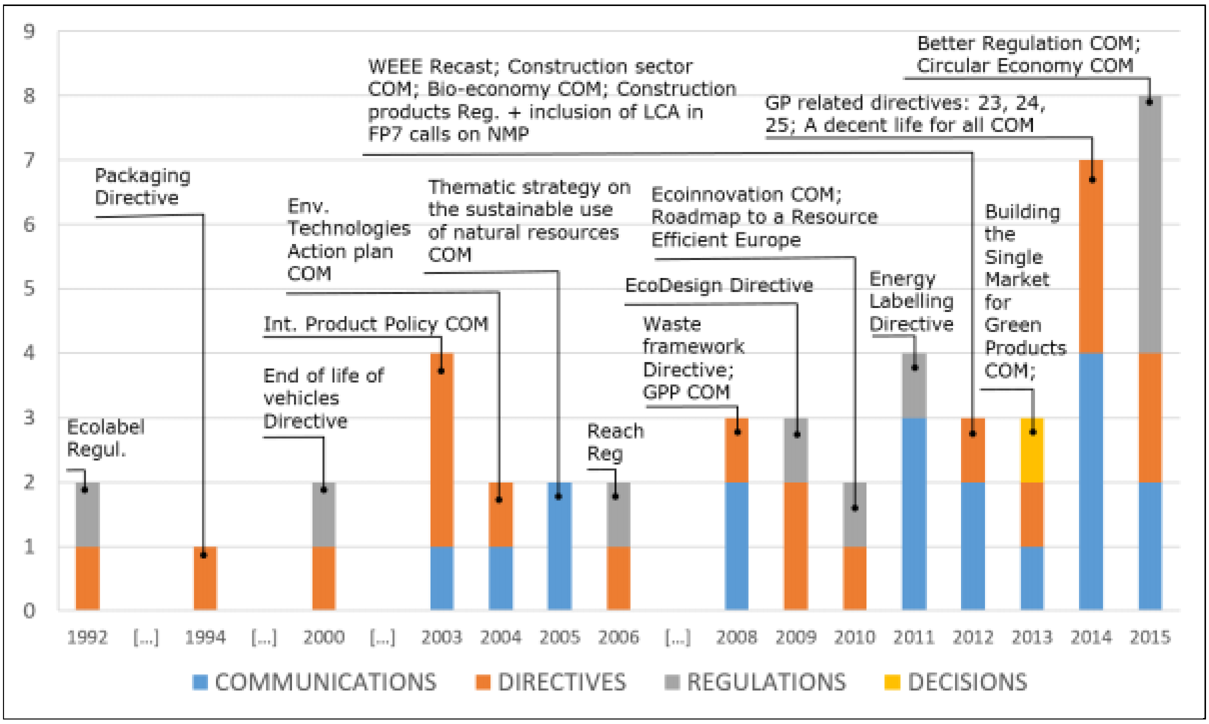


Source: [2-8], adapted from [2-10]

2.2.1.3 *Methods developed for EU policy making*

LCA was originally used to support decision making in a business context. Over time, however, it has developed into an important tool for policymaking. For example, Ecopoints 2006 (called often Ecological Scarcity Method) was developed assuming an established environmental policy framework. The method was originally applied to Swiss environmental targets, but the updated version takes into account developments in European legislation. Another important method is 'The Methodology study for Eco-design of Energy-using Products' (MEEuP). The method allows evaluating the eligibility of various energy-using products (EuP) over their life-cycle under the Eco-design of EuP Directive 2005/32/EC. The quantitative assessment includes specific impact assessment factors for inventory data and technical parameters for EuPs while ensuring consistency within the existing legislation. In general, life cycle thinking plays an important role in supporting different EU polices. Figure 2-3 shows some example policies with emphasis on life-cycle considerations.

Figure 2-3. Examples of EU policies integrating LCA in the period between 1992 and 2015



Source: [2-11]

2.2.1.4 *Methods used outside of Europe*

Japanese experts developed the LIME method. Although based on various inputs from around the world, the weighting reflects the environmental conditions of Japan, thereby limiting its use. Nevertheless, the collaboration with LIME served as a basis for the development of the IMPACT 2002+ method.

In the USA, the EPA has developed an impact assessment tool – TRACI - which represents the conditions in the USA. Similarly, LUCAS was developed as a method adapted to the Canadian context. It builds on existing methods such as TRACI and IMPACT 2002+, which are re-parameterized and further developed to better assess Canadian life cycle inventories.

2.2.2 The related ISO standards

ISO 14040:2006 (Environmental management – Life cycle assessment – Principles and framework) provides a general description of LCA and presents the purpose of the assessment. This standard, together with ISO 14044:2006 (Environmental management – Life cycle assessment – Requirements and guidelines), specifies the requirements for each of the four phases of LCA, as presented in Table 2-1. These two standards replace the original series of ISO standards focusing on each phase individually - 14040:1997 (LCA-Principals and guidelines), ISO 14041:1998 (LCA-Life Inventory Analysis), ISO 14042:2000 (LCA-Impact Assessment) and ISO 14043:2000 (LCA-Interpretation).

In addition to ISO 14040/44 there are other specific assessment frameworks for environmental assessment on product level such as:

- ISO 14067:2018 (Greenhouse gases – Carbon footprint of products – Requirements and guidelines for quantification). The standard is part of the ISO 14060 series, which provides guidelines for quantification, monitoring, reporting and verification of greenhouse gas (GHG) emissions. The ISO 14067 describes the methodology for quantification of the carbon footprint of a product, based on the LCA specified in ISO 14040/44.
- ISO 14025:2006 (Environmental labels and declarations – Type III environmental declarations – Principles & procedures) specifies principles and requirements for developing environmental declarations using predetermined parameters based on the ISO 14040 series of standards. The standard is intended for use

in business-to-business communication to enable comparisons between environmental aspects of a product or service products fulfilling the same function.

2.3 References for Chapter 2

[2-1] Life cycle assessment: Principles and practice, SAIC, EPA/600/R-06/060, May 2006.

[2-2] ILCD Handbook, General guide for Life Cycle Assessment - Detailed guidance, EUR 24708 EN, EC JRC 2010.

[2-3] A critical view on the current application of LCA for new technologies and recommendations for improved practice, Giesen et al., Journal of Cleaner Production, 259, 120904, 2020.

[2-4] Life Cycle Assessment: a review of the methodology and its application to sustainability, Curran M.A., Current Opinion in Chemical Engineering, 2:273–277, 2013.

[2-5] ILCD Handbook, Framework and requirements for Life Cycle Impact Assessment models and indicators, EUR 24586 EN, EC JRC 2010.

[2-6] Dreicer, M., Tort, V., Manen, P. (1995). ExternE, Externalities of Energy, Vol. 5 Nuclear, Centre d'étude sur l'Evaluation de la Protection dans le domaine nucléaire (CEPN), edited by the European Commission DGXII, Science, Research and development JOULE, Luxembourg.

[2-7] Garnier-Laplace, J.C., Della-Vedova, R. Gilbin, D. Copplestone, J. Hingston and P. Ciffroy, (2006). First derivation of Predicted-No-Effect Values for freshwater and terrestrial ecosystems exposed to radioactive substances. Environ. Sci. & Techno., 40, 6498-6505.

[2-8] ILCD Handbook, Recommendations for Life Cycle Impact Assessment in the European context, EUR 24571 EN, EC JRC 2011.

[2-9] ILCD Handbook, Analysis of existing Environmental Impact Assessment methodologies for use in Life Cycle Assessment, EC JRC 2010.

[2-10] Frischknecht, R., Braunschweig, A., Hofstetter, P., Suter P., Human health damages due to ionising radiation in life cycle impact assessment, Environmental Impact Assessment Review 20 (2000), 159–189.

[2-11] Life cycle assessment for the impact assessment of policies, Sala S., Reale F., Cristobal-Garcia J., Marelli L., Pant R., EUR 28380 EN, 2016.

3 Summary of results from state-of-the art LCA studies on nuclear energy

3.1 Introduction

Each generation wants to exercise the right to enjoy the benefits of modern industrialized society. On the other hand, there is a growing recognition of the need to implement measures to combat climate change, and to do it without delay, because mankind does not have much time left for action. The principle of "sustainable development" offers a viable solution for this dilemma: current generations can satisfy their economic/human development needs and they can enjoy the blessings of modern technology without destroying the environment and exhausting its resources, as well as without compromising similar rights of future generations. The well-known Brundtland report[12] defined sustainable development succinctly as meeting the needs of the present without compromising the ability of future generations to meet their own needs.

However, nothing comes free and all industrial activities come with hazards which may cause environmental damage if not controlled properly. Electricity generation activities are no exception, no matter the technology applied. Some electricity generation activities do not inflict significant harm during the operation phase itself, but rather during the associated upstream and downstream processes such as fuel mining, facility construction and dismantling, waste treatment and disposal phases. Consequently, a complete lifecycle assessment (LCA) is required in order to provide a full understanding of the impact of a particular technology on sustainable development objectives. As mentioned in Chapter 2, sustainability assessments generally address three pillars: economic development, social development and environmental protection. There is a substantial body of literature available on the assessment of sustainability of various electricity generation technologies. A significant number of sustainability indicators have been developed to facilitate comparison between technologies. These indicators address, and are categorised according to, the three aforementioned pillars.

The Taxonomy Regulation [3.1-1] sets up a framework for the development of an EU classification system ("EU Taxonomy") of environmentally sustainable economic activities for investment purposes. It establishes six environmental objectives:

- climate change mitigation;
- climate change adaptation;
- the sustainable use and protection of water and marine resources;
- the transition to a circular economy;
- pollution prevention and control;
- the protection and restoration of biodiversity and ecosystems.

For an economic activity to be included in the EU Taxonomy, it must contribute substantially to at least one environmental objective and do no significant harm to the other five (see also [3.1-2 & 3]).

In order to have an objective picture of the potential hazards and resource depletion characteristics of nuclear energy compared to various other electricity generation technologies, and to place nuclear energy in the overall impact landscape, Chapter 3.2 provides a concise overview of some representative lifecycle impact assessment (LCIA) studies investigating the dominant electric power production methods. Although a substantial body of literature exists on the assessment of sustainability of different electricity generation technologies, not many studies address nuclear energy. Moreover, the review presented in Chapter 3.2 is mainly limited to studies in which the lifecycle impact of nuclear electricity generation is assessed and compared with other electricity generation technologies in the same study. This helps to ensure that technologies are compared using the same assessment methodologies and consistent assumptions. In addition, studies that review, compile and statistically compare results from many other sustainability assessments are also considered.

As the objectives established in the Regulation are environmental objectives, the overview of existing representative lifecycle impact assessment (LCIA) studies and sustainability assessments provided in Chapter 3.2 below deal predominantly with the sustainability indicators of the environmental protection pillar.

[12] Our Common Future, World Commission on Environment and Development (WCED), 1987, also known as the 'Bruntland Report'.

However, indicators from the other two pillars are compared where they address aspects of the environmental objectives of the Regulation.

3.1.1 References for Chapter 3.1

[3.1-1] Regulation (EU) 2020/852 of the European Parliament and of the Council of 18 June 2020 on the establishment of a framework to facilitate sustainable investment, and amending Regulation (EU) 2019/2088

[3.1-2] Financing a Sustainable European Economy, Technical Report, Taxonomy: Final report of the Technical Expert Group on Sustainable Finance, March 2020

[3.1-3] Taxonomy Report, Technical Annex, Updated methodology & Updated Technical Screening Criteria, March 2020

3.2 Comparison of impacts of various electricity generation technologies

Following a short review of nuclear energy's current and projected share in electricity generation, this section compares the environmental impact of nuclear energy with other generation technologies. The comparison is organised according to the environmental objectives of the Taxonomy Regulation. Lifecycle impacts of nuclear energy are reviewed in order, firstly, to assess its contribution to the climate objectives, namely, to climate change mitigation. Having confirmed that nuclear energy can contribute substantially to climate change mitigation, this section then goes on to compare nuclear energy with other electricity generation technologies[13] from the point of view of the requirement to do no significant harm to the four non-climate environmental objectives:

- the sustainable use and protection of water and marine resources;
- the transition to a circular economy;
- pollution prevention and control;
- the protection and restoration of biodiversity and ecosystems.

3.2.1 Nuclear energy's share in global and EU electricity generation

According to the International Atomic Energy Agency (IAEA) [3.2-1], at the end of 2018 there were altogether 451 nuclear power plant (NPP) units in operation all over the world with a total electricity generating capacity of 396.9 GW. The LWR[14] type is dominant with 353.9 GW installed capacity[15] (89% of total installed capacity: 71% PWR and 18% BWR), while about half of the remaining 11% is generated in PHWR units such as the Canadian CANDU design. The rest is produced in gas-cooled reactors (2%), LWGRs (2%, also called RBMK) and Fast Breeder Reactors (1%). LWRs represent an even greater proportion of the installed capacity of reactors under construction (94%, of which 85% PWR and 9% BWR). For the near-term future, new investments are expected to follow similar patterns.

Due to their dominance, LWR type nuclear power reactors figure predominantly in existing LCA analyses.

According to [3.2-2], in 2018 this almost 400 GW nuclear capacity delivered about 10% of the global electricity supply. In 2017, the situation was similar, as illustrated in Figure 3.2-1 showing the relative contributions of the various fuel types to the total electricity generated[16] in the world, the OECD[17] countries and the EU-28, based on data taken from [3.2-3] and [3.2-4].

According to the 2018 World Energy Outlook published by the International Energy Agency (IEA), in 2017 [3.2-3], the total electricity generation of the world amounted to 25 640 TWh[18]. As shown in Figure 3.2-1, the worldwide share of nuclear was 10.4%. The combined share of low carbon generation technologies (i.e.

[13] Some of which are included in the Taxonomy

[14] Reactor-type acronyms introduced in this paragraph: LWR – Light Water Reactor; PWR- Pressurised Water Reactor; BWR – Boiling Water Reactor; PHWR – Pressurised Heavy Water Reactor; LWGR – Light Water Graphite-moderated Reactor; RBMK (Russian acronym) – Reaktor Bolshoy Moshchnosti Kanalnyy, "high-power channel-type reactor", CANDU – Canada Deuterium Uranium.

[15] Installed capacity is the maximum instantaneous output of electricity that an installation is normally able to produce, usually given in units of Watts (W) or multiples thereof, e.g. kW, MW or GW. Electricity generation, on the other hand, refers to the amount of electricity that has actually been produced over a specific period of time. This may be measured in Watt-hours (Wh) or, multiples thereof, e.g. kWh, MWh, GWh or TWh (terawatt-hours).

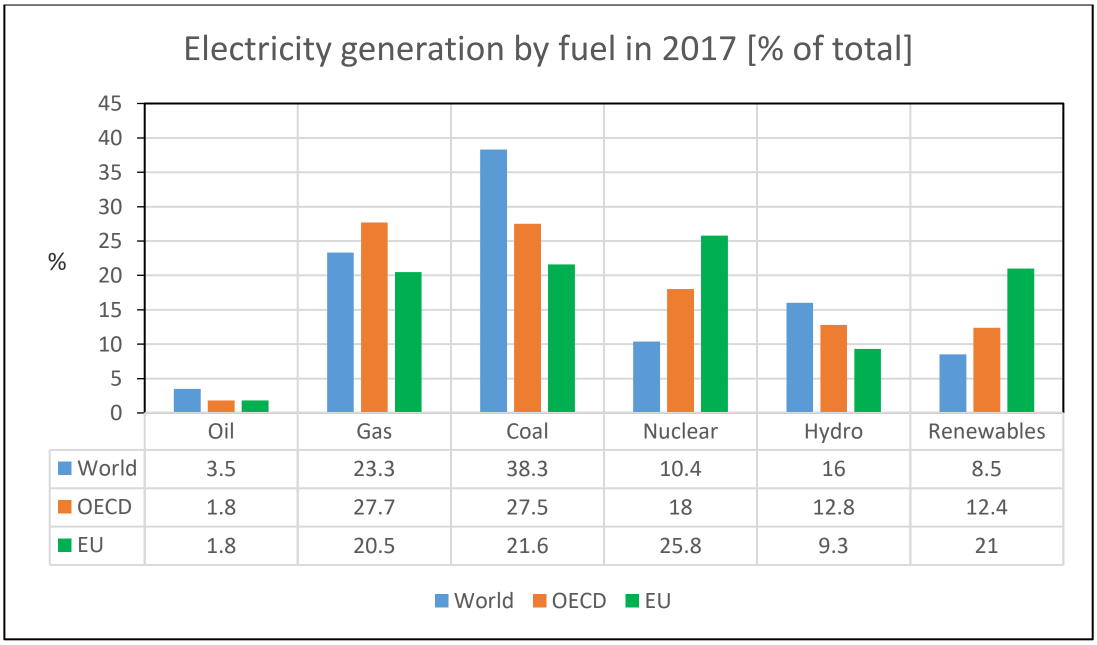
[16] See footnote 15

[17] Organisation for Economic Cooperation and Development

[18] See Table 1.4 in [3.2-3]

renewables, hydro and nuclear) amounted to about 35% of the total world generation. The remaining 65% was generated by the combustion of fossil fuels (coal, oil and gas), thereby contributing significantly to global warming and emitting considerable amounts of other pollutants that are important from an environmental and public health perspective.

Figure 3.2-1. Electricity generation by fuel type in 2017

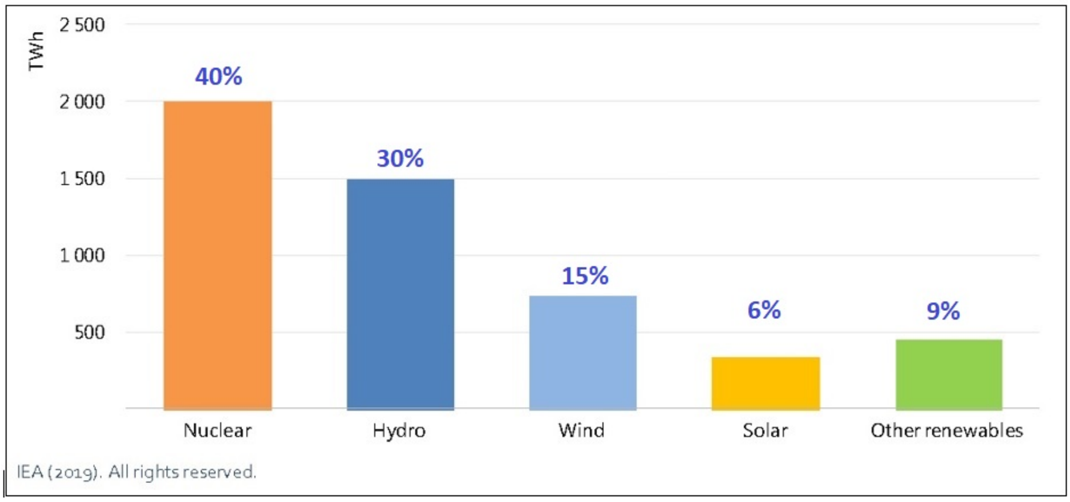


	Oil	Gas	Coal	Nuclear	Hydro	Renewables
World	3.5	23.3	38.3	10.4	16	8.5
OECD	1.8	27.7	27.5	18	12.8	12.4
EU	1.8	20.5	21.6	25.8	9.3	21

Source: Ref. [3.2-4]

Globally nuclear accounted for about 30% of low carbon electricity, second to hydro's 46%, whereas in OECD countries, the corresponding figures were about 42% for nuclear and about 30% for hydro. The situation in the EU-28 is particularly interesting, because the share of low carbon generation technologies in 2017 amounted to 56% of the total. Nuclear accounted for almost half (46%) of those low carbon sources (see [3.2-4]). In the EU, the share of hydro is relatively low in the low carbon generation area (about 17%), but the high share of wind and solar (which amounts to about 37%) somewhat balances the picture.

Figure 3.2-2. Low carbon electricity generation in advanced[19] economies by source in 2018



Source: Ref. [3.2-2]

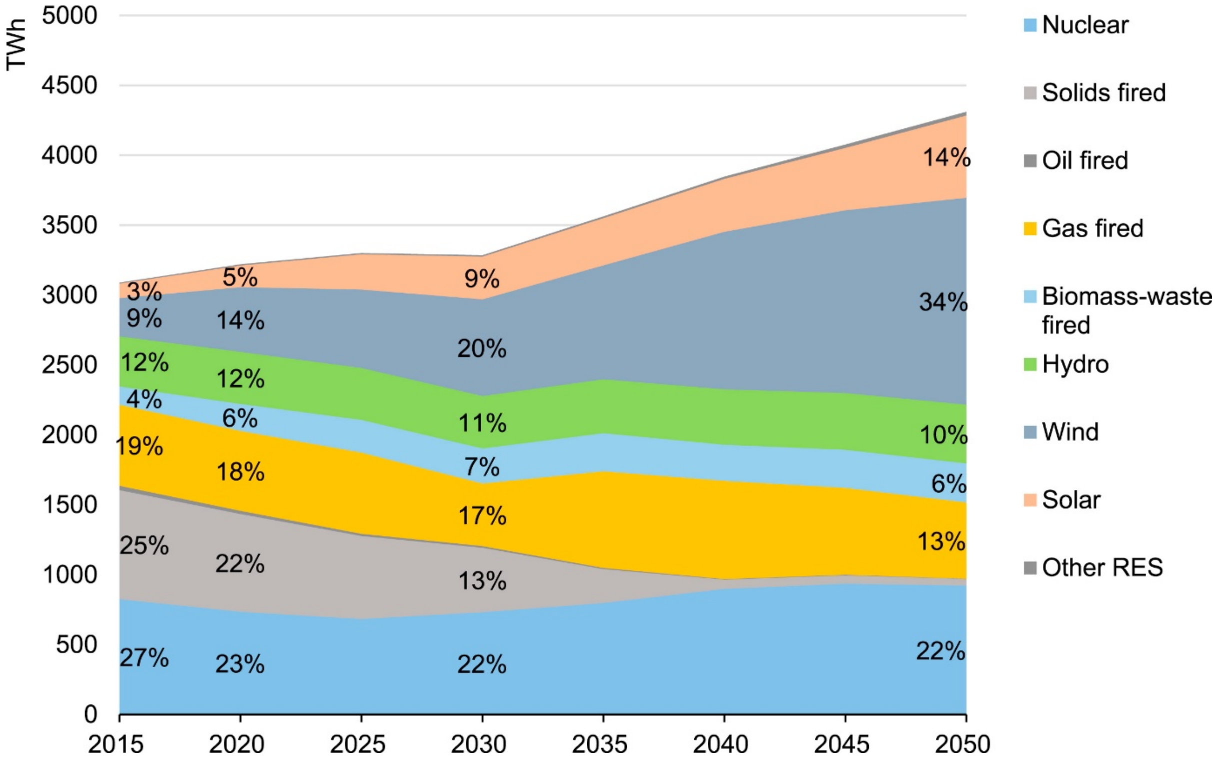
[19] Advanced economies are the most developed countries having a GDP per capita above a certain threshold.

The sectoral composition of low carbon electricity supply in the “advanced economies” is shown in Figure 3.2-2, illustrating that nuclear – with its 2000 TWh delivered energy – is the most important low carbon electricity supplier also in these countries and it accounts for about **40%** of low carbon electricity (see [3.2-2] for further details).

The above figures show that currently nuclear is producing at least 30% of low carbon electricity worldwide and more than 40% in the advanced economies.

The electricity supply technologies presented in the Technical Annex to the Taxonomy Report emerged from the investment needs specifically related to scenarios developed by the EC in order to meet the EU energy and climate 2030 targets[20]. These scenarios have been quantified using the PRIMES energy systems model[21]. The model simulates prospective energy consumption and energy supply in the EU. Figure 3.2-3 shows the projection of the electricity generation by fuel under the core policy scenario - EUCO30 - adopting climate, energy and transport policies for 2030 and the long-term milestone to reduce GHG emissions in the EU at least by 80% in 2050[22]. The analysis presents an almost constant share of nuclear in the electricity supply mix over the studied horizon.

Figure 3.2-3. Projection of the electricity generation by source in the EU



Source: Ref. [3.2-5]

The projected evolution of the energy system is highly dependent on technology assumptions. Figure 3.2-4 provides insights on the projected installed capacity of nuclear power plants in the EUCO 30 scenario. The study envisages new build projects as well expectations on lifetime extensions (referred to in the graph as retrofitting) reviewed by the relevant experts, industry representatives and stakeholders[23].

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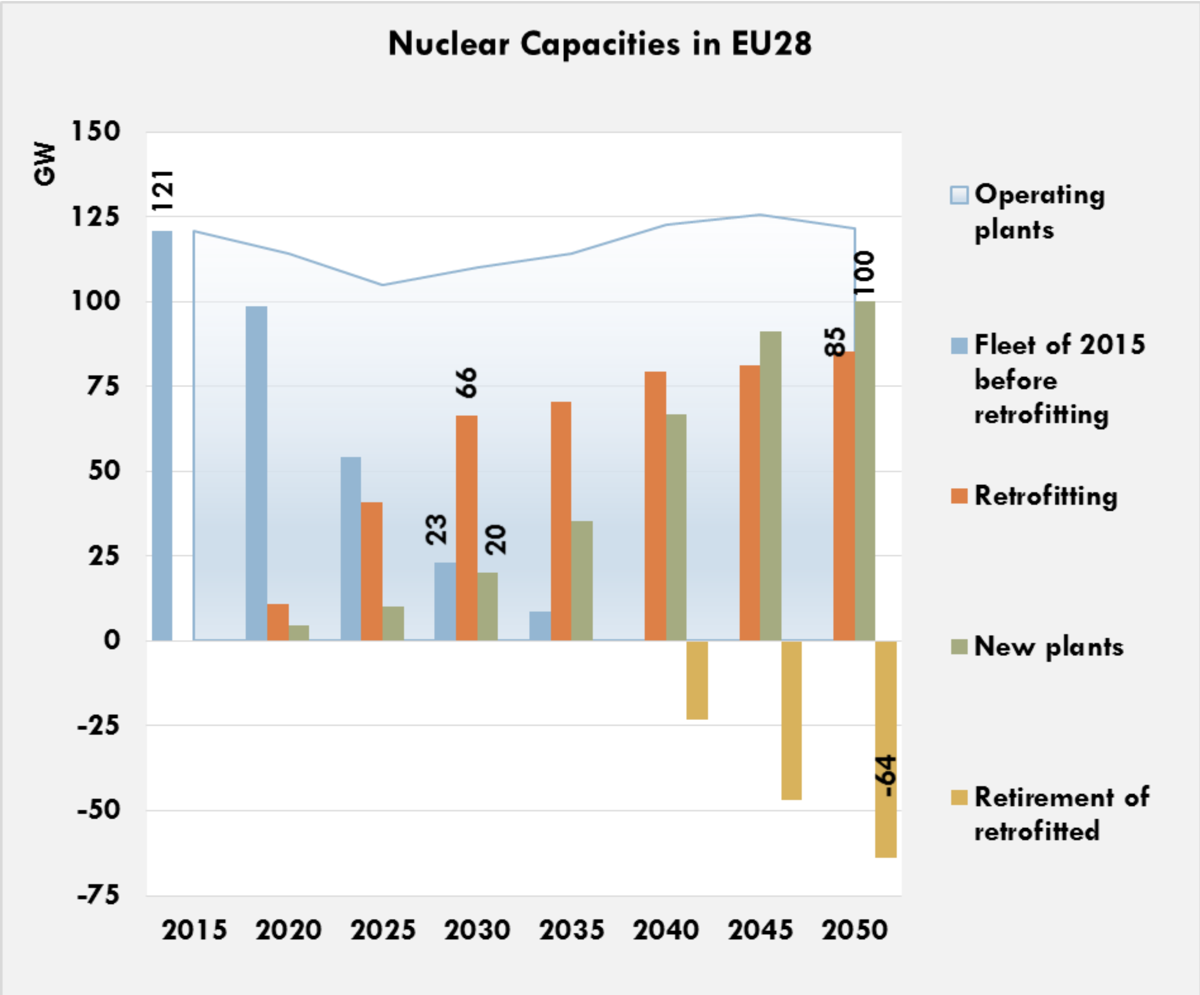
20 The European Council agreed to the 2030 strategy with targets on reducing greenhouse gas emissions by at least 40%, increasing the share of renewable energy to at least 27%, and achieving an energy efficiency improvement of at least 27%: https://www.consilium.europa.eu/en/policies/climate-change/2030-climate-and-energy-framework/

21 http://www.e3mlab.eu/e3mlab/index.php?option=com_content&view=category&id=35%3Aprimes&Itemid=80&layout=default&lang=en

22 Since then, the Commission has set out its vision for a climate-neutral EU by 2050. This objective is at the heart of the European Green Deal (https://ec.europa.eu/info/strategy/priorities-2019-2024/european-green-deal_en).

23 Details conducted under the ASSET project: https://ec.europa.eu/energy/sites/ener/files/documents/2018_06_27_technology_pathways_-_finalreportmain2.pdf

Figure 3.2-4. Evolution of the nuclear installed capacity in the EU



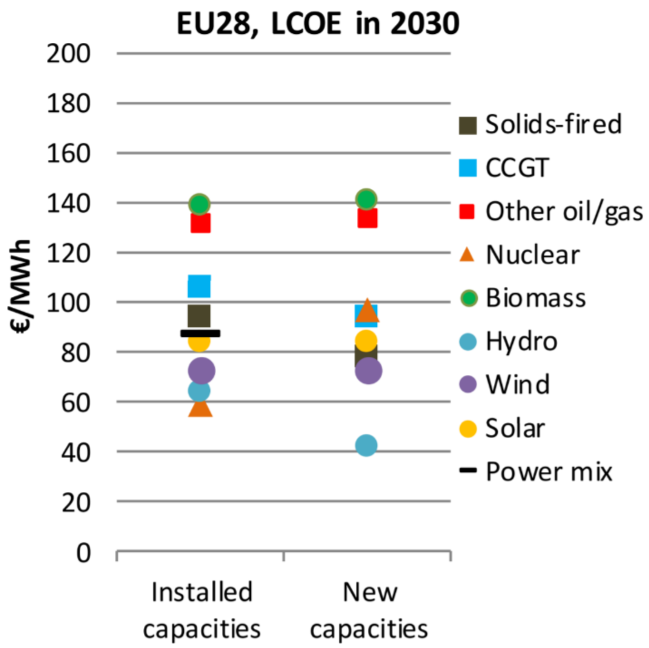
Source: Ref. [3.2-5]

Nuclear is the most capital-intensive baseload technology and therefore, as shown in the figure above, retrofitting of the existing fleet is a favourable option in the mid-term. Extending the lifetime of the existing nuclear generation capacities often involves significant works in order to replace ageing components and improve safety to meet higher safety requirements and expectations of the regulatory authorities. However, despite these additional costs, lifetime extension of existing plants remains an economically very attractive option and one that is already implemented or planned in several EU Member States. Regarding new build, some Member States are already undertaking, or are planning, the construction of new large nuclear power plant projects. Moreover, there is an increasing interest in smaller scale nuclear power reactors, so-called Small Modular Reactors (SMRs).

Figure 3.2-5 shows the generation costs of different technologies. Considering the existing capacities, nuclear power represents the lowest generation costs in 2030. The cost increases when considering new installed capacities, but nuclear remains competitive and close to the levelised cost of the current power mix. However, as mentioned above, nuclear energy is highly capital-intensive, and this presents certain difficulties to investors for financing the construction of new large nuclear power plants, which has become more challenging in the last three decades, as energy markets have been deregulated. According to the IAEA[24], to encourage nuclear development despite these difficulties, innovative approaches to financing and support policies are being pursued, including partial investment or loan guarantees from the government.

[24] https://www.iaea.org/topics/funding-and-finance

Figure 3.2-5. Levelised cost of electricity in the EU



Source: [25]

3.2.2 Contribution to climate change mitigation

The Technical Expert Group (TEG), in its Taxonomy Report Technical Annex [3.2-6], clearly recognised that nuclear energy has near-to-zero greenhouse gas emissions in the energy generation phase, and it did not express any doubts that nuclear energy can make a substantial contribution to climate change mitigation, one of the six environmental objectives of the Taxonomy Regulation. Consequently, it is not intended to dedicate a significant part of this chapter to demonstrating the contribution of nuclear energy to climate change mitigation. Nevertheless, the TEG report mentions only the electricity generation phase of nuclear energy, whereas the whole lifecycle should be considered when assessing any particular technology's contribution to climate change mitigation. It is useful therefore, to illustrate by at least one typical comparison from the open literature, how nuclear energy compares with other technologies with regard to lifecycle greenhouse gas emissions.

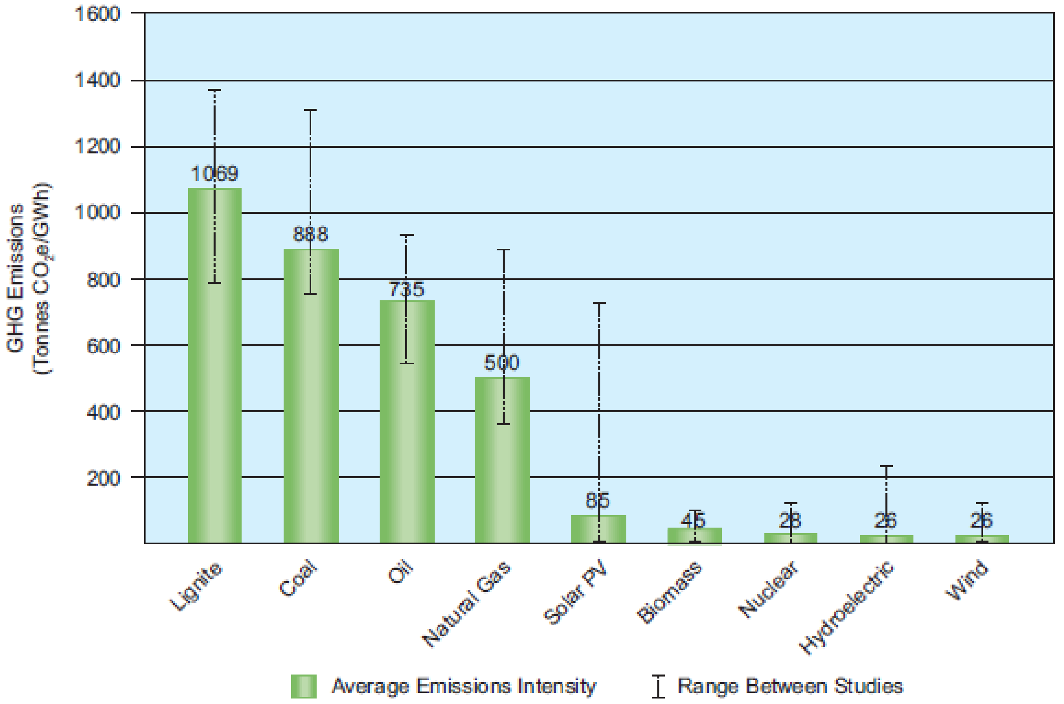
Figure 3.2-6, from reference [3.2-7], is the result of a secondary research compilation of twenty-one credible[26] sources in which lifecycle GHG emissions of different electricity generation technologies have been assessed.

The figure shows that lifecycle GHG emissions from nuclear energy are among the lowest of all the technologies, comparable with (or slightly greater than) wind and hydroelectricity and lower than solar PV. That this is typical of the results from other credible LCAs can be seen from references 3.2-8, 9, 10, 11 among many others.

Some variation can be seen in the values of GHG emissions for nuclear energy provided in the literature. Among the reasons for the variations, assumptions regarding the fuel enrichment process and the grade of uranium ore extracted in the mining stage can have a major impact on the assessed lifecycle emissions. Enrichment via the gaseous diffusion process requires a significant amount of energy input, and if it is assumed that this energy is supplied by burning fossil fuels, or even by the current energy mix, the resulting GHG emissions for the nuclear energy lifecycle can be significant. It may be more reasonable when assessing the climate mitigation potential of the nuclear energy chain to assume that the electricity required is produced by the resulting nuclear power or a future decarbonised mix. More importantly, the gaseous diffusion process has been phased out and replaced by the centrifuge enrichment process, which is up to 50 times less energy costly than the gaseous diffusion process [3.2-12, 13].

[25] https://ec.europa.eu/energy/sites/ener/files/epc_report_final_1.pdf
[26] Studies published by governments and universities were sought out, and industry publications used when independently verified.

Figure 3.2-6. Lifecycle GHG emissions intensity of electricity generation technologies



Source: Ref. [3.2-7]

According to [3.2-14, 15], the current world mean production uranium ore grade in 2009 was of the order of 0.12% uranium oxides (U_3O_8). In general, the grades of exploitable metallic ores will fall globally as the higher grade reserves are extracted first and are progressively depleted. As the grade of the available uranium ore falls, a greater amount of energy (and other material inputs) will be required in the mining and milling stage to extract the same amount of U_3O_8. If uranium ore grade declines by a factor of ten, then energy inputs to mining and milling increase by at least a factor of ten [3.2-16, 17]. Furthermore, it is generally accepted that uranium recovery in the mining and milling stage decreases as ore grade falls, although it is recognised that further work is needed to quantify this effect more accurately. In the lifecycle analysis for nuclear electricity generation performed in [3.2-15], the estimated level of GHG emissions for the mining and milling stage was 1.3 gCO_2-eq/kWh for an assumed ore grade of 0.15% U_3O_8. However, an assumed ore grade of 0.01% U_3O_8 resulted in significantly higher GHG emissions in the mining and milling stage, increasing the lifecycle GHG emissions by about 26 gCO_2-eq/kWh. Even lower ore grades result in correspondingly larger GHG emissions. Some current LCA analyses provided in the literature have assumed lower grade ores than are currently available or likely to be available on a reasonable time horizon, thus resulting in higher assessed GHG emissions for the nuclear lifecycle. According to [3.2-15], current world uranium resources are projected to remain above a grade of 0.01% U_3O_8 for the next 50 years based on predicted nuclear power annual growth rates of 1.9%.

Lifecycle GHG emissions for the existing French nuclear reactor fleet in 2010, at that time using the gaseous diffusion process supplied by nuclear energy, was assessed to be 5.29 gCO_2-eq/kWh [3.2-8]. Uranium ore grades corresponded to the current production from the mining activities supplying the French fuel cycle, which were all higher than 0.1% [3.2-18]. According to [3.2-8], nuclear power plants (including construction, operation and decommissioning) are responsible for 40% of the lifecycle GHG emissions, uranium mining for 32% and enrichment 12%.

According to [3.2-10, 14], lifecycle GHG emissions for a future EPR (European Pressurised-water Reactor), using the centrifuge enrichment process, have been estimated to be very similar to that estimated in the above study, at 4.25 gCO_2-eq/kWh.

According to the foregoing, lifecycle GHG emissions from nuclear electricity generation are comfortably within the 100 gCO_2-eq/kWh emissions intensity threshold proposed by the TEG for electricity generation, and will remain so for at least the next 50 years, thereby satisfying the TEG definition for a substantial contribution to climate change mitigation.

From the wider system perspective, nuclear contributes further to climate change mitigation through synergy with renewable energy technologies. In an interconnected electricity system, each power plant interacts with others through the same grid. Nuclear is the major dispatchable low carbon source of electricity next to hydro. Being used as baseload technology, it provides flexible operation to complement the intermittent renewable energy sources. Thus, wind and solar deploy more efficiently. On the one hand, this avoids use of highly carbon-intensive generation technologies often used for a backup. On the other hand, this integration, together with the electricity storage, brings benefits to the electricity grid, by minimising short-term disruptions.

3.2.3 DNSH to the sustainable use and protection of water and marine resources

In accordance with article 17 of the Taxonomy Regulation, an economic activity shall be considered to cause significant harm to the sustainable use and protection of water and marine resources where that activity is detrimental:

(i) to the good status or the good ecological potential of bodies of water, including surface water and groundwater; or

(ii) to the good environmental status of marine waters.

Fresh water is a precious resource and its use needs to be managed sustainably. All energy generation technologies consume water to some extent, but those based on thermal technologies, including nuclear energy (as well as renewable technologies based on thermal energy, like concentrating solar power and biomass), have relatively high water consumption requirements when compared to non-thermal renewable technologies[27].

While water is consumed in all lifecycle stages of most energy technologies, for those based on thermal energy, the vast majority is consumed as cooling water during the operation of the power plants. This is particularly the case for nuclear energy [3.2-8]. The exception is biomass, for which, in addition to the water consumed by the power plants, very large amounts of water may be consumed during the production of the feedstock, depending on factors such as the type of crop, geographic location, local climate and crop management techniques [3.2-11]. For the nuclear energy lifecycle, while water consumption at the mining stage is small in comparison to the operation of power plants, it nevertheless has to be carefully considered, as mining and milling activities are often located in dry and arid areas where it is especially important to preserve available water sources. Moreover, water consumption is strongly dependent on the mining practices employed. In-situ leaching (ISL) techniques consume larger amounts of water than other mining techniques [3.2-8].

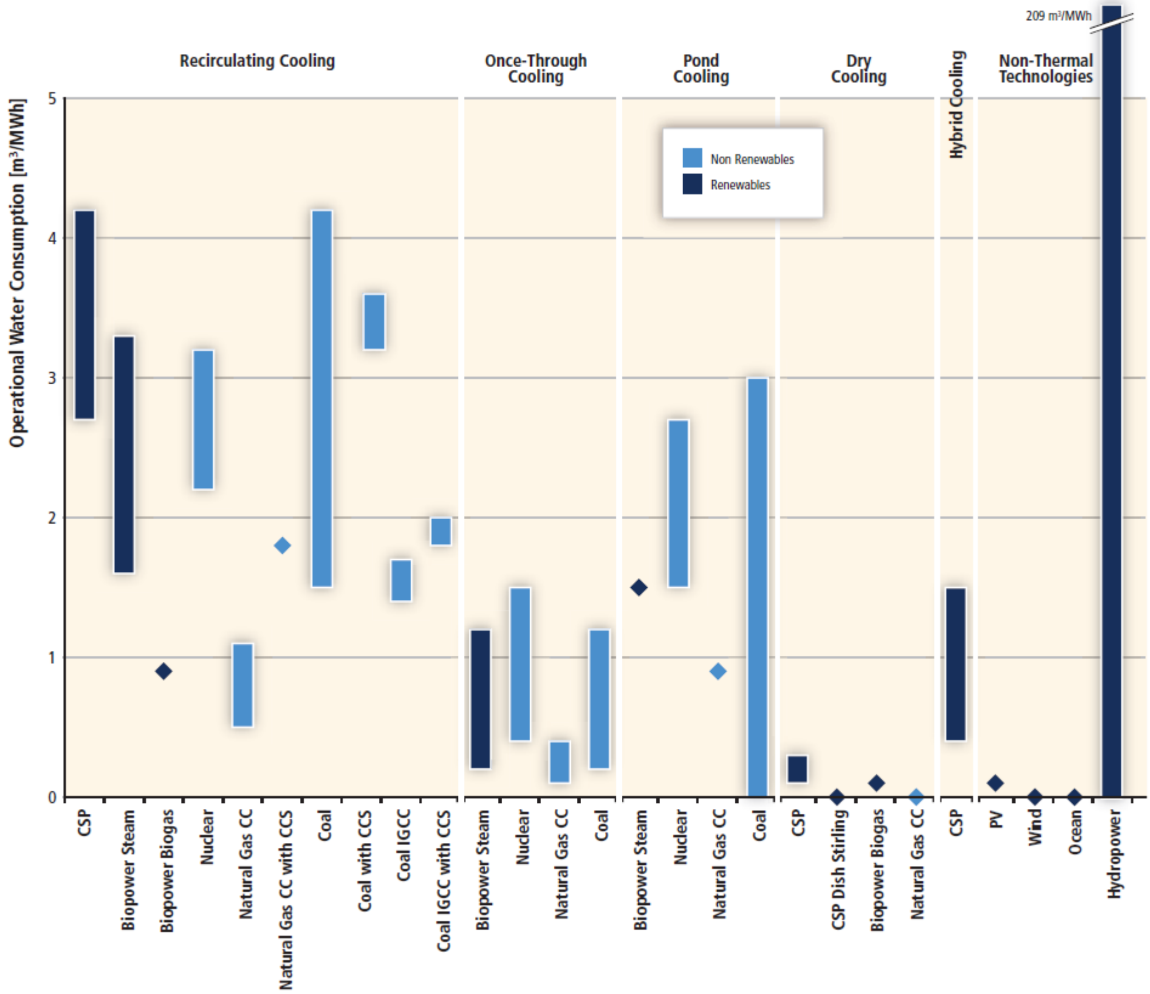
Common sustainability indicators for water usage of energy generation technologies are water withdrawal and water consumption. Withdrawal is the amount of fresh water removed or diverted from ground or surface waters (even if some is returned), while consumption is the amount lost from the immediate environment through evaporation, incorporation into products, take-up by crops, consumption by humans or animals or otherwise removed.

For power plants based on thermal energy, the water consumption depends strongly on the chosen cooling technology. Plants utilising once-through cooling withdraw large volumes of water but consume very little as most of it is returned to the same watercourse with a higher temperature. The temperature increase or absolute discharge temperature is subject to statutory limits. Many nuclear power plants are located at the coast and use seawater for cooling in a once-through system. Such plants neither withdraw, nor consume, significant amounts of fresh water. Nuclear power plants also commonly employ recirculating cooling, using evaporative cooling towers, or pond cooling, both of which require make-up water to compensate for losses due to evaporation. In both cases, water consumption is greater than for plants employing once-through cooling technology.

Figure 3.2-7 [Ref 3.2-11] compares water consumption data for the operation phase of different electricity generation plants, taking into account the use of different cooling technologies. The figure aggregates and presents data from a large number of studies reported in the available literature.

[27] The exception, regarding non-thermal renewable technologies, is hydropower, for which large quantities of water may be lost due to evaporation from the surface of the hydroelectric reservoirs.

Figure 3.2-7[28]**.** Ranges of rates of operational water consumption by thermal and non-thermal electricity-generating technologies (m^3/MWh)[29]



N:	18	4	1	5	4	1	16	2	7	3	1	3	3	9	1	1	1	7	11	2	1	2	4	2	2	1	4
Sources:	11	3	1	5	4	1	8	1	2	1	1	3	3	4	1	1	1	2	4	2	1	2	2	2	2	1	2

Source: Ref. [3.2-11]

It can be seen from the figure that while nuclear energy consumes significant amounts of water compared to renewable technologies like solar PV, wind and ocean energy, it is comparable to or better than concentrating solar power (CSP), hydropower and biomass[30]. These latter technologies are not excluded from the taxonomy, nor is a particular cooling technology specified in the technical screening criteria for these technologies. The water consumption associated with nuclear energy does not therefore constitute a reason for exclusion of nuclear energy from the taxonomy. Water usage in the power generation phase of the nuclear energy lifecycle is discussed further in Chapter 3.3.7 and in the related TSC in Chapter 5 and Annex 4.

In addition to water withdrawal and consumption, electricity generation may also affect the quality of both fresh and marine waters through chemical, thermal and radioactive pollution.

A number of sustainability indicators for comparing chemical pollution and its potential impacts on water ecosystems have been used in lifecycle assessments in the literature. The more common ones include direct

[28] CSP – Concentrating Solar Power; CC – Combined Cycle; CCS – Carbon Capture & Storage; IGCC - Integrated Gasification Combined Cycle; PV - Photovoltaic

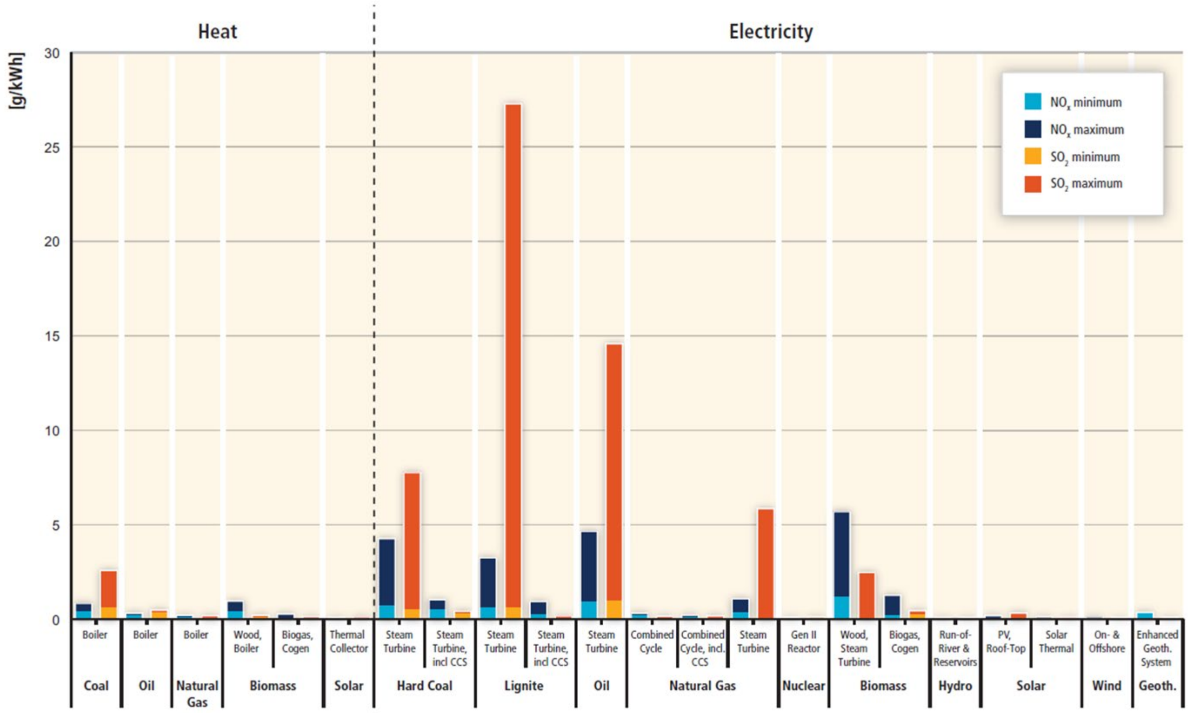
[29] Based on a review of available literature. Bars represent absolute ranges from available literature, diamonds single estimates; N represents the number of estimates reported in the sources. Refer to the original reference for further notes and information on the methods and references used in the literature review.

[30] Especially when taking into account water consumption for the production of feedstock.

emissions of nitrous oxides (NO_x) and sulphur dioxide (SO_2), as well as impact indicators for acidification, eutrophication and eco-toxicity (fresh water and marine eco-toxicity)[31].

Figure 3.2-8 [Ref 3.2-11] compares NO_x and SO_2 data for the lifecycle of different heat and electricity generation technologies and clearly shows that nuclear energy, based on current Generation II power plants, along with wind and hydro have relatively very low emissions of these substances compared to fossil fuel technologies. Among the technologies included in the Taxonomy, natural gas, biomass and solar PV all have more lifecycle emissions of both NO_x and SO_2 than nuclear energy.

Figure 3.2-8. Cumulative lifecycle emissions of NO_x and SO_2 per unit of energy generated for current heat and electricity supply technologies[32]



Source: Ref. [3.2-11]

Acidification potential refers to the compounds that are precursors to acid rain. These include sulphur dioxide (SO_2), nitrogen oxides (NO_x), nitrogen monoxide (NO), nitrogen dioxide (N_2O), and other various substances. Acidification potential is usually characterized by SO_2-equivalence (g SO_2-eq/kWh$_e$). Atmospheric emissions of these acidifying substances can persist in the air for some days allowing their transport over very large distances, and during which time they undergo chemical conversion into acids (sulphuric and nitric). Deposition of the primary pollutants sulphur dioxide, nitrogen dioxide and ammonia (NH_3), along with their reaction products, leads to changes in the chemical composition of the soil and surface water. This process interferes with ecosystems, leading to what is termed 'acidification'.

Eutrophication is the gradual increase in the concentration of phosphorus, nitrogen, and other minerals and plant nutrients in aquatic ecosystems resulting in over-enrichment that can give rise to excessive growth of algae and depletion of oxygen that supports healthy underwater life. The indicator for eutrophication potential is expressed in grams phosphate equivalent per unit of electricity generated (g PO_4^{3-}-eq/kWh$_e$). Some methodologies calculate freshwater and marine eutrophication potentials separately. As phosphorous is the key limiting nutrient for freshwater eutrophication, its units are g P-eq/kWh, whereas for marine water, nitrogen is most often the key limiting nutrient, so that the units of marine eutrophication are g N-eq/kWh.

Stamford & Azapagic [3.2-9], as well as Treyer & Bauer [3.2-23], in their lifecycle sustainability assessments of electricity options for the UK and UAE respectively, compared a comprehensive range of mid-point

[31] Some of these indicators (NO_x, SO_2, acidification) are important not only in respect of water ecosystems, but also in relation to air pollution, soil quality and terrestrial ecosystems.

[32] Data from [3.2-19, 20, 21]; traditional biomass use not considered. Figures for coal and gas power chains with CCS are valid for near-future forecasts [3.2-22].

environmental impact indicators for several electricity generation technologies. The results for acidification and eutrophication potentials are provided in Figure 3.2-9. Data of Poinssot et al [3.2-8] for nuclear energy are also included in the figure for comparison[33].

It can be seen that nuclear energy provides the lowest contribution to acidification compared to the other technologies included in the comparison. With regard to eutrophication, nuclear energy also performs better than the other technologies for the combined eutrophication indicator of the CML methodology as well as for the freshwater eutrophication calculated according to the ReCiPe methodology. Only for marine eutrophication, the ReCiPe methodology calculates a slightly higher contribution than natural gas and the renewable technologies, while still almost an order of magnitude lower than oil-based electricity generation. It can also be seen that the results of Poinssot et al [3.2-8], calculated with data of a completely different origin and using their own methodology, compare extremely well with the results of the other investigators. Importantly, they also provide a detailed breakdown of the contribution to each indicator from the different phases of the lifecycle of the nuclear energy chain. Mining is responsible for 82% of the acidification potential, while reactors (construction, operation and decommissioning) contribute the next biggest share at 8%. Regarding eutrophication, mining with 53%, enrichment 17%, reactor operation 14% and reprocessing 11% are the main contributors.

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[33] The data of Stamford & Azapagic [3.2-9], Treyer & Bauer [3.2-23] and Poinssot et al [3.2-8] are compared in several figures in the remainder of Chapter 3.2. Some basic data relating to these studies is given below:

Stamford & Azapagic [3.2-9]:
LCIA methodology: CML 2001.
Electricity generation technologies: Coal, Natural gas (CCGT*), Nuclear (PWR*), Wind (Offshore), Solar PV*.
Nuclear energy: Future PWR for the UK operating on a once-through (open) fuel cycle. Centrifugal enrichment.
Data from EcoInvent 2.2 database.
The ranges indicated in the figures represent the results from sensitivity studies. For nuclear energy, the sensitivity studies investigated the use of MOX* and different mixes of gaseous diffusion and centrifuge enrichment (from 0 to 30% diffusion).

Treyer & Bauer [3.2-23]:
LCIA methodology: ReCiPe 'midpoint' impact indicators.
Electricity generation technologies: Oil[a], Natural gas (conventional), NGCC* (current) [a], NGCC (future), CCS*, Nuclear (PWR)[a], Solar PV (building) [a], Solar PV (open ground), CSP*, Wind (Onshore) [a].
Nuclear energy: PWR (EPR*), Uranium extraction via 50% in-situ leaching, 30% underground mining, 20% open pit mining; Centrifugal enrichment only; Once-through (open) fuel cycle.
Data from EcoInvent 3.1 database.

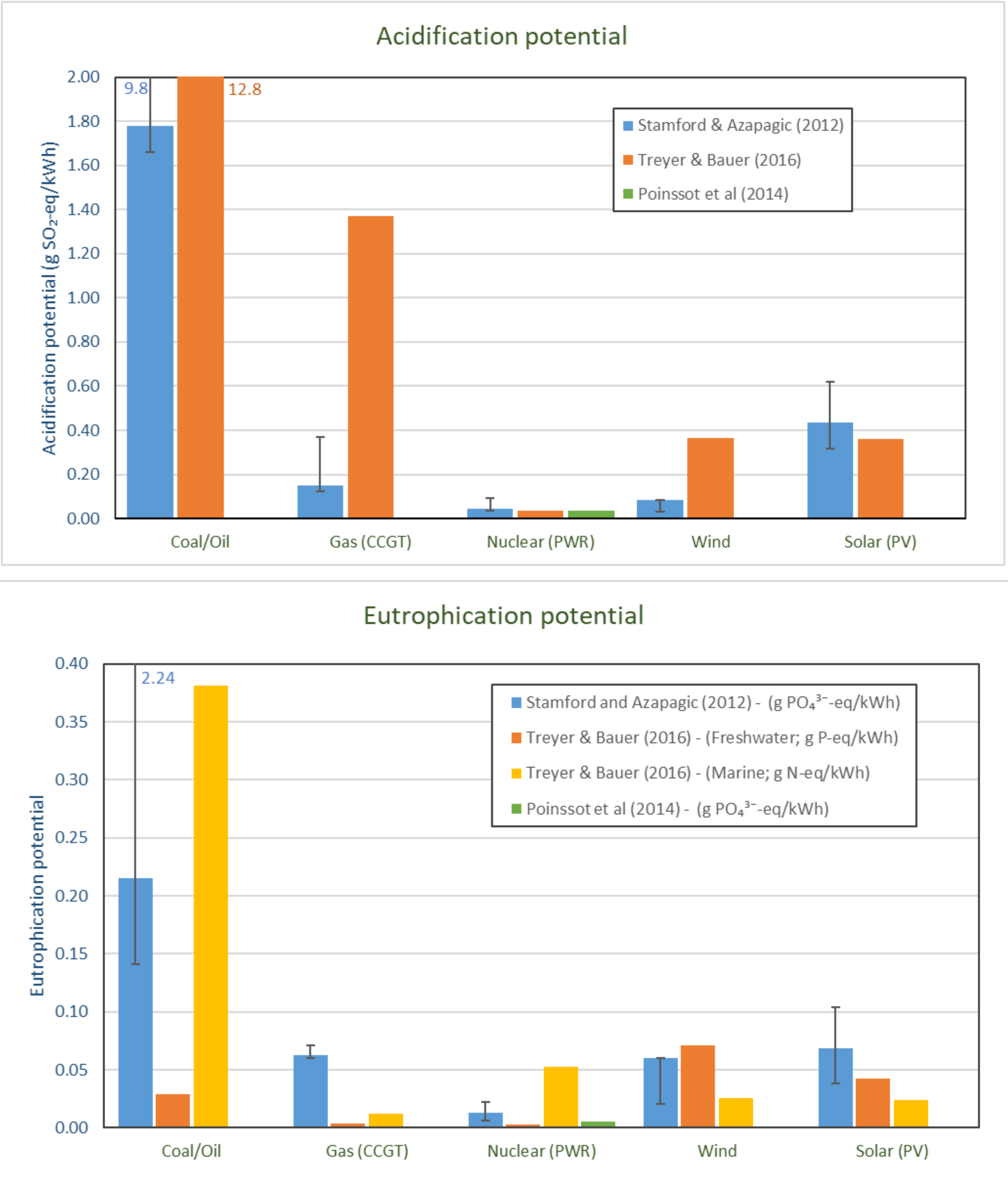
Note regarding comparisons between the data from the above two references: Due to environmental conditions, the potential of solar energy in UAE is high compared to the UK, whereas the potential for wind is low compared to the UK; this will have an impact on the calculated indicators particularly for these two technologies.

Poinssot et al [3.2-8]
LCIA methodology: NELCAS (CEA proprietary tool).
Electricity generation technologies: Nuclear (PWR).
Nuclear energy: Current French nuclear fleet; plutonium recycling in MOX fuel. Data from publicly available annual environmental reports of the different French nuclear installations.

Refer to the original references for further information on data and assumptions.

* CCGT – Combined cycle gas turbine; NGCC – Natural gas combined cycle; CCS – Carbon capture & Storage; PV – photovoltaic; PWR – Pressurised-water reactor; CSP – Concentrating Solar Power; MOX – Mixed-oxide (uranium & plutonium) fuel; EPR – European Pressurised-water Reactor.
[a] included in the comparisons in Chapter 3.2
None of the fossil-based technologies included in the comparisons in Chapter 3.2 include CCS.

Figure 3.2-9. Acidification and Eutrophication potentials of electricity generation technologies



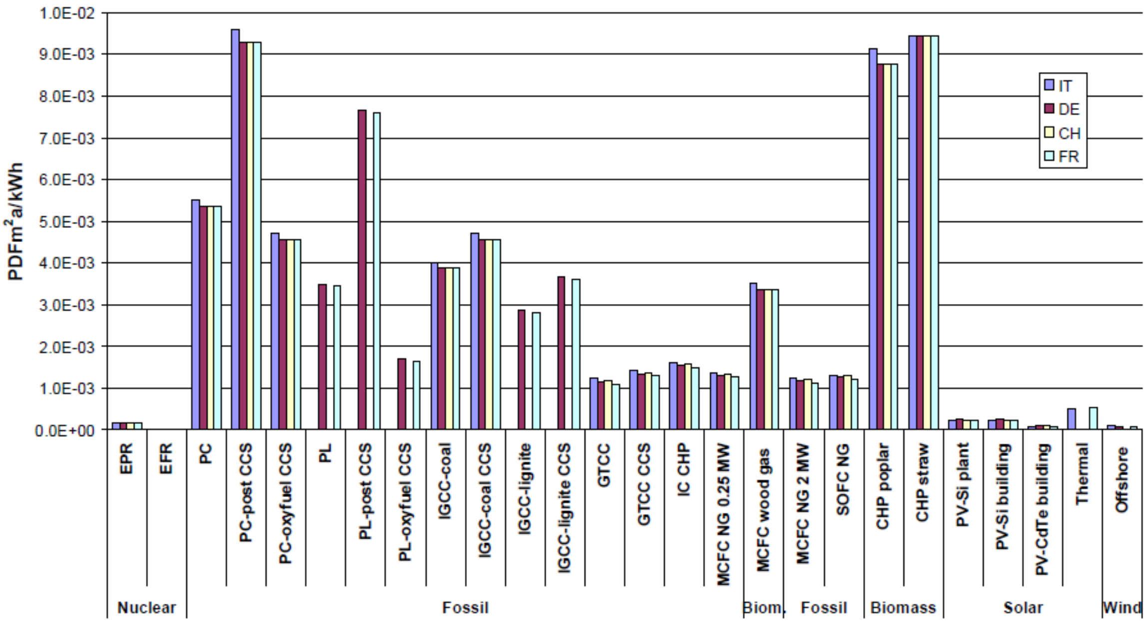
Data from: [3.2-9], [3.2-23], [3.2-8]

Acidification and eutrophication potentials were also compared in the NEEDS project [3.2-10], in this case using a single combined end-point indicator[34] quantifying the loss of species (flora & fauna) due to the release of substances to air, water, and soil. The indicator is given in terms of Potentially Disappeared Fraction of species on 1 m^2 of earth surface during one year (PDFm^2a) per kWh electricity produced. The comprehensive comparison is shown in Figure 3.2-10 (from [3.2-10]).

Nuclear energy is represented by a current generation III PWR (European Pressurised-water Reactor) and a future fast breeder reactor option based on the European Fast Reactor (EFR). Data are presented for Italy, Germany, Switzerland and France, taking into account local conditions for each technology. Again, nuclear energy can be seen to be one of the best performers for this specific indicator.

[34] Calculated following the methodology of the Life Cycle Impact Assessment method Eco-indicator 99 [3.2-24] and covering complete energy chains.

Figure 3.2-10. Results of the environmental impact indicator: Acidification and eutrophication[35]



Source: Ref. [3.2-10]

Water ecosystems are also damaged by toxic chemical releases, including heavy metals, volatile organic compounds (VOCs) and particles. Various ecotoxicity indicators have been used in sustainability assessments to compare technologies in terms of the toxic damage potential of their lifecycle chemical emissions.

Freshwater aquatic ecotoxicity potential (FAETP) refers to the impact on fresh water ecosystems, as a result of emissions of toxic substances to air, water and soil. Marine ecotoxicity refers to impacts of toxic substances on marine ecosystems. Both indicators are expressed as grams 1,4-dichlorobenzene equivalents/kWh (g 1,4-DCB-eq/kWh).

Stamford & Azapagic [3.2-9], as well as Treyer & Bauer [3.2-23], compared both fresh water and marine ecotoxicity potentials[36] of several electricity generating technologies. The results are provided in Figure 3.2-11.

With regard to freshwater ecotoxicity, nuclear energy is again the best performer according to Treyer & Bauer, whereas the results of Stamford & Azapagic rank natural gas as best, with the other technologies fairly evenly matched, although nuclear has the potential to be comparable with gas according to the sensitivity studies. The data of Poinssot et al again compare very well with the data of Treyer & Bauer and the lower bound data of Stamford & Azapagic. Concerning nuclear, the bulk[37] of the impact is due to metals such as vanadium, copper and beryllium coming from uranium mill tailings. Regarding marine ecotoxicity, nuclear is again ranked best (Treyer & Bauer – ReCiPe methodology) or second best (Stamford & Azapagic – CML methodology[38]) along with natural gas.

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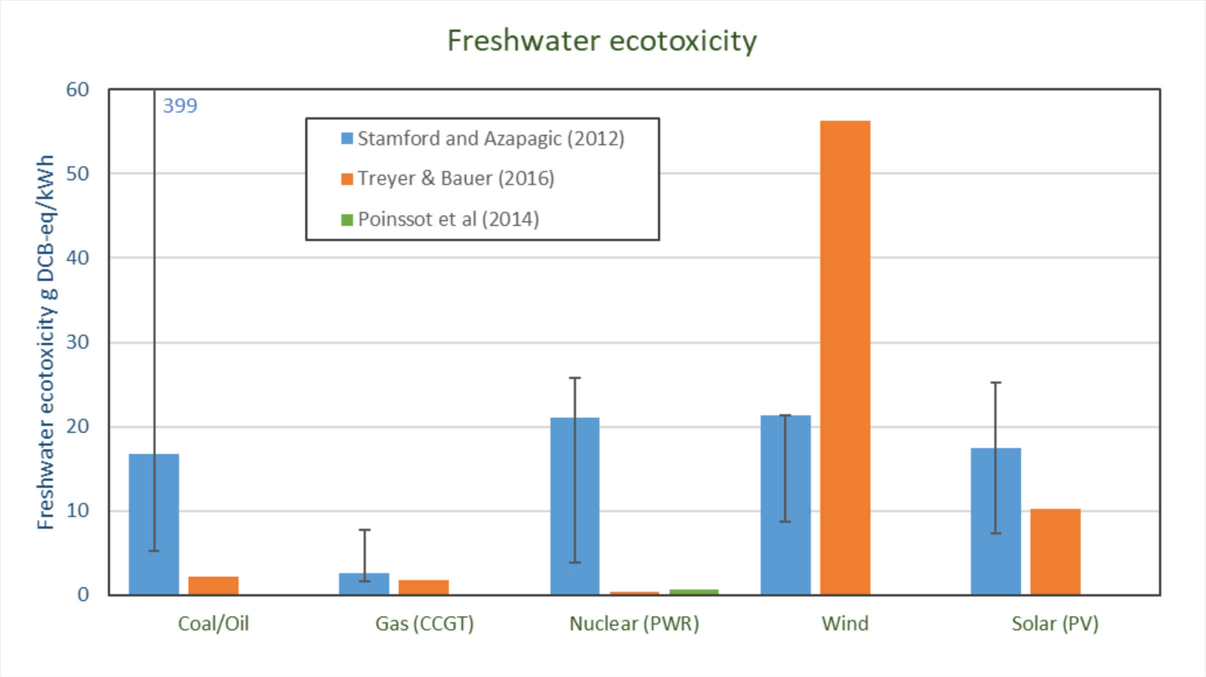
[35] EPR – European Pressurised-water Reactor; EFR – European Fast Reactor; PC – Pulverised coal; PC-post CCS - Pulverized Coal with post combustion Carbon Capture and Storage; PC-oxyfuel CCS - Pulverized Coal with oxyfuel combustion and CCS; PL - Pulverized Lignite; PL-post CCS - Pulverized Lignite with post combustion Carbon Capture and Storage; PL-oxyfuel CCS - Pulverized Lignite with oxyfuel combustion and CCS; IGCC-coal - Integrated Gasification Combined Cycle coal; IGCC-coal CCS - Integrated Gasification Combined Cycle coal with CCS; IGCC-lignite - Integrated Gasification Combined Cycle lignite; IGCC-lignite CCS - Integrated Gasification Combined Cycle lignite with CCS; GTCC - Gas Turbine Combined Cycle; GTCC CCS- Gas Turbine Combined Cycle with CCS; IC CHP - Internal Combustion Combined Heat and Power; MCFC NG 0.25 MW - Molten Carbonate Fuel Cells using Natural Gas 0.25 MW; MCFC wood gas - Molten Carbonate Fuel Cell using wood derived gas 0.25 MW; MCFC NG 2 MW - Molten Carbonate Fuel Cells using Natural Gas 2 MW; SOFC NG - Solid Oxide Fuel Cells using Natural Gas 0.3 MW; CHP poplar - Combined Heat and Power using short rotation coppiced poplar; CHP straw - Combined Heat and Power using straw; PV-Si plant - Photovoltaic, ribbon crystalline Silicon - power plant; PV-Si building - Photovoltaic, ribbon crystalline Silicon - building integrated (rooftop); PV-CdTe building - Photovoltaic Cadmium Telluride – building integrated (rooftop); Thermal - Concentrating solar thermal – power plant.

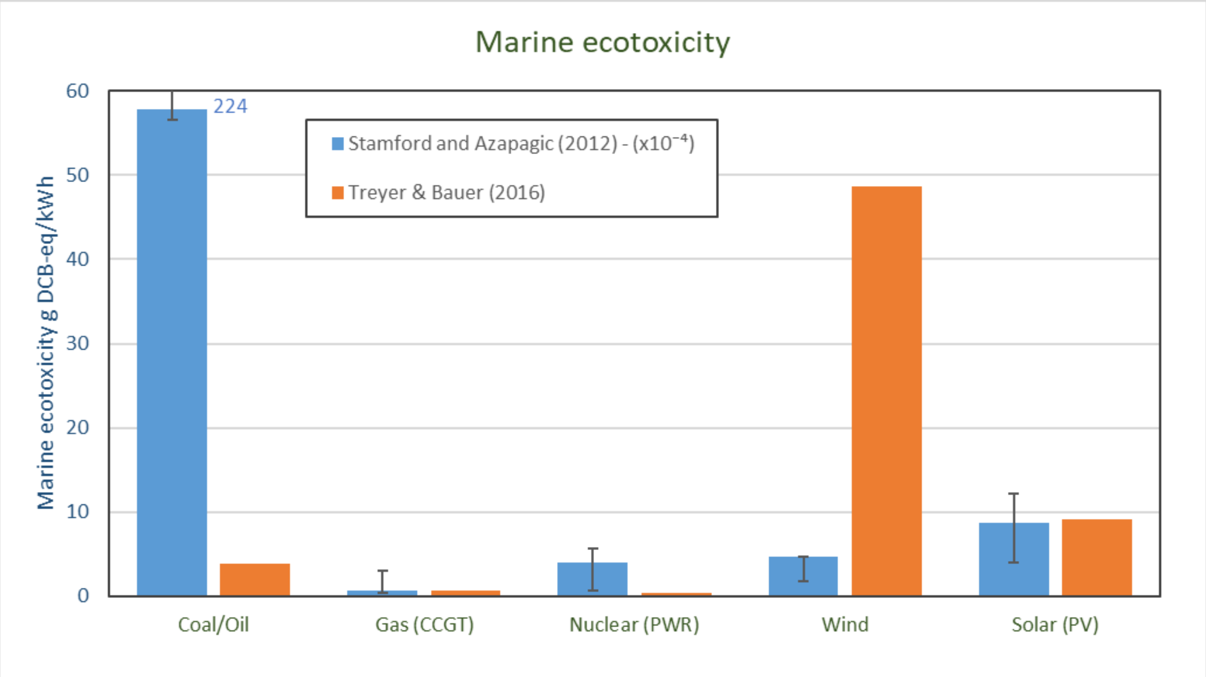
[36] Calculated according to the CML 2001 Impact Assessment Methodology providing 'midpoint' impact indicators.

[37] More than 70% according to Stamford & Azapagic [3.2-9]; 99% according to Poinssot et al [3.2-8]

[38] Note that the CML methodology produces significantly larger values for the marine ecotoxicity than the ReCiPe methodology. The values calculated by Stamford & Azapagic have been multiplied by 10^{-4} to allow them to be reported on the same scale in Figure

Figure 3.2-11. Aquatic ecotoxicity potentials of various electricity generation technologies





Data from [3.2-9], [3.2-23], [3.2-8]

Aquatic ecotoxicity associated with nuclear energy would not therefore appear to constitute a reason for exclusion of nuclear energy from the taxonomy as it is comparable with, or better than, other technologies included in the Taxonomy. However, the dominant contribution of mining and milling to freshwater ecotoxicity will be further discussed in Chapter 3.3.1 and in the related TSC in Chapter 5 and Annex 4.

With regard to thermal pollution of water bodies, nuclear power plants using once-through cooling systems withdraw water and return it at increased temperature. Elevated temperatures in the receiving water bodies can negatively affect aquatic ecosystems. There is little information on the assessment of thermal pollution of water bodies in the lifecycle sustainability assessments in the literature. However, thermal pollution is

3.2-11. This is a feature of the methodology, as calculations made with the two methodologies using identical lifecycle inventory data also exhibit such large differences in the results (Stamford [3.2-25]).

tightly controlled and measures are taken to maintain temperature increases within acceptable limits in order to avoid harm to the aquatic ecosystems. In periods of drought or heatwaves, it has sometimes been necessary to reduce power or shutdown nuclear power plants in order to keep thermal pollution of water bodies within the statutory limits. However, thermal pollution is not unique to nuclear energy and other electricity generation based on thermal technology and using water for cooling have similar effects. Compliance with EU water legislation is the guarantee of absence of significant harm.

Nuclear power plants may have to operate at reduced power or shut down in cases of extreme prolonged dry weather or high ambient temperature, when cooling water intake levels become too low or when the limits on the temperature of water returned to watercourses is exceeded. However, this does not pose any safety risk and is a very rare occurrence, as very extreme weather conditions are taken into account in the design of the plants.

There is no commonly used impact indicator specifically to characterise radiological pollution of water bodies. The commonly used ionising radiation impact indicator characterises the human health impact of radiation reaching the human body through all relevant pathways. This is discussed more in Chapter 3.2.5. Radiological releases to the environment are subject to strict limits. More information on related EU legislation, including the Euratom Basic Safety Standards Directive and the Euratom Drinking Water Directive is provided in Annex 1.

In summary, there is no evidence that nuclear energy does more harm to the sustainable use and protection of water and marine resources than other energy technologies included in the Taxonomy. However, with regard to this environmental objective, water consumption during the operation of nuclear power plants and the contribution of uranium mining and milling to pollution of water bodies will be discussed further in Chapters 3.3.7 and 3.3.1 respectively. Related TSC are discussed in Chapter 5 and Annex 4.

3.2.4 DNSH to the transition to a circular economy, including waste prevention & recycling

In accordance with article 17 of the Taxonomy Regulation, an economic activity shall be considered to cause significant harm to the transition to a circular economy, including waste prevention and recycling, where:

(i) that activity leads to significant inefficiencies in the use of materials or in the direct or indirect use of natural resources such as non-renewable energy sources, raw materials, water and land at one or more stages of the lifecycle of products, including in terms of durability, reparability, upgradability, reusability or recyclability of products;

(ii) that activity leads to a significant increase in the generation, incineration or disposal of waste, with the exception of the incineration of non-recyclable hazardous waste; or

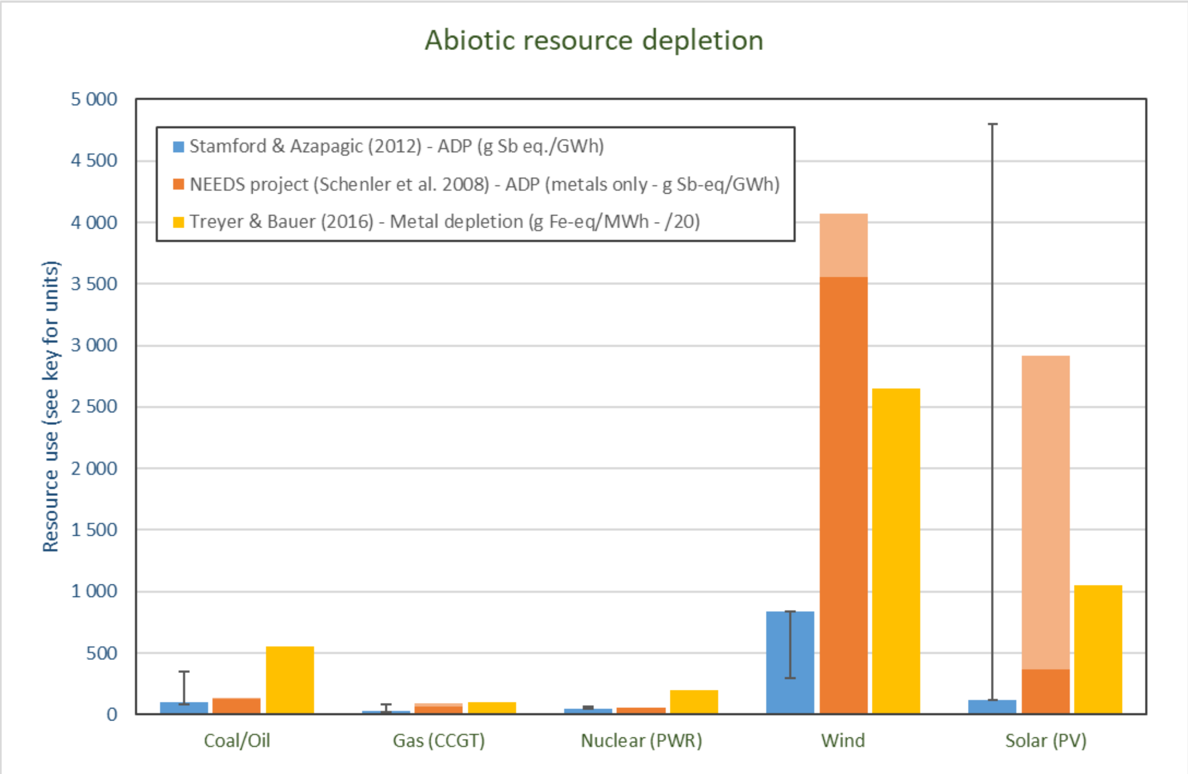
(iii) the long-term disposal of waste may cause significant and long-term harm to the environment.

Abiotic Depletion Potential (ADP) is an indicator frequently used in lifecycle assessments to characterise the utilisation of natural resources. Abiotic depletion refers to the depletion of non-living (abiotic) resources such as metals, minerals and fossil energy. The scarcity of the different natural resources used is a factor in the calculation of the indicator. It is measured in kilograms of Antimony (Sb) equivalents reflecting the scarcity of the different resources relative to the reference ore (antimony). Clearly, technologies having lower values of depletion potential are better from the point of view of sustainability.

There is a paucity of published data comparing ADP for nuclear energy with other energy generating technologies. Data from three studies are compiled in Figure 3.2-12. Stamford & Azapagic [3.2-9] provide central estimates plus ranges corresponding to sensitivity analyses. This is the ADP-elements indicator following the CML methodology and relates to the depletion of metal and non-metal mineral resources. The data from the NEEDS project [3.2-26], also calculated according to the CML methodology, is however limited only to the use of metallic ores, as is the data of Treyer & Bauer [3.2-23]. The latter is calculated according to the ReCiPe methodology and is given in units of iron-equivalent instead of antimony-equivalent. Maximum and minimum values from the NEEDS data are shown in the figure (dark and light bars). These maximum and minimum values correspond only to national differences in the implementation of the different technologies, except for solar PV. The minimum values for solar PV correspond to the use of CdTe panels, whereas the maximum values correspond to the use of Si panels. Stamford & Azapagic consider PV panels according to the average world mix[39]. The sensitivity analyses of Stamford & Azapagic investigated different end-of-life recycling rate assumptions as well as different installation situations (on building facades and flat roofs instead of slanted roofs). Solar PV shows relatively high sensitivity to the different assumptions.

Despite the differences in absolute values for Solar PV and wind, there is a clear ranking of technologies, with nuclear and gas having the lowest ADP followed by coal/oil, solar and wind.

Figure 3.2-12. Use of natural resources



Data from [3.2-9], [3.2-26] and [3.2-23]

In addition to the use of abiotic metals and mineral resources, the same authors also provide assessments of fossil fuel resource use of the different electricity generation technologies. The respective data are shown in Figure 3.2-13.

Nuclear and wind have very low fossil fuel use. The lifecycle of solar PV has a slightly higher fossil fuel usage than wind and nuclear.

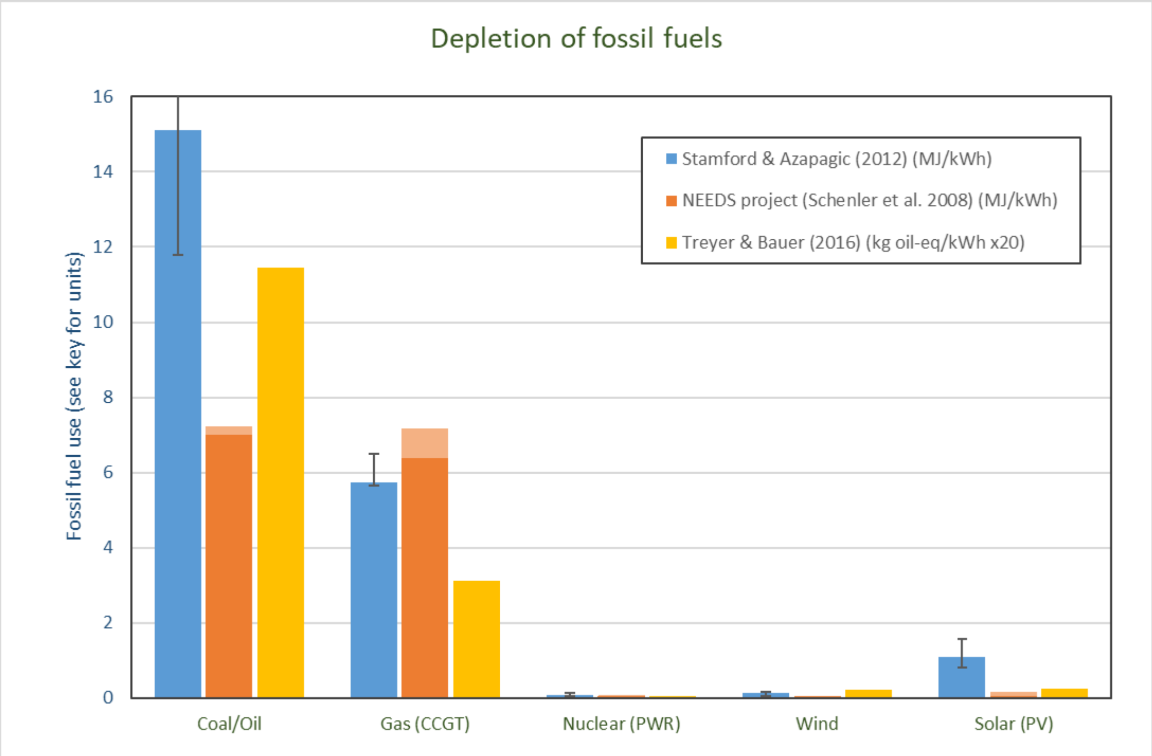
Of course, nuclear is the only technology with significant use of uranium resources. Current thermal reactor technologies are only capable of utilising a small fraction of the potential energy contained in the mined natural uranium. Advanced reactors utilising a fast-neutron spectrum operating in a closed fuel cycle40 would be capable of extracting around 50 times more energy from the natural uranium, but these reactors are not yet deployed on a commercial scale. The utilisation of uranium in current reactors in an open fuel cycle, in which the spent fuel is disposed of in a final repository, therefore results in the disposal of plutonium and uranium-238 that could potentially be used to generate energy in a future closed fuel cycle. However, for every metric ton of natural uranium feed, only about 120-130 kg of enriched uranium fuel for use in current reactors is produced. The remaining 870-880 kg end up as depleted uranium in the enrichment tails. This depleted uranium is retained, and can be utilised in future advanced reactors. The already accumulated stocks of depleted uranium, when used in a closed fuel cycle with advanced reactors, will be sufficient for several

39 98.4% Si panels of various types (mono-crystalline, multi-crystalline, amorphous and ribbon panels and laminates – refer to the original reference for details) and 1.6% CdTe and CIGS (cadmium-indium-gallium-selenide)

40 See Chapter 3.3.5 for more information on open and closed fuel cycles and the material content of spent LWR fuel.

centuries of nuclear power generation at present global levels. Currently known reserves of uranium, used in the same way, extend this time frame to a few millennia.

Figure 3.2-13. Use of fossil fuels

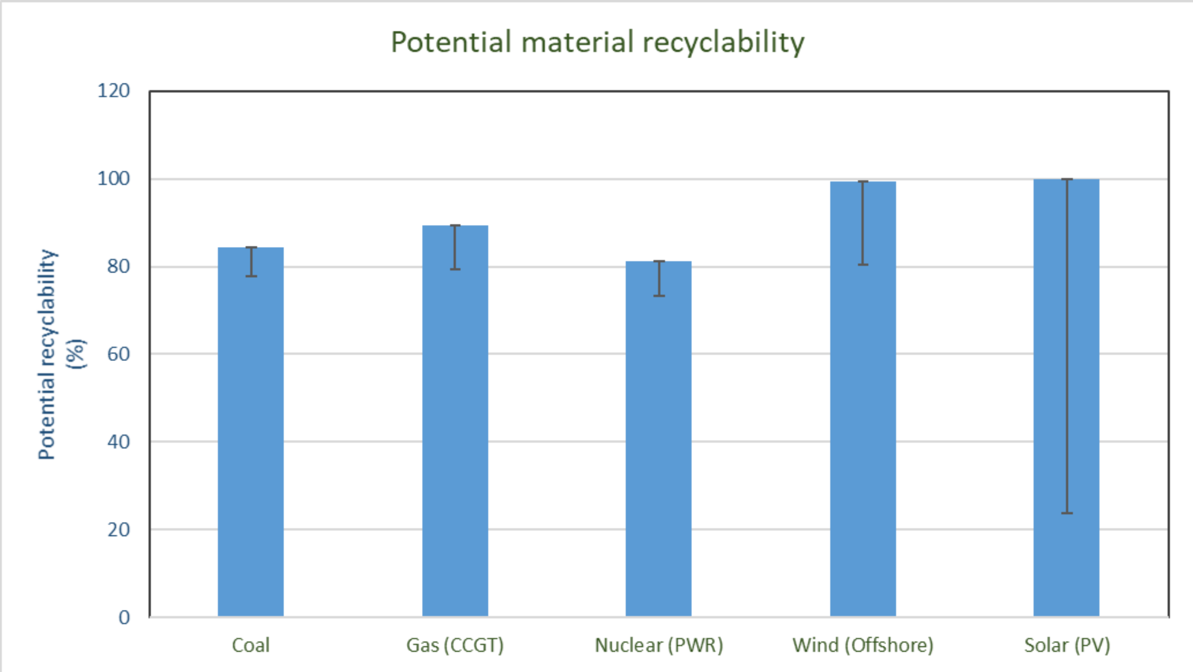


Data from [3.2-9], [3.2-26] and [3.2-23]

Recyclability of materials is also an important factor when considering efficiency in the use of natural resources. Stamford & Azapagic [3.2-9] also calculated the potential material recyclability ratios for the different technologies. These are shown in Figure 3.2-14. The calculation is based on the inventory of the different materials used in plant construction and their potential recyclability (e.g. most metals are 100% recyclable, concrete is 79.4% recyclable, etc.). For nuclear energy, the fact that a proportion of the materials become too activated for reuse is taken into account in the assessment, and this proportion, which is less than 5%, is excluded. However, the central estimates for the other technologies do not take into account the recoverability or ease of recovery of the materials for recycling. This is taken into account in the sensitivity analyses, which explore the effect on the recyclability of the materials if current UK demolition recycling rates are considered for the major components. As the authors point out, decommissioning an offshore wind farm typically involves leaving a mass of steel in the seabed to reduce cost and minimise disruption to benthic life, and for typical solar PV modules, solar glass coated in metal oxides makes up a large part of total mass and this may pose recycling difficulties. Consequently, the high potential recycling rates calculated for wind and solar may be difficult to achieve in practice and this is reflected in the range of values resulting from the sensitivity analyses (also shown in Figure 3.2-14).

Recycling rates for coal, gas and nuclear are similar, with the limits being due largely to the extensive use of concrete, which is considered only 79.4% recyclable. The rate for nuclear is slightly lower also due to the small proportion of activated materials that are not recyclable.

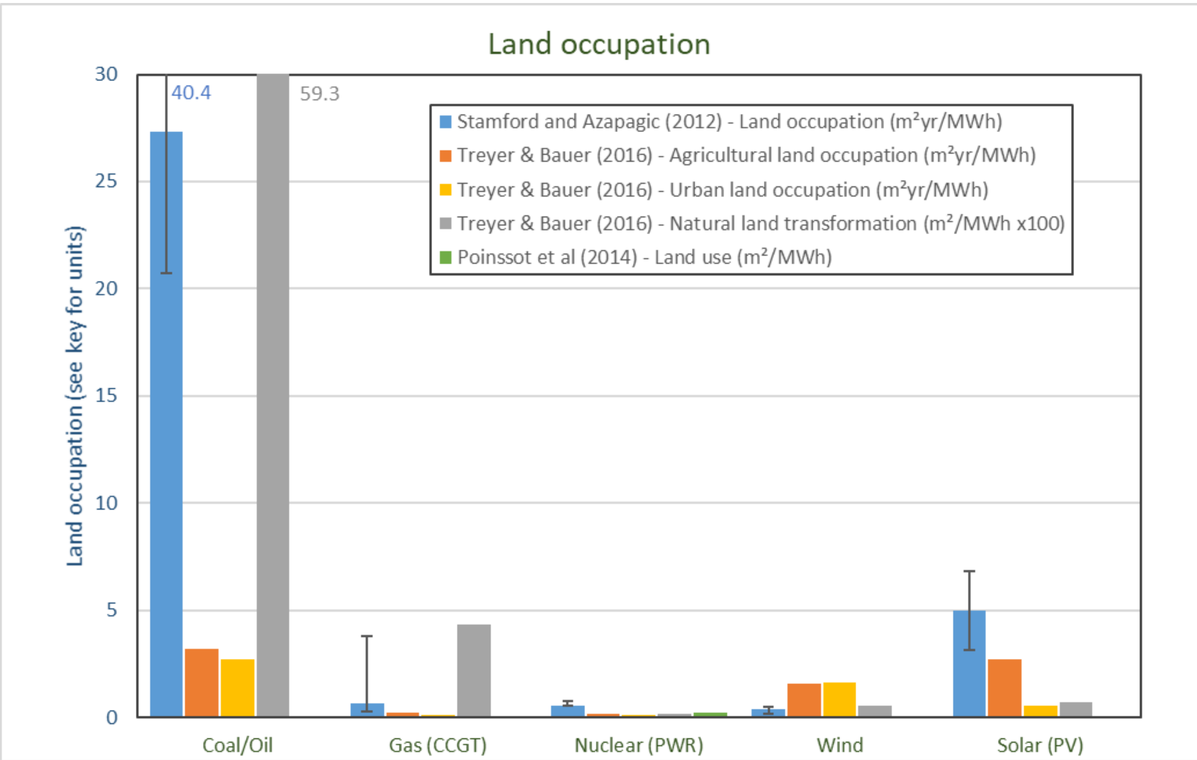
Figure 3.2-14. Potential material recyclability



Data from [3.2-9]

Stamford & Azapagic [3.2-9], as well as Treyer & Bauer [3.2-23], also calculated the land occupation of the various technologies. The data are shown in Figure 3.2-15, along with the land use data calculated by Poinssot et al [3.2-8] for the nuclear energy chain.

Figure 3.2-15. Land occupation



Data from [3.2-9], [3.2-23] and [3.2-8]

Lifecycle land occupation of coal power is significantly greater than the other technologies. The second largest is solar PV. According to Stamford & Azapagic [3.2-9], for solar PV, 95% of the land occupation is associated with the production of the metals for the manufacture of the panels and 5% for the panel manufacturing sites. The authors assumed installation of panels on roof spaces, which does not therefore contribute to the land occupation as this space is not in competition with other potential uses. Solar farms would be expected to have a greater land occupation. Land occupation by offshore wind (Stamford & Azapagic), nuclear and gas are negligible. Onshore wind (Treyer & Bauer) has greater land occupation.

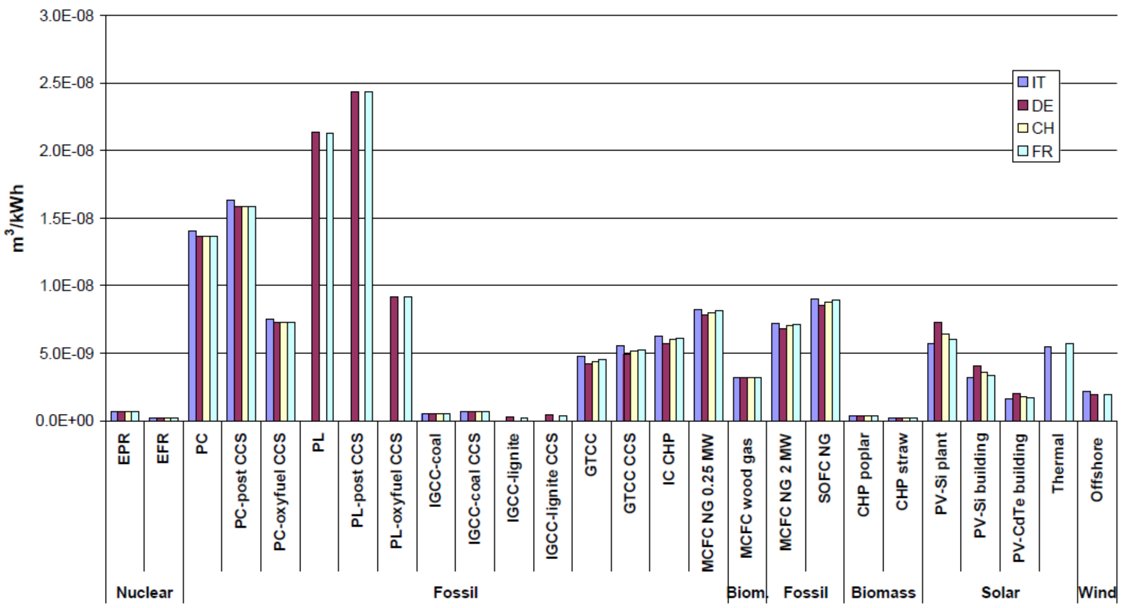
With regard to water usage of different energy generation technologies, this has already been discussed in Chapter 3.2.3 above and will not be repeated here.

Also important under the environmental objective related to the transition to a circular economy are the waste streams from the different energy technologies that require storage in repositories.

Waste streams were considered and compared in the NEEDS project [3.2-10], in this case using two separate indicators, one for chemical wastes and one for radioactive wastes. The indicators are given simply in units of m^3 of waste requiring storage/disposal[41] in a repository per unit of electricity generated (m^3/kWh), and so it does not provide any measure of the potential harm to humans or nature, neither does it reflect the confinement time necessary to prevent future damage to the environment. The comprehensive comparisons are shown in Figures 3.2-16 & 17 (from [3.2-10]).

It can be seen that nuclear energy produces relatively small quantities of chemical wastes requiring storage, even compared to renewable technologies. Of course, nuclear energy produces the largest amount of radioactive wastes[42]. Radioactive waste quantities produced by the European Fast Reactor (EFR) are considerably less than for the European Pressurised-water Reactor (EPR) as the fuel is recycled so spent fuel assemblies do not go into the waste stream. In volumetric terms, the amount of radioactive waste produced by nuclear energy operated on the basis of PWRs (EPR) is somewhat less than the amount of chemical waste requiring storage/disposal in a repository produced by some fossil technologies and comparable with (slightly higher than) the amount of chemical waste from some solar PV technologies.

Figure 3.2-16. Chemical waste volumes from different electricity generation technologies[43]



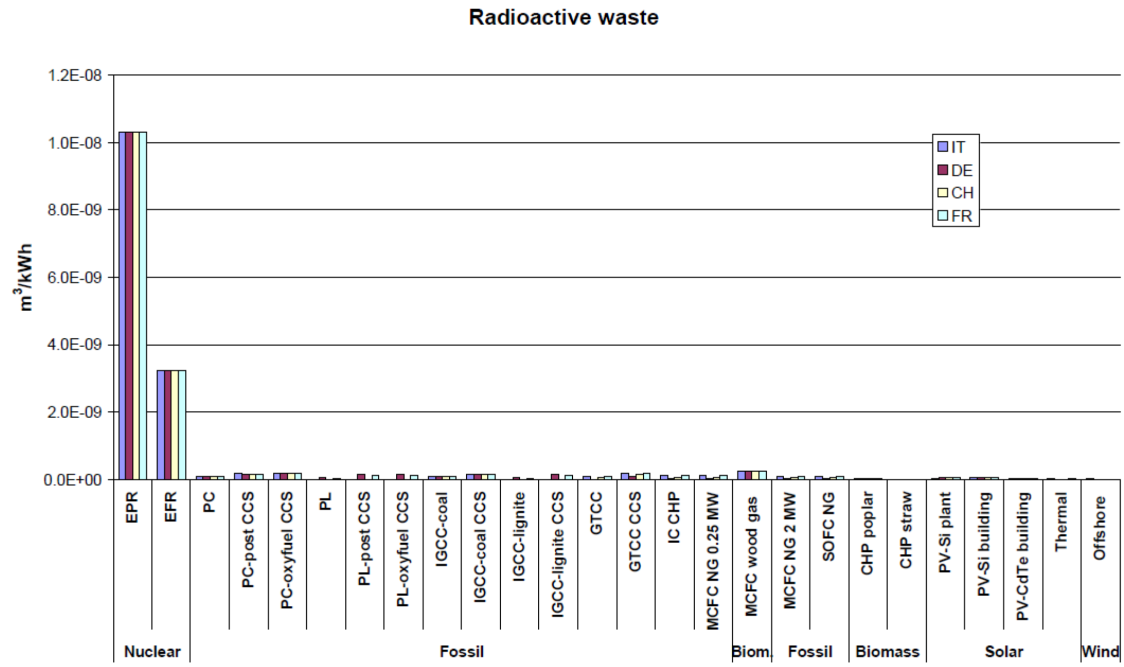
Source: [3.2-10]

[41] This is the volume occupied in the final repository or underground deposit.

[42] Note that the radioactive wastes associated with the non-nuclear technologies shown in the figure mainly reflect the fact that nuclear energy is part of the energy mix supplying the different technologies with part of their energy needs.

[43] For the key to the technologies, refer to footnote 35.

Figure 3.2-17. Radioactive waste volumes from different electricity generation technologies[44]



Source: [3.2-10]

Note that some countries (e.g. France) do not consider spent fuel to be waste. Spent fuel comprises large amounts of recoverable uranium and plutonium that can be used in fast breeder reactor fuel. While fast breeder reactors are not deployed yet on a large-scale commercial basis, they are very much an option for the future for some countries, and so the uranium and plutonium within the spent fuel is considered a valuable resource. Poinssot et al [3.2-8] calculates the total amount of radioactive waste requiring geological disposal at about 1.5 m^3/TWh_e for the current French nuclear fleet with plutonium recycled once in MOX fuel. This strategy reduces the amount of waste requiring geological disposal, which is almost an order of magnitude less than the amount shown in Figure 3.2-17. This reflects the fact that spent fuel elements (including spent MOX fuel elements) are not included in the waste stream in France.

In summary, there is no evidence that nuclear energy does more harm to the transition to a circular economy, including waste prevention and recycling, than other energy technologies included in the Taxonomy. However, with regard to radioactive wastes specifically, clearly nuclear energy produces larger quantities than other generation technologies. Radioactive waste and its management will be discussed in detail in Part B of this report.

3.2.5 DNSH to pollution prevention and control

In accordance with article 17 of the Taxonomy Regulation, an economic activity shall be considered to cause significant harm to pollution prevention and control where:

(i) that activity leads to a significant increase in the emissions of pollutants into air, water or land, as compared with the situation before the activity started.

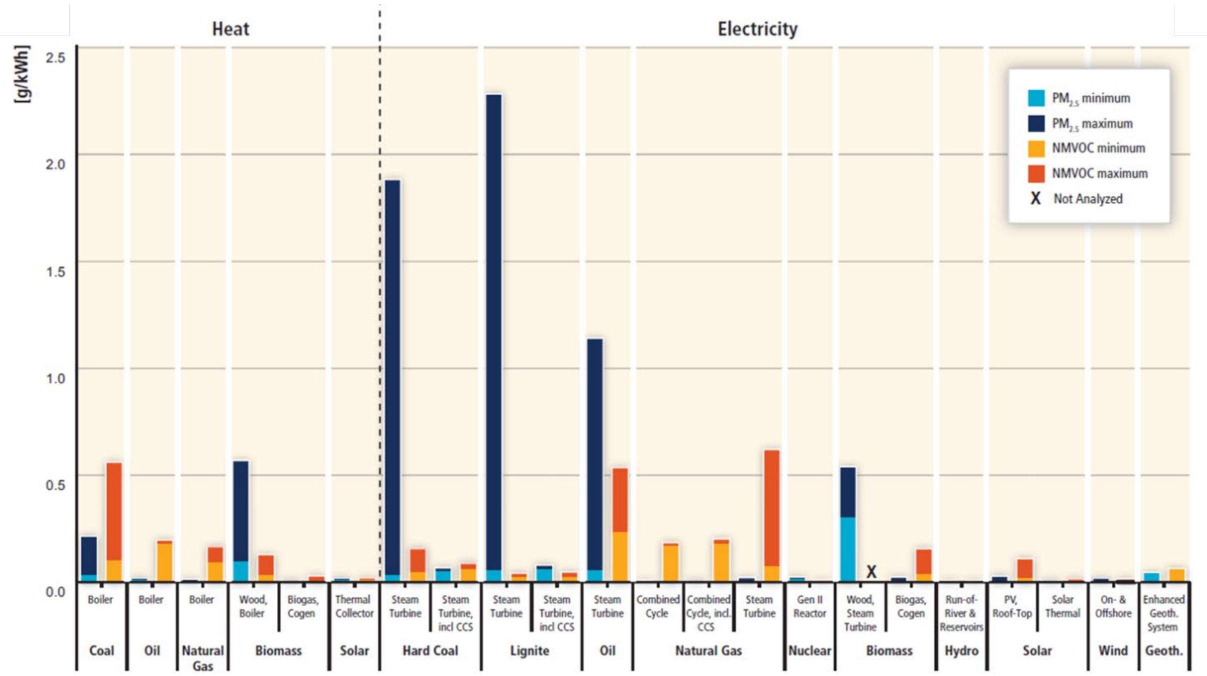
Pollutants having a specific impact on water and marine resources are discussed in Chapter 3.2.3 and will not be considered again here. This section will deal with other pollutants and some of the related sustainability indicators used to characterise their impacts on human health and the environment. A number of such indicators for comparing chemical pollution and its potential impacts have been used in lifecycle assessments in the literature. The more common ones include direct emissions of particulate matter (PM), nitrous oxides (NO_x), sulphur dioxide (SO_2) and non-methane volatile organic compounds (NMVOC), as well as impact indicators for ozone layer depletion, photochemical smog, human toxicity potential and human health.

[44] For the key to the technologies, refer to footnote 35.

Indicators to measure the potential for radiation effects on health and environment have also been developed and compared.

Direct emissions of NO_x and SO_2 were discussed in Chapter 3.2.3 (Figure 3.2-8), where it was seen that nuclear energy compares very favourably to a range of other electricity generation technologies, including renewables. Figure 3.2-18 [Ref 3.2-11] compares PM and NMVOC data for the lifecycle of the same range of (heat and) electricity generation technologies and clearly shows that nuclear energy, based on current Generation II power plants, has very low emissions of these substances compared to fossil fuel technologies and is comparable with renewable technologies. Fossil fuel technologies have the largest emissions of both PM and VOC.

Figure 3.2-18. Cumulative lifecycle emissions of NMVOC and PM2.5 per unit of energy generated for current heat and electricity supply technologies[45]



Source: [3.2-11]

Ozone Depletion Potential (ODP) represents the potential of depletion of the ozone layer due to the emissions of chlorofluorocarbon compounds and chlorinated hydrocarbons. The ODP of the different contributing substances are converted to an equivalent quantity of CFC-11 and the indicator is expressed in units of µg CFC-11 eq/kWh.

Photochemical Oxidant Creation Potential (POCP) or photochemical smog is caused by the creation of ozone from volatile organic compounds (VOCs) and nitrogen oxides in the presence of sunlight. Although ozone is critical in the high atmosphere to protect against ultraviolet light, low-level ozone is implicated in impacts as diverse as crop damage and increased incidence of asthma and other respiratory complaints. POCP is usually expressed relative to the oxidant creation potential of ethylene and is expressed using the reference unit, g C_2H_4 eq/kWh.

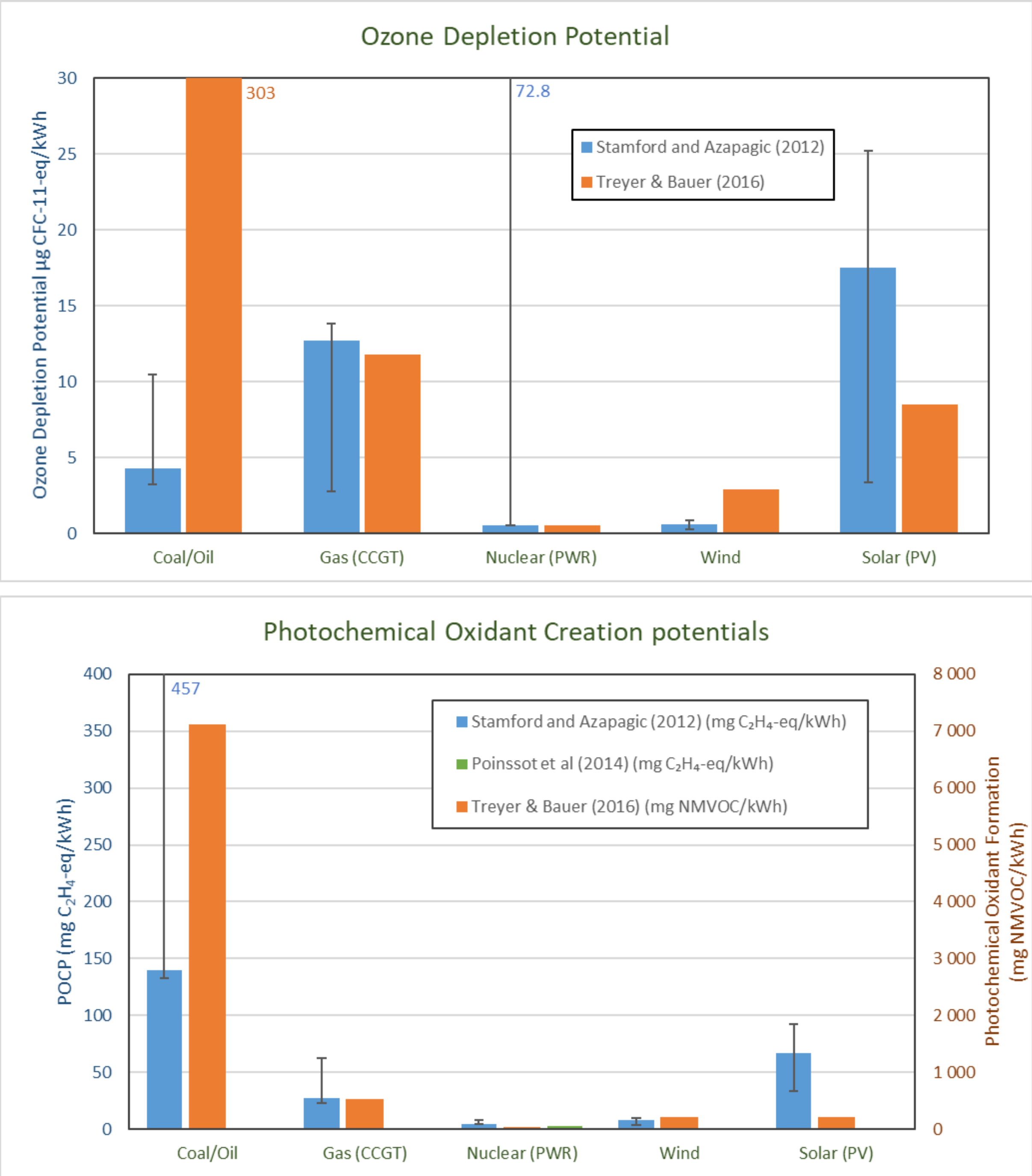
The ODP and POCP calculated by Stamford & Azapagic [3.2-9] and Treyer & Bauer [3.2-23] for several electricity generation technologies, and the POCP of Poinssot et al [3.2-8] for nuclear energy, are provided in Figure 3.2-19.

Nuclear energy is the best performer in both categories. According to the authors of [3.2-9], the calculated ODP upper bound of 73 µg CFC-11 eq/kWh coming from the sensitivity studies for nuclear energy is anomalous. It results from considering the impact of enriching fuel using the diffusion process, but is based

[45] Data from Bauer, 2008; Viebahn et al., 2008; Ecoinvent, 2009; traditional biomass use not considered. Figures for coal and gas power chains with CCS are valid for near-future forecasts (Bauer et al., 2009).

on United States Enrichment Corporation (USEC) diffusion plants, which were still using Freon as coolant. This is no longer relevant, as USEC's diffusion plants are no longer operational.

Figure 3.2-19. Ozone Depletion & Photochemical Oxidant Creation potentials of electricity technologies

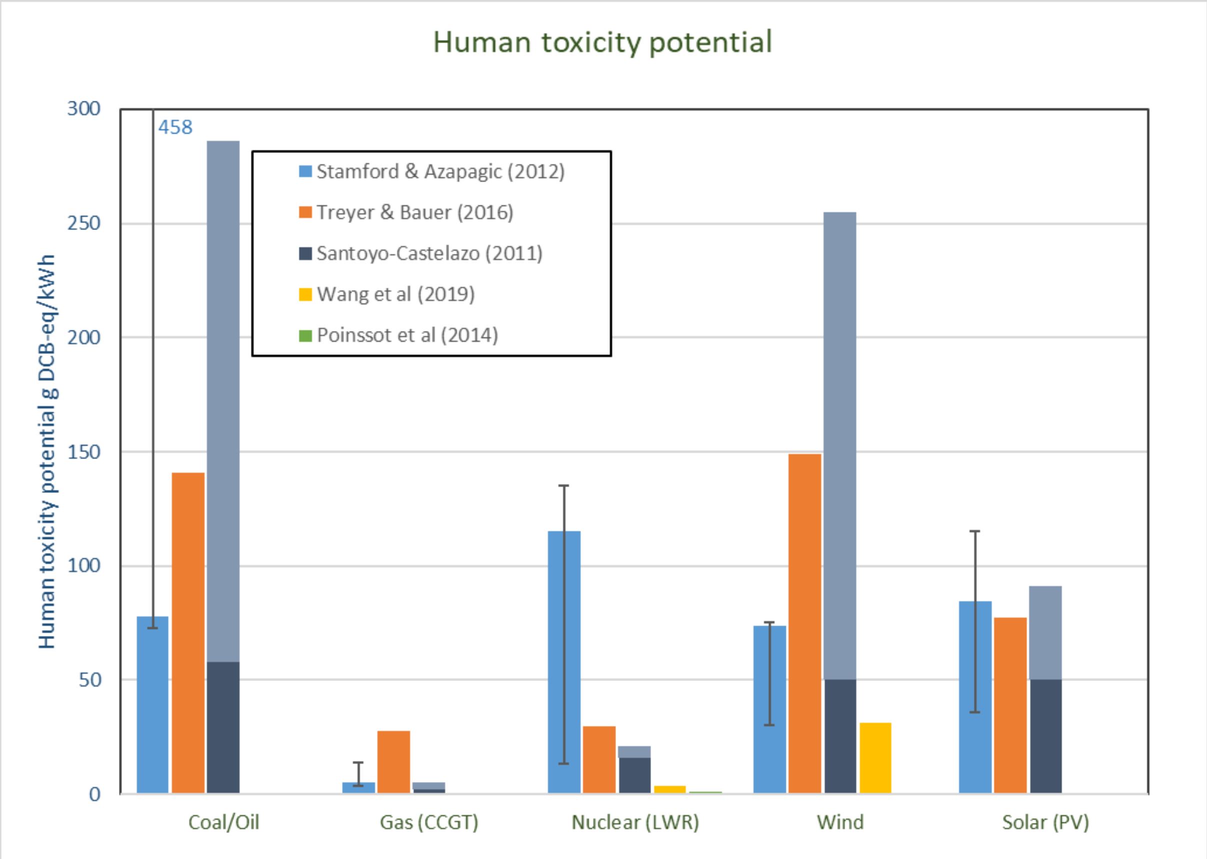


Data from [3.2-9], [3.2-23], [3.2-8]

Human toxicity potential (HTP) is a measure of the effect of toxic substances on human health considering all exposure routes for all chemicals for an infinite timeframe. Important contributing substances include heavy metals as well as particulate matter, SO_x and NO_x emissions, volatile organic compounds (VOC) and chlorinated organic compounds among others. The indicator used to categorise human toxicity potential is measured in 1,4-dichlorobenzene equivalent/kWh.

There is some variability in the published data comparing HTP for nuclear energy with other energy generating technologies. Data from several recent studies are compiled in Figure 3.2-20. Stamford & Azapagic [3.2-9] provide central estimates plus ranges corresponding to sensitivity analyses while Santayo-Castelazo [3.2-27] provides minimum and maximum values corresponding to the ranges of assumptions for the different technologies.

Figure 3.2-20. Human toxicity potential of different electricity generation technologies[46]



Data from [3.2-9, 23, 27, 28 and 8]

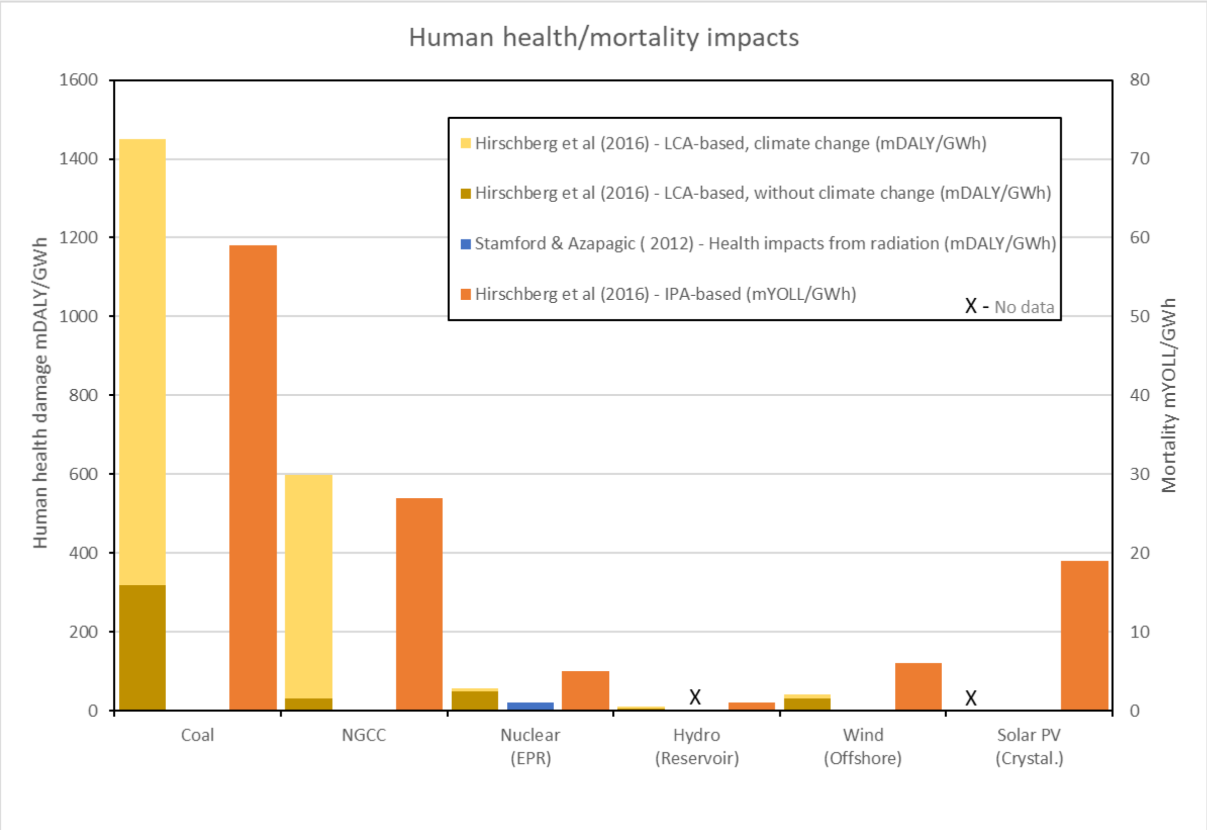
The general trend observed in Figure 3.2-20, with the exception of the Stamford & Azapagic result for nuclear, is that of the five technologies considered, gas has the lowest HTP followed by nuclear, then solar PV, wind and coal/oil all having higher values of HTP. The central estimate of Stamford & Azapagic for nuclear energy is high compared to the values obtained by other authors, although their lowest estimate from the sensitivity studies is well aligned with the data from the others. These sensitivity studies considered the recycling of plutonium in MOX fuel (from 0% to 8% MOX) and varying proportions of fuel enriched by gaseous diffusion (from 0% to 30%). The authors report that heavy metals, including arsenic and chromium, the bulk of which comes from the uranium mining and milling operations, contributes substantially to the HTP. Plutonium recycling in MOX fuel will reduce the need for fresh uranium from mining operations, thereby reducing the related contribution to the HTP. Poinssot et al [3.2-8], also report that the main contributor (99%) to HTP for the nuclear energy chain is mining, but has molybdenum as the main source, followed by selenium and vanadium. There are clearly some discrepancies between the methodologies with regard to the characterisation of the contribution of the different elements to the toxicity potential. Stamford & Azapagic specifically point out that there is currently disagreement between LCA impact methodologies over HTP results, with CML and IMPACT2002+ methodologies giving higher HTP values for nuclear and solar PV, whereas Eco-Indicator 99, EDIP2003 and RECIPE methodologies all show coal as having the highest human health impact. The data of Treyer & Bauer [3.2-23] is calculated according to the RECIPE methodology.

46 Nuclear: Stamford & Azapagic [3.2-9]: future UK PWR; Santoyo-Castelazo [3.2-27]: Swiss PWR & BWR; Wang et al [3.2-28]: existing gen. II Chinese PWR; Treyer & Bauer [3.2-23]: European Pressurised-water Reactor (EPR); Poinssot et al [3.2-8]: current French reactor fleet. Wind: Stamford & Azapagic [3.2-9]: offshore; Santoyo-Castelazo [3.2-27]: both onshore and offshore; others: onshore. Data of Wang et al (nuclear and wind only). See also footnote 33.

This point is also illustrated in Figure 3.2-21, which presents end-point human health impacts for selected energy technologies calculated by Hirschberg et al [3.2-29] based on lifecycle impact analysis using the ReCiPe method[47]. The indicator resulting from this analysis is measured in units of DALYs (Disability Adjusted Life Years = years of life lost + years lived with a disability) per unit of electricity generated. The effects included are climate change (presented separately in Figure 3.2-21 – light gold bars), and human toxicity, ionizing radiation, photochemical oxidant formation, and particulate matter formation (presented together – dark gold bars). Note that ionizing radiation is not included in the human toxicity potential data presented in Figure 3.2-20. Stamford & Azapagic [3.2-9] calculated the radiation health effects separately, also in units of DALYs per unit of electricity generated, and these are also shown in Figure 3.2-21 (blue bars)[48]. For nuclear, they represent about one-third of the total human health impact (including radiation effects) calculated by Hirschberg et al (2016), although comparisons should be made with caution due to the different methodologies used by the two sets of authors.

Also shown on the figure are mortality data calculated by Hirschberg et al for the same technologies using an Impact Pathway Approach (IPA) based on methods developed in the European Union-funded ExternE research project [3.2-30]. The mortality impacts are quantified in terms of Years of Life Lost (YOLLs) per unit of electricity generated.

Figure 3.2-21. Human health and mortality impacts from different electricity generation technologies



Data from [3.2-29] and [3.2-9]

The authors note that the results of IPA and LCIA are not directly comparable. The approaches to estimation differ considerably, with LCIA results depending on the choice of a particular LCIA-method and to a higher

[47] The LCA calculations were made for three different social perspectives called Hierarchist (H), Egalitarian (E) and Individualist (I). The results presented here are only for the Hierarchist perspective, which can be interpreted as a kind of balanced compromise between the other two more extreme perspectives. The absolute results vary appreciably depending on the chosen perspective, but the ranking of the technologies is little affected, except for gas, whose impact is in the same range as nuclear and renewables under the Egalitarian perspective but considerably worse under the other perspectives. The Egalitarian perspective is less dominated by impacts due to climate change [3.2-29].

[48] Radiation health impacts calculated by Stamford & Azapagic [3.2-9] are included for all technologies except hydro, but as the values are very low compared to nuclear, they do not register on the graph when plotted using the same scale as the LCA data of Hirschberg et al (2016).

extent on subjective elements related to the various social perspectives while not allowing simulation of site-specific effects (as opposed to IPA). The health impact estimators have different scopes, i.e. YOLLs derived using IPA are a subset of DALYs generated using LCIA. The estimates based on LCIA cover not only health impacts of major pollutants but also the highly uncertain ones caused by the climate change; the latter are not included in IPA-estimates. However, while the absolute values calculated using the different methodologies are not comparable, the ranking of the different technologies is very similar for both methodologies.

The results presented here concern normal operation. Human health impacts of accidents are discussed in Chapter 3.5

According to the information presented in the foregoing, pollution arising from the whole lifecycle associated with the use of nuclear power for electricity production, and its effects on the environment and human health, is low when compared to fossil-based energy sources and is comparable with, or better than, some renewable technologies included in the Taxonomy. This includes health effects from radiation. Based on the above, nuclear energy cannot be considered to do significant harm to pollution prevention and control.

3.2.6 DNSH to the protection and restoration of biodiversity and ecosystems

In accordance with article 17 of the Taxonomy Regulation, an economic activity shall be considered to cause significant harm to the protection and restoration of biodiversity and ecosystems where that activity is:

(i) significantly detrimental to the good condition and resilience of ecosystems; or

(ii) detrimental to the conservation status of habitats and species, including those of Union interest.

Biodiversity is an essential factor for the well-being of the earth's ecosystems. Loss of biodiversity is regarded as a long-term problem negatively affecting the natural functioning of the ecosystems, which in many cases (e.g. agriculture, tourism, etc.) poses a valuable or even essential commodity for human society.

In the preceding sections, several LCIA indicators that characterise potential damage to biodiversity and ecosystems have already been compared for different energy generation technologies. These include indicators for acidification, eutrophication and ecotoxicity, among others. These will not be further discussed here.

In this section, three further indicators used in the literature to characterise potential impacts on the protection and restoration of biodiversity and ecosystems are compared for different electricity generation technologies.

The first is terrestrial ecotoxicity. Terrestrial ecotoxicity potential (TETP), in general terms, refers to the impact on non-human living organisms of terrestrial ecosystems resulting from lifecycle emissions of toxic substances to air, water and soil. Similar to aquatic ecotoxicity potential, the indicator used to categorise terrestrial ecotoxicity potential is measured in terms of mass of 1,4-dichlorobenzene equivalent per unit of electricity generated. This indicator has been calculated and compared between different electricity generation technologies by Stamford & Azapagic [3.2-9], using the CML 2001 methodology, and by Treyer & Bauer [3.2-23], using the ReCiPe methodology.

Ecotoxicity potentials were also compared in the NEEDS project [3.2-10]. Here a single indicator was used which quantifies the loss of species (flora & fauna) due to the release of ecologically toxic emissions to air, water, and soil. The indicator is given in terms of Potentially Disappeared Fraction of species on 1 m^2 of earth surface during one year ($PDFm^2a$) per unit of electricity produced. It follows the methodology of the Life Cycle Impact Assessment method Eco-indicator 99 [3.2-24] and covers complete energy chains.

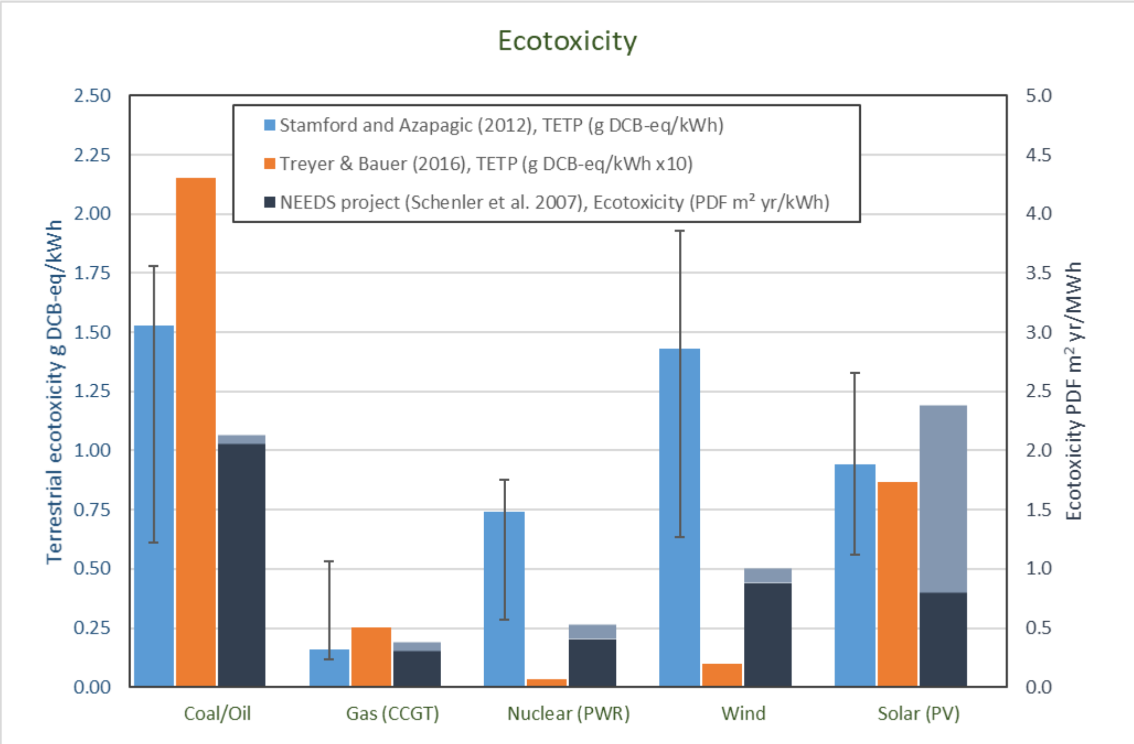
The above two indicators are shown together in Figure 3.2-22. Maximum and minimum values from the NEEDS data are shown in the figure (dark and light bars). These maximum and minimum values correspond only to national differences in the implementation of the different technologies, except for solar PV. The minimum values for solar PV correspond to the use of CdTe panels, whereas the maximum values correspond to the use of Si panels. Stamford & Azapagic consider PV panels according to the average world mix[49]. The data for Treyer & Bauer [3.2-23] relates to multi-crystalline Si panels.

The results of Stamford & Azapagic [3.2-9] and Schenler et al [3.2-26] both indicate that natural gas has the lowest impact, followed by nuclear in second place. Solar PV, wind and coal are all more damaging, with the

[49] Comprising 98.4% Si panels of various types (mono-crystalline, multi-crystalline, amorphous and ribbon panels and laminates – refer to the original reference for details) and 1.6% CdTe and CIGS (cadmium-indium-gallium-selenide)

ranking depending on the indicator chosen and the type of solar panel technology assumed. The data of Treyer & Bauer [3.2-23], calculated using the ReCiPe methodology, have significantly lower values all round (the data represented by the bars in the chart have been multiplied by a factor of 10). The ranking is also different, with nuclear energy having the lowest impact, followed by wind, gas, solar and oil.

Figure 3.2-22. Ecotoxicity

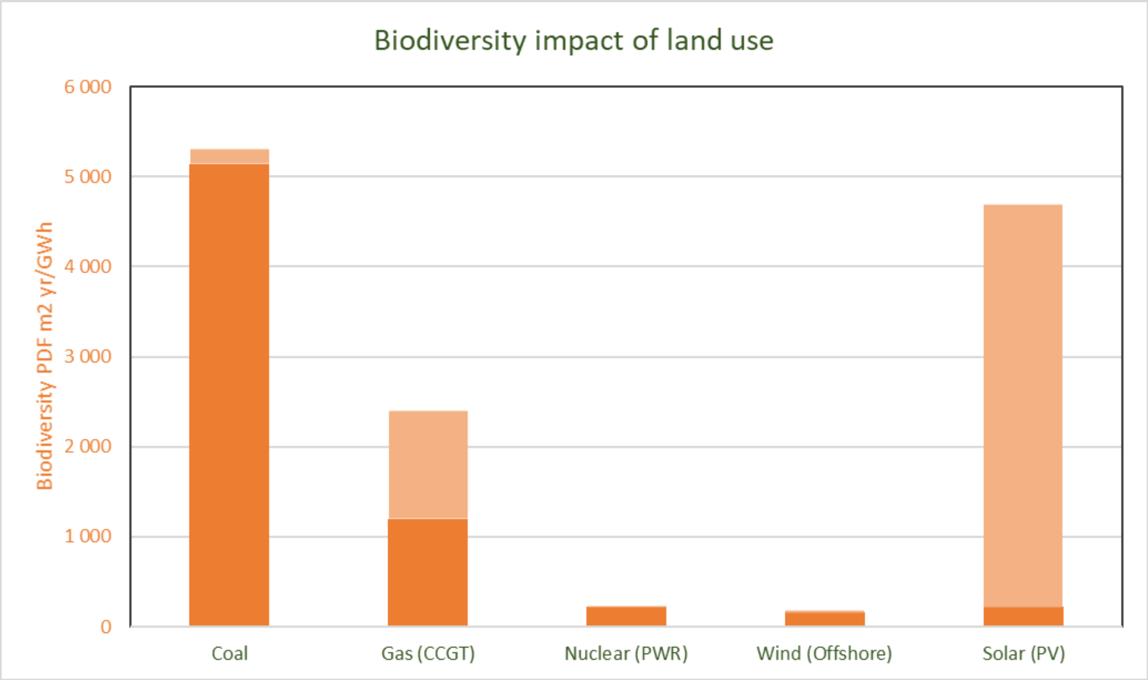


Data from [3.2-9], [3.2-23] and [3.2-26]

The third impact indicator to be discussed in this section is biodiversity impact of land use, which has been calculated in the NEEDS project [3.2-26]. Human land use, i.e. changing the natural state of land by human activities, is one of the potential reasons for loss of biodiversity, i.e. loss of species. The indicator quantifies the loss of species (flora & fauna) due to land use. It is given in terms of Potentially Disappeared Fraction of species on 1 m^2 of earth surface during one year (PDFm^2a) per unit of electricity produced. It follows the methodology of the Life Cycle Impact Assessment method Eco-indicator 99 [3.2-24] and covers complete energy chains. Data for the same set of electricity generation technologies are shown in Figure 3.2-23 from [3.2-26]. The explanation of the minimum and maximum values is the same as for the NEEDS data in Figure 3.2-22.

The biodiversity impact of land use is calculated to be low for nuclear, wind and solar PV based on CdTe panels. The impact from gas is an order of magnitude larger than nuclear and wind, with solar PV based on Si panels higher again, and coal having the most damaging impact.

Figure 3.2-23. Biodiversity impact of land use



Data from [3.2-26]

According to the information presented in the foregoing, nuclear energy compares favourably with competing technologies with regard to its impact on biodiversity taking into account the whole lifecycle. There is no evidence that nuclear energy does more harm to the protection and restoration of biodiversity and ecosystems than other energy technologies included in the Taxonomy.

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3.3 Detailed assessment of the impacts of nuclear energy in its various lifecycle phases

Open and closed nuclear fuel cycles

In order to understand the various technological processes in the lifecycle of nuclear energy, one must first be familiarized with the commonly used nuclear fuel cycle types of today. Basically two fuel cycle types are used today, the “open” and the “partially closed”, which are also often referred to as “once through cycle” (OTC) and “twice through cycle” (TTC), respectively.

One has to distinguish between the “partially” closed fuel cycle as it applies to currently dominant (thermal neutron spectrum) reactor technology, which is limited to twice-through, and the “fully” closed cycle, which allows multi-recycling of spent fuel but requires fast reactors (FRs) to be integrated in the reactor fleet. Multi-recycling in FRs – which have been demonstrated but not yet widely deployed on a commercial basis – would result in a huge reduction in the need for uranium mining. In the current chapter references to the “closed” fuel cycle include the “partially” closed cycle, i.e. to the TTC. Which type is meant, if relevant, is clear from the context or data under discussion.

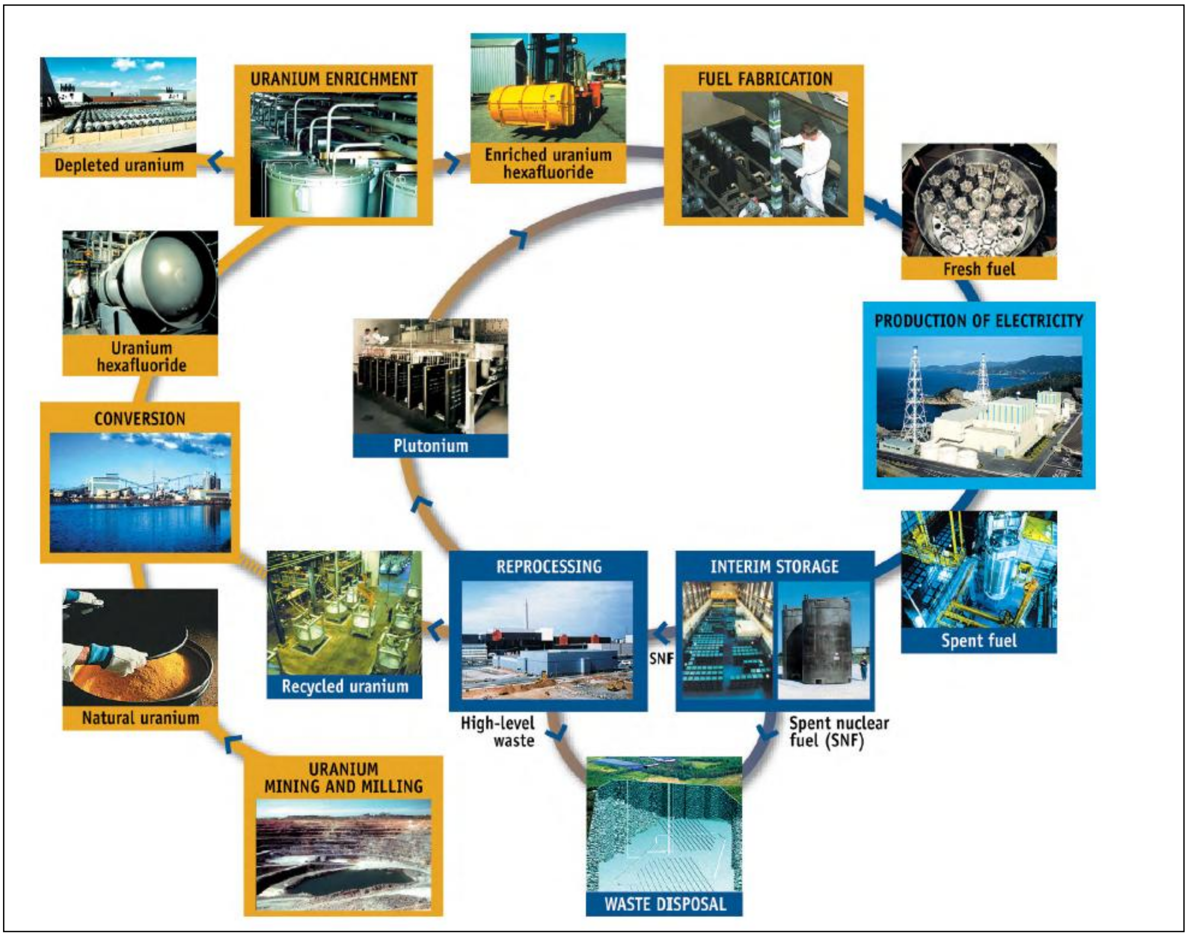
Note: the vast majority of current power reactors utilize thermal (low-energy) neutrons to maintain the controlled chain reaction in their active core, but there exist reactor designs which are based on fission induced by fast (high-energy) neutrons. While thermal reactors utilize ^{235}U isotope (the only naturally occurring uranium isotope to undergo fission induced by thermal neutrons), the fast reactors make use of the ^{238}U uranium isotope, as well. In addition to ^{238}U, the nuclear fuel of the fast reactors contains ^{235}U and ^{239}Pu isotopes, the ^{235}U is usually enriched to 20%[50]. A fast breeder reactor (FBR) is a special fast reactor version, which is able to produce (“breed”) more fuel than consumed. The most important breeding reaction is the ^{238}U absorption reaction, when fissile ^{239}Pu is produced from the ^{238}U isotope. The efficiency of the breeding process is characterized by the breeding ratio parameter: if the breeding ratio exceeds 1 then the reactor produces more fissionable fuel than it consumes, i.e. more ^{239}Pu is produced from ^{238}U than burned in the original fuel containing ^{235}U and ^{239}Pu. Those materials (e.g. ^{238}U and ^{232}Th) which are able to produce fissile material when irradiated by neutrons are called fertile materials. A remarkable feature of the fast reactors is that they can utilize the uranium about 60 times more efficiently than the current thermal reactors.

The main steps in the two different cycles are illustrated in Figure 3.3.1-1, which shows that the main difference between the two cycles is the destiny of the spent nuclear fuel (SF) after it has been utilized in a nuclear power plant for electricity production. After the burnt fuel is removed from the reactor, in both cases the SF is first stored in wet storage facilities (borated water-filled pools) for some years, until the remnant heat of the fuel decreases to a level appropriate for dry storage. The wet storage is followed by another storage period in a so called interim storage facility, where the SF is kept under safe conditions for several years (sometimes for several decades), usually in dry vaults or in special casks (containers). Here the removal of the remnant heat from the fuel is ensured by air cooling.

After this interim storage period the closed and open fuel cycles diverge: in the closed cycle the SF is transported to a reprocessing facility, where the spent fuel is disassembled, the fuel rods are cut and the ceramic fuel pellets are dissolved in acidic solutions. According to the World Nuclear Associaton (WNA, see https://world-nuclear.org/) the used LWR nuclear fuel (having an average burnup level) contains about 96% of uranium (98.5% of which consists of the ^{238}U isotope), about 3% of stable fission products and 0.9% of plutonium. The reprocessing aims to recover the uranium and plutonium still present in the spent fuel and reuse them to manufacture new nuclear fuel. The plutonium directly goes to the fuel manufacturing factory to be used for the fabrication of MOX (mixed oxide) fuel. The reprocessed uranium (abbreviated as RepU) is transported to the uranium conversion factory, where it is mixed with the “fresh” uranium, directly coming from the yellow cake production facility. The highly radioactive waste remaining after the chemical dissolution processes is first solidified, then it is vitrified into borosilicate glass, which is sealed into stainless steel cylinders. These “drums” are then stored waiting for their disposal in an appropriate final disposal facility. Note that the spent MOX fuel is not reprocessed further and hence the meaning of the TTC (twice through cycle) name becomes understandable: in this type of closed cycle the nuclear fuel is placed into the reactor for electricity production twice. Actually the TTC is only a partially closed cycle, because in a real closed cycle the MOX fuel should be reprocessed many times (further details of the “fully” closed fuel cycle are given in Chapter 3.3.5 on reprocessing).

[50] For thermal reactors the average enrichment of uranium is around 4%, with 5% as maximum enrichment.

Figure 3.3.1-1. The scheme of the open and closed nuclear fuel cycles



Source: [3.3.1-1]

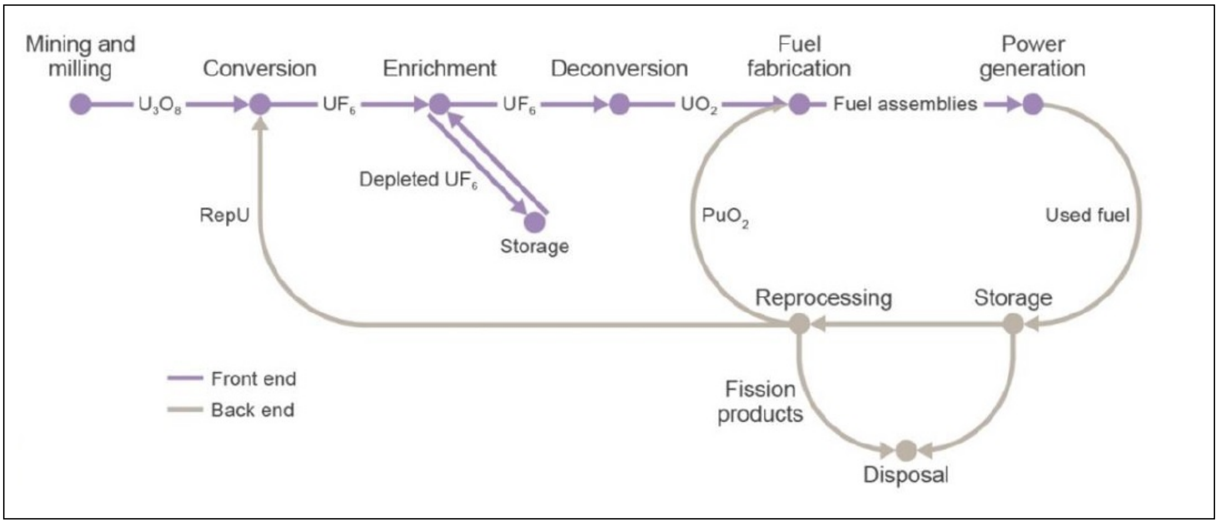
It has to be noted that about 30% of the total amount of SF produced globally in the nuclear power plants has been reprocessed, saving large amount of direct uranium mining capacity.

In the open cycle there is no reprocessing step and the SF is to be disposed at a final disposal facility, because the SF in the open cycle is not processed further. The very first facility of this kind is planned to be operable in Finland by 2025. Currently all the spent fuel utilized in open cycles so far around the world are waiting for the construction of the appropriate national disposal facilities. Now the OTC (once through cycle) name is also clear: in this cycle the nuclear fuel is placed into the reactor for electricity production only once.

The first part of the nuclear fuel cycle (i.e. mining and milling, conversion, enrichment and fuel fabrication) is called together the "front end" of the cycle, while the remaining steps (i.e. all steps after the used fuel has been removed from the reactor) form the "back end" (see Figure 3.3.1-2).

Regardless of whether open or closed, the nuclear fuel cycle starts with the uranium mining and milling phase, which is described in the next subchapter.

Figure 3.3.1-2. Scheme of the "front end" and "back end" parts of the nuclear fuel cycle



Source: [3.3.1-2]

3.3.1 Uranium mining and uranium ore processing

3.3.1.1 Description of the underlying technologies and processes

In the following we restrict ourselves to the description of the basic technological details only, in order to make the potential environmental and human health impacts of uranium mining and milling understandable for the reader. A very detailed and professional description of the topic is provided e.g. at the website of the World Nuclear Association, see https://world-nuclear.org/.

According to the WNA, in 2018 the global uranium production was about **53 500 metric tons** of uranium, with Kazakhstan alone supplying about 40% of the total. About 90% of the global production comes from seven countries: Kazakhstan (40%); Canada (13%); Australia (12%); Namibia (10%); Niger (5%); Russia (5%) and Uzbekistan (4%).

The above figures show that Europe is not a significant uranium supplier any more, although for example East Germany had large uranium mining operations before 1990 and the whole history of uranium mining started in the Czech Republic at the Jáchymov underground mine at the end of the XIXth century. Note that Madame Curie used a large amount of pitchblende ore (uraninite) from Jáchymov when she isolated radium and discovered polonium.

In 2018, this 53 500 metric tons of uranium was sufficient to cover the decisive portion of fuel supply needs of the 451 NPP units operated at that time around the world and providing approximately 400 GW electric power. The remainder was covered by reprocessed fissile materials.

In comparison, a coal-fired power plant of 1 GW electric power consumes 9000 metric tons of coal per day! (See e.g. https://energyeducation.ca/encyclopedia/Coal_fired_power_plant).

Uranium is naturally occurring in the Earth's crust almost everywhere, its average concentration is **2.8 ppm** (parts per million, 10^{-6}). Mining of uranium needs uranium deposits with much higher concentration: the ore grade in the largest currently cultivated mines ranges from **0.12%** (1200 ppm) at Ranger mine (Australia) to **14.7%** at Cigar Lake mine (Canada). In Canada deposits with even 20% grade were discovered.

Three naturally occurring isotopes are present in natural uranium: ^{238}U (99.275%), ^{235}U (0.72%) and ^{234}U (0.0054% = 54 ppm).

Currently the mining of uranium ore is carried out by using three different methods.

Open-pit mines are cultivated at those places where the uranium ore is abundant in layers close to the surface of the Earth. Large open pit mines are in operation e.g. in Namibia and Niger.

Figure 3.3.1-3. Uranium mining in Jáchymov around 1935

Source: [3.3.1-6]

Figure 3.3.1-4. Pitchblende ore

Source: [3.3.1-7]

Deep-pit (underground, UG) mines are constructed at those places where the uranium-rich layers are located deep (sometimes several hundred meters) below the surface. Two of the largest mines (Cigar Lake in Canada and Olympic Dam in Australia) are underground mines.

Chemical leaching is a mining method which is gaining ground gradually: the basic principle of the “in-situ leaching” (ISL) method is that a liquid substance containing acidic (sulphuric acid) or alkaline (sodium carbonate) chemicals is pumped below the surface, into the sand or sandstone layer containing the uranium ore. The scheme of the ISL is shown in Figure 3.3.1-5. The pumped-in liquid slowly dissolves the minerals containing uranium, then – after some time – it is driven back to the surface, where it is further processed to extract the dissolved uranium. The waste liquid substance remaining after the processing is pumped back under the surface. This method does not cause “landscape wounds” to remediate as open-pit mines do and does not produce large amount of waste rock, which is characteristic to underground mines. However, it produces large quantities of waste water containing aggressive chemicals, and these wastes must be managed properly in order to avoid damage to the environment. Also the original quality of the groundwater must be restored after the leaching operations have been terminated. Due to the intense pumping of liquids characterizing this method, it is more energy intensive than the two other mining technologies. The energy for pumping is often available only from large diesel generators, as uranium mining sites are often located at distant off-grid locations with no electricity transmission lines (see Refs. [3.3.1-5, -13, -16 and -17] for more details). However, despite its high energy demand (i.e combustion of 7 litres diesel oil to extract 1 kg U_{nat}), a definite advantage of the ISL is the lack of tailings and waste rock piles, because there is no excavation and ore milling in this technology.

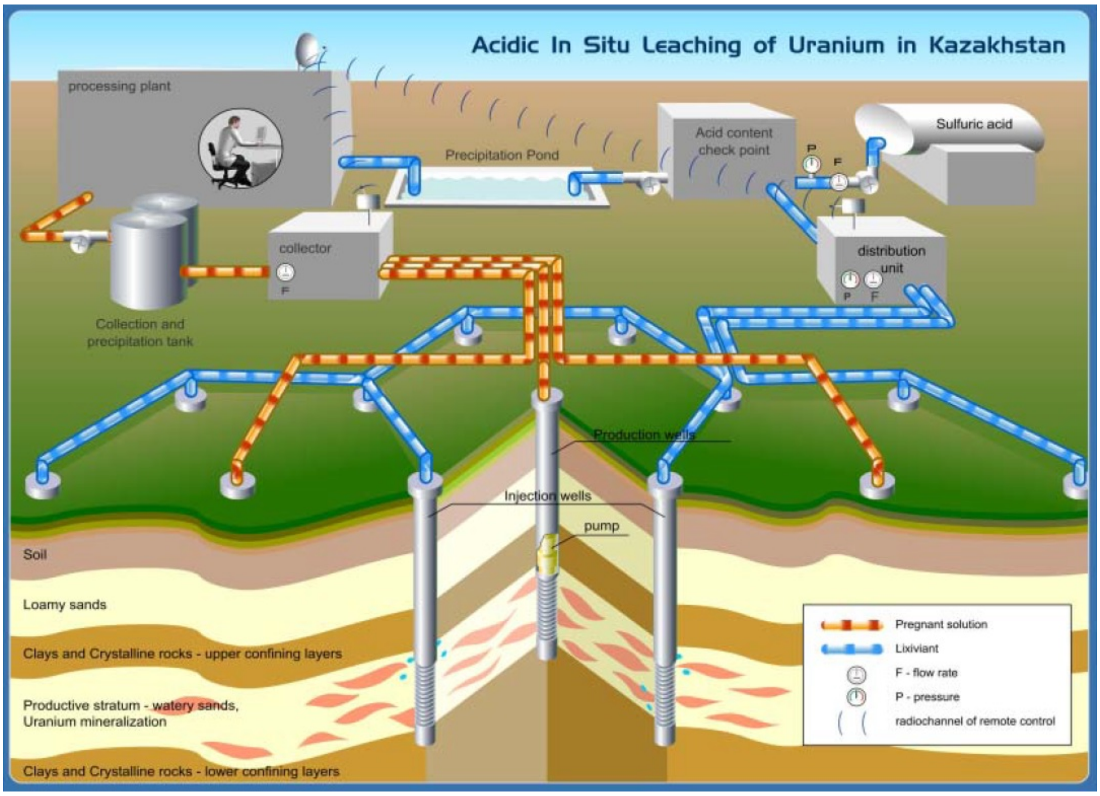
The uranium mines in Kazakhstan almost exclusively use the ISL technology, but it is prevalent in the USA, as well, where it is called “in situ recovery” (ISR). As of today, globally almost **50%** of uranium is mined using this method and due to the advantages discussed above it is applied more and more, wherever it is feasible.

The uranium ore obtained from the mine then undergoes further processing phases, which are characteristic for conventional mines excavating metalliferous minerals. First the uranium ore is milled into small pieces in an ore mill, then it is selected, cleaned and dried. This process yields a fine powder substance which undergoes the following further chemical processing steps:

- by using a suitable chemical solvent material, the uranium present in the ore-powder is solved into an alkaline, acidic or peroxide solution (most often sulphuric acid is used),
- the uranium solution is then separated from the other components,
- in the last step the uranium is precipitated and dried in an oxide form (**U_3O_8**).

The dried uranium precipitate has bright yellowish colour, which is why the end-product is called “yellowcake”. The uranium heavy metal content of the yellowcake is usually more than **75%**.

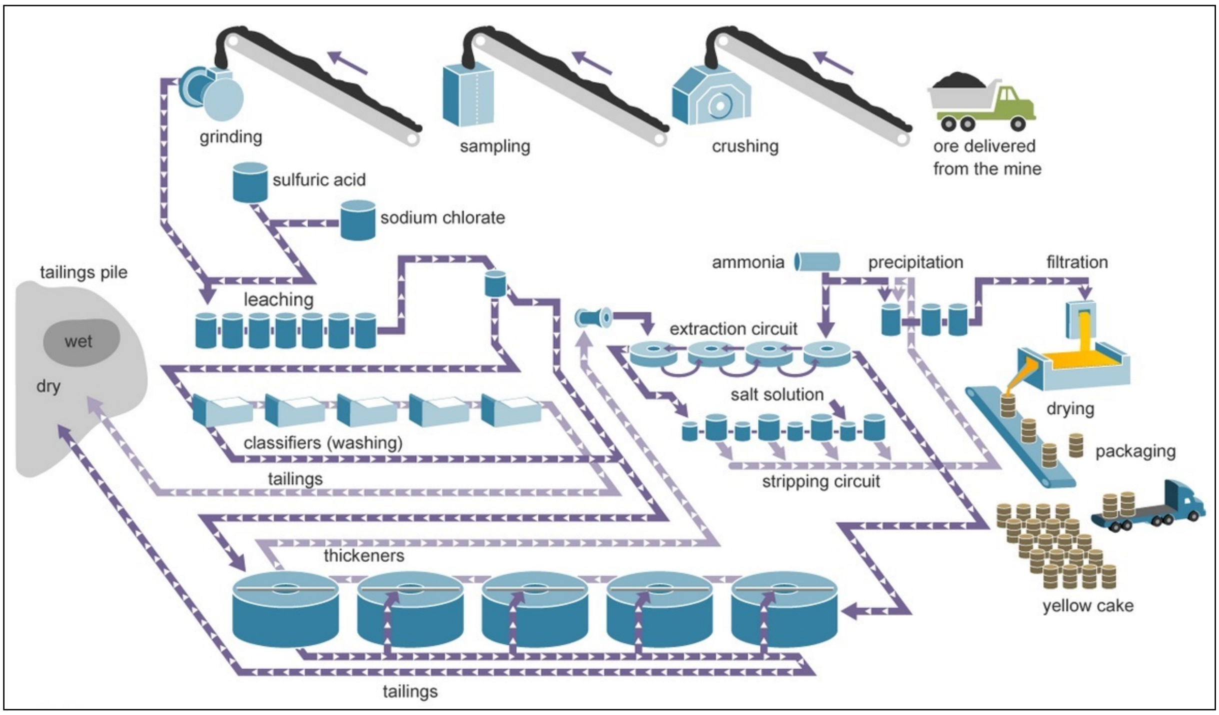
Figure 3.3.1-5. Scheme of the application of acidic ISL technology in Kazakhstan



Source: [3.3.1-17]

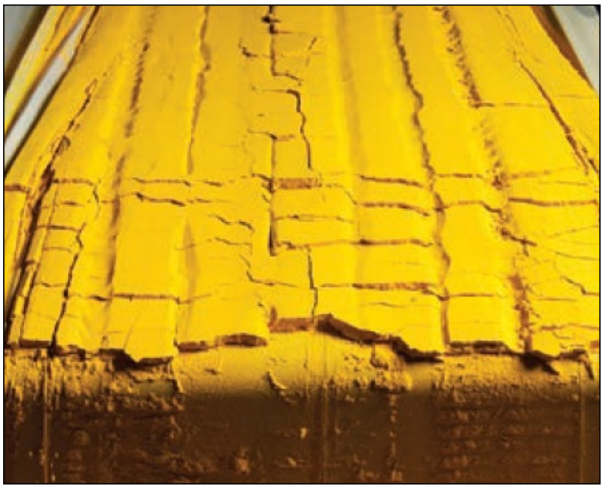
Figure 3.3.1-6 shows the scheme of a uranium mill, receiving raw ore from the mine (usually located close to the mill) and packing yellowcake into transport containers at the end of the process. Figure 3.3.1-7 illustrates how the yellowcake looks at the end of the production line.

Figure 3.3.1-6 Scheme of a uranium mill



Source: U.S. Energy Information Administration, https://www.eia.gov/

Figure 3.3.1-7. Yellowcake on the production line



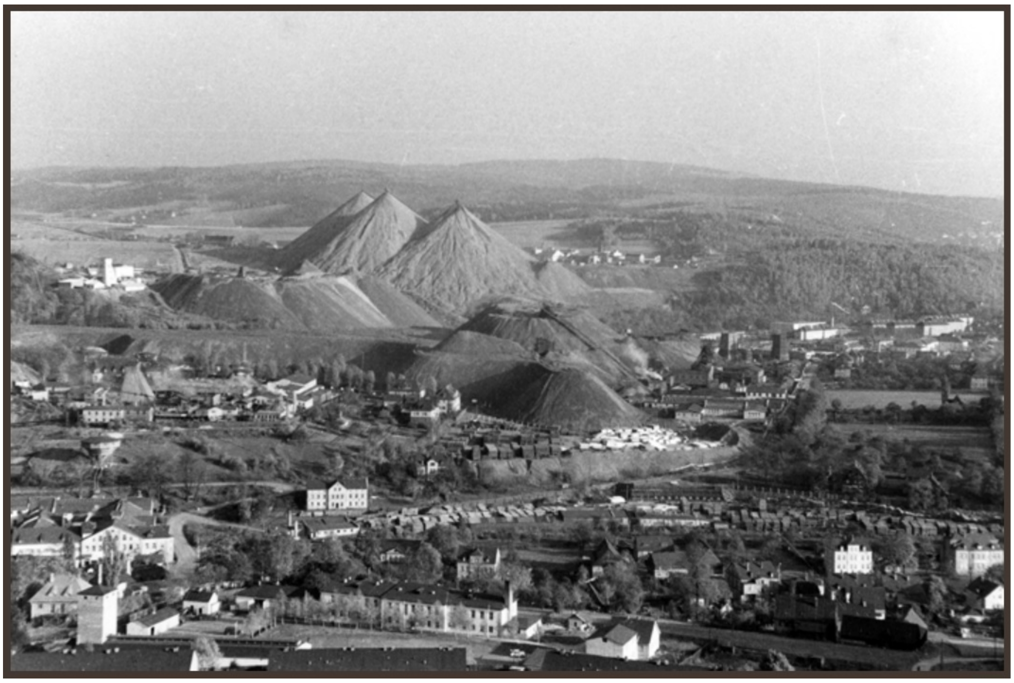
Source: https://www.orano.group/en

3.3.1.2 Identification of key potential impacts on the environment and human health

3.3.1.2.1 Historical background

Uranium became a strategic material during World War II due to its role in manufacturing nuclear weapons. The race for the exploration and exploitation of uranium resources continued during the Cold War period, as well, when the superpowers stockpiled huge numbers of nuclear warheads containing mainly plutonium produced in military reactors running on uranium fuel. In this period national security considerations were of prime importance and environmental aspects of uranium mining were secondary or tertiary. Figure 3.3.1-8 illustrates the way some uranium mines were operated in this period by showing the tailings of the Schlema-Alberoda underground mine located in the former East Germany. These huge uranium ore tailings – sometimes formed very close to the dwellings – have now been removed and the affected area has been completely remediated. Note that the USA and France have also now completely remediated the uranium mill tailings at their closed down mining sites.

Figure 3.3.1-8. Tailings at the Schlema-Alberoda site (Germany, former GDR) in 1991



Source: [3.3.1-3]

A similar situation is shown in Figure 3.3.1-9 depicting the tailings of an abandoned uranium mine in Tajikistan. The remediation of the Taboshar site is currently in progress with international assistance coordinated by the EBRD and the European Commission.

Figure 3.3.1-9. Tailings at the Taboshar site (abandoned uranium mine in Tajikistan)

Source: European Commission (DG DEVCO - now DG INTPA)

This adverse situation changed radically after 1990, when the nuclear arms race abated, although uranium is still considered as strategic material for national defence. The new circumstances allowed uranium mines to gradually introduce environmentally friendly technologies and establish responsible operational practices which gave priority to the protection of human health and minimization of environmental impacts. Parallel to these technological improvements the relevant legal framework and associated regulations developed substantially, including regulatory competence and oversight capabilities. As a consequence, the uranium mines of today must adhere to strict radiation and industrial safety rules and must satisfy relevant environmental standards and regulations (see Chapter 3.3.1.4 “Legal background and regulations” for a detailed list of relevant standards and regulations).

3.3.1.2.2 Non-radioactive impacts

If non-radioactive impacts are considered, uranium mines and uranium ore mills are very similar to the conventional mining operations excavating metalliferous minerals. The environmental impacts of these mines and mills can be summarized as follows:

1. piling up large amounts of rock waste (in case of underground operations);
2. creation of large tailings;
3. inflicting landscape damage (landscape “scars” in case of open-pit operations);
4. presence of various heavy metals in the washwater;
5. hazardous seepages due to the utilization of large quantities of aggressive liquid chemicals;
6. excessive dusting;
7. damage and pollution caused by transport operations;
8. significant water consumption and waste water production.

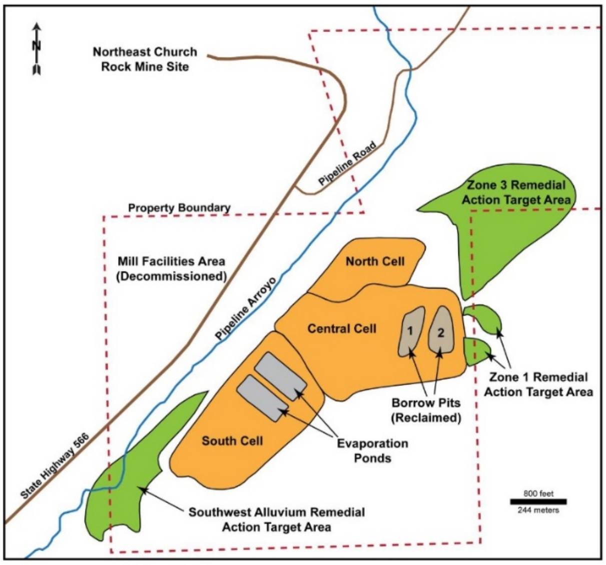
Abandoned or improperly constructed uranium mill tailings can lead to significant contamination of the soil, surface waters and groundwaters, if a proper containment of the tailings is not established or maintained. A failed or improper containment can cause dispersal of radioactive dust, erosion of the tailing ramps or large discharge of contaminated water or sediments (from sedimentation ponds). Characteristic reasons for sudden containment failures (e.g. for dam breaches) are soil/ground movements, leakages, pond overfilling and earthquakes (see Ref. [3.3.1-8]). However, application of long-lasting and leak-proof bottom liners and well-sealing tailing covers can efficiently prevent the occurrence of hazardous environmental impacts (see Chapter 3.3.1.5 “Identification of applicable means to avoid or mitigate the impacts” for the details).

Figure 3.3.1-10. Broken dam of the Church Rock and its remediation

The broken Church Rock tailings dam in July 1979

Source: Wikipedia/Commons

The map of the Church Rock site to remediate



Source: US NRC

The two remediated evaporation ponds in 2020

Source: Google Maps (Imagery ©2021 Maxar Technologies, NMRGIS, USDA Farm Service Agency, Map data ©2021)

Figure 3.3.1-10 illustrates a major tailings dam breach and its remediation. In July 1979 at the Church Rock (New Mexico, USA) uranium mining site the dam of the evaporation pond used for storing uranium mill wastewater was broken and released about 1100 metric tons of solid tailings waste and more than 400 000 m^3 of effluent into the nearby Puerco River. The released acidic effluent was contaminated by radioactive isotopes (e.g. ^{238}U, ^{230}Th, ^{226}Ra) and toxic metals (e.g. lead and arsenic), as well as sulphates. This event is considered as the largest single release of liquid radioactive waste in the USA (see [3.3.1-18] for details). In 1982 the uranium mine was closed and the site was gradually cleaned and remediated under the supervision of the US EPA (Environmental Protection Agency).

3.3.1.2.3 Radioactive impacts

If radioactive impacts are considered, then uranium mining and milling operations first of all must pay special attention to eliminating the <u>human health hazards</u> represented by the omnipresent radon gas and the radioactivity in the tailings.

Workers working in underground uranium mines are the most exposed group to the radon hazard: inhalation of radon and radioactive dust was responsible for developing lung cancer among people working in early uranium mines (it was even called "Jáchymov miners disease"; see [3.3.1-6] for details).

Depending on the uranium ore grade, the average specific activity of radon in unvented tunnels of underground uranium mines is between 10 kBq/m^3 and 1 MBq/m^3. These tunnels are intensively vented when workers are present[51].

Radon gas is present at low concentration in the natural uranium ore and it is released to the atmosphere during the uranium mining and milling operations. When the uranium ore is processed, most of the uranium isotopes are removed from the rock or sand by chemicals. Huge waste stockpiles (so called tailings) are formed from the remaining material, which still contains – after some months of decay – about 75% of the radioactivity of the natural uranium ore due to those isotopes in the uranium decay chains, which were not removed. In the tailings the ^{230}Th (thorium) is the dominant long-lived isotope, which decays with a half-life of 77000 years to ^{226}Ra (radium) followed by ^{222}Rn (radon). This radon gas and its progeny[52] emanate from the tailings as the thorium and radium decays.

Reference [3.3.1-8] provides a comprehensive overview of the environmental impacts represented by the uranium mine tailings. Depending on the grade of the uranium ore mined at the given site, the specific activity of the tailings ranges from 1 MBq/t to 100 MBq/t or higher. If these mines pile up for example one million metric tons of new tailings annually then they "surface" between 1 and 100 TBq of activity above the ground. Note that as long as the ore stays under the ground the radiation (including the radon gas) is sealed and shielded. The ISL extraction process does not produce tailings, therefore the following considerations are not applicable to this particular mining technology.

Tailings built up in the surroundings of open pit and underground mines may represent multiple risks to the environment and human health for the following reasons:

1. emission of direct gamma radiation (mainly from the radium);
2. emanation of radon gas;
3. dispersion of radioactive dust, taken by the winds to the surrounding areas;
4. contamination of surface and groundwaters by heavy metals present in the tailings;
5. acidifying groundwater due to the high sulphide content of the tailings' material.

In the early decades of uranium mining, mill tailings were constructed without bottom liners and surface coverage (coverage is applied to prevent dusting and radon emissions). This often caused significant contamination of the air, soil, as well as surface and groundwaters around the tailings.

When the tailings are not covered, part of the radioactivity in the tailings emanates to the air as radon or is carried away with the dust blown by the winds. Another fraction can be washed away with the rainwater and can infiltrate the groundwater or the water bodies nearby (rivers or lakes). Radioactivity in the air can be

51 The Basic Safety Standards Directive (Council Directive 2013/59/Euratom of 5 December 2013) sets the reference level for the annual average radon activity concentration in air at workplaces to not more than 300 Bq/m^3.

52 Radon (^{222}Ra) is a naturally occurring radioactive gas with a half-life of 3.8 days. When radon in air decays, it forms a number of short-lived radioactive decay products, known as radon progeny, which include ^{218}Po, ^{214}Po, ^{214}Pb and ^{214}Bi. The inhalation of radon and its progeny have been recognised as a cause of lung cancer by international radiation and health protection organizations.

inhaled as gas or dust (particulate matter), while radioactivity escaping into the water bodies can be ingested through drinking or eating (e.g. fish or crops irrigated with the water). Both cases represent significant potential harm to humans and the biota and their avoidance needs appropriate mitigation measures.

Waste rocks may also contain low concentrations of uranium and radionuclides from the uranium decay chain causing radiation levels above the normal background. In these cases, an adequate covering preventing weathering and erosion may be required. Any seepage from the waste rock piles must be analysed and, where appropriate, collected and treated.

Contaminated mine water can be produced by extraction and temporarily lowering of the water table during mining operations where ore deposits are located below the ground water table, dewatering of underground and open pit mines, runoff of surface waters or seasonal rainfalls in the mining area. The radioactivity of this water generally originates from the dissolution of soluble uranium, thorium, radium and lead ions.

As far as possible, the contaminated water is recycled in the mining and the milling operations. Water that is not recycled must be either contained at the site or released under controlled conditions as is, or after treatment to the environment in accordance with the established standards for maximum concentrations of specific contaminants in discharged water.

The decommissioning of a uranium mill will also generate large amounts of radioactively contaminated scrap, which have to be disposed of in a safe manner.

ISL does not involve excavation and, therefore, there is no waste rock or tailings materials produced. However, the generated sludge and evaporate salts must be safely disposed of. Because of the specific activity of the radioactively contaminated wastes from the ISL operations, the solid waste generated in some jurisdictions is considered LLW that must be disposed of in an approved waste disposal facility.

Figure 3.3.1-11 compares the importance of various operational aspects for the dominant uranium mining techniques in the "production" phase (see Ref. [3.3.1-3] for the complete analysis).

Figure 3.3.1-11. Comparison of the importance of various operational aspects for the three main uranium mining technologies in the "production" lifecycle phase

Aspects	Production		
	ISL	Open-cut	UG
Operational			
Worker H & S (radiation)	**	***	*****
(conventional)	****	****	*****
Public safety (radiation)	**	***	**
Water/quality (surface)	**	****	**
(groundwater)	*****	***	***
Tailings	n/a	****	****
Waste rock	*	****	***

n/a = Not applicable; * Very low importance; ** Low importance;
*** Normal importance; **** High importance; ***** Critical importance.

Source: [3.3.1-3]

The comparison shows that for the ISL the only critically important operational aspect is preserving (restoring) the quality of the groundwater. The ISL uses chemicals (acids or alkalines) to extract uranium (see Figure 3.3.1-5) and the restoration of the neutral pH in the aquifers leached with chemicals is usually carried out by flushing the depleted underground with water until acceptable groundwater concentrations are attained (see Ref. [3.3.1-16]).

In case of an underground mine ensuring radiation and conventional safety for the miners is of prime importance, but tailings also represent important issues to deal with. In case of open-pit mines the conventional safety for the miners, the quality of the surface waters, the tailings and the waste rock are of main importance. It can also be deduced that the environmentally least harmful technique is the ISL, while open-pit mining represents the largest environmental load. As mentioned earlier, currently around 50% of the uranium is mined using the ISL technique globally and its application is continuously gaining ground.

3.3.1.3 Summary of lifecycle analysis results for the uranium mining and milling phase

In order to illustrate the characteristic environmental impacts of the uranium mining and milling phase and their important contribution to the impact of the complete nuclear energy lifecycle, this sub-chapter presents the breakdown of several radioactive and non-radioactive impact indicators between the different lifecycle phases of nuclear-based electricity generation.

The data presented result from the study reported in [3.3.1-11]. This study analyses the entire French nuclear fleet presently consisting of 56 operating units. The French nuclear reactors are operated in a “limited” closed fuel cycle (i.e. in a twice-through cycle, when the spent nuclear fuel is being reprocessed once) and in 2019 they produced 52% of the total nuclear electricity generated in the EU-28 countries.

Illustrative data taken from the study are presented in **Annex 2** (*Summary of LCA results for all lifecycle phases of nuclear energy*), along with data from other selected studies. Note that here only graphical illustrations and textual explanations are given, all numerical values are tabulated in Annex 2. The analysis of the data in Annex 2 reveals the following characteristics of the mining and milling lifecycle phase:

— Closed cycle (TTC) / Non-radioactive impact indicators

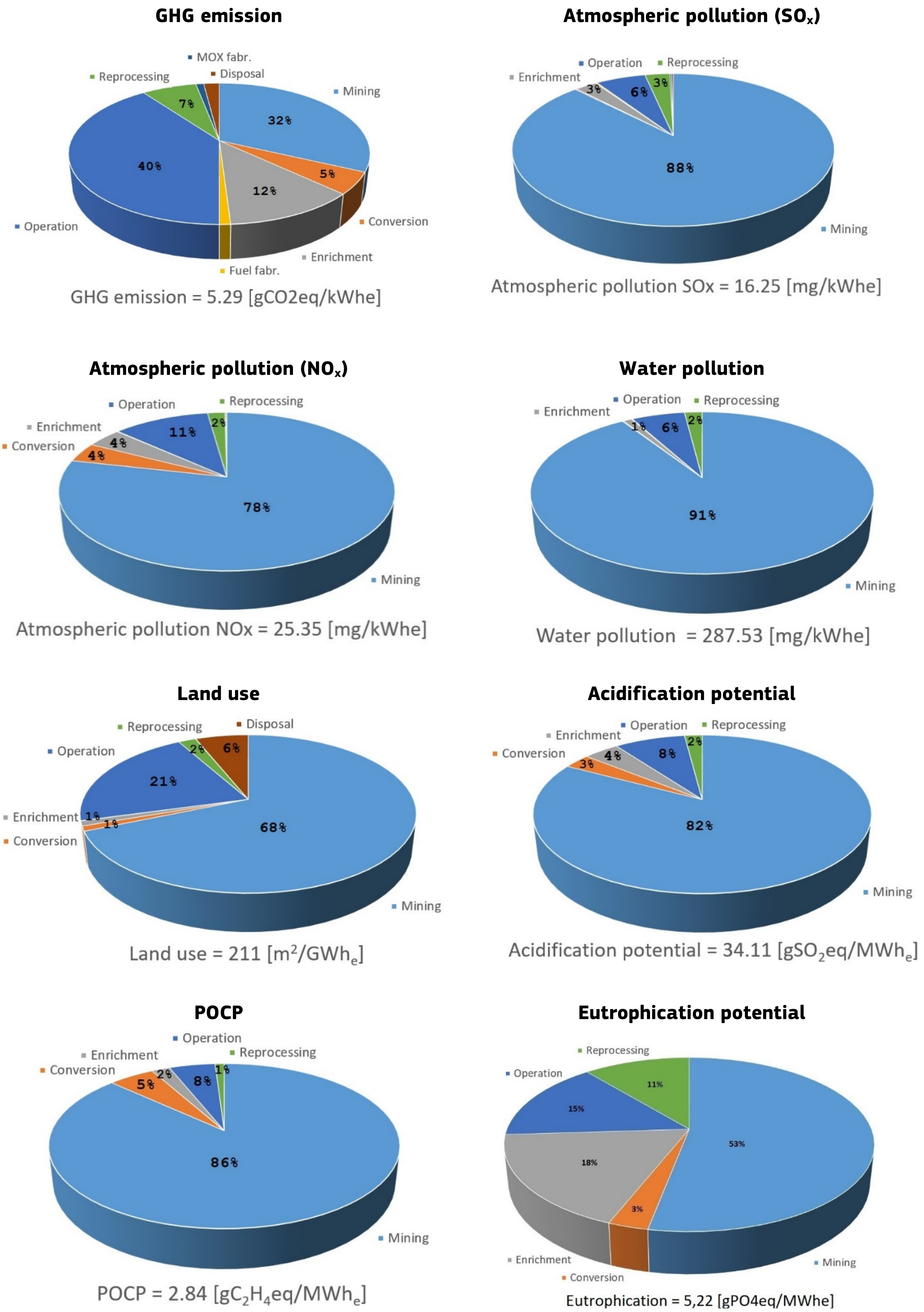
  If the whole nuclear lifecycle is considered, then uranium mining has large contribution (≈**32%**) to the total GHG emission and **dominates** the following impacts: SO_x emission (≈**88%**), NO_x emission (≈**78%**), water pollution (≈**91%**) and land use (≈**68%**). Mining is almost exclusively (≈**99%**) responsible for the potential eco-toxicity and human toxicity impacts and also **dominates** the acidification (≈**82%**), ozone creation (≈**86%**) and eutrophication (≈**53%**) potentials. Mining does not have significant share in the water consumption, water withdrawal and production of technological waste impacts.

— Closed cycle (TTC) / Radioactive impact indicators

  Due to the emission of radon, uranium mining is responsible for about **55%** of the total **gaseous** radioactive emissions during the total nuclear lifecycle (reprocessing provides the rest). No significant liquid radioactive emissions can be attributed to mining.

  If solid radioactive wastes are considered, mining produces only VLLW (stored in form of tailings and residual waste), although in rather large quantities (around **600 000 m^3/year**, assuming that 60% of the uranium required to deliver the average French 400 TWh_e/year is mined in underground and open pit mines). Note that in the ISL mining process tailings are not produced.

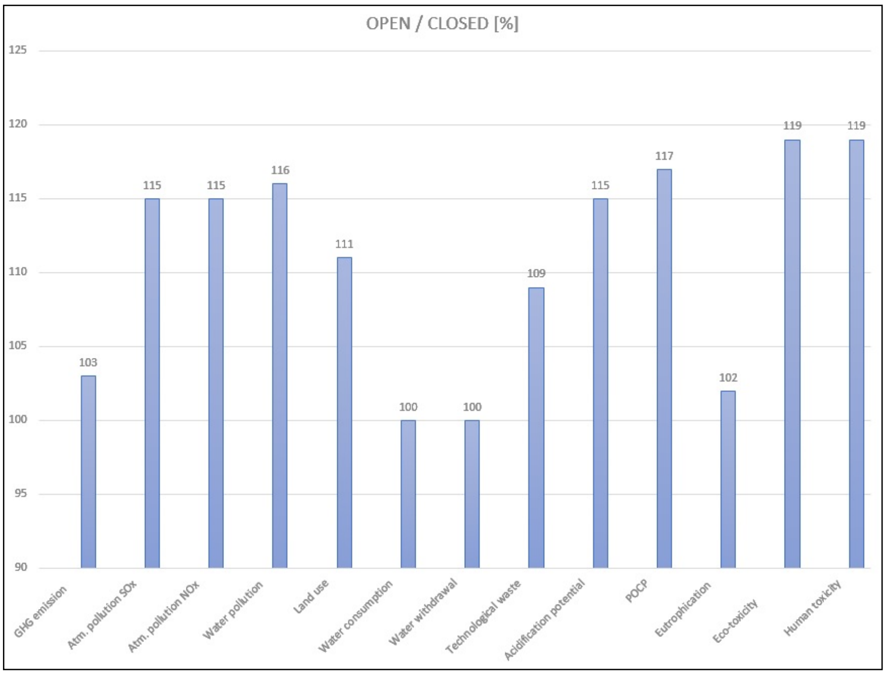
Figure 3.3.1-12. Important impact contributions from the mining & milling phase (closed cycle)



Data from [3.3.1-11]

— Open cycle / Non-radioactive impact indicators

Figure 3.3.1-13. Comparison of non-radioactive impact indicators corresponding to the total LC of nuclear energy for the open and closed cycles



Data from [3.3.1-11]

Figure 3.3.1-13 shows the ratio of impact indicators for the open and closed cycles. It can be seen that the open cycle always has somewhat larger or equal impact, but in most cases the figures do not differ significantly (the largest deviation is about +20%). It is obvious that the use of a TTC or an OTC fuel cycle does not fundamentally change the environmental impacts of uranium mining and milling, it merely results in a difference in the amount of mining required for each kWh of electricity produced (TTC needs less U-mining because it utilizes also fissile material obtained by reprocessing).

— Open cycle / Radioactive impact indicators

There are significant differences in the radioactive impact indicators, e.g. the closed cycle produces about **50%** more gaseous radioactive emissions during the total LC, mainly due to the extra releases during the reprocessing phase (see Figure 3.3.1-14). The closed cycle produces about **ten times** more liquid radioactive releases, again due to the reprocessing phase. Also, the amount of ILW-LL is more than **three times** higher here, due to the reprocessing. On the other hand, the open cycle produces somewhat more VLLW due to the higher amount of uranium mined and generates about **three times** more HLW, because here the spent fuel is not reprocessed.

Note that in the open cycle, uranium mining is responsible for almost **100%** of the total gaseous radioactive emissions, because here there is no fuel reprocessing phase where additional significant radioactive gas releases occur (gaseous emissions during operating phase are negligible if compared to the radon releases).

Figure 3.3.1-14. Comparison of radioactive impact indicators corresponding to the total LC of nuclear energy for the closed and open cycles

Data from [3.3.1-11]

3.3.1.4 Legal background and regulations

As described in detail Annex 1, the nuclear and radiation safety and security aspects of various lifecycle phases of nuclear energy are regulated in the EU by the Directives listed below:

- Nuclear Safety Directive (NSD) – Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014;
- Basic Safety Standards (BSS) – Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation;

  Note that the BSS are in-line with the current ICRP recommendations, which are of global validity (see the document "The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4), 2007", [3.4-12]);
- Radioactive Waste Directive – Council Directive 2011/70/Euratom establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste;
- Transport Directive – Council Directive 2006/117/Euratom of 20 November 2006 on the supervision and control of shipments of radioactive waste and spent fuel;
- Water Directive – Council Directive 2013/51/Euratom laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption.

Mining activities always represent an intrusion to the natural environment, therefore fulfilment of the requirements laid down in the relevant EU Directives controlling and limiting environmental impacts of such intrusive operations is of prime importance. The most important EU Directives related to the protection of the environment are:

- Environmental Impact Assessment (EIA) Directive – Directive 2014/52/EU of 16 April 2014 amending Directive 2011/92/EU on the assessment of the effects of certain public and private projects on the environment;
- Strategic EIA Directive – Directive 2001/42/EC of the European Parliament and of the Council on the assessment of the effects of certain plans and programmes on the environment;
- Air Quality Directive – Directive 2008/50/EC of the European Parliament and of the Council on ambient air quality and cleaner air for Europe;

In addition to the above Directives of general scope, uranium mining and milling activities must also conform to the following specific EU Directives:

- Mining Waste Directive – Directive 2006/21/EC of the European Parliament and of the Council of 15 March 2006 on the management of waste from extractive industries and amending Directive 2004/35/EC;
- Environmental Liability Directive – Directive 2004/35/CE of the European Parliament and of the Council of 21 April 2004 on environmental liability with regard to the prevention and remedying of environmental damage.

Note that EU Directives are implemented into the national regulations in each EU Member State. In addition, the corresponding national laws often refer to relevant IAEA safety standards, as well as associated International Conventions.

Concerning possible transboundary effects of the mining activities and public involvement in the site selection, facility design, construction and exploitation, as well as mine closure and site remediation operations the Espoo and Aarhus conventions are relevant:

- Espoo Convention – Espoo Convention on Environmental Impact Assessment in a Transboundary Context (February 26, 1991);
- Aarhus Convention – Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters, Aarhus, Denmark, 25 June 1998

In each Member State the national legal system – incorporating the above listed EU Directives – contains specific laws on mining and extraction activities. Some of these “mining laws” were first introduced back in the 18th or 19th century and provide very detailed regulations on the exploration of minerals, mining rights and licences, compensation of mining damages, etc.

Note that the actual <u>licensing</u> of mining facilities must always be carried out according to the local (national) regulations in effect in the specific country involved, therefore the regulations laid down in these national laws are also to be strictly followed by the mining companies.

Companies operating in the various areas of civil nuclear energy (e.g. design, construction and operation of nuclear facilities, manufacturing of nuclear materials or components, extraction of raw materials, etc.) are obligated by law to obtain a certification according to internationally recognized quality, environmental, as well as health and safety management standards, such as

- ISO 9001:2015 – Quality Management System;
- ISO 14001:2015 – Environmental Management System;
- ISO 45001:2018 – Health and Safety Management at Work (replacing ISO 18001).

As a rule, the appropriateness of the internal governance of a civil company operating in a specific area of nuclear energy is proven by demonstrating that the company uses internationally recognized management systems to manage nuclear and industrial safety, radiation protection, technological & radioactive waste handling and environmental protection tasks during all phases of the activity concerned. The audit is carried out by an accredited body and repeated periodically.

As mentioned before, “green mining” or “sustainable mining” gradually gains ground also in the uranium mining industry. The principles and practices of environmental friendly mining are being promoted by the International Council on Mining and Metals (ICMM). Mining companies that decided to operate as a “sustainable mine” must adhere to the ICMM principles of sustainable development. These ICMM principles were integrated into the following policy document of the Word Nuclear Association (WNA):

Sustaining Global Best Practices in Uranium Mining and Processing: Principles for Managing Radiation, Health and Safety, and Waste and the Environment

(https://www.world-nuclear.org/our-association/publications/position-statements/best-practice-in-uranium-mining-(1).aspx).

A recent – and important – development in the regulation of mine tailings management activities is the publication of a new industrial standard by the ICMM in August 2020:

Global Industry Standard on Tailings Management (GlobalTailingsReview.org).

According to the Preamble, the new standard "*strives to achieve the ultimate goal of zero harm to people and the environment with zero tolerance for human fatality. It requires Operators to take responsibility and prioritise the safety of tailings facilities… It also requires the disclosure of relevant information to support public accountability.*"

It is expected that the introduction and consistent application of this new standard will contribute to the improvement of the global tailings situation significantly and will support the mitigation of mine tailings-related impacts on the environment and human health. The proposed standard covers the management of tailings produced by all types of mining activities, but its provisions are well applicable for uranium mines, as well.

Another important provider of related guidelines is the International Finance Corporation (IFC), which is a member of the World Bank Group and focuses on financing private investments in less-developed countries. IFC developed a series of guidelines on the proper management of EHS (environment, health and safety) in the projects financed by IFC. The most relevant IFC guides in connection with the Taxonomy are as follows:

- IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007
- IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks and Impacts, IFC, January 2012
- IFC Performance Standard 6 – Biodiversity Conservation and Sustainable Management of Living Natural Resources, IFC, January 2012
- IFC EHS Guidelines – Mining, IFC, December 2007
- IFC EHS Guidelines for Thermal Power Plants, IFC, December 2008

The industrial sector specific IFC EHS guidelines are available at

https://www.ifc.org/wps/wcm/connect/topics_ext_content/ifc_external_corporate_site/sustainability-at-ifc/policies-standards/ehs-guidelines.

The IFC EHS guide for mining is an excellent collection of "performance levels & measures that are generally considered to be achievable in new facilities by existing technology at reasonable costs". Although this guide is not written specifically for uranium mines, it can be well used as a technical reference document to identify Good International Industry Practice (GIIP) also in this sector. It covers all important aspects of sustainable mining, including water use and water quality, protection against acid generation, management of waste rock dumps & tailings, hazardous waste treatment, land use and biodiversity, air quality, noise & vibration and energy use.

Note that the Taxonomy Report Technical Annex [3.3.1-19] frequently refers (in the TSC tables) to IFC performance standards as documents containing adequate criteria, especially IFC Standard 1 (*Assessment and Management of Environmental and Social Risks and Impacts*) and IFC Standard 6 (*Biodiversity Conservation and Sustainable Management of Living Natural Resources*) is used at many places. The application of IFC standards may be a valid option in non-EU countries, because they are already accepted and applied in these areas.

3.3.1.5 Identification of applicable means to avoid or mitigate the impacts

This subchapter specifies procedures, methods and best (leading) practices which are applicable to eliminate or mitigate the potentially harmful impacts identified in the previous subchapter. It is proven in the industrial practice that by the application of appropriate methods the key potential impacts can be controlled and their harm can be kept well below the applicable regulatory limits.

In case of tailings, the application of long-lasting and leak-proof bottom liners and well-sealing tailing covers can efficiently prevent the occurrence of hazardous environmental impacts, such as acid drainage, fugitive dusting or radon gas escape. In general, tailings with properly functioning containment do not have harmful environmental impacts, these impacts are present only if the integrity of the containment is breached or such protection had not been constructed in the past.

The most important prevention / mitigation measures for uranium mines and mills are as follows:

- construction of bottom liners under the tailings to prevent groundwater contamination;
- covering of tailings (e.g. by a layer of clay) to prevent radon emanation and dusting;

- prevention of the erosion, movement, sliding, instability of the tailings;
- minimizing storm water run-off to avoid exposure of polluted areas to water;
- avoid acid rock drainage (a phenomenon characteristic to rock waste dumps) by proper sealing and covering of the rock dumps;
- monitoring the tailings area to detect contaminated seepages and leakages (deployment of groundwater wells, monitoring stations, etc.);
- monitoring the stability and structural health of the tailings and their containment structures (e.g. dams or walls) to detect their degradation and avoid accidents;
- applying ISL technology, if the long-term quality of the groundwater can be preserved;
- reduce the use of diesel fuel to avoid GHG and NO_x emissions;
- rehabilitation (reclamation) of tailings of closed or abandoned mines.

The above list is not exhaustive, a more complete list can be found e.g. in the IFC EHS Guidelines – Mining, IFC, December 2007 document.

Evaluation and summary

This section summarizes the results of the detailed analysis performed in the previous subchapters. The first part gave a brief description of the most used uranium mining technologies of today. Then potentially harmful impacts on the environment and human health were identified, treating non-radioactive and radioactive impacts separately. The next subchapter illustrated – by using the results of adequate lifecycle analysis studies – the contribution of the uranium mining and milling phase to the total lifecycle impact of nuclear energy and compared some technological options (e.g. effect of open and closed fuel cycles and the difference between the PWR and BWR designs). The next part was devoted to the laws, directives and regulations ensuring that the uranium mining and milling activities are carried out with the minimum possible impact on the environment and human health. Full compliance with the regulations listed here is required in the corresponding TSC (see Annex 4) in order to ensure that industrial activities related to uranium mining and milling will exert only acceptable environmental effects and will not represent a threat to the health of the workers and the population.

The final part listed industrial processes and best practices which are regularly used to eliminate or mitigate the potentially harmful impacts of uranium mining and milling. It is demonstrated by the best available technologies of today that by the application of adequate practices the impacts can be controlled and their magnitude can be kept well below the applicable regulatory limits.

Considering non-radioactive impacts in closed fuel cycles, uranium mining & milling has significant contribution to the total GHG emission and dominates the following impacts: SO_x and NO_x emissions, water pollution and land use. Almost 100% of the total eco-toxicity and human toxicity impacts over the whole nuclear lifecycle is connected to mining and milling and this phase also dominates the acidification, ozone creation and eutrophication potentials. On the other hand, mining & milling does not have significant effect on the water consumption, water withdrawal and production of technological waste (see Figure 3.3.1-12 for the numerical values and graphics).

If radioactive impacts in closed fuel cycles are considered then – due to radon emissions – uranium mining is responsible for 55% of the total gaseous radioactive emissions during the total lifecycle and the remaining 45% is provided by the reprocessing (see Chapter 3.3.5 for details). Note, that no significant liquid radioactive emissions can be attributed to mining and milling. In case of solid radioactive wastes, mining produces only VLLW (stored in form of tailings and residual waste), but in rather large quantities.

If closed and open fuel cycles are considered, then there is no large difference between the non-radioactive impacts, as the maximum deviation is about 20% (see Figure 3.3.1-13). On the other hand, there are significant differences in the radioactive impact indicators, mainly due to the reprocessing phase. The closed cycle produces about 50% more gaseous radioactive emissions and about ten times more liquid radioactive releases. Also due to reprocessing, the amount of ILW-LL is more than three times higher in the closed cycle. On the other hand, the open cycle produces more VLLW due to the higher amount of uranium mined and generates about three times more HLW, because the spent fuel is not reprocessed (see Figure 3.3.1-14). In the open cycle uranium mining is responsible for almost 100% of the total gaseous radioactive emissions, because here there is no contribution from reprocessing phase (gaseous emissions during operating phase are negligible if compared to the radon releases).

Evaluating the potential environmental and human health impacts of uranium mining and milling, one can conclude that the decisive factor is the grade of the uranium ore extracted. The lower the grade the higher the envisaged impacts, because mining a low grade uranium ore requires more energy (very often from fossil sources, because at remote mining sites only diesel generators can be used) and produces more rock waste and tailings.

The most important health concerns of uranium mining and milling is the radiotoxicity of rock waste piles and tailings (including the dispersal of fugitive radioactive dust), as well as radon gas emissions. If potentially harmful environmental impacts are considered, then the most important concerns are related to the pollution of water resources (including the quality of groundwater, which is a prime concern for ISL mining), as well as acidification, eco-toxicity and human toxicity.

In the following tables we relate the above identified potentially harmful impacts to the objectives of TEG and list appropriate mitigation measures to prevent or mitigate these impacts.

Table 3.3.1-1 shows the assignment ("matching") of the six environmental objectives applied by the TEG to some impact indicators widely used in related LCA studies.

Table 3.3.1-1. Matching the LCA indicators with the TEG environmental objectives

Environmental objective	Evidences of significant harm according to Taxonomy Regulation (EU) 2020/852 of 18 June 2020	Corresponding LCA impact indicators	
		Non-radioactive	**Radioactive**
climate change mitigation	significant GHG emissions	- GHG emissions	-
climate change adaptation	increased adverse impact of the current and expected climate, on the activity itself or for other people, nature and assets	- GHG emissions - water withdrawal	-
sustainable use and protection of water & marine resources	activity is detrimental to the good status, or the good ecological potential of water bodies, including surface- and groundwater	- water pollution - wtr consumption & withdrawal - acidification p. - eutrophication p. - ecotoxicity - human toxicity	- liquid RA releases
	or to the good environmental status of marine waters	- water pollution (marine waters) - acidification p. - eutrophication p. - ecotoxicity - human toxicity	- liquid RA releases
transition to a circular economy, including waste prevention and recycling	leads to significant inefficiencies in the use of materials or in the direct or indirect use of natural resources (such as non-renewable energy sources, raw materials, water and land) at one or more stages of the life cycle of products	- wtr consumption & withdrawal - land use - depletion of natural resources	-
	leads to a significant increase in the generation, incineration or disposal of waste	- production of TW	- solid RW production
	the long-term disposal of waste may cause significant & long-term harm to the environment	- production of TW - ecotoxicity - human toxicity	- solid RW production
pollution prevention and control	leads to a significant increase in the emissions of pollutants into air, water or land, as compared to the situation before the activity started	- atmospheric pollution - water pollution - ozone creation p. - ecotoxicity - human toxicity	- gaseous RA releases - liquid RA releases
protection and restoration of biodiversity and ecosystems	significantly detrimental to the good condition and resilience of ecosystems detrimental to the conservation status of habitats and species	- water pollution - wtr consumption & withdrawal - land use - acidification p. - eutrophication p. - ozone creation p. - ecotoxicity	- gaseous RA releases - liquid RA releases

RA = radioactive; RW = radioactive waste; TW = technological waste; p. = potential

Table 3.3.1-2 shows the importance of the potential impacts associated with the mining & milling lifecycle phase on the TEG environmental objectives.

Table 3.3.1-2. Importance of mining & milling impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		Prevention or mitigation of potentially harmful impacts	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	+++	Decreasing fossil fuel consumption	Diesel fuel use at remote locations
Water withdrawal	++	Optimization of water use	-
Water consumption	++	Optimization of water use	-
Water pollution	+++++	Tailings management	Containment
Ecotoxicity	+++++	Tailings management & covering	Containment & dusting prevention
Human toxicity	+++++	Idem	Idem
Land use	++++	Reclamation of land	Full remediation
Atmospheric pollution	++++	Covering of tailings	Dusting prevention
Acidification pot.	++++	Waste rock management	Avoid acid drainage
Eutrophication pot.	++++	Runoff water control	-
Ozone creation pot.	++++	Control of NOx emission	Control of diesel fuel use
Production of TW	++++	Selective waste management	-
Depletion of resources	+++	Exploring new uranium deposits	Application of novel technologies
Production of solid RW	++++	Stabilization and capping	VLLW only
Gaseous RA releases	+++++	Covering of tailings	Radon retention
Liquid RA releases	**N/A**	-	-

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

As it can be seen from Table 3.3.1-2, uranium mining and milling activities do not represent significant challenge to the climate change mitigation and adaptation TEG objectives.

However, they can significantly challenge the four remaining environmental objectives, as most of the LCA indicators can exert “high” or “critical” impacts on all these four objectives.

These challenges can be averted, as there are appropriate measures – using existing technology at reasonable costs – to prevent the occurrence of the potentially harmful impacts or mitigate their consequences (see the “appropriate measures” column in Table 3.3.1-2).

The due application of these measures is ensured by satisfying the related Technical Screening Criteria.

TSC for the uranium mining & milling activities are provided in Chapter 5 and Annex 4 of the present report (*Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy*).

3.3.1.6 References for Chapter 3.3.1

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3.3.2 Conversion to uranium hexafluoride (UF_6)

3.3.2.1 Description of the underlying technologies and processes

The final product of the uranium mining and milling phase is the yellowcake, which contains about 75% uranium oxide. However, the yellowcake still has a very long way to go before it is ready to be loaded into the core of a reactor as nuclear fuel to produce energy. The next step is the conversion of uranium oxide to uranium hexafluoride gas (UF_6 or “hex”), which can later be used in the uranium enrichment process as input.

The yellow cake, constituted by U_3O_8 is refined to obtain high purity UO_2 or UO_3 by means of chemical processes. The U_3O_8 is dissolved in nitric acid, then the solution is filtered to remove suspended impurities and treated with solvents to obtain an aqueous solution called uranyl nitrate liquor (UNL). Depending on the process, the UNL is subject to different physicochemical transformations such as dehydration and denitration, precipitation, filtration and calcination.

In the final step the purified product is converted into uranium tetrafluoride (UF_4), which is a solid material called “green salt” in the uranium industry.

Figure 3.3.2-1. Picture of green salt in a laboratory

Source: Wikipedia

Figure 3.3.2-2. UF_6 transport cylinders on their way to the enrichment plant

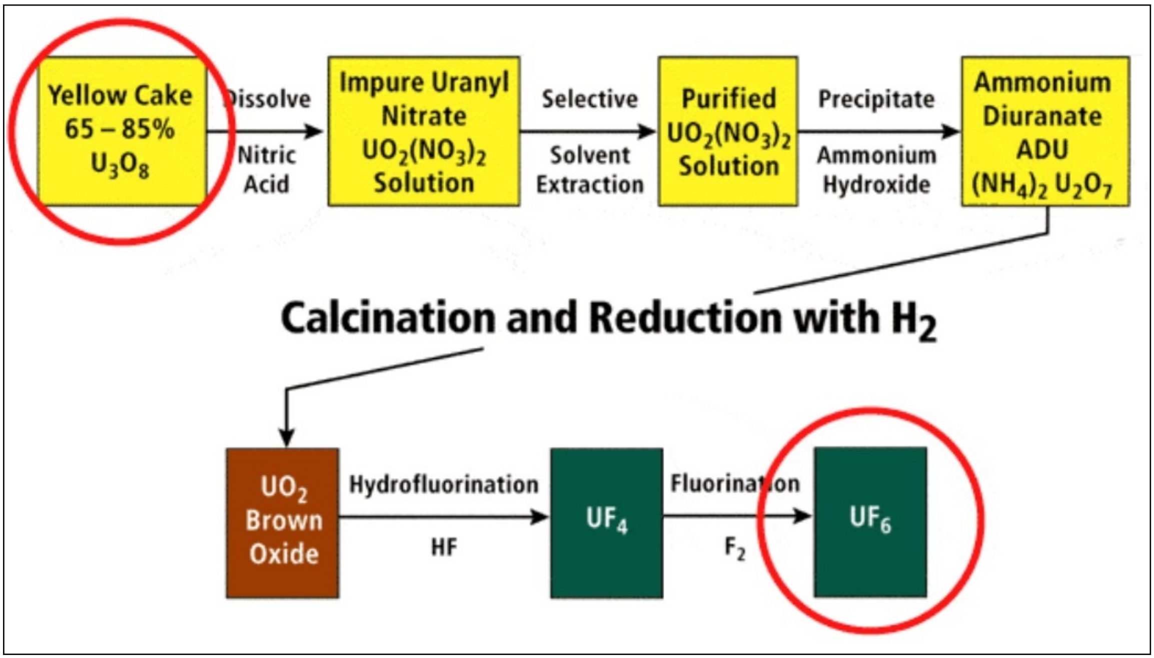
Source: Word Nuclear Transport Institute

Figure 3.3.2-3. View of the Comurhex II facility in Malvési, France

Source: Orano

Basically there are two technologies to carry out the conversion of uranium oxide to uranium hexafluoride on an industrial scale. The first process is carried out in two different factories: the first plant converts yellowcake to either uranium tetrafluoride (UF_4) or uranium trioxide (UO_3), while the second plant converts UF_4 or UO_3 to uranium hexafluoride (UF_6). The other technology converts yellowcake to UF_6 at a single facility by making use of the so called dry process.

Figure 3.3.2-4. Scheme of the wet conversion process



Source: https://web.evs.anl.gov/uranium/guide

The transformation of uranium oxide to UF_4 is carried out by a method called hydrofluorination, which can be performed either by a wet or a dry process.

In the wet process (see Figure 3.3.2-4) the uranium oxide reacts with aqueous hydrofluoric acid (HF) and the UF_4 is recovered by a precipitation process, where ammonium hydroxide is used.

In the dry process the reaction is realized with gaseous HF and the hydrofluorination occurs in excess HF.

In a common last process stage UF_4 is fluorinated by using either calcium fluoride (in fluidized bed reactors) or fluorine (in flame-tower reactors) to obtain UF_6 which is then distilled for purification.

Finally the UF_6 gas is loaded into steel transport cylinders. When cooled to room temperature, the UF_6 gas solidifies inside the cylinder. These cylinders (see Figure 3.3.2-2) are then transported to the enrichment facilities.

Today there are five major global suppliers of uranium conversion services: Orano/Comurhex (France), Cameco (Canada), Converdyn (USA), Rosatom/TVEL (Russia) and CNNC (China).

Table 3.3.2-1 shows the details and the 2019 production data for these companies, according to WNA (see https://world-nuclear.org for more details).

Orano (France) applies the wet conversion process and performs its conversion operations at two sites: the Comurhex Malvési plant converts uranium oxide into UF_4 (and uranium metal), while the Comurhex Tricastin factory in Pierrelatte produces UF_6 from the UF_4 manufactured at Malvesi.

Cameco (Canada) also uses the wet conversion process and it manufactures UF_6 for LWRs and UO_2 for the CANDU PHWRs. Cameco performs its operations at two sites: the Blind River (Ontario) uranium refining facility manufactures UO_3 from U_3O_8, while the conversion to UF_6 takes place at a facility located at Port Hope (Ontario).

Table 3.3.2-1. Global capacities for providing uranium conversion services in 2019 (Source: WNA)

Company	Locations	Nominal capacity [tU as UF_6]	Capacity utilization in 2019 [tU]
Orano (France)	Pierrelatte & Malvési	15 000	**2 500** (17% / 7%)([1])
CNNC (China)	Lanzhou & Hengyang	15 000	**10 000** (67% / 29%)
Cameco (Canada)	Blind River & Port Hope	12 500	**10 000** (80% / 29%)
TVEL (Russia)	Glazov & Seversk	12 500	**12 000** (96% / 35%)
ConverDyn (USA)	Metropolis	7 000	**0**([2])
Total	**57 500**	62 000	**34 500** (56% / 100%)

(1) Percentage of the nominal ("nameplate") capacity / Percentage of the total annual global production

(2) Note that in 2019 CoverDyn (USA) did not deliver products because in 2018 the company's plant in Metropolis (Illinois, USA) had been temporarily shut down for a major refurbishment.

While Orano and Cameco both use the wet process that requires two different facilities, the ConverDyn (USA) company applies a special process called "dry fluoride volatility conversion process", first introduced by Honeywell, USA. This process is capable of converting yellowcake to UF_6 at a single facility and it also produces very clean UF_6 gas with 99.99% or higher purity. The Honeywell dry conversion process consists of five main steps: feed preparation, reduction, hydrofluorination, fluorination, and distillation.

TVEL (Russia) uses the wet process, as well, and its production facilities are located in Glazov (UF_4 production) and Seversk (UF_6 production).

CNNC (China) also applies the wet process and – according to the World Nuclear Association – China intends to be self-sustaining in the front-end of nuclear cycle by 2030, by constructing all necessary facilities with the required capacity.

3.3.2.2 Identification of key potential impacts on the environment and human health

3.3.2.2.1 Non-radioactive impacts

The factories where uranium conversion takes place are large chemical – metallurgical plants and the conversion process itself uses a large quantity of toxic chemicals (e.g. HNO_3 – nitric acid; HF – hydrofluoride; F_2 – fluorine gas), as well as auxiliaries (e.g. CaF_2 – calcium fluoride; KOH – potassium hydroxide and ammonium hydroxide – NH_4OH). Therefore potential environmental and human health hazards characteristic of large chemical plants (e.g. accidental emission of toxic gases, chemical explosions, release of toxic liquids, etc.) also apply here and must be prevented / mitigated adequately.

The end-product of the conversion process is the uranium hexafluoride, which is stored in large steel transport cylinders (see Figure 3.3.2-2). In these cylinders the UF_6 is stored as a solid substance. If the integrity of a cylinder is lost (e.g. due to intense corrosion or fire) then it may represent the following hazards (see https://web.evs.anl.gov/uranium/guide for more details):

- Uncontrolled release of UF_6 to the environment, potentially affecting the health of plant workers and the public. The most important health hazard is the inhalation of the highly corrosive hydrogen fluoride gas, which is generated when UF_6 reacts with air moisture.
- Uranylfluoride (UO_2F_2) can also be formed when UF_6 and air moisture is in contact (UO_2F_2 is a solid substance which can be dispersed in the air and it has toxic effects if inhaled).
- If several UF_6 cylinders are leaking simultaneously then a rapid release and dispersal of large quantities of UF_6 can happen, potentially affecting a large number of people downwind.

3.3.2.2.2 Radioactive impacts

Generally, the radioactive impacts of the conversion phase are limited, because the process mostly deals with substances having specific activities corresponding to the NORM (naturally occurring radioactive materials), TENORM (technologically enhanced NORM) and VLLW (very low level radioactive waste). However, these materials may represent a threat to human health if inhaled or ingested.

In addition to VLLW, some LILW-SL (short-lived, low and intermediate level waste) are also produced during the wet process in the fluidized bed reactors where fluorination of UF_4 takes place by using calcium fluoride. During the process the CaF_2 is contaminated with short-lived decay products coming from the ^{238}U decay chain.

The plant personnel may receive direct radiation impacts – through gamma radiation[53] – during handling and/or inspection of the UF_6 storage cylinders. These impacts must be duly monitored and controlled.

Refining yields low level radioactive waste, as the uranium used in the process is natural uranium. Solid radioactive waste is constituted by non-reusable steel drums used to transport the uranium ore concentrate to the refining plant (steel drums are washed and reused while in good condition), non-soluble substances and filters which are treated to recover uranium before being packed and treated, dried slugs from liquid effluent treatment, and other miscellaneous materials from maintenance and other operations, such as gloves, cloths, rugs, etc. The process also yields liquid waste, such as the uranium ore concentrates (UOC) drums washwater, and other aqueous solutions containing different chemical compounds. The liquid wastes are treated to recover uranium and other elements, and to remove the contaminants before being released to the environment. Sludges from the treatment of liquid waste are dried and treated to recover uranium prior to packaging and managing it as solid radioactive waste. Calcination gases are treated and filtered to recover uranium and other elements before releasing them to the atmosphere.

Table 3.3.2-2 shows the low level radioactive waste produced in the refining process.

Table 3.3.2-2. Radioactive waste generated by the refining of 1000 metric tons of uranium[54]

Arisings	**Amount**	**Classification**	**Comments**
Steel drums	70 t	Reuse/recycle or waste if damaged	UOC drums
Non-soluble and filter aid	50 t	Waste	All processes (depends on the nature of UOC). To be managed as solid low level radioactive waste.
Liquid effluent Sludges	3000 – 10000 m^3 300 t	Waste	All processes (depends on the nature of UOC).
Liquid nitrate	200 t	By-product	

The radioactive waste produced in the fluidized bed reactors is calcium fluoride contaminated with uranium short-lived daughter products of ^{238}U. After decay, the uranium is recovered through dissolution with nitric acid and solvents, and recycled as UNL which is incorporated again to the refining process. The clean calcium fluoride can be reused or disposed of as non-radioactive waste.

Table 3.3.2-3 shows the waste generated in the conversion to UF_6.

Table 3.3.2-3. Waste generated by the conversion of 1000 metric tons of UF_6[55]

Arisings	**Amount**	**Classification**	**Comments**
Solid CaF_2	10 t	Treatment	Fluidized bed
Sludges with small amounts of U	20 – 50 t	Treatment	Wet process
Sludges without U	30 t	Reuse or non-radioactive waste	Wet process

[53] The gamma dose rate at the external surface of a UF_6 storage cylinder varies between 4 and 1200 μSv/h, depending on whether enriched natural U, recycled U or enriched recycled U is stored in the cylinder (see http://www.wise-uranium.org/ruxfw.html#ENR for details).

[54] International Atomic Energy Agency, Minimization of Waste from Uranium Purification, Enrichment and Fuel Fabrication, IAEA-TECDOC-1115, IAEA, Vienna (1999).

[55] International Atomic Energy Agency, Minimization of Waste from Uranium Purification, Enrichment and Fuel Fabrication, IAEA-TECDOC-1115, IAEA, Vienna (1999).

3.3.2.3 Summary of lifecycle analysis results for the conversion lifecycle phase

In general the conversion phase has rather limited contribution to the various impact indicators and it is obviously not a dominant contributor to any impact indicator (see [3.3.2-1] for the detailed LCA results - TTC fuel cycle - and Table A.2-1 in Annex 2).

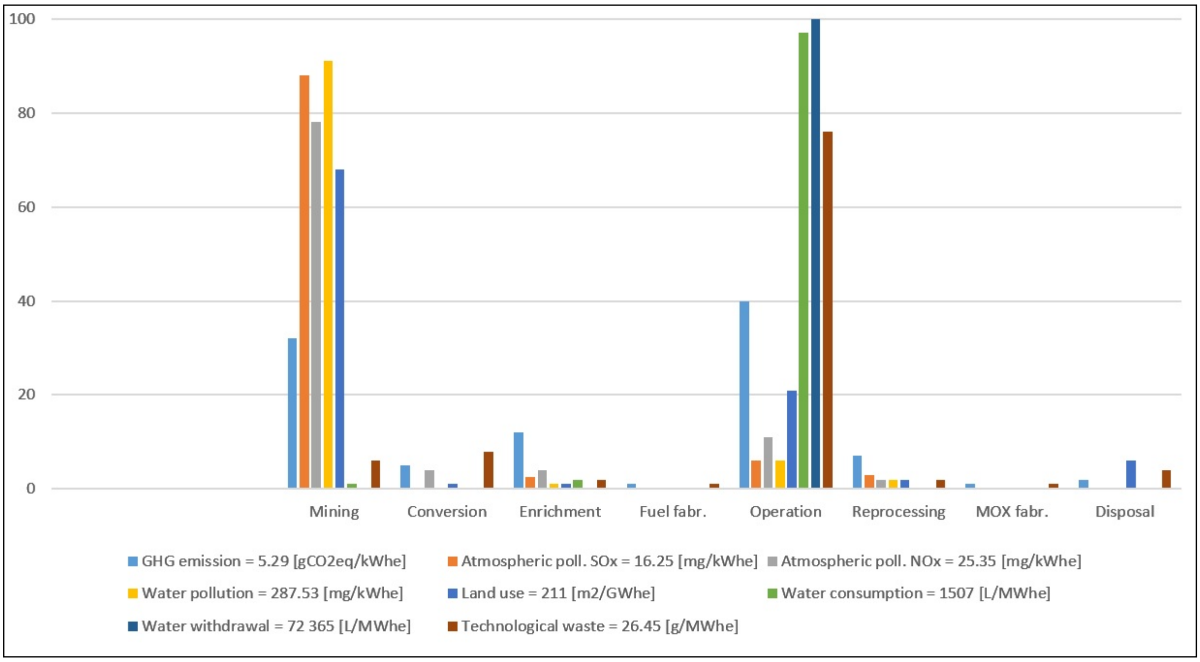
If the whole nuclear lifecycle is considered, then conversion has negligible contribution (**<1%**) to the SO_x emissions, water pollution, land use, water consumption and withdrawal, eco-toxicity and human toxicity. It has some contribution to the total GHG emission (≈**5%**), NO_x emissions (≈**4%**), technological waste (≈**8%**), acidification potential (≈**3%**), POCP (≈**5%**), eutrophication potential (≈**3%**), and production of solid radioactive waste (ILW-SL, about **4%**).

Figure 3.3.2-5 shows the relative contributions to the non-radioactive impact indicators for all lifecycle phases of nuclear energy in the case of closed (TTC) fuel cycle. It is clear from the picture that the main contributions are given by the mining & milling and operation phases. The other phases contribute much less and this is also true for the conversion phase.

Figure 3.3.2-6 shows the same data for the potential impact indicators. One can see that the dominant contribution comes from the mining & milling phase and the contribution from the conversion phase is insignificant.

If the closed cycle is compared to the open cycle (OTC) then no significant differences can be detected for any impact indicators, basically the same contributions can be observed (see Table A.2-1 in Annex 2).

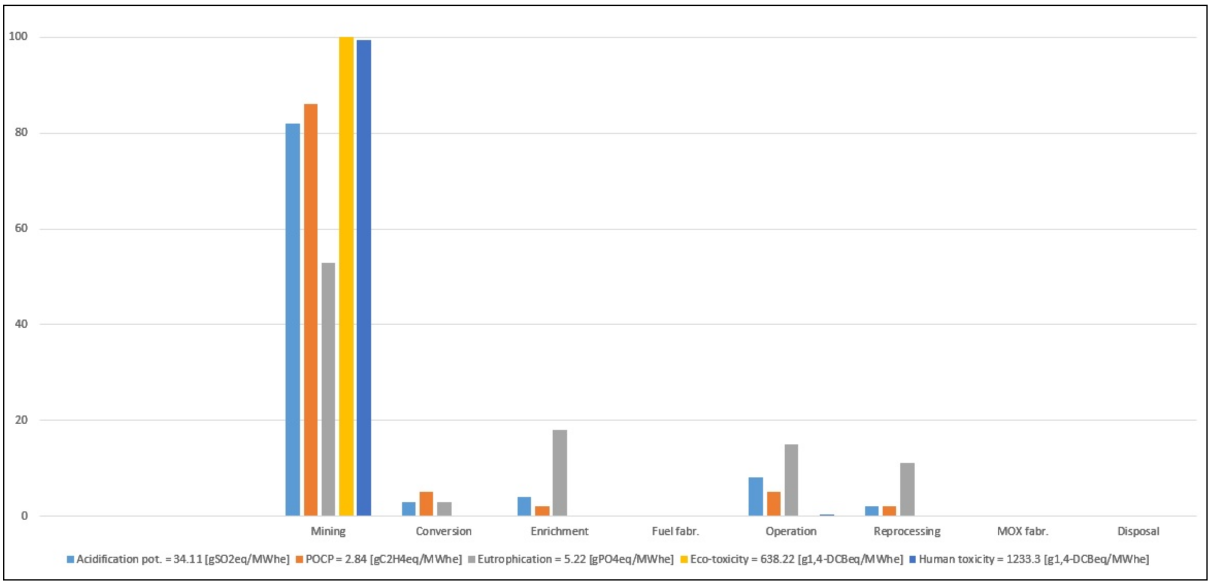
Figure 3.3.2-5. Relative contributions (in percentage) of the various lifecycle phases of nuclear energy to the non-radioactive impact indicators in case of the closed cycle



The numerical values after the impact indicators show the sum for all LC phases (total impact).

Data from [3.3.2-1]

Figure 3.3.2-6. Relative contributions (in percentage) of the various lifecycle phases of nuclear energy to the potential impact indicators in case of the closed cycle



The numerical values after the impact indicators show the sum for all LC phases (total impact).

Data from [3.3.2-1]

The conversion phase does not contribute significantly to the radioactive impact indicators either. It has some liquid radioactive emission but it is not significant if compared to the liquid emissions in the operation phase. The same is true for the solid VLLW generation: it is negligible if compared to the mining & milling phase. The short-lived ILW generation is not negligible in the conversion phase, but it is rather low: about **4%** in the closed and about **5%** in the open cycle.

3.3.2.4 Legal background and regulations

The EU regulations corresponding to nuclear and radiation safety and security aspects of various lifecycle phases of nuclear energy are discussed in Chapter 3.3.1, here the details are not repeated. In addition to specific considerations outlined in Chapter 3.3.1 on the application of ISO 9001:2015 (Quality Management System); ISO 14001:2015 (Environmental Management System) and ISO 45001:2018 (Health and Safety Management at Work) at the conversion facilities, the below listed regulations and standards are applicable to nuclear facilities carrying out conversion activities.

- REACH regulation - Regulation (EC) No 1907/2006 of the European Parliament and of the Council of 18 December 2006 concerning the Registration, Evaluation, Authorisation and Restriction of Chemicals (REACH)
- National Emission Reduction Commitments Directive – Directive (EU) 2016/2284 the European Parliament and of the Council of 14 December 2016 on the reduction of national emissions of certain atmospheric pollutants

Related guidelines of the International Finance Corporation (IFC):

- IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007
- IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks and Impacts, IFC, January 2012
- IFC Performance Standard 6 – Biodiversity Conservation and Sustainable Management of Living Natural Resources, IFC, January 2012

3.3.2.5 Identification of applicable means to avoid or mitigate the impacts

This subchapter specifies procedures, methods and best (leading) practices which are applicable to fully avoid or mitigate the potentially harmful impacts identified in the previous subchapter.

3.3.2.5.1 Non-radioactive impacts

As the uranium conversion factories are large chemical – metallurgical plants dealing with toxic gases and liquids, the best industrial practices and technological solutions to eliminate potential hazards associated with the handling and use of such substances are to be applied.

In case of the end-product of the conversion process (i.e. UF_6 stored in transport cylinders) the highest potential risk is represented by those accidents when the integrity of the cylinders is lost. Therefore all UF_6 cylinder handling and storage operations must be conducted in a manner that minimizes the chances of accidents.

3.3.2.5.2 Radioactive impacts

Although the specific activity of the materials handled during the conversion process is usually low, these substances may represent a threat to human health if inhaled or ingested. Therefore the conversion technology has to apply strict health protection and worker's safety measures to avoid such effects.

If radiological environmental impacts are considered, then prevention of water and air pollution by radioactive materials is the main protection measure to avoid the emergence of such effects.

The ILW-SL waste generated during the wet process in the fluorination process (i.e. CaF_2 contaminated with short-lived decay products of the ^{238}U decay chain) can be cleaned and disposed of by the following method. After a certain waiting period, which ensures that short-lived isotopes disappeared from the contaminated calcium fluoride, the uranium can be recovered and the remaining CaF_2 can either be reused or disposed of as technological (non-radioactive) waste.

Direct radiation impacts to which the plant personnel is potentially exposed (e.g. during handling and/or inspection of the UF_6 storage cylinders) are controlled by the usual radiation protection and dose monitoring procedures commonly applied at nuclear facilities.

3.3.2.6 Evaluation and summary

This section summarizes the results of the detailed analysis performed in the previous subchapters.

As already mentioned, the conversion phase has limited contribution to the various impact indicators and it is not a dominant contributor to any impact indicator. This is true for the open and closed fuel cycles, as well.

In Chapter 3.3.1 Table 3.3.1-1 shows the assignment ("matching") of the environmental objectives applied by the TEG to some impact indicators widely used in related LCA studies. This table is not repeated here, but it must be duly considered when interpreting Table 3.3.2-4 below.

Table 3.3.2-4 shows the importance of the potential impacts associated with the conversion lifecycle phase on the TEG environmental objectives.

It can be seen from Table 3.3.2-4 that uranium conversion activities do not represent significant challenge to any of the TEG objectives.

The existing minor challenges (e.g. waste generation) can be averted, as there are appropriate measures – using existing technology at reasonable costs – to prevent the occurrence of the potentially harmful impacts or mitigate their consequences (see the "appropriate measures" column in Table 3.3.2-4).

Note that Technical Screening Criteria were not developed for this activity, because this lifecycle phase does not represent a dominant contribution in any of the impact categories used in our study.

Table 3.3.2-4. Importance of conversion phase impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		Prevention or mitigation of potentially harmful impacts	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	++	Decreasing fossil fuel consumption	-
Water withdrawal	+	Optimization of water use	-
Water consumption	+	Idem	-
Water pollution	+	Application of best practices	-
Ecotoxicity	+	Idem	-
Human toxicity	+	Idem	-
Land use	+	Reclamation of land	Full remediation
Atmospheric pollution	++	Application of best practices	-
Acidification pot.	++	Idem	-
Eutrophication pot.	++	Idem	-
Ozone creation pot.	++	Idem	-
Production of TW	++	Selective waste management	-
Depletion of resources	++	Optimization of use of chemicals	-
Production of solid RW	++	Recovering contaminated CaF_2	VLLW and ILW-SL only
Gaseous RA releases	++	Proper handling of UF_6 cylinders	Only accidental releases
Liquid RA releases	++	Application of best practices	-

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

3.3.2.7 References for Chapter 3.3.2

[3.3.2-1] Ch. Poinssot, et al.: Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211

3.3.3 Uranium enrichment

3.3.3.1 Description of the underlying technologies and processes

The step following the conversion phase is the enrichment of uranium, which means increasing the concentration of the ^{235}U isotope in the uranium. Considering that commercial PWRs use nuclear fuel containing uranium enriched to maximum 5 wt% in the ^{235}U isotope, appropriate isotope enrichment techniques must be applied, because the natural uranium contains only 0.711 wt% of ^{235}U.

The end-product of the conversion step is UF_6 (uranium hexafluoride or "hex"), which has several advantageous properties if the technological needs of the enrichment phase are considered (see [3.3.3-1] for details):

— Fluorine contains only one isotope, therefore the UF_6 contains only one fluorine isotope;

— The compound can be handled at reasonable temperatures and pressures;

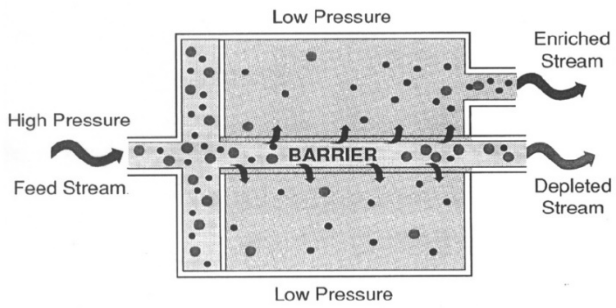
— The compound is water soluble, but at room temperature it is a white crystalline solid. When heated it is vaporized and turns to gas which fits better to the enrichment process.

The uranium in the UF_6 compound can be enriched in ^{235}U by using two very different industrial processes: one method relies on diffusion and the other applies high rotation-speed centrifuges. Both methods utilize the small mass difference between the ^{235}U and the ^{238}U isotopes.

3.3.3.1.1 Gaseous diffusion method

Industrial scale gaseous diffusion (see Figure 3.3.3-1) makes use of a well-known process called molecular effusion, in which a contained, pressurised gas escapes from a tank through miniscule holes having diameters considerably smaller than the mean free path of the gas molecules in question.

Figure 3.3.3-1. Scheme of a gaseous diffusion stage



Source: [3.3.3-1]

Figure 3.3.3-2. A transport cylinder is prepared for handling at the plant

Source: urenco.com

According to Graham's law of effusion, if a mixture of two gases (having molar masses M_1 and M_2) undergoes effusion under the same temperature and pressure, then the ratio of effusion flows is inversely proportional to the square root of the molar mass ratios: $Q_{E1}/Q_{E2} = \sqrt{(M_2/M_1)}$. If this equation is applied to the $^{235}UF_6$ and $^{238}UF_6$ gas mixture then $Q_{235}/Q_{238} = \sqrt{(352/349)} =$ **1.00429** (the molar mass of fluorine is 19). This number is the theoretical ^{235}U – ^{238}U separation factor (or enrichment ratio) which is associated with a single diffusion stage. The stages are connected one after another, forming a cascade in the hall of a gaseous diffusion plant (see Figure 3.3.3-3).

Figure 3.3.3-3. A cascade of diffusors at the Georges Besse (France) enrichment plant

Source: Orano

The <u>theoretical</u> minimum number of diffusion stages, N_{min}, required to achieve a given enrichment level can be calculated from the enrichment of the final product, the enrichment of the depleted uranium (usually called "tailings") and the enrichment ratio, or separation factor, α, as follows (see [3.3.3-1] for details):

$$\mathbf{N_{min} = \ln(R_P/R_T)/\alpha_0}$$

Here α_0 is the separation gain (α-1). In theory $\boldsymbol{\alpha_0}$ **= 0.00429** for the gaseous diffusion process; $\mathbf{R_P = {}^{235}C_P/(1 - {}^{235}C_P)}$; $\mathbf{R_T = {}^{235}C_T/(1 - {}^{235}C_T)}$; where $\mathbf{{}^{235}C_P}$ and $\mathbf{{}^{235}C_T}$ correspond to the ^{235}U concentration (wt%) of the final product and the tailings, respectively. If $^{235}C_P$ = 0.04 (4 wt%) and $^{235}C_T$ = 0.002 (0.2 wt%) then N_{min} = 707.

In a real enrichment plant a diffusion stage contains several thousands of thin tubes having porous (or membrane) walls (barriers) and the "feedstock" is pumped through these tubes. Due to the efficiency of the barrier the real separation gain in an industrial facility is much lower than the α_0 theoretical value, in practice the applicable barriers will not have a separation factor higher than α = **1.0022** (see [3.3.3-4] for details). Using the real α separation factor, the actual number of stages will be 1379, therefore about **1400** diffusion stages have to be constructed to achieve the **4%** enrichment, which is the average fuel enrichment in PWRs.

As described in Chapter 3.3.2, at the end of the conversion process the UF_6 material is stored at room temperature in the transport cylinders in solidified form. When starting the enrichment process, uranium transport cylinders are heated in an autoclave (see Figure 3.3.3-2), generating heated UF_6 gas that is fed into the diffusion process.

Gaseous diffusion plants (see Figure 3.3.3-4 for example) used to be very energy intensive. As already mentioned, first the storage cylinders must be heated up before feeding the UF_6 gas into the system. The pressure driving the separation process is created by high pressure compressors, but each stage causes a certain pressure loss which has to be compensated by repeatedly compressing the UF_6 gas, before entering the next stage in the cascade. This repeated compression heats up the gas, therefore it must be cooled before entering the next stage. This multiple pumping and cooling through several hundreds of diffusion stages requires an extremely large amount of energy, e.g. a US gaseous diffusion plant with an annual capacity of 10 million SWU[56] requires about 2700 MW of electrical power [3.3.3-3].

[56] SWU = separative work unit (a unit characterizing the capacity of the uranium enrichment plants, the detailed explanation is given later in this subchapter).

In France, the Georges Besse diffusion enrichment plant – located at the Tricastin site and now retired – had been supplied by electricity from three NPP units located at the same site, in order to ensure low-cost electricity for this very energy intensive technology. As a comparison, the new Georges Besse II centrifugal plant with the same enrichment capacity consumes only 50 MW_e.

Figure 3.3.3-4. View of the Portsmouth gaseous diffusion uranium enrichment plant

Source: DOE, USA

Since June 2013 no gaseous diffusion plants are used for providing enrichment services in any country because they have all been gradually shut down around the world.

The last two large facilities to close were the Georges Besse I plant (France) in 2012, and the Paducah plant (USA), which stopped operation in May 2013 (see [3.3.3-4] for details). Currently all enrichment services are provided by facilities utilizing the ultracentrifuge technology.

3.3.3.1.2 Ultracentrifuge method

When UF_6 gas is placed in a centrifuge then the centrifugal force acting on the gas molecules with larger mass is larger than the force acting on the lighter gas molecules. As a consequence, UF_6 molecules containing the heavier ^{238}U isotope drift to the outer wall of the centrifuge, while UF_6 molecules with the lighter ^{235}U isotope tend to stay in the middle region (i.e. around the vertical axis) of the centrifuge. In practice an industrial gas centrifuge (often called "ultracentrifuge") is a long, slim vertical cylinder, encapsulated in a closed tank under vacuum and it rotates with very high speed (between 50 and 70 krpm). The system is fed by UF_6 gas and as a result of the fast centrifuging the concentration of the heavier UF_6 molecules increases towards the outer wall of the centrifuge. In the long cylinder the flow paths are arranged in such a manner, that the heavier gas moves towards the bottom of the tank, while the lighter gas moves to the top, allowing separation of the "products" at the bottom and at the top of the centrifuge. The gas enriched in the ^{235}U isotope is then fed to the next centrifugal-stage, while the gas depleted in the ^{235}U isotope is driven back to the beginning of the whole process (see Figure 3.3.3-5). At the end of the above described process, 10-15% of the original uranium quantity is obtained as enriched uranium, while 85-90% remains as depleted uranium (note that the ^{235}U-concentration in the depleted uranium is much lower than 0,711 wt% characterizing natural uranium, usually it is between 0,2 or 0,35 wt%). A significant advantage of the centrifugal enrichment over the gaseous diffusion method is its energy consumption: the centrifugal method requires about 50 times less energy than the gas diffusion method. In addition, the separation factor for a centrifugal stage can be significantly higher than for a gaseous diffusion stage, because the radial separation factor is proportional to the absolute mass difference between ^{238}U and ^{235}U isotopes rather than the ratio of the molecular masses, as in the gaseous diffusion process.

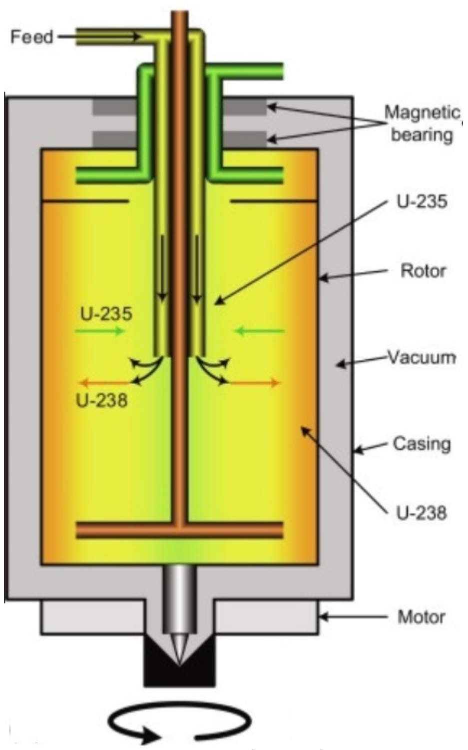
The theoretical radial separation factor for a centrifuge can be expressed as

$$\alpha = \exp\,[(M_2 - M_1)\cdot v^2/2RT]$$

where $M_2 - M_1 = 3$ g/mol (molar mass difference between the $^{238}UF_6$ and $^{235}UF_6$ gases); **v** is the peripheral velocity of the rotating cylinder, **T** is the temperature in °K and **R** is the universal gas constant (8.314 J/°K/mol). If we take a cylinder with 20 cm diameter at 300 °K (27 °C) and rotate it with 30 krpm speed, then $\alpha = 1.0611$ is obtained. If the rotation speed is increased to 50 krpm, then $\alpha = 1.1788$ is the result (see [3.3.3-6] for more details). In a real installation the separation factors are considerably lower, but still much higher than for the gaseous diffusion method.

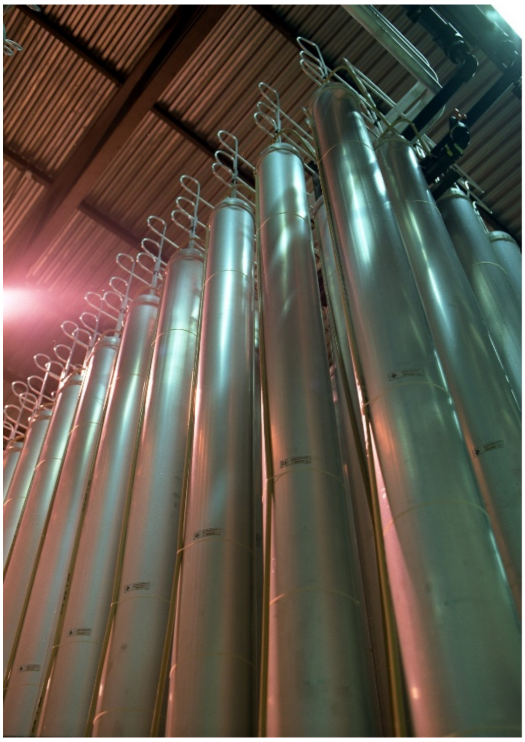
The UF_6 substance arrives at the centrifugal enrichment plant in the same standard transport cylinders as to the gaseous enrichment plant. Before feeding it to a centrifuge cascade (see Figure 3.3.3-6) it is also heated up in autoclaves and used in gaseous form in the enrichment process.

Figure 3.3.3-5. Scheme of ultracentrifugal separation



Source: [3.3.3-5]

Figure 3.3.3-6. An ultracentrifuge cascade



Source: urenco.com

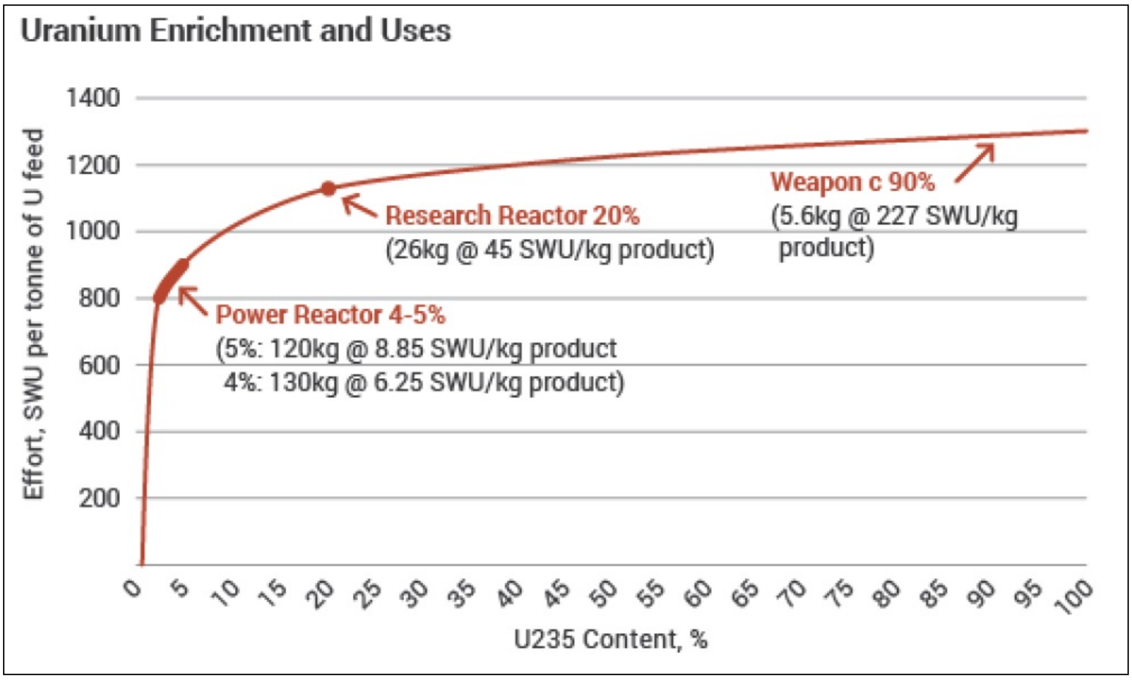
The separation factor for a centrifugal stage is larger than for a diffusion stage, therefore one needs considerably fewer centrifuges to achieve a given enrichment level. However, the enrichment capacity (material throughput) of a centrifugal stage is much smaller than for the diffusion case, therefore in a real enrichment plant centrifuge cascades operating in serial and parallel arrays are applied. The serial centrifuges work to multiply the separation effect, while the parallel cylinders provide the required magnitude (throughput) of the separative work.

At the end of the process the "product" (i.e. the gas enriched in the ^{235}U isotope) is compressed, cooled and stored in transport cylinders as a solid substance. The "tailings" (i.e. the gas depleted in ^{235}U isotope) undergoes the same procedure and is finally also stored in transport cylinders in solidified form. The product cylinders are shipped to the fuel manufacturer, while the tailings can either be re-enriched or deconverted to a chemically stable uranium oxide (U_3O_8) or shipped to a final disposal facility. Note that the deconversion of UF_6 to U_3O_8 produces hydrofluoric acid as a by-product, which can be marketed.

3.3.3.1.3 Separative work unit

The separative work unit (SWU) is a unit commonly applied to characterize the capacity of the uranium enrichment plants. The SWU indicates the energy input relative to the amount of uranium processed, the degree to which it is enriched and the level of depletion of the remnant substance (the "tails"). The "kilogram separative work" unit measures the quantity of separative work performed to enrich a given amount of uranium ("feedstock") to a certain amount, when feed and product quantities are expressed in kilograms (see [3.3.3-7] for details). Figure 3.3.3-7 shows the dependence of SWU on the ^{235}U enrichment level of the final product, assuming a given amount of feedstock (one metric ton of natural uranium). The dependence curve is strongly nonlinear, in the low enrichment region (below 5%) one has to invest a lot of effort to achieve the 4-5% enrichment levels commonly used for PWR and BWR fuel. For enrichment levels above about 20% the curve almost reaches a plateau, i.e. above a certain enrichment level it requires much less effort to increase the enrichment considerably.

Figure 3.3.3-7. Dependence of SWU on the ^{235}U concentration in the product



Source: [3.3.3-7]

Table 3.3.3-1. Worldwide enrichment plants between 2013 and 2020 – Operational and planned capacities given in thousand SWU/year

Country	Company & plant	2013	2015	2018	2020
France	Orano: Georges Besse II	5 500	7 000	7 500	7 500
Germany-NL-UK	Urenco: Gronau (D); Almelo (N L); Capenhurst (UK)	14 200	14 400	13 900	13 620
Russia	Rosatom/Tenex: Angarsk, Seversk, Novouralsk, Zelenogorsk	26 000	26 578	28 215	27 654
USA	Urenco: New Mexico	3 500	4 700	4 600	4 540
China	CNNC: Hanzhun, Lanzhou	2 200	5 760	6 750	6 750
Others	Japan, Argentina, Brazil, India, etc.	150	162	135	140
Total SWU/year		51 550	58 600	61 100	60 200

Source: [3.3.3-7]

Currently enrichment services are provided exclusively by plants based on the centrifuge technology and Russia alone provides almost 50% of the global capacity.

Note that as of today the USA does not operate an enrichment facility of its own: after shutting down the large diffusion plants in Oak Ridge, Portsmouth and Paducah there were ambitious plans to establish a sufficiently large centrifuge enrichment plant, but these plans were not realized due to various reasons (e.g. cheap global market prices for enrichment services).

3.3.3.2 Identification of key potential impacts on the environment and human health

3.3.3.2.1 Non-radioactive impacts

Uranium enrichment plants use gaseous UF_6 as working material and deliver two different types of end-products, both stored in standard transport cylinders. One of the end-products is the enriched uranium itself, which is then transported to the nuclear fuel fabrication facility where it is subjected to further processing steps and usually becomes uranium oxide (UO_2) fuel. The other end-product is the depleted uranium (DU), also called "tailings". The concentration of the ^{235}U isotope in the enrichment tailings is between 0.20 and 0.35 wt% and the cylinders containing this material are either transported to a long-term storage location (waiting there for later use, see Figures 3.3.3-8 and -9) or shipped to a "deconversion" facility where the UF_6 gas is chemically decomposed to yield uranium oxide (U_3O_8) and hydrofluoric acid.

By the end of the last century large stocks of depleted uranium transport cylinders were piled up at storage yards close to the enrichment plants. A considerable part of this stockpile came from the enrichment operations related to national defence, i.e. from producing nuclear weapons. The EIA for a planned DU-deconversion plant in the USA ([3.3.3-8]) states that in 2004 the US DOE had an inventory of about 700 000 metric tons of DU, stored in about 60 000 transport cylinders at the Paducah, Portsmouth and Oak Ridge sites (see Figure 3.3.3-9 to illustrate the size of the stockpile at the Paducah site). Besides the USA, other countries (e.g. France, UK and Russia) also stored large quantities of DU, and large projects were initiated to decrease the size of these stockpiles considerably. These projects had to deal with large amounts of contaminated steel (i.e. the transport cylinders themselves) and had to deconvert or re-use the solid UF_6 material stored in the cylinders (see section "*Identification of applicable means to avoid or mitigate the impacts*" for the possible methods to accomplish these goals).

Figure 3.3.3-8. Storage of cylinders with depleted UF_6 at the Portsmouth site

Source: DOE, USA

Unlike uranium conversion plants, uranium enrichment facilities do not regularly use large amounts of additional toxic chemicals during the enrichment process. However, the working material of the enrichment process is the gaseous UF_6, which forms hydrofluoric acid (a very corrosive substance) when in contact with moisture. Adequate measures must be implemented throughout the whole process, in order to avoid leakages.

The chemical toxicity of UF_6 is more significant than its radiological toxicity, therefore protective measures required in an enrichment plant are similar to those valid in other chemical factories dealing with the production of fluorinated chemicals.

If a deconversion process is applied to treat the depleted uranium then this process produces HF (hydrogen fluoride) and hydrofluoric acid (HF in aqueous solution). The deconversion factory is separated from the enrichment plant and as a rule regulations for chemical factories using toxic chemicals apply to them.

Figure 3.3.3-9. Aerial view of the depleted UF_6 cylinder storage yard at the Paducah gaseous diffusion plant (USA), storing about 38 000 used cylinders in 2001

Source: www.robertharding.com

However, potential problems associated with leaking or damaged transport cylinders may also occur in an enrichment plant, therefore the considerations and risks related to the integrity of the cylinders also apply here (see the corresponding section in Chapter 3.3.2 on uranium conversion).

As already mentioned, the diffusion enrichment plants were very energy intensive and indirectly they were responsible for large CO_2 emissions, if the electricity had not been ensured from low-carbon electricity production sources (e.g. hydro or nuclear). On the contrary, a gas centrifuge plant requires about 50 times less electric power to supply the same separative work as a diffusion plant, therefore concerns related to the extensive CO_2 emissions potentially associated with uranium enrichment are no longer present. By 2013 diffusion enrichment plants have been closed around the world and after this date only gas centrifuge plants provide enrichment services.

3.3.3.2.2 Radioactive impacts

In the enrichment phase – similar to the conversion phase – radioactive impacts are limited, because the process deals with substances having specific activities corresponding to the NORM, TENORM and VLLW levels. VLLW includes small amounts of alumina and sodium fluoride produced in the chemical traps of the purification system to retain small amounts of UF_6 carried along hydrofluoric acid and other non-condensable gases, adsorption and filtering media, scrap metal, clothing, rags, dried slugs from treatment of liquid effluents, and oil and sludge from maintenance and decontamination activities. The enrichment process involves materials containing only natural and long-lived radioactive isotopes, because in the applied technology, formation of highly radioactive isotopes (e.g. by nuclear fission or by neutron irradiation of materials) does not occur.

The only exception is when reprocessed uranium (REPU) is being enriched, because in this case the REPU is first purified[57] to eliminate all short-lived (and therefore highly radioactive) impurities and then it is converted to UF_6. For technological reasons the REPU enrichment is performed only in centrifugal plants, where there are only minor differences between the enrichment of natural uranium and REPU. These include some extra measures when handling the "product" and the application of dedicated cascades, where appropriate radiation shielding is applied (see [3.3.3-9] for more details).

[57] Note that presently this purification process can be carried out only at a Russian facility.

The materials used in the process represent a threat to human health if inhaled or ingested, therefore the enrichment technology must apply strict health protection and worker's safety measures to avoid such effects.

If potential environmental impacts are considered, then prevention of water and air pollution by radioactive materials is the main protection measure to avoid the emergence of such effects.

The plant personnel may also receive direct radiation impacts – through gamma radiation – during handling and inspecting the UF_6 storage cylinders. These impacts are controlled by the usual radiation protection and dose monitoring procedures commonly applied at nuclear facilities.

As mentioned above, the enrichment process generates large amounts of depleted uranium which can be considered as a by-product for future use or as waste. UF_6 can be stored in steel containers for long periods of time (i.e. for decades), provided that there is a suitable periodic surveillance programme in place to ensure the long-term integrity of the containers. Alternatively it can be "deconverted" to depleted U_3O_8, which is a more stable substance, better suited for storage or disposal, allowing also the recovery of high purity hydrofluoric acid for industrial use. Deconversion can also save a significant amount of uranium mining. Alternatively, the HF is neutralized into CaF_2 for storage or for industrial use.

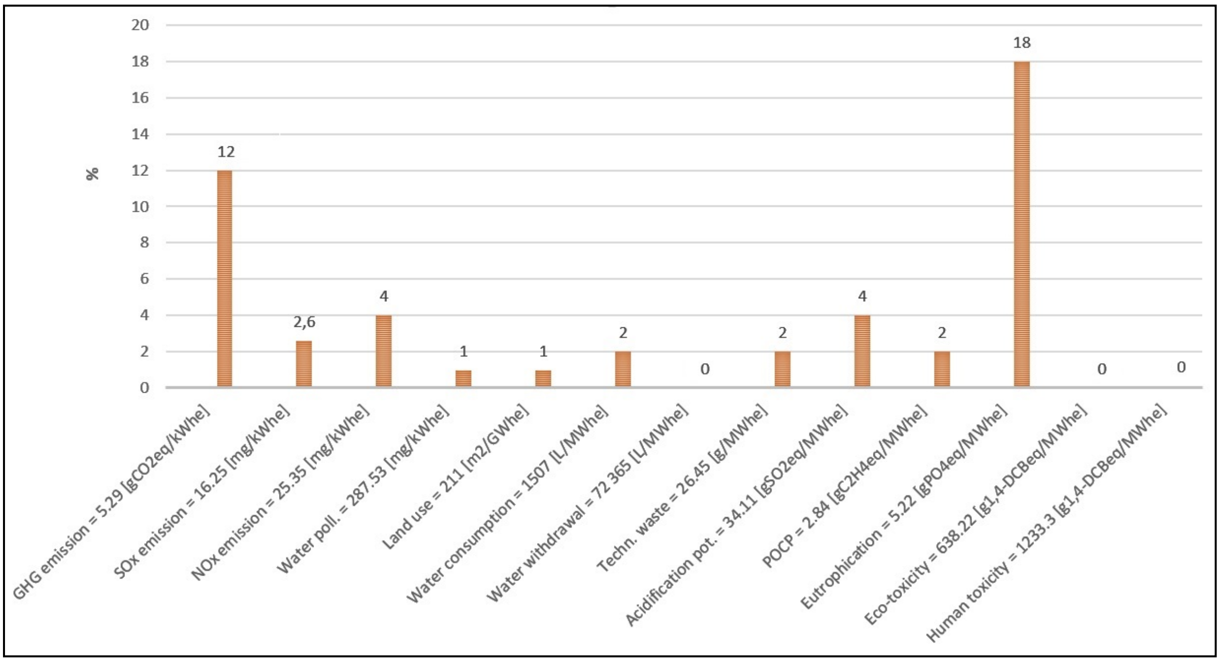
3.3.3.3 Summary of lifecycle analysis results for the enrichment lifecycle phase

In general the enrichment phase has moderate contribution to the various impact indicators and it is not a dominant contributor to any impact indicator (see [3.3.3-1] for the detailed LCA results - TTC fuel cycle - and Table A.2-1 in Annex 2).

If the whole nuclear lifecycle is considered, then enrichment has negligible contribution (**≤1%**) to the water pollution, land use, water withdrawal, eco-toxicity and human toxicity. It has some contribution to the SO_x emissions (≈**3%**) and NO_x emissions (≈**4%**), water consumption (≈**2%**), technological waste (≈**2%**), acidification potential (≈**4%**), POCP (≈**2%**).

It has larger than 10% contribution only to the total GHG emission (≈**12%**) and the eutrophication potential (≈**18%**).

Figure 3.3.3-10. – Relative contributions (in percentage) of the enrichment phase to the non-radioactive and potential impact indicators in case of the closed cycle



Data from [3.3.3-10]

Figure 3.3.3-10 shows the relative contributions to the non-radioactive and potential impact indicators for the enrichment lifecycle phase of nuclear energy in the case of a closed (TTC) fuel cycle. It is clear from the picture that the enrichment phase provides significant contribution only to the "GHG emission" and the "Eutrophication potential" impact indicators. For the other impact indicators the contribution is less than 5% and often even negligible.

If the closed cycle is compared to the open cycle (OTC) then no significant differences can be detected for any impact indicators, basically the same contributions can be observed (see Table A.2-1 in Annex 2).

The enrichment phase does not contribute significantly to any of the radioactive impact indicators. There are no atmospheric or liquid radioactive discharges and no significant amount of solid radioactive waste is produced in any waste category. Note that the depleted uranium is usually not considered as radioactive waste, because later it is either deconverted to uranium oxide and HF (hydrofluoric acid) or reused again for enrichment.

3.3.3.4 Legal background and regulations

The EU regulations corresponding to nuclear and radiation safety and security aspects of various lifecycle phases of nuclear energy are discussed in Chapter 3.3.1, here the details are not repeated. In addition to specific considerations outlined in Chapter 3.3.1 on the application of ISO 9001:2015 (Quality Management System); ISO 14001:2015 (Environmental Management System) and ISO 45001:2018 (Health and Safety Management at Work) the below listed regulations and standards are applicable to nuclear facilities carrying out enrichment activities.

- REACH regulation - Regulation (EC) No 1907/2006 of the European Parliament and of the Council of 18 December 2006 concerning the Registration, Evaluation, Authorisation and Restriction of Chemicals (REACH)
- National Emission Reduction Commitments Directive – Directive (EU) 2016/2284 the European Parliament and of the Council of 14 December 2016 on the reduction of national emissions of certain atmospheric pollutants

Related guidelines of the International Finance Corporation (IFC):

- IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007
- IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks and Impacts, IFC, January 2012
- IFC Performance Standard 6 – Biodiversity Conservation and Sustainable Management of Living Natural Resources, IFC, January 2012

Note that the application of IFC standards may be preferred in non-EU countries, because their validity is not restricted to Europe.

3.3.3.5 Identification of applicable means to avoid or mitigate the impacts

This subchapter specifies procedures, methods and best (leading) practices which are applicable to fully avoid or mitigate the potentially harmful impacts identified in the previous subchapters.

3.3.3.5.1 Non-radioactive impacts

Uranium enrichment plants can be considered as large chemical plants dealing with gaseous UF_6 as working material. Since UF_6 in contact with water produces hydrofluoric acid, any UF_6 leakage or air ingress has to be avoided in the process. In order to avoid such harmful leakages by default, in most areas of an enrichment plant the pressure of the UF_6 gas is maintained below atmospheric pressure and double containment protection is provided for those areas where the use of higher pressures is unavoidable. Effluent and venting gases are also collected and adequately treated (see [3.3.3-7] for more details).

In addition, where toxic gases and liquids are handled in an enrichment plant, best industrial practices and technological solutions must be applied to eliminate potential hazards associated with the handling and use of such substances.

The UF_6 feedstock, the enriched end-product and the remaining depleted uranium (tailings) are all stored in standard transport cylinders. For these cylinders the highest potential risk is represented by those accidents when the integrity of the cylinders is lost. Therefore all UF_6 cylinder handling and storage operations must be conducted in a manner that minimizes the chances of accidents.

3.3.3.5.2 Radioactive impacts

Although the specific activity of the materials handled during the enrichment process is usually low, these substances represent a threat to human health if inhaled or ingested. Therefore the enrichment technology has to apply strict health protection and worker's safety measures during the whole process to avoid such effects.

If radiological environmental impacts are considered, then prevention of water and air pollution by radioactive materials is the main protection measure to avoid the emergence of such effects.

Direct radiation impacts to which the plant personnel are potentially exposed (e.g. during handling and/or inspection of the UF_6 storage cylinders) are controlled by the usual radiation protection and dose monitoring procedures commonly applied at nuclear facilities. As a special case when REPU is enriched, special radiation protection (shielding) measures must be applied for those centrifuge cascades where the REPU is processed.

In case the depleted uranium is deconverted then this activity is performed in a factory which is separated from the enrichment plant, but it is usually located at the same site. Deconversion of UF_6 is routinely performed by reacting the UF_6 gas with water steam to achieve "defluorination". This reaction produces uranyl fluoride (UO_2F_2), which is then further reacted with steam and hydrogen to produce U_3O_8 powder and hydrogen fluoride (HF) vapour. The U_3O_8 powder is suitable for long-term safe storage in containers. The gaseous HF is cooled and liquefied as hydrofluoric acid, which is a marketable product (see www.urenco.com for more details). For example, the Pierrelatte facility (Tricastin, France) handles and stores depleted uranium recovered from defluorination after enrichment and from processing of used nuclear fuel. Uranium from used enrichment components is recovered at the Socatri plant, which is the radioactive waste management facility at Tricastin, but also treats industrial discharges from the Tricastin site. The above description illustrates that the depleted uranium and the used components of an enrichment plant (usually contaminated by uranium deposits) can be treated in a safe manner, without producing intermediate- and high-level radioactive wastes.

3.3.3.6 Evaluation and summary

This section summarizes the results of the detailed analysis performed in the previous subchapters.

As already mentioned, the enrichment phase has limited contribution to the various impact indicators and it is not a dominant contributor to any impact indicator. This is true for both the open and closed fuel cycles.

In Chapter 3.3.1 Table 3.3.1-1 shows the assignment ("matching") of the environmental objectives applied by the TEG to some impact indicators widely used in related LCA studies. This table is not repeated here, but it must be duly considered when interpreting Table 3.3.3-2 below.

Table 3.3.3-2 shows the importance of the potential impacts associated with the enrichment lifecycle phase on the TEG environmental objectives. Note that only centrifugal enrichment was considered in the table, because after 2013 no diffusion enrichment plants are in operation.

It can be seen from Table 3.3.3-2 that uranium enrichment activities do not represent a significant challenge to any of the TEG objectives.

The existing challenges (e.g. proper handling of depleted uranium stocks) can be adequately managed as there are appropriate measures – using existing technology at reasonable costs – to prevent the occurrence of potentially harmful impacts or mitigate their consequences (see the "appropriate measures" column in Table 3.3.3-2).

Note that Technical Screening Criteria were not developed for this activity, because this lifecycle phase does not represent a dominant contribution in any of the impact categories used in our study.

Table 3.3.3-2. Importance of enrichment phase impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		Prevention or mitigation of potentially harmful impacts	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	++	Limiting fossil fuel consumption	The energy intensive diffusion method had been phased-out
Water withdrawal	+	Optimization of water use	-
Water pollution	+	Application of best practices	-
Ecotoxicity	+	Idem	-
Human toxicity	+	Idem	-
Land use	+	Reclamation of land	Full remediation
Water consumption	+	Application of best practices	-
Atmospheric pollution	+	Idem	-
Acidification pot.	+	Idem	-
Ozone creation pot.	+	Idem	-
Eutrophication pot.	++	Limiting chemical releases	-
Production of TW	++	Defluorinated DU is stored as TW	U_3O_8 powder can be safely stored in proper containers for long time
Depletion of resources	++	Reconversion / re-enrichment of DU	DU is usually treated as an asset, saving significant U-mining needs
Production of solid RW	++	Reconversion / re-enrichment of DU	Only TENORM and VLLW levels
Gaseous RA releases	+	Application of best practices	Insignificant gaseous RA releases
Liquid RA releases	+	Idem	Insignificant liquid RA releases

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

3.3.3.7 References to Chapter 3.3.3

[3.3.3-1] https://web.evs.anl.gov/uranium/guide

[3.3.3-2] USNRC Technical Training Center, Uranium Enrichment Processes Training material

[3.3.3-3] J.R. Lamarsh & A.J. Baratta: Introduction to Nuclear Engineering, 3rd edition, Prentice-Hall, Inc., Upper Saddle River, New Jersey 07458, USA, 2001

[3.3.3-4] S. Philippe & A. Glaser: Nuclear archaeology for gaseous diffusion enrichment plants, Science & Global Security, 22: 27-49, Taylor & Francis Group, 2014

[3.3.3-5] R.L. Murray & K.E. Holbert: Isotope separators, Nuclear Energy (8th edition), 2020

[3.3.3-6] M.G. Ragheb: Isotopic separation and enrichment, January 2007 (https://www.researchgate.net/publication/242126786_ISOTOPIC_SEPARATION_AND_ENRICHMENT)

[3.3.3-7] World Nuclear Association (WNA), Uranium enrichment (www.world-nuclear.org/)

[3.3.3-8] Final Environmental Impact Statement for construction and operation of a depleted uranium hexafluoride conversion facility at the Portsmouth, Ohio site, DOE/EIS-0360, June 2004

[3.3.3-9] Use of Reprocessed Uranium: Challenges and Options, IAEA Nuclear Energy Series No. NF-T-4.4, IAEA, Vienna, 2009

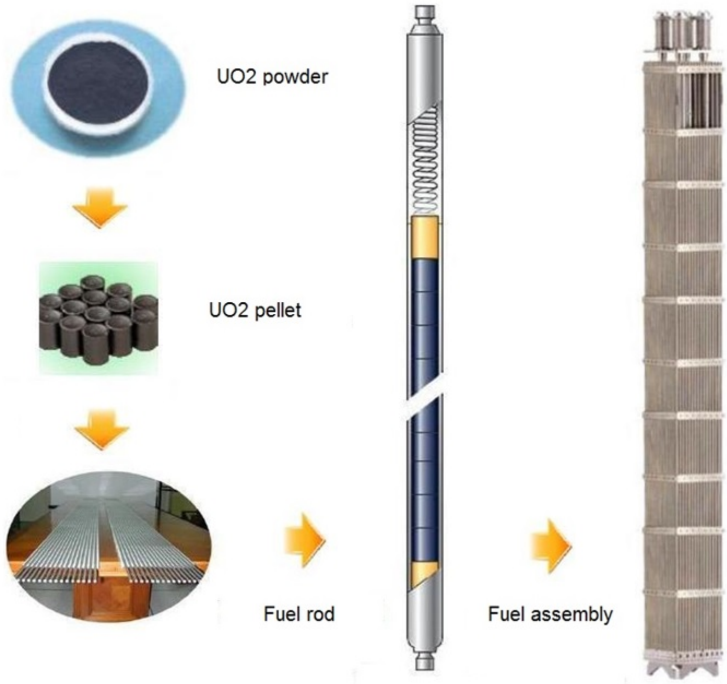
[3.3.3-10] Ch. Poinssot, et al.: Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211

3.3.4 Fabrication of UO_2 fuel – manufacturing fuel rods and fuel assemblies

3.3.4.1 Description of the underlying technologies and processes

The processes for fabrication of nuclear fuel, the fuel rods and the fuel assemblies are well established. The fabrication is performed inside closed environments (e.g. controlled pressure systems, glove boxes, etc.), and release of radioactive materials is prevented.

Figure 3.3.4-1. Main steps in the fabrication of nuclear fuel for power reactors



Source: JRC

Fuel pellets are made by mechanical compaction of enriched UO_2 powder in a press. These powders are milled before the compaction to obtain the required particle size, and to recycle the scraps from the fuel fabrication process. After pressing the pellets are sintered at high temperature (approximately 1700°C) and in reducing atmosphere (Ar/H_2) to yield a dense material with the required grain size. The pellets are then ground to the right diameter in a centreless grinder. Finally, the pellets are inspected and checked for defects, and pellets that do not conform to the specification are removed from the batch and recycled together with the waste powder from the grinding.

The ceramic UO_2 fuel pellets are enclosed in a metallic tube of a zirconium alloy, the cladding. Currently several types of alloys are used, mainly Zr-Sn (e.g. Zircaloy-2 & Zircaloy-4) and Zr-Nb (e.g. E110, M5, Zr2.5Nb, ZIRLO). These tubes are manufactured from nuclear grade zirconium, i.e. with very low hafnium content, as hafnium exhibits very high neutron absorption. The material production requires specific separation techniques as these two elements are chemically very similar. The tubes are finally filled with inert gas (helium) and hermetically sealed by welding.

The fuel pins are arranged in a quadratic or hexagonal geometry in the final fuel assemblies, in which a support structure with bottom and top nozzles and intermediate grids keeps the fuel rods in place, and assures correct positioning in the reactor core. A fuel assembly may also contain guide tubes for control rods which contain neutron absorbing material (e.g. boron, cadmium, hafnium or indium), and which are used to regulate the power of the reactor. The central position is often reserved for an instrumentation channel containing in-core detectors for measuring neutron flux along the assembly or the temperature of the coolant at the outlet of the assembly.

Figure 3.3.4-2. Illustrative views (from left to right) of the PWR, VVER and BWR fuel assemblies

Sources: www.nuclear-power.net; www.tvel.ru; www.westinghousenuclear.com/sweden

3.3.4.2 Identification of key potential impacts on the environment and human health

- The fabrication of nuclear fuel is a mechanical process, with minimal liquid waste streams, which limits the risks of dispersion into the aquatic environment. The gases of the sintering process are filtered to remove particulates before release, and only trace quantities may be released into the air. The exhaust air will also contain traces of radon – a gaseous decay product in the decay chain of the uranium isotopes.
- Since the fabrication of fuel is done at high-temperatures, the fuel production is an energy-intensive process. However, the required energy relative to the amount of energy generated from the fuel is very small.
- The collective annual public dose (normalized to the electricity production) from enrichment and fuel fabrication has been estimated as 0.003 man-Sv/(GWa), which is less than 1% of the total collective public dose resulting from the entire nuclear fuel cycle (see pages 170 and 173 of [3.3.4-1]). The public dose contribution from the fuel fabrication activities is therefore negligible.

The fuel fabrication process includes routes for the recovery and recycling of the chemical compounds and uranium. Uranium is recovered from pellets rejected in the quality controls, from precipitates in the conversion process, from filters, scrap from machining operations, and dust collection. The chemical compounds and the uranium are reincorporated in the material flows, where appropriate, for reuse and recycle. These routes include treatments to remove impurities, dissolve uranium with acids, filtrate suspended solids, extract uranium with solvents, and concentrate, and precipitate the final product. Fuel manufacturing facilities also process metal waste which is recycled. Other solid radioactive waste includes cloths, rags, decontamination residues, filters and dried slugs.

Figure 3.3.4-3. Workbench for assembling fuel assemblies in the Lingen fuel factory of Framatome (Germany)

Source: www.framatome.com

Table 3.3-1. Waste generation in fuel fabrication (per 1000 t U)

Material	Amount	Classification	Process/origin
Ammonium fluoride solution	4 000 m^3	By-product	AUC
Ammonium nitrite solution	5 000 m^3	By-product	AUC/ADU
Extraction residues	10 m^3	Material for treatment	AUC/ADU
Sludges	1 m^3	Material for treatment	AUC/ADU
Hydrogen fluoride	1 000 t	By-product	IDR
Zircaloy	1 t	Material for treatment	FA fabrication
Stainless steel	1 t	Material for treatment	FA fabrication
Miscellaneous metal scrap	40 t	Material for treatment	FA fabrication
Vent filters	100 – 200 m^3	Material for treatment	All
Mixed burnable waste	300 m^3	Material for treatment	All

Legend

AUC = ammonium uranyl carbonate process; ADU = ammonium diuranate process

IDR = Integrated Dry Route process; FA = fuel assembly

Source: [3.3.4-3]

3.3.4.3 Legal background and regulations

Fuel fabrication factories operate within the limits that are defined in the nuclear licence granted to the owner of the facility, based on the safety analysis report and the environmental impact study. The licence specifies, among others, the maximum allowable releases in alpha activity and total activity in aqueous and gaseous effluents, which are key to the potential impacts on the environment and human health. The true releases are generally well below the limits. As an example, the 2019 report for the Framatome Romans site, where around 700 metric tons of uranium are processed per year, shows that the liquid effluents in the years 2017-2019 contained 5, 7 and 11 MBq of alpha activity, respectively, compared to an authorised release of 7000

MBq (see Ref. [3.3.4-2]). The numbers for the atmospheric release of alpha-activity in these three years were 0.09, 0.08 and 0.09 MBq, respectively, compared to an authorised release of 210 MBq.

3.3.4.4 Identification of applicable means to avoid or mitigate the impacts

Nuclear installations, such as fuel fabrication plants, are subjected to periodic controls, audits and environmental monitoring. Controls and continuous improvements of processes and operational practices further reduce the potential impacts.

3.3.4.5 Evaluation and summary

The UO_2 fuel fabrication activities have an insignificant contribution to the environmental and human health impacts of the nuclear fuel cycle.

Note that Technical Screening Criteria were not developed for this activity, because this lifecycle phase does not represent a dominant contribution in any of the impact categories used in our study.

3.3.4.6 References for Chapter 3.3.4

[3.3.4-1] UNSCEAR – Sources, Effects and Risks of Ionizing Radiation, 2016 Report

[3.3.4-2] https://www.framatome.com/businessnews/liblocal/docs/3_Actualites/Dossiers/Rapport-Framatome_TSN_2019.pdf

[3.3.4-3] Edwards, C.R., Oliver, A.J. Uranium processing: A review of current methods and technology. JOM 52, 12–20 (2000). https://doi.org/10.1007/s11837-000-0181-2

3.3.5 Reprocessing of spent nuclear fuel

3.3.5.1 Description of the underlying technologies and processes

3.3.5.1.1 Composition of spent LWR[58] fuel and the motivation for reprocessing

Natural Uranium comprises about 0.7% of the fissile isotope ^{235}U, 99.3% ^{238}U and traces of ^{234}U.

The proportion of the fissile isotope present in natural uranium is not sufficient to sustain a nuclear fission chain reaction in light water reactors. The fuel for these reactors generally contains fuel pellets manufactured from uranium dioxide (UO_2), also known as uranium oxide (UOX), in which the uranium has been enriched to around 3 - 5% in the ^{235}U isotope.

Following its irradiation and power generation in a light water reactor, the fuel is removed and cooled in a spent fuel storage pool for a period of time. A fuel element typically spends around 3 - 4 years inside the reactor and several more years, depending on the spent fuel management strategy, in a spent fuel pool.

Following its discharge from the reactor, a fuel element contains typically[59] [3.3.5-1]:

- ~ 95.5% Uranium (U)
- ~ 1% Plutonium (Pu)
- ~ 3.4% fission products (FP)
- < 0.1% other transuranic elements (known as Minor Actinides – mainly Neptunium - Np, Americium - Am and Curium - Cm)

As the production of power in the reactor results from fission, the uranium in the spent fuel is depleted in the fissile ^{235}U isotope. The residual content of ^{235}U is typically less than 1% and close to that of natural uranium [3.3.5-1, 2], the exact value depending on the initial fuel enrichment and the amount of power produced by the fuel during its stay in the reactor (the burnup rate)[60].

[58] The discussion is limited to spent fuel from light water reactors as these account for almost 90% of the global nuclear installed capacity. Similar considerations apply to the fuel from some other reactor types. Spent fuel from PHWRs like the CANDU reactors, which use natural (un-enriched) uranium fuel, are less attractive for reprocessing due to the low proportion of ^{235}U and Pu present.

[59] These are typical values, but may vary depending a number of parameters including the initial fuel composition, irradiation conditions and length of time in the reactor.

[60] Note also that about one-third of the fission events in a thermal power reactor is due to fission of ^{239}Pu, produced through neutron capture by ^{238}U.

On its discharge from the reactor, the spent fuel generates a significant amount of residual heat and contains a broad range of fission products (FP), which are the main contributors to a high level of radioactivity. During several years of storage in the spent fuel pool the decay of the more highly radioactive FPs, having short half-lives, results in a considerable reduction in both the residual heat of the fuel and its radioactivity level.

3.3.5.1.2 Open and closed fuel cycles

Techniques for reprocessing of irradiated uranium were developed in the 1940s for military purposes. Today, reprocessing is a mature technology that has been practised at industrial scale in the civil nuclear industry for four decades.

The main motivation for reprocessing spent fuel is to recover the uranium and plutonium for reuse, as these materials contain significant energy potential. Reprocessed uranium (RepU) can be re-enriched and recycled as UOX fuel. Recovered plutonium is incorporated into mixed-oxide (MOX – U+Pu oxide) fuel which may be recycled once[61] in current (thermal neutron) reactors.

The ability to reprocess spent fuel and recover the useful components gives rise to a number of options for the nuclear fuel cycle:

— **The open fuel cycle**[62], in which spent UOX fuel elements, after cooling for some years in an interim storage facility, are encapsulated in a disposal container for disposal in a geological repository. No reprocessing of fuel takes place.

— **The partially closed fuel cycle**[63], in which spent UOX fuel elements are reprocessed and recovered Pu and U are reused in MOX fuel, which is recycled once in current reactors. The spent MOX fuel elements then follow the same processes as the spent UOX fuel in the open cycle, i.e. interim storage and encapsulation in a disposal container for disposal in a geological repository.

— **The fully closed fuel cycle**, in which spent fuel is repeatedly reprocessed and recycled in nuclear power reactors. For the fully closed fuel cycle, advanced reactors operating with a fast neutron spectrum (fast neutron reactors or fast reactors) are required[64].

Today, after recycling once in current reactors, spent MOX fuel elements go into interim storage, after which there are two options: direct disposal as described above under the partially closed fuel cycle, or further reprocessing[65].

The main fuel cycle options are summarised in Table 3.3.5-1, from [3.3.5-3]. Partitioning and transmutation mentioned under the fourth option in Table 3.3.5-1 is briefly discussed in Chapter 3.3.5.1.3.

The partially closed fuel cycle as currently practised allows a saving in the requirement for fresh natural uranium and therefore in the associated mining activities. The quantification of this saving varies among studies. According to [3.3.5-3], for the partially closed fuel cycle with single recycling of plutonium in thermal neutron reactors as practised today, about 11% more electricity is produced per metric ton of natural uranium. If the reprocessed uranium is also recycled as nuclear fuel, an additional 10% electricity can be generated per metric ton of natural uranium. According to [3.3.5-2], savings of up to 30% in natural uranium requirements can be achieved with recycling of uranium and plutonium in current reactors.

In the partially closed fuel cycle, the majority of the RepU is not recycled. Its main potential can be realised by future fast neutron reactors in which the predominantly fertile content of the RepU can be transformed into fissile isotopes and burned in the same reactors. If fast neutron reactors are used with full recycling of plutonium and uranium, current uranium reserves would permit at least 5 000 years of operation at present global levels of nuclear power generation [3.3.5-3]. Uranium mining in this case will become much less significant.

[61] A second recycling in LWRs is feasible, but multiple recycling of plutonium in present day LWRs is limited as the fraction of fissile plutonium isotopes decreases at each recycling.

[62] Also referred to as the once-through cycle; see chapter 3.3.1.

[63] Also referred to as the twice-through cycle; see chapter 3.3.1.

[64] In Europe, prototype and commercial scale demonstration fast neutron reactors have been developed, built and operated, but fast reactors are not yet commercially available. They remain under development for future deployment.

[65] For a possible second recycling in LWRs or as part of a fully closed fuel cycle in the future.

Table 3.3.5-1. Characteristics of the open fuel cycle and main different levels of closing the fuel cycle

Type of cycle	Type of reactor	Treatment of spent fuel	Re-use of spent fuel material	Waste requiring geological disposal
Open fuel cycle	thermal neutron reactors	Storage, encapsulation and disposal	none	all the spent fuel after one cycle
Partially closed cycle *(one cycle of extraction of uranium and plutonium)*	thermal neutron reactors	spent fuel is reprocessed for extraction of uranium and plutonium	first cycle: re-use of plutonium and depleted uranium for MOX fuel, re-use of reprocessed uranium	conditioned high level waste and compacted fuel cladding spent MOX fuel
		spent recycled fuel (MOX fuel) is stored for later disposal	no second cycle	waste from reprocessing and fuel fabrication
Fully closed cycle *(repeated extraction of uranium and plutonium)*	fast neutron reactors and thermal neutron reactors	repeated reprocessing (also of spent recycled fuel) for extraction of plutonium and uranium	plutonium and uranium from different re-use cycles and depleted uranium are mixed to allow fabrication of recycled fuel	conditioned high level waste and compacted fuel cladding
				waste from reprocessing and fuel fabrication
Fully closed cycle + Partitioning and Transmutation *(repeated cycles of partitioning, followed by transmutation of long-lived residues)*	fast neutron reactors or waste burners	repeated reprocessing including partitioning	full re-use of plutonium and uranium 'burning' of long-lived residues (transmutation)	residual conditioned high level waste and compacted fuel cladding
				waste from reprocessing and fuel fabrication

Source: [3.3.5-3]

3.3.5.1.3 Impact of reprocessing on the generation of radioactive waste

In addition to the primary objective of recovering valuable energy resources, reprocessing brings benefits in terms of the quantities, heat load and radiotoxicity of radioactive wastes requiring geological disposal.

As the uranium and plutonium are recovered from the spent fuel, only the fission products and minor actinides, representing about 3.5% of the spent fuel, remains as high-level waste (HLW). This is generally immobilised in glass blocks which are capable of providing for confinement of the waste over very long time spans. Additional long-lived, intermediate level waste (ILW-LL) is also generated during reprocessing, comprising the structural materials of the fuel elements, like cladding, end caps and so on, as well as some technological process wastes.

As mentioned in Chapter 3.3.5.1.2 above, in the open fuel cycle, spent fuel elements are encapsulated in a disposal container for disposal in a geological repository.

In comparison to the open cycle, a partially closed cycle is not expected to give a major reduction of the footprint of a geological repository, as there will be a need to also dispose of the spent recycled MOX fuel elements. For a fully closed cycle, with total recycling of the plutonium and uranium, the needed repository size for the high level waste is reduced by 40% [3.3.5-3].

A process complementary to the fully closed cycle is 'partitioning and transmutation' in which not only plutonium and uranium, but also the other long-lived radiotoxic residues (the minor actinides and some of the fission products) are separated and extracted (i.e. 'partitioning'). Their transformation into short-lived products (i.e. 'transmutation') would generate waste that decays over much shorter timeframes. This would be done by adapted fast neutron reactors or in dedicated 'waste burning' reactors. Development of partitioning and transmutation is currently only at an experimental scale.

If, in addition to the fully closed fuel cycle, partitioning and transmutation is applied, the high level waste volumes could be reduced even further. Moreover, the vitrified waste will contain mainly short-lived components (while the long-lived components are recycled or consumed), thus producing less heat [3.3.5-4]. This would contribute to further reduce the footprint of the required geological disposal facility due to the lower waste volumes and a closer packing of the waste.

3.3.5.1.4 Impact of reprocessing on the lifecycle costs of nuclear energy

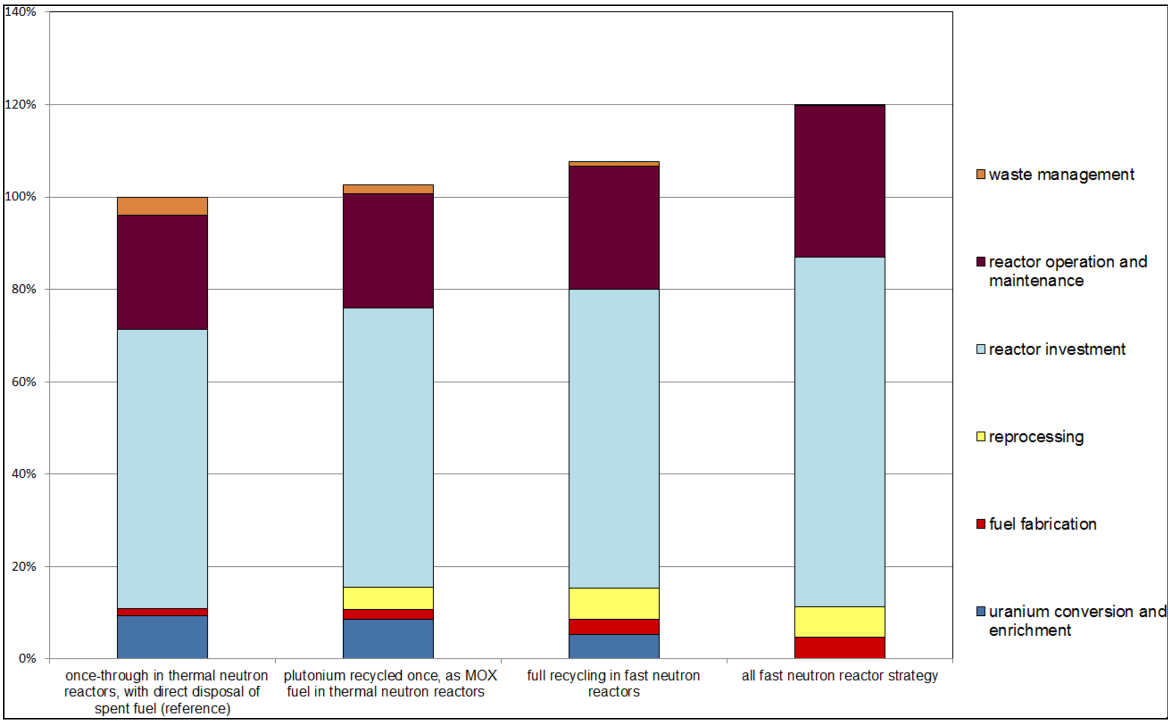
When addressing the economics of nuclear energy it has to be kept in mind that the largest component of the cost is the capital cost of the nuclear power plants; most studies agree that the total fuel cycle expenditures (including front end and back end) typically account for about 10 to 20% of the overall energy production costs [3.3.5-3].

There are however uncertainties associated with the cost estimates and the elaboration of the respective financing schemes.

The OECD Nuclear Energy Agency compared a variety of open, partially closed and closed cycles in 2006 [3.3.5-5]. The results of this study, summarised in Figure 3.3.5-1 for some of the options considered, indicate a maximum increase in generating costs of 20% compared to the open cycle. The uncertainties, however, are in excess of this difference.

Current cost estimates generally favour the open fuel cycle, but it is rather marginal compared to the twice-through cycle in which Pu is recycled once in MOX fuel in current reactors. Closing the fuel cycle reduces the costs for the front end of the cycle (less uranium acquisition, processing and enrichment) but the savings do not totally balance the costs of the additional steps and facilities as mentioned in the previous sections. In this context, the assumed future uranium price is important and the impact of closing the fuel cycle is frequently presented in the form of a uranium break-even price.

Figure 3.3.5-1. Relative cost estimates for alternative types of fuel cycles[66]



Sources: [3.3.5-3] & [3.3.5-5]

Besides the (variable) uranium price, significant uncertainties affect the cost estimates for both open and closed cycles.

Closing of the fuel cycle is clearly a strategy having long term implications so that not only present uranium prices are important for the economics, but also projections of their future evolution.

Furthermore, while economic considerations are mainly based on free-market uranium prices that reflect the actual and expected abundance of uranium mainly for present and near future generations, sustainability considerations would tend towards preserving natural resources also for future generations.

66 It should be noted that all fuel cycle steps were included in the study but are not indicated separately in the figure. Some steps are aggregated and incorporated into previous or subsequent steps.

3.3.5.1.5 Impact of reprocessing on non-proliferation

The use of nuclear materials for solely civil purposes is controlled worldwide by the application of IAEA safeguards, acting under the Non-Proliferation Treaty[67] (see also Annex 1, Chapter 2.2). Within the EU, the control is complemented by the Euratom safeguards inspections. The Euratom inspectors verify the declared uses of nuclear materials at facility level, while the IAEA mandate also extends to verifying the absence of undeclared activities and diversion. In addition, the material and the facilities have to be secured from non-state sabotage or theft through physical protection measures.

While for the front-end of the fuel cycle the control is dedicated to uranium and the enrichment process, the safeguarding and physical protection measures on the back-end of the cycle are concentrated on plutonium, which is the main fissile component of spent fuel. It is of note that the plutonium discharged from commercial nuclear power plants is of poor quality in respect of fabrication of efficient atomic weapons due to its isotopic composition[68]. The plutonium is nevertheless submitted to all applicable international control measures; the reasoning is that even low grade materials could be of interest.

A non-proliferation benefit of the open cycle in the short term is that the sensitive material, the plutonium, is not separated from the spent fuel. Moreover, the spent fuel is, to a certain extent, "self-protecting" over the first 100 years after discharge from a reactor. The radiation levels are so high[69] that it is practically impossible to manipulate fuel elements without specialised equipment. Nevertheless, the fuel assemblies in interim storage facilities are submitted to safeguards and physical protection measures to ensure that they are not diverted and that they remain intact.

For the closed cycle options, the short term safeguarding of fuel recycling facilities and their protection is much more demanding, especially from the moment the plutonium is separated from the rest of the fuel. Safeguards concepts for the reprocessing facilities at Sellafield and La Hague were developed in the 1980s, and implementation in the late 1990s included the installation of Euratom safeguards laboratories at the facility sites [3.3.5-6]. More recently, a conceptually similar on-site laboratory is being operated at the Rokkasho facility in Japan. Increased surveillance and verifications, also by measurement of samples from the process flow, have to be implemented. In order to reduce the quantity of separated plutonium, alternative reprocessing techniques are under development, where the plutonium is not extracted separately from the spent fuel, but together with the uranium.

However, in the case of a fully closed cycle, essentially all fissile material is, in the end, re-used and consumed, which is beneficial in respect of the long term proliferation risk. With full recycling, also the front-end uranium enrichment process, which is particularly sensitive, is reduced to a minimum. And at the backend, geological repositories are mainly limited to the disposal of high-level waste, which will not require long term safeguards controls.

In the case of a partially closed cycle, in which fuel is recycled once and spent recycled (MOX) fuel is considered a waste to be disposed of, safeguards and physical protection considerations for the recycling is similar to the closed cycle and for disposal similar to the open cycle, except that the plutonium composition of spent MOX fuel is degraded and it is therefore less sensitive for proliferation.

3.3.5.1.6 Reprocessing operations

In Europe, nuclear fuel reprocessing is carried out at two sites at La Hague in France and at Sellafield in the United Kingdom, although closure of the Thermal Oxide Reprocessing Plant (THORP) at the end of 2018 brought an end to reprocessing of UOX fuel at Sellafield for currently operating reactors. The Magnox fuel reprocessing plant, for reprocessing the remaining fuel from the shut-down Magnox reactors, remains the only operational reprocessing facility at the site and is scheduled to close in 2021. Facilities for reprocessing spent nuclear fuel are also operated in India, Japan and Russia. In the USA, three civil reprocessing plants have been built, but a 1977 change in government policy, which ruled out all US civilian reprocessing as part of US non-proliferation policy, put an end to reprocessing operations [3.3.5-2]. China is pursuing the construction of a civil reprocessing plant [3.3.5-7].

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[67] "Treaty on the non-proliferation of nuclear weapons", IAEA INFCIRC 140, 1970.

[68] In comparison with "weapons grade plutonium", the plutonium discharged from most of the civil reactors generates a relatively high neutron radiation and generates heat, linked to its composition.

[69] For most of the reactors, the radiation of the fuel remains very high during about 100 years after discharge, at a level that would be lethal for operators in a few hours; for some type of reactors (e.g. CANDU Heavy Water Reactor) the discharged fuel will only exceed that radiation level for a few years.

Although a number of processes have been developed for reprocessing of nuclear fuel, the focus in this report is on the current large scale industrial process called PUREX (plutonium uranium extraction) and its environmental impacts. PUREX is a hydrometallurgical process used in all commercial reprocessing plants and involves several stages.

Following dismantling of the fuel elements, the fuel pin bundles are chopped into sections a few centimetres long and then dissolved in an aqueous solution of concentrated nitric acid.

The dissolved uranium and plutonium are then separated in the aqueous nitric acid stream from the fission products and minor actinides by a solvent extraction process, using tributyl phosphate dissolved in kerosene or dodecane. The uranium is then separated from the plutonium in further chemical processes that finally produce UO_2 and PuO_2 in powder form.

After the separation and extraction of the U and Pu, the minor actinides and fission products from the spent fuel remain in solution. This solution is concentrated and vitrified in preparation for final disposal. In a future development of the process, further separation of the minor actinides may take place as part of a partitioning and transmutation strategy (see Chapter 3.3.5.1.3).

Fuel cladding and other structural elements in Zircaloy and stainless steel are not dissolved in the nitric acid and can be separated out. These are compacted in steel drums. In addition to these structural elements, small amounts of other elements that are resistant to the dissolution in nitric acid are recovered by settling, centrifuging or by filtration and conditioned for disposal [3.3.5-1,8]. In Europe, these solids are categorized as intermediate-level wastes (ILW) that require deep geologic disposal.

Liquid waste streams from the reprocessing operations are subject to filtration, evaporation, and other treatments to reduce the residual radioactive substances. However, a small fraction of the radioactivity originally present in the used fuel is discharged from the plant in the liquid effluent stream.

Shearing of the fuel pins, as well as dissolution of the fuel in the nitric acid, release gaseous fission products including the noble gases krypton (Kr) and xenon (Xe), as well as iodine (I)[70] and carbon-14 (^{14}C) in the form of CO_2. The gas stream is scrubbed prior to release and ensures that statutory emission limits are respected.

The noble gases krypton and xenon are released to the environment. They do not contribute significantly to the radiation dose of the workers or the public. The total radiation doses to members of the public from reprocessing operations in Europe are very low, as will be shown in Chapter 3.3.5.2.1, below (see in particular Figures 3.3.5-6 and 3.3.5-7).

3.3.5.2 Identification of key potential impacts on the environment and human health

3.3.5.2.1 Environmental impact of fuel cycle options

A closed or partially closed fuel cycle requires the construction, operation and eventual decommissioning of reprocessing plants, which are not required in the case of the open fuel cycle. This will bring additional environmental impacts to the nuclear energy lifecycle.

On the other hand, as mentioned in Chapter 3.3.5.1.2, implementation of the partially-closed fuel cycle, as practised today, can bring savings in the requirements for fresh natural uranium of 20 – 30%, and much more significant savings can be realised in the case of a fully closed fuel cycle. Consequently, the environmental impact from the mining stage[71] will be reduced. Moreover, reductions in the volumes, thermal loads and radiotoxicity of the radioactive waste requiring final disposal, that are associated with the different options of the closed fuel cycle discussed in Chapter 3.3.5.1.3, reduce the required excavated volumes of final repositories which also bring reductions in environmental impacts.

Poinssot et al [3.3.5-9] provide data on the environmental footprint for the existing French reactor fleet and fuel cycle, assuming both once-through and twice-through fuel cycles. The once-through fuel cycle does not involve reprocessing. The twice-through cycle included, at that time, reprocessing and recycling of plutonium in MOX fuel making up one-third of the fuel elements in the core of 22 reactor units representing 31.2% of the installed capacity of the French fleet. Re-enriched reprocessed uranium was also utilised. The resulting

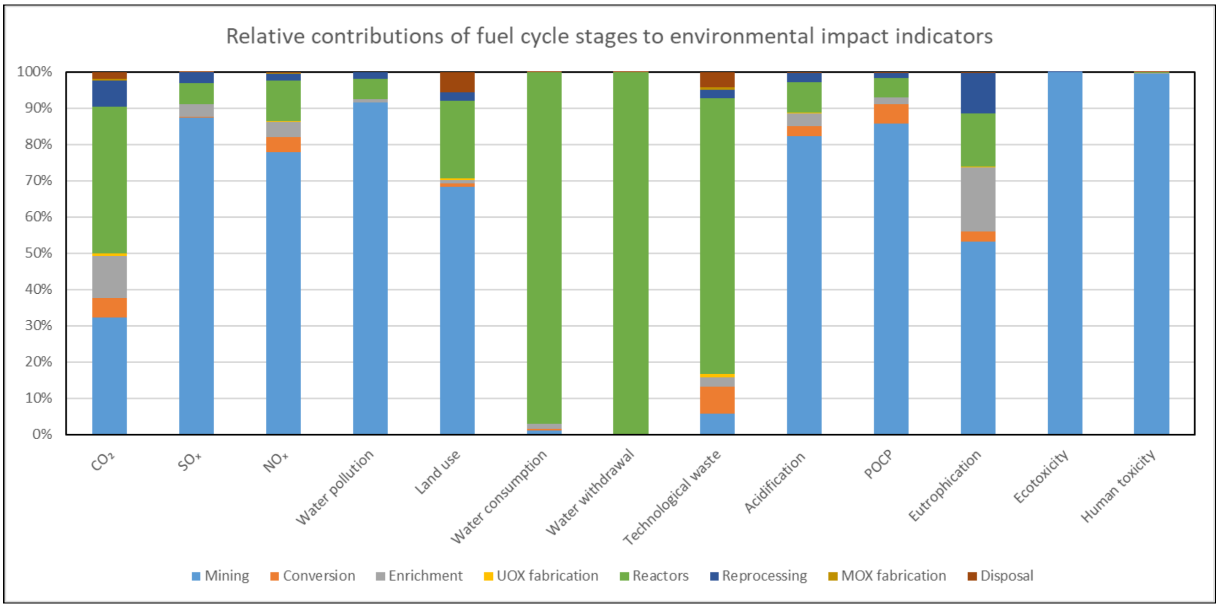
70 Isotopes ^{127}I, which is stable, and ^{129}I, which has a very long half-life (15.7 million years). The isotope ^{131}I is not relevant as it has a half-life of 8 days and disappears during storage of the fuel prior to reprocessing.

71 And to a lesser extent also from the uranium conversion and enrichment stages, as recycling of plutonium in MOX fuel will reduce the need for enriched uranium, whereas the use of re-enriched reprocessed uranium still utilises the conversion and enrichment stages.

saving on the requirements for fresh natural uranium was estimated at about 14.5% and the saving on conversion and enrichment at about 9%.

Figure 3.3.5-2 shows the relative contributions from the different steps of the fuel cycle to the full range of non-radiological environmental impact indicators calculated by Poinssot et al for the current French twice-through cycle. It can be seen that, with the exception of water withdrawal, water consumption and technological waste, the front end of the fuel cycle (mining, conversion, enrichment and UOX fabrication), and mining in particular, is the dominant contributor to the other ten indicators. Consequently, reducing these operations can bring significant reductions to the environmental impact of the nuclear energy lifecycle.

Figure 3.3.5-2. Relative contribution of each step of the fuel cycle to environmental impact indicators



Data from Poinssot et al [3.3.5-9]

To investigate the impact of closing the fuel cycle, Poinssot et al also calculated lifecycle impacts for the French fleet assuming an open fuel cycle, without reprocessing. The ratio of the impact indicators for the twice-through versus open fuel cycle resulting from this study are presented in Figure 3.3.5-3. This shows that the twice-through cycle, as practised in France, always has a lower non-radiological environmental impact than the open fuel cycle.

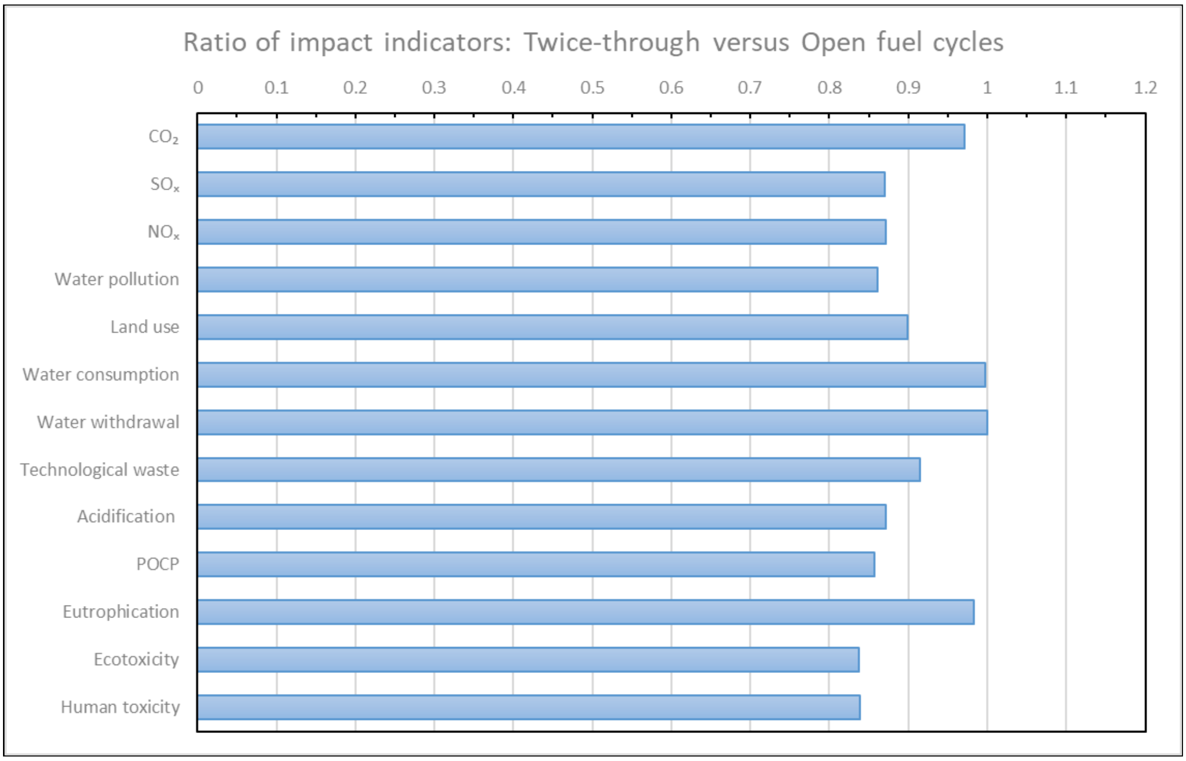
The situation is slightly different when it comes to radiological indicators. Figure 3.3.5-4 shows the ratio of lifecycle generated volumes of different categories of solid radioactive waste as well as the ratio of radioactive releases to the environment for the French twice-through versus open fuel cycle.

With regard to solid radioactive waste, the volume of very low-level waste (VLLW) is lower for the twice-through cycle, as 99% of these wastes are generated in the mining stage.

Short-lived, low and intermediate-level waste (LILW-SL), in the case of the twice-through cycle, is mainly from reactor operation (75%) and reprocessing (19%), with 65% of the total coming from dismantling of the installations at the end of life. With little contribution from mining, the total volume of LILW-SL is slightly higher under the twice-through cycle than the open fuel cycle.

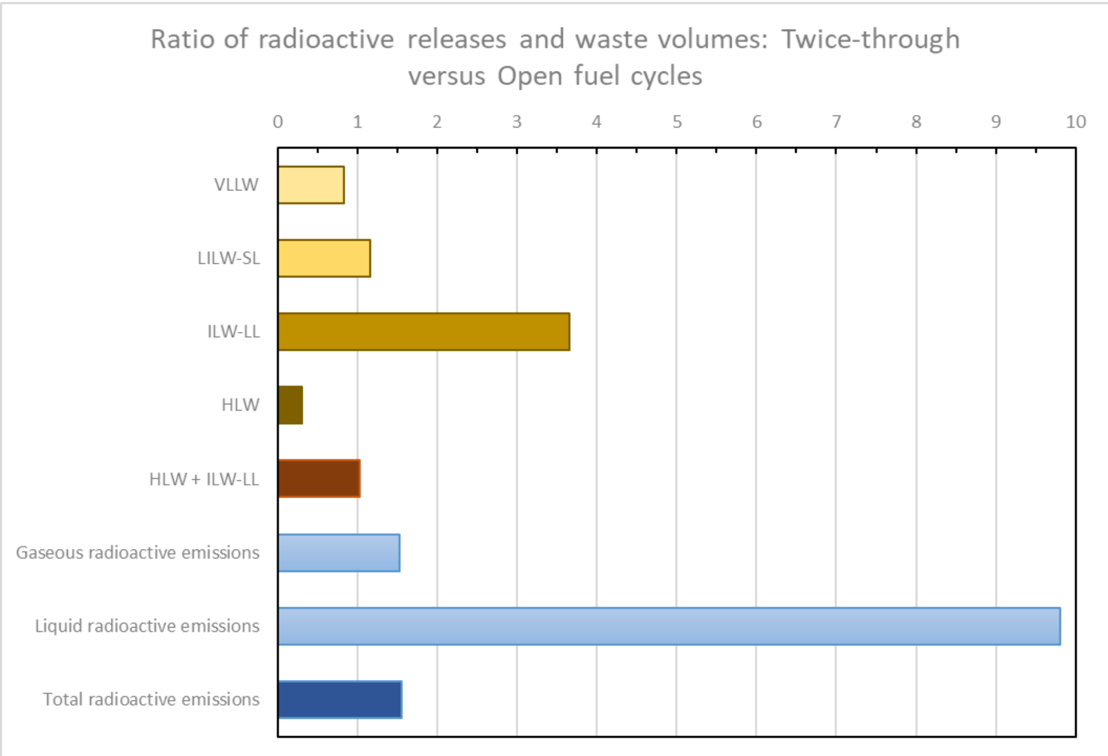
Long-lived, intermediate-level waste (ILW-LL), in the case of the twice-through cycle, is again mainly from reactor operation (25%) and reprocessing (62%), but in this case reprocessing dominates. Dismantling of the installations at the end of life contributes about 42% of the total. The additional process waste from reprocessing operations includes the structural materials of the fuel elements, cladding, end caps, etc. With little contribution again from mining and a large contribution from reprocessing, the total volume of ILW-LL is higher under the twice-through cycle than the open fuel cycle.

Figure 3.3.5-3. – Ratio of non-radioactive impact indicators corresponding to the total LC of nuclear energy for the twice-through versus the open fuel cycles



Data from [3.3.5-9]

Figure 3.3.5-4. – Ratio of radioactive impact indicators corresponding to the total LC of nuclear energy for the twice-through versus the open fuel cycles



Data from [3.3.5-9]

High-level waste (HLW), in the case of the twice-through cycle, is only produced by the spent fuel reprocessing operations. It includes the fission products and minor actinides which are vitrified and stored in canisters for final disposal. Compared to the open cycle, in which fuel elements are encapsulated for disposal without reprocessing, the total volume of HLW is reduced considerably[72].

In France, HLW and ILW-LL are intended for disposal in geological repositories. The combined volume of these two categories of waste for the open and twice-through fuel cycles are as follows:

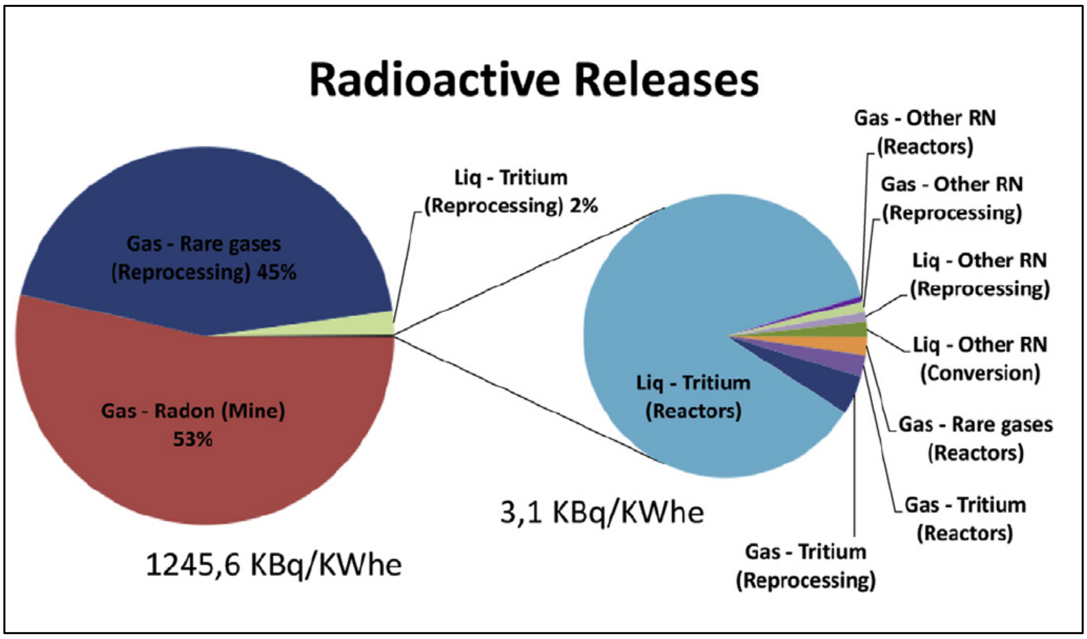
— Open fuel cycle: **1.49 m^3/TWh_e** (0.32 m^3/TWh_e ILW-LL & 1.17 m^3/TWh_e HLW)

— Twice-through: **1.53 m^3/TWh_e** (1.18 m^3/TWh_e ILW-LL & 0.36 m^3/TWh_e HLW)

It can be seen that the total volume of waste to be disposed of in a geological repository is not very different for the two fuel cycle options. However, HLW requires a greater excavated volume and surface area of geological repository than ILW-LL and also contributes more to the long-term radiotoxicity. As a result, according to [3.3.5-9], the estimated repository volume is 3.4 times higher for the open fuel cycle compared to the twice-through cycle.

With regard to radioactive releases, both gaseous and liquid releases are greater for the twice-through cycle, although it will be shown below that this has a very minor impact on radiation doses to members of the public. Gaseous releases are about 53% higher in the twice-through cycle, whereas liquid releases are almost a factor of 10 greater. However, as liquid releases make up only a very small proportion of total radioactive releases, the 10-fold increase has only a small impact on the total radioactive releases, which are around 56% higher for the twice-through cycle. Poinssot et al [3.3.5-9] showed the distribution of the radioactive releases to the environment (in kBq/kWh_e) from the different stages of the nuclear fuel cycle and for different nuclide groups (see Figure 3.3.5-5, reproduced from [3.3.5-9]).

Figure 3.3.5-5. – Radioactive releases from the nuclear energy twice-through lifecycle



Source: [3.3.5-9]

The main contributors to the lifecycle radioactive releases for the twice-through cycle are:

— Radon gas from uranium mining: 53.4%

— Noble gases (mainly krypton) from reprocessing: 44.4%

— Liquid tritium from reprocessing: 2%

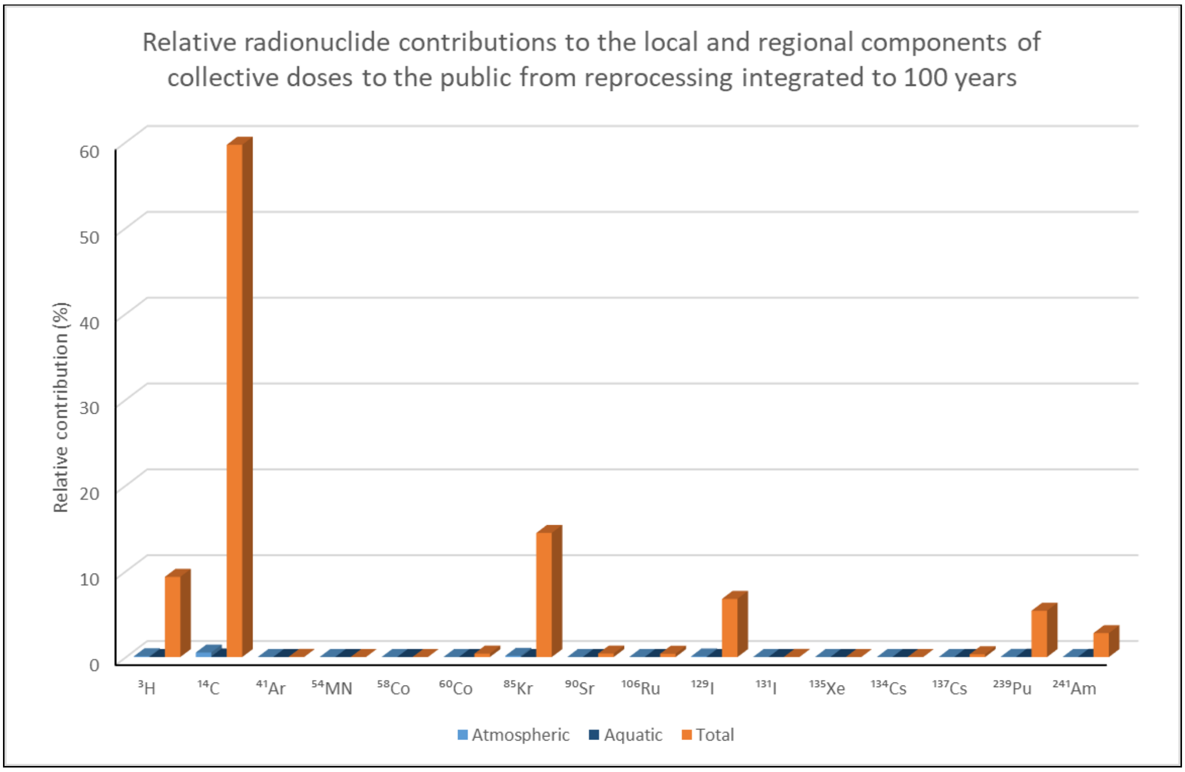
[72] It should be noted that in this French case, spent MOX fuel is not sent for disposal, as it would be in a partially-closed fuel cycle. It is sent to intermediate storage for possible future use in a fully closed fuel cycle. If spent MOX fuel would be encapsulated for geological disposal, the reduction in the volume of HLW would not be so significant.

The other approximately 0.2% comprises mainly (>80%) liquid tritium releases from nuclear reactor operation, with the remainder divided between tritium gas released from reprocessing and reactor operation, noble gas released from reactor operation, and carbon 14 and other radionuclides released during reprocessing, reactor operation and uranium conversion.

However, the effects of radioactive releases in terms of doses to members of the public vary considerably for different radionuclides and different release pathways (releases to atmosphere, freshwater or marine environments). UNSCEAR [3.3.5-10] has developed a methodology for assessing the radiation exposures of the general public[73] (annual doses in Sv/year or man.Sv/year) from discharges of radionuclides to the environment (in Bq/s), based on the use of dose calculation factors for unit discharge rates of radionuclides to atmosphere, to the different freshwater environments (small rivers and large rivers/lakes) and to a marine environment.

Figure 3.3.5-6 shows the relative contributions to the collective dose to local and regional populations from the different radionuclides released to the environment during reprocessing. The data is from UNSCEAR [3.3.5-10] and is based on release data for the la Hague reprocessing plant in France for the year 2010.

Figure 3.3.5-6. – Relative radionuclide contributions to the local and regional components of collective doses to the public from reprocessing



Data from [3.3.5-10]

It can be seen that while ^{85}Kr is responsible for almost 90% of radiological releases in (kBq/kWh$_e$) from the reprocessing stage, it contributes less than 15 % to the dose (in Sv) to the public. Carbon-14 is released in much smaller quantities, but contributes more than 50 % to the public dose. The differences are related to the chemical properties of the isotopes and to their decay characteristics.

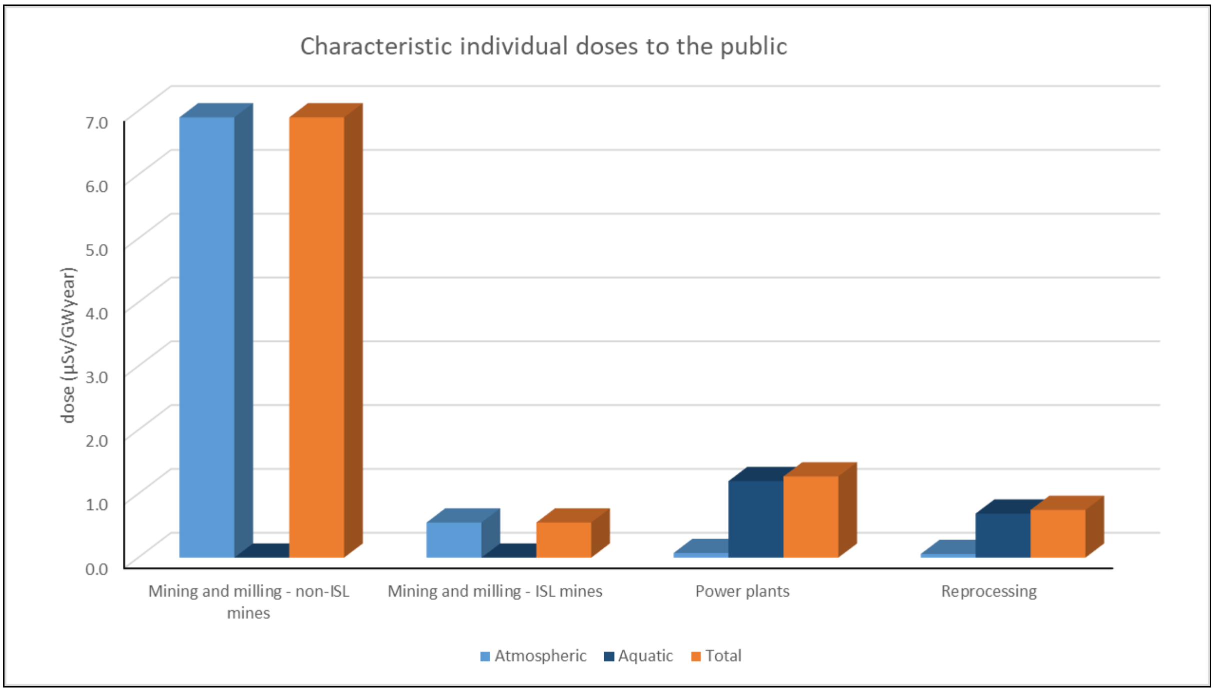
UNSCEAR also calculated public doses per unit of generated electricity from the mining, reactor operation and reprocessing stages of the nuclear energy lifecycle [3.3.5-10]. These are shown in Figure 3.3.5-7.

The dominant contributor is conventional uranium mining. However, mining performed by the in-situ leaching (ISL) technique results in a strong reduction in radiological emissions and doses to the public, as was already pointed out in Chapter 3.3.1. The dose to the public from reprocessing is seen to be less than the dose due to the operation of power plants. Carbon-14 discharges are an important contributor to the dose to the public

[73] Doses to individuals as well as collective doses to population groups.

from nuclear power plants, as they are for the reprocessing stage [3.3.5-10]. In all cases, the doses are very low, at just a few μSv/year per GW of electricity generation. The legal dose limit for members of the public in the EU from all authorised activities is 1 mSv/year, while the average dose to the public from natural background radiation and medical uses of radiation amount to about 3 mSv/year (see Chapter 3.4).

Figure 3.3.5-7. – Summary of characteristic individual[74] doses to the public in Europe normalized to electricity generated in 2010 for mining and milling, electricity generation from nuclear power reactors, and fuel reprocessing (μSv/GW.year)



Data from [3.3.5-10]

The modern European plants are able to contain their environmental effluents within strict regulatory limits, and as shown above the resulting doses to members of the public are very low. However, this was not always the case in the past. During the 1970s, radioactive discharges from the reprocessing facilities, particularly those at Sellafield, were relatively high.

In 2005, the IAEA reported on the results of worldwide marine radioactivity studies [3.3.5-11]. It noted that the authorized water-borne radioactive discharges to the Irish Sea from Sellafield[75], became measurable in most parts of the North East Atlantic and also in the Arctic Ocean. After the mid-1970s, Sellafield began to substantially reduce the discharges and the impact of the reductions on the measured levels of ^{137}Cs in surface water can be seen in Figure 3.3.5-8[76], which shows that substantial improvements had already been achieved by the mid-1990s. Signing of the OSPAR[77] convention in 1998 gave further impetus to progressive and substantial reductions of discharges from the reprocessing plants.

In 2018, the discharges at Sellafield were very substantially lower (for ^{137}Cs and ^{90}Sr, to approximately 0.1% and 0.2% respectively of their peaks in the 1970s). According to the Sellafield Limited 2018 annual environment monitoring report [3.3.5-12], all discharges in 2018 were well within authorised discharge limits, which are set taking into account the relevant parts of the Euratom Basic Safety Standards Directive. The estimated individual annual dose to members of the critical groups of the local population in 2018 was about 0.1 mSv (compared to 2.5 to 3 mSv/year in the 1970s and 1980s [3.3.5-13]). The marine pathway contributed 89 μSv, equally shared between seafood consumption and external radiation from beach occupancy. The estimated dose through the terrestrial pathway was 12 μSv, the main contributions coming from consumption

[74] The characteristic individuals are those living 5 km from the points of discharge with behaviour indicative of the majority of people living the area.

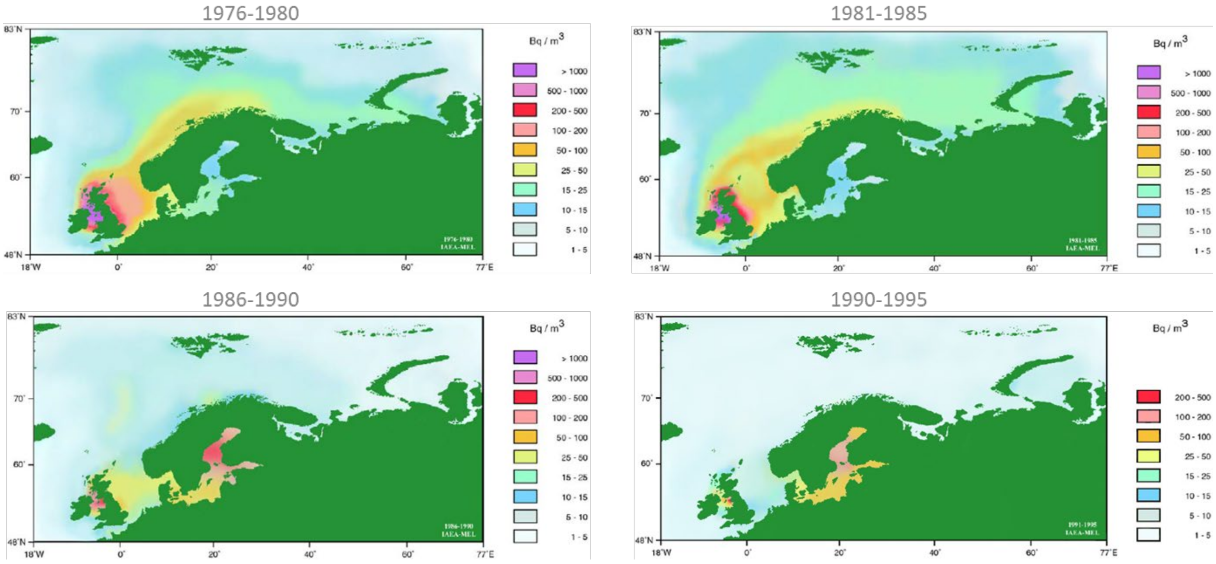
[75] First of all of ^{137}Cs, but also other radionuclides, notably plutonium isotopes, americium and technetium.

[76] Note: The increased levels of ^{137}Cs in the Baltic sea after 1986 are due to the accident at Chernobyl.

[77] The Oslo/Paris (OSPAR) convention: Convention for the Protection of the Marine Environment of the North-East Atlantic, see https://www.ospar.org/convention.

of terrestrial foodstuff (4.4 μSv) and direct radiation from the plants on site (3.9 μSv). A significant part of the estimated total dose was due to radionuclides resulting from historic rather than 2018 discharges.

Figure 3.3.5-8. ^{137}Cs in surface water of European seas



Source: [3.3.5-11]

Similarly, the 2019 annual environmental monitoring report for the La Hague site [3.3.5-14], shows that all discharges in 2017, 2018 and 2019 were well within the limits authorised by the French regulations. The calculated annual dose to the most exposed group of the local population as a result of the 2019 discharges to the environment was 0.0142 mSv, 96% of which was due to the gaseous discharges. This is less than 1.5% of the limit of 1 mSv/year set by the French regulations for the public dose from all authorised activities, and less than 0.5% of the average dose from natural sources of radiation to the French population. The most exposed group comprised local farmers living close to the site, and downwind according to the dominant wind direction, and consuming local agricultural products. The most exposed group of the population with regard to the liquid discharges to the sea comprised professional fishermen living at the coast close to the site and consuming the local seafood. The annual dose in this case was calculated to be 0.0067 mSv, 52% due to the liquid discharges and 48% due to the gaseous discharges.

The conclusions of the IAEA report on the results of worldwide marine radioactivity studies [3.3.5-11] are that although the ocean contains the majority of the anthropogenic radionuclides released into the environment, the radiological impact of this contamination is low. Radiation doses from naturally-occurring radionuclides in the marine environment (e.g. ^{210}Po) are on the average two orders of magnitude higher.

The results confirm that the dominant source of anthropogenic radionuclides in the marine environment is global fallout. The total ^{137}Cs input from global fallout was estimated to be 311 PBq for the Pacific Ocean, 201 PBq for the Atlantic Ocean, 84 PBq for the Indian Ocean and 7.4 PBq for the Arctic Ocean. For comparison, about 40 PBq of ^{137}Cs was released to the marine environment from the Sellafield and Cap de la Hague reprocessing plants, the majority in the 1970s and early 1980s.

The Chernobyl accident contributed about 16 PBq of ^{137}Cs to the sea, mainly the Baltic and Black Seas. The worldwide average concentration due to global fallout is about 2 Bq/m^3.

Changes in radionuclide concentrations in water profiles with time in the North Atlantic and Pacific Oceans were also studied. A clear decrease of radionuclide concentrations in surface water was observed due to transport of radionuclides to medium water depths.

3.3.5.2.2 Impact of reprocessing on the waste streams

Reprocessing aims at the separation of valuable energy resources (uranium and plutonium) to recycle them, and at the reduction of high level waste (HLW) volume and radiotoxicity prior to final disposal. In Europe, the only reprocessing process actually applied at industrial level is the PUREX process in which U and Pu (~97% of the spent fuel) are recovered. Both U and Pu may be recycled, in which case U goes back to a conversion plant, prior to a new enrichment, and Pu to a dedicated Mixed Oxide (MOX) nuclear fuel fabrication plant.

The impact of reprocessing on the different waste streams can be summarised as follows:

Gaseous: during the shearing and dissolution of the fuel rods, gaseous radionuclides are released in the off-gases. The biggest part are noble gases (fission products in the spent fuel) which are released into the atmosphere in a controlled manner, respecting authorisation and regulation thresholds. Their environmental and human potential impact is negligible as they are chemically inert and do not interact with biological molecules. Other off-gases (e.g. iodine) having a potential environmental and/or radiological impact can be trapped in scrubbers and treated.

Liquid to solid: the highly radioactive liquid waste after separation of U and Pu contains the remaining minor actinides (MA) and fission products (FP), responsible for about 99% of the radioactivity of the spent nuclear fuel. This 'high level waste' (HLW) is calcined, vitrified in a boro-silicate matrix and stored in special glass canisters awaiting final disposal in a deep geological repository. Reprocessing one metric ton of spent nuclear fuel produces about 0.15 m^3 of high level solid waste. Other liquid waste from aqueous partitioning is cemented and put into dedicated waste canisters. This cemented waste is classified as ILW-LL (Intermediate Level Waste – Long Lived).

Other solids: after the shearing and dissolution of the fuel rods, the separated metallic structural materials and claddings are compacted and put into waste drums. This waste is classified as ILW-LL.

A comparative study of the once-through cycle (OTC) with the twice-through cycle (TTC; nuclear fuel being reprocessed once) concluded that the geological deep repository (GDR) volume needed for the OTC is about 3.4 times higher than the GDR volume needed for the TTC. This is mainly explained by the lower HLW volume in the TTC (see [3.3.5-9] for details).

More elaborated fuel cycle scenarios, leading to further reductions of waste volumes and radiotoxicity, are being studied and envisaged but are not yet developed at industrial scale. Further research is being performed, both on the partitioning side (e.g. highly efficient separation of MA and/or some long-lived fission products) and on the transmutation side (e.g. in dedicated reactors with a fast neutron spectrum and/or in 'accelerator driven systems' (ADS)).

3.3.5.3 Legal background and regulations

The most relevant international treaties and agreements dealing with environmental aspects of reprocessing are the following (see Annex 1):

- OSPAR Convention - The Convention for the Protection of the Marine Environment of the North-East Atlantic, (25 March 1998)
- Espoo Convention – Espoo Convention on Environmental Impact Assessment in a Transboundary Context (February 26, 1991);
- Aarhus Convention – Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters, Aarhus, Denmark, 25 June 1998

At EU level, the following directives are particularly relevant for reprocessing plants in the EU:

- Nuclear Safety Directive (NSD) – Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014;
- Basic Safety Standards (BSS) – Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation;

  Note that the BSS are in-line with the current ICRP recommendations, which are of global validity (see the document "The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4), 2007");
- Radioactive Waste Directive – Council Directive 2011/70/Euratom establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste;
- Transport Directive – Council Directive 2006/117/Euratom of 20 November 2006 on the supervision and control of shipments of radioactive waste and spent fuel;

- Water Directive – Council Directive 2013/51/Euratom laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption.

The construction, operation and decommissioning of a reprocessing plant can potentially have significant effects on the environment and therefore the following environmental EU legislation is relevant in the case of new projects, or changes to existing projects, related to reprocessing plants:

- Environmental Impact Assessment (EIA) Directive – Directive 2014/52/EU of 16 April 2014 amending Directive 2011/92/EU on the assessment of the effects of certain public and private projects on the environment;
- Strategic EIA Directive – Directive 2001/42/EC of the European Parliament and of the Council on the assessment of the effects of certain plans and programmes on the environment;
- Air Quality Directive – Directive 2008/50/EC of the European Parliament and of the Council on ambient air quality and cleaner air for Europe;

In relation to the EIAs, the IAEA has developed specific guidance for the preparation of environmental impact assessments in the nuclear power sector:

- Managing Environmental Impact Assessment for Construction and Operation in New Nuclear Power Programmes, NG-T-3.11 IAEA Nuclear Energy Series, Vienna 2014
- Strategic Environmental Assessment for Nuclear Power Programmes: Guidelines, NG-T-3.17 IAEA Nuclear Energy Series, Vienna 2018

Companies operating in the various areas of civil nuclear energy (e.g. design, construction and operation of nuclear facilities, manufacturing of nuclear materials or components, extraction of raw materials, etc.) are obligated by law to obtain a certification according to internationally recognized quality, environmental, as well as health and safety management standards, such as

- ISO 9001:2015 – Quality Management System;
- ISO 14001:2015 – Environmental Management System;
- ISO 45001:2018 – Health and Safety Management at Work (replacing ISO 18001).

As a rule, the appropriateness of the internal governance of a civil company operating in a specific area of nuclear energy is proven by demonstrating that the company uses internationally recognized management systems to manage nuclear and industrial safety, radiation protection, technological & radioactive waste handling and environmental protection tasks during all phases of the activity concerned. The audit is carried out by an accredited body and repeated periodically.

3.3.5.4 Identification of applicable means to avoid or mitigate the impacts

3.3.5.4.1 Non-radioactive impacts

As reprocessing plants are large chemical plants dealing with toxic gases and liquids, the best industrial practices and technological solutions to eliminate potential hazards associated with the handling and use of such substances are to be applied.

3.3.5.4.2 Radioactive impacts

Radioactive releases to the atmosphere are subject to legal limits, set in agreement with international guidance so that radiation will not result in any harm for the population or the environment. Operators continuously monitor the effluents and report the data obtained to the regulatory authority.

As mentioned in Chapter 3.3.5.1.6, liquid waste streams from the reprocessing operations are subject to filtration, evaporation, and other treatments to reduce the residual radioactive substances. The gas stream from the reprocessing operations is processed by cleaning systems, including scrubbers and HEPA (High Efficiency Particulate Air) filters prior to release. The noble gases krypton and xenon are released to the environment. They do not contribute significantly to the radiation dose of the workers or the public. The applied processes ensure that statutory emission limits are respected.

3.3.5.5 Summary and conclusions

Commercial scale reprocessing of spent nuclear fuel for civil purposes is now a mature technology that has been practised for several decades.

During the 1970s and 1980s, radiological emissions from reprocessing plants were significantly higher than at present. In more recent years, European reprocessing plants have been operating well within the discharge limits set by national authorities and the doses to the reference groups of the population due to radiological emissions from those plants are well below the statutory limits and very small compared to the individual doses due to natural background radiation.

UNSCEAR has estimated the doses to the public from reprocessing to be lower than the doses due to emissions from nuclear power plants. In both cases the doses are very low compared to legal limits for public exposure and compared to the dose from natural radiation.

Nuclear fuel reprocessing in a closed nuclear fuel cycle generally leads to a lower environmental impact of the nuclear energy lifecycle compared to the open fuel cycle due mainly to the reduced need for uranium mining that arises from closing the fuel cycle.

As can be seen from Table 3.3.5-2, nuclear fuel reprocessing activities do not represent a significant challenge to the TEG objectives. Releases of radioactive substances to the environment are maintained well within statutory limits using existing technology, thereby resulting in a very low impact on human health.

In the light of the above analysis it can be concluded that industrial activities associated with reprocessing of spent nuclear fuel do not represent significant harm to human health or to the environment. They do not represent significant harm to any of the TEG objectives, provided that the associated industrial activities satisfy appropriate Technical Screening Criteria.

TSC for the spent nuclear fuel reprocessing activities are provided in Chapter 5 and Annex 4 of the present report (*Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy*).

Table 3.3.5-2. Importance of nuclear fuel reprocessing impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		**Prevention or mitigation of potentially harmful impacts**	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	++	Decreasing fossil fuel consumption	-
Water withdrawal	+		-
Water consumption	+		-
Water pollution	++	Application of best practices	-
Ecotoxicity	++	Idem	-
Human toxicity	++	Idem	-
Land use	++	Reclamation of land	Full remediation
Atmospheric pollution	+++	Application of best practices	-
Acidification pot.	+		-
Eutrophication pot.	+		-
Ozone creation pot.	+		-
Production of TW	+++	Application of best practices	-
Depletion of resources	++	Application of best practices	-
Production of solid RW	+++	Application of best radioactive waste management practices	Reprocessing does not add significant quantities of waste to the nuclear energy lifecycle. It separates and processes what is generated in other stages.
Gaseous RA releases	++++	Application of best practices for trapping and removing radioactive substances from the waste stream.	Regulatory release limits apply.
Liquid RA releases	++++	Idem.	Idem

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

3.3.5.6 References for Chapter 3.3.5

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3.3.6 Fabrication of MOX fuel

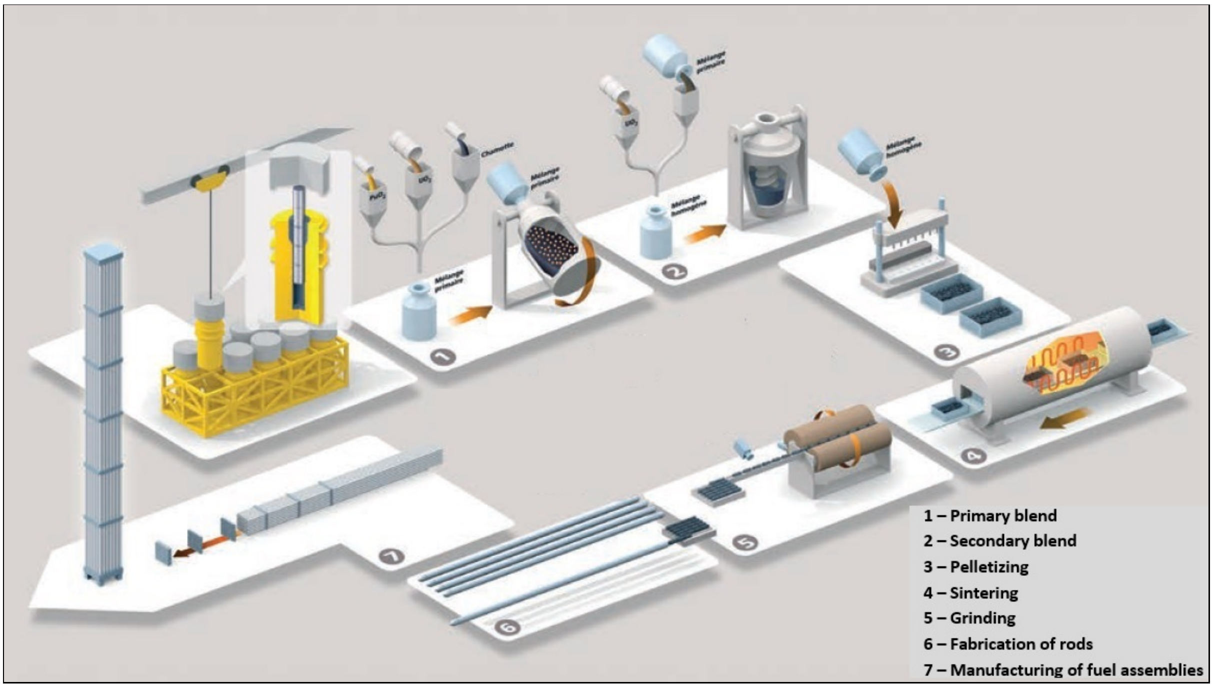
3.3.6.1 Description of the underlying technologies and processes

The processes for fabrication of Mixed OXide (MOX) nuclear fuel for LWRs is not very different from that of UO_2 fuel, the major difference being the starting materials. For MOX fuel, separated plutonium oxide from used fuel is the main fissile source, and not enriched uranium. The separation of plutonium from used fuel is described in Chapter 3.3.5. The separated plutonium is composed of the isotopes 238-242, of which 238, 239 and 241 are fissile, the major component being ^{239}Pu, around 50-60 wt%.

MOX fabrication is also a mechanical process, in which plutonium oxide powder and uranium oxide (depleted or natural) powder are mixed in a two-stage process, the so-called MIMAS process [3.3.6-1]. In the first stage, a master blend is produced with about 30 wt% plutonium by micronisation of the two powders. In the second stage, the master blend is mixed and milled together with uranium oxide to obtain the required isotopic composition. Again, this is a dry and mechanical process, in which dust and air-borne particulates form the prime risk.

In the next step, fuel pellets are made by compaction of the MOX powder in a mechanical press and subsequent sintering at high temperature (approximately 1700ºC) and reducing atmosphere (Ar/H_2), followed by grinding to the right diameter in a centreless grinder, inspection and selection. Due to the much higher radiotoxicity of plutonium and the gamma radiation from the ^{241}Am isotope (decay product from ^{241}Pu), the fabrication process has to be performed in alpha-tight and pressure controlled glove boxes, shielded with lead glass. Most process steps are automatized.

Figure 3.3.6-1. Scheme of the MOX fabrication process at the Melox facility in Marcoule, France



Source: [3.3.6-2]

3.3.6.2 Identification of key potential impacts on the environment and human health

— Due to its higher radiotoxic effects, fabrication of MOX fuels requires strict working conditions, stricter than UO_2. In the production facility three barriers are present:

  - The under-pressured glove boxes, to confine the material and avoid contamination. In addition, lead-glass shielding reduces the radiation dose. The air from the glove box is filtered with high efficiency.
  - The under-pressure laboratory, in which an eventual contamination can be contained. Again the air from the laboratory is filtered with high efficiency.
  - The reinforced building to protect the installation from external influences.

— The potential impacts of MOX fabrication are not different from those of UO_2 fuel listed in Chapter 3.3.4, but the higher radiotoxicity of plutonium translates into smaller quantities, which can be processed in a single batch.

— The manufacturing process includes routes for the recovery of chemicals, Pu and U, which are reused. Due to the radiological characteristics of Pu and its progeny, additional radiation protection measures are incorporated in the manufacturing process. Waste produced in MOX fabrication installations include U, Pu, zirconium and stainless steel, filters, transuranium elements, and miscellaneous waste.

3.3.6.3 Legal background and regulations

As noted in Chapter 3.3.4, fuel fabrication factories operate within the limits that are set by the nuclear licence. The data for the MELOX facility in France can be used as an example. According to the 2019 report for the Orano Melox site, where around 100 metric tons of heavy metal are processed per year, the liquid effluents in the years 2017-2019 contained less than 0.35, 0.38 and 0.48 MBq of alpha activity, respectively, compared to an authorised release of 2400 MBq (see Ref. [3.3.6-2]). The numbers for the atmospheric release of alpha-activity in these three years were less than 0.04, 0.01 and 0.01 MBq, respectively, compared to an authorised release of 7.4 MBq.

3.3.6.4 Evaluation and summary

The MOX fuel fabrication activities have an insignificant contribution to the environmental and human health impacts of the nuclear fuel cycle.

Note that Technical Screening Criteria were not developed for this activity, because this lifecycle phase does not represent a dominant contribution in any of the impact categories used in our study.

3.3.6.5 References for Chapter 3.3.6

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3.3.7 Nuclear power plant operations (production of electricity)

The power generation phase includes the construction, operation and decommissioning of nuclear power plants. A discussion of the impacts associated to the long term operation of these facilities has also been included in this section.

3.3.7.1 Description of the underlying technologies and processes

Commercial power reactors may be grouped in five different technological families[78], according to the coolant and moderator mediums used, and other technological choices.

Pressurised Water Reactors (PWR) and Boiling Water Reactors (BWR) use water both as cooling medium and as moderator. PWRs use two separate circuits to generate steam, while BWRs generate the steam directly in the reactor vessel. They use low enriched uranium as fuel. Collectively known as Light Water Reactors (LWR), they dominate the world market, with 90 % of the total capacity installed. They have been built by nearly all countries with commercial nuclear power programmes.

The third type by market share is the Pressurised Heavy Water Reactor (PHWR), which uses heavy water (deuterium oxide, $^{2}H_2O$) as moderator, and often as coolant as well. Canada operates 19 PHWR reactors and India 18 units, while Argentina, South Korea, Romania, China and Pakistan are currently running one or several reactors of this type. Unlike LWRs, heavy water plants can use natural uranium as fuel.

The other two reactor technologies are the Advanced Gas-cooled Reactors (AGR), operated in the United Kingdom with graphite as moderator and carbon dioxide as coolant, and the Light Water Graphite Reactor (LWGR), also known as RBMK type, moderated by graphite and cooled by water, of which 10 units are operated by Russia[79].

Table 3.3.7-1. Current market share per reactor technology

Reactor technology	PWR	BWR	PHWR	LWGR	AGR
Installed capacity (%)	73%	17%	6%	2%	2%

3.3.7.1.1 NPP construction

The construction of a nuclear power plant is a major infrastructure project, requiring extensive site preparation works, as well as the movement of large amounts of materials (clay, sand, stone, steel, etc.), the erection of buildings and the installation of numerous mechanical and electrical equipment. From the pouring of the first nuclear concrete to the start of commercial operation, successful nth-of-a-kind projects require at least five years to complete, but much longer periods have been observed for 1st-of-a-kind reactors, in particular if sufficient expertise in the management or engineering project teams was missing, or when the supply chain was weak or inexperienced in nuclear projects.

All processes involved in the construction of a nuclear power plant can be grouped in the following phases:

— <u>Pre-construction activities</u>. This phase includes the selection of the site, the design and licensing activities (like the Environmental Impact Assessment for example), as well as other site studies, procurement of

[78] In addition to these five groups, two fast breeder reactors are operated by Russia.

[79] Three additional graphite-moderated small units (11 MWe) are operated by Russia at a remote northern location

long-lead items and other management-related tasks. It involves mostly office work, and its environmental impact can be expected to be very minor compared with the subsequent phases.

- Site infrastructure. It includes any intervention on the site not requiring a construction licence from the nuclear safety authority. Some examples of activities on this phase are the construction or adaptation of site access infrastructures, excavation and earth movement of conventional parts, construction of temporary buildings or utility connections.
- Construction. Once the future operator is granted a licence, the actual construction of the NPP can start. This phase involves mainly civil works (excavations, laying of concrete foundations, erection of concrete and steel buildings, including the containment building, cooling towers or cooling channel), mechanical works (installation of large equipment, like the reactor vessel, steam generators, turbogenerators, etc., as well as smaller items like pumps, tanks, heat exchangers and piping, involving extensive welding activities), electrical works (installation of high, medium and low voltage transformers, switchboards, batteries, cabling, etc.) and I&C works (installation of instrumentation and control items, process computers, control room, etc.). The installation of numerous other auxiliary systems, like fire protection or heating and ventilation, complete this phase.
- Commissioning. Once the assembly of a system is completed, its commissioning may begin. It includes some preparatory activities, like inspections, cleaning or flushing, followed by different validation tests carried out at different levels. The construction and commissioning phases may have an overlap of several months, as construction activities in some systems may continue in parallel with the commissioning of already completed systems. In the latest stages of the commissioning, fresh fuel is loaded into the reactor vessel, enabling the latest integrated plant performance tests, leading to commercial operation.

Figure 3.3.7-1. Unit 3 of Flamanville NPP under construction (France, 2010)

Source: Wikipedia/Commons [80]

3.3.7.1.2 NPP operation

The design of most reactors currently operating assumed a service life of 30-40 years, but experience shows that service life extensions up to 60 or 80 years can be achieved subject to certain conditions (see below). During this long period, the majority of plants are constantly running at full power. They are shut down only for refuelling (typically several weeks every one or two years, a period also used for the maintenance operations that cannot be completed during operation), or in case of malfunctions. Some reactor technologies

[80] Original source: http://www.panoramio.com/photo/54209290 panoramio; author: http://www.panoramio.com/user/440234 schoella; unmodified, licensed under the Creative Commons Attribution 3.0 Unported licence.

(PHWR and LWGR) can be refuelled during operation. Although most NPPs are operated constantly at full power (baseline load) for economic reasons (as the operating cost is typically very low compared with the initial investment), load following (that is, the adjustment of the electrical output to the varying demand) has been practiced safely and reliably by the French nuclear fleet for years and to a lesser extent in Germany and other countries, with a good operating experience [3.3.7-43].

Most of the potential environmental impacts of NPP operation are related either to radioactive emissions or to the use of cooling water. The main technological aspects related to these issues are therefore discussed here.

The nuclear reactions that take place in the reactor core generate a certain amount of radionuclides, as a result of different processes: fissions of U or Pu atoms, neutron absorption by the fuel (creating the actinides), neutron absorption by structural materials and impurities in or around the core (called activation products) or radioactive decay of the previous isotopes.

The radionuclides produced are mostly contained inside the fuel rods, and processed with the management of the spent fuel; however, a minimal part cannot be contained inside the fuel, either because they leak through the fuel cladding or because they are generated outside the fuel (case of the activation products). In either case, most radionuclides remain within the structural materials (and will be later treated as solid waste during the decommissioning phase) or can be removed by the waste management systems of the nuclear plants, so that the radioactivity released to the environment during normal operation is minimised, and in any case below the authorised limits. The gaseous and liquid effluent streams from the site are constantly monitored, and additional samples from the environment at different locations around the facility are taken and analysed by the utility operating the plant and regulatory authorities usually conduct their own independent measurements.

As discussed below, the radionuclides that make up the majority of the radioactivity released by nuclear power plants during their normal operation are ^{14}C, tritium and some noble gases, like ^{85}Kr. The processes leading to the emissions of the first two are described here in more detail.

<u>Carbon-14 generation</u>

Carbon-14 is produced mainly by activation of nitrogen and oxygen contained (sometimes as impurities) in the fuel, the moderator or the coolant. If graphite is used as moderator, the activation of ^{13}C atoms is also an important generation path. The total amount produced depends on many factors (fuel enrichment, relative masses of fuel and moderator, concentration of nitrogen impurities and operating temperatures), and therefore is different at each plant. Part of the ^{14}C generated is retained in the fuel or in the reactor materials, and will therefore be treated at the back end of the fuel cycle, and the remaining part will be carried by the coolant and eventually released to the environment.

In PWRs, the ^{14}C is carried away from the core by the coolant, as dissolved CO_2, and eventually the CO_2 will mix with the gaseous nitrogen blanketing the primary system, and will join the flow to the primary offgas treatment system vents, which are the main release pathway (around 70% of the total release) [3.3.7-46]. Other paths are the steam generator blowdown tank vent, the turbine gland seal condenser exhaust or the ventilation exhausts of different buildings.

In BWRs, the radioactive CO_2 is carried directly to the turbine, and therefore the main release path (> 99%) is the condenser steam air ejector [3.3.7-46].

Heavy water reactors generate higher amounts of ^{14}C, mostly within the heavy water used as moderator and coolant. However more than 90% of it is removed by ion exchange resins, and stored as radioactive waste at the site or at a disposal facility. A small fraction (3.9%) is released to the atmosphere from the moderator cover gas and the annulus gas systems [3.3.7-46].

The generation of ^{14}C in AGRs takes place mostly within the graphite, from where a small fraction goes into the CO_2 coolant, together with the ^{14}C generated directly in the coolant. The total inventory present in the coolant is released to the atmosphere, either by leakages or by periodic purges.

<u>Tritium generation</u>

Tritium may be produced by ternary fission (in the few fissions where three fission products are obtained, instead of the usual two), or by activation of ^{2}H, ^{3}He, ^{6}Li and ^{10}B present in the coolant, moderator or other materials present in the reactor.

Ternary fission is the main generation path in LWRs, although part of it is retained in the fuel or fuel cladding and only a fraction leaks into the coolant. In addition to the ternary fissions, the main activation process in

LWRs is that of boron. As PWRs have boric acid dissolved in the coolant to control reactivity, the tritium generating rates are higher than in BWR reactors. In both cases the amounts of tritium released are very low and are often directly released into the environment without further processing.

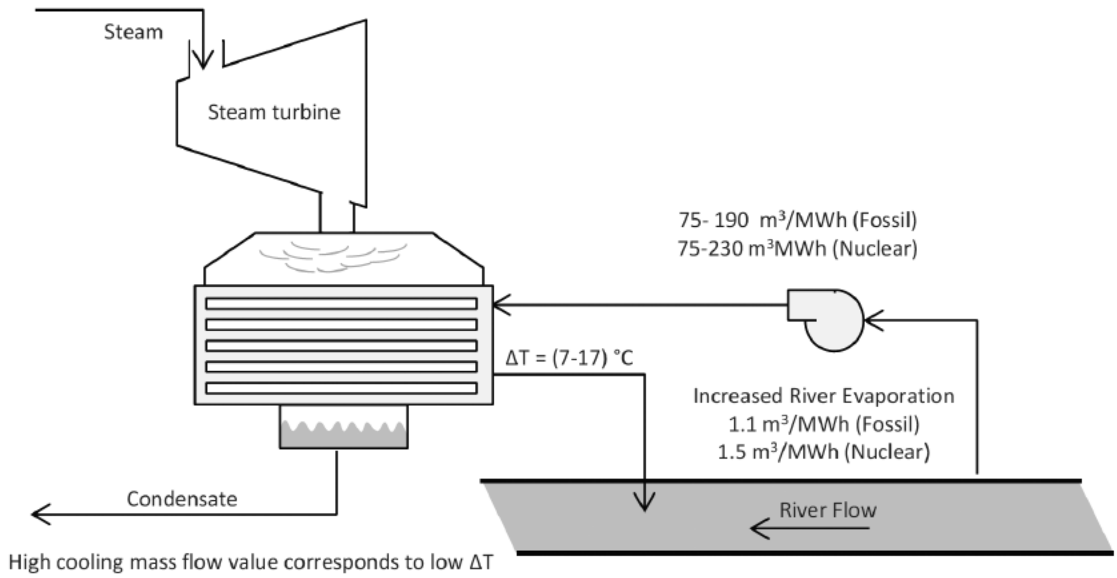
Heavy water reactors generate tritium in the moderator, and from there it can leak also to the coolant. However, as heavy water leaks are mostly recovered, releases to the environment are low, although typically higher than in the case of LWRs.

Cooling water systems

The operation of any thermal power plant requires releasing to the environment the part of the thermal energy generated that is not transformed into electrical power. For a given thermal output, the cooling required is determined by the thermodynamic efficiency of the plant, which is typically 32-36% in the case of nuclear power plants. The cooling is ensured through the use of cooling water systems. Additional water is needed for dissipating heat from spent fuel stored in pools or from the fuel in the vessel during plant shutdown, as well as industrial and potable water for conventional uses at the plant, however these requirements are negligible compared to the main plant cooling, and are not discussed here. Different types of main cooling water systems exist [3.3.7-37, -38]:

— Once-through wet cooling. Large amounts of water are taken from a natural source (most often the sea, but also big lakes or rivers), run through the plant condenser and then returned to the environment a few degrees warmer (which causes a certain evaporation offsite when the flow cools downstream, around 1% of the total flow). If available, this method usually yields the lowest cost. However, if this type of system is used with lakes or rivers, the legal restrictions on the maximum discharge temperature may induce a limit on the maximum power available in hot summer conditions, reducing total plant output and flexibility. Screens and mechanical filters are installed on the water intake to remove debris and marine life from the flow, which requires a small fraction of that flow (about 0.4%) to flush the filters and clean the screens.

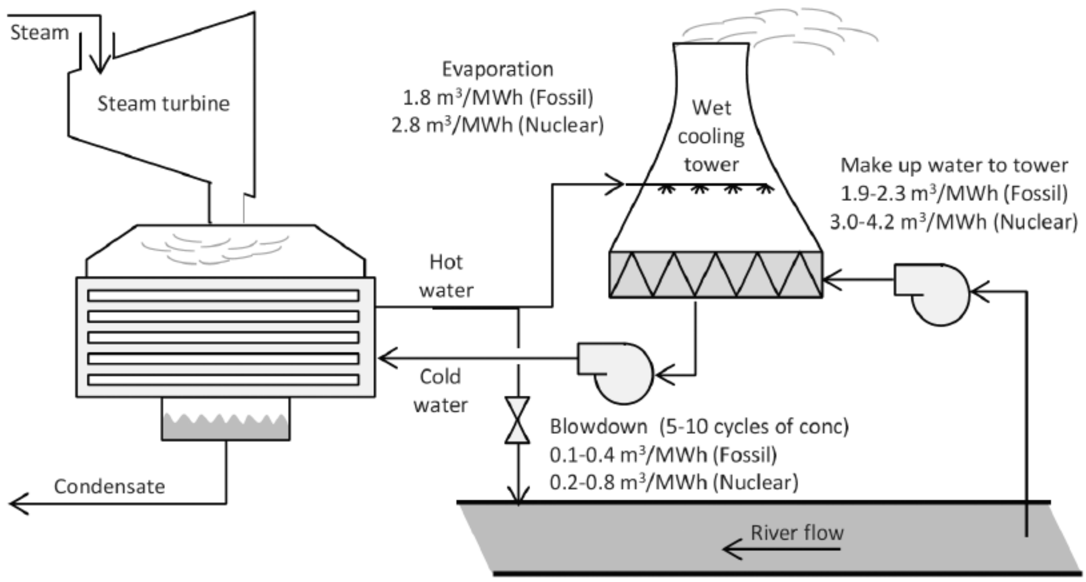
Figure 3.3.7-2. Once-through cooling water system



Source: [3.3.7-37]

— Recirculating wet cooling. In this case water heated in the plant condenser is driven to cooling towers, where it is cooled and partially evaporated. As evaporated water is replaced with additional make-up water, the concentration of impurities in the circuit tends to increase, requiring to bleed the system and hence use additional make-up flow. Therefore, the water consumed is the sum of the evaporation flow and the bleeding flow. Cooling towers may employ either natural draft (chimney effect) or mechanical draft using large fans (enabling a much lower profile, but requiring power, typically 1% of the net plant power). Some plants may replace or combine cooling towers with artificial or natural ponds to minimise the evaporative losses, at the expense of requiring more land.

Figure 3.3.7-3. Recirculating cooling system with cooling towers



Source: [3.3.7-37]

- Dry cooling. Dry cooling transfers heat from steam or water to air by forced flow through finned metal tubes, not by evaporation, thus substantially reducing water consumption. However dry cooling systems reduce plant efficiency (a decrease of about 2% averaged over the year) and increase the capital investment. For these reasons, dry cooling has not been used so far by the nuclear industry (with the exception of experimental or very small reactors) and only rarely by conventional power plants (mostly in South Africa), but it has been lately proposed within the design of some small modular reactors as an alternative for sites where water is very scarce.

Currently, 45% of all NPPs use once-through sea water cooling, 15% use once-through systems on lakes, 14% use once-through systems on rivers and 26% use cooling towers (although some plants have hybrid systems).

See [3.3.7-37] for an extensive description of cooling water systems employed by nuclear stations.

3.3.7.1.3 Lifetime extension or Long-Term Operation (LTO) of nuclear power plants

Nuclear power plants are typically designed for a specific operational lifetime, typically 30 to 40 years for the Generation II NPPs[81] that are operating today, many of which are approaching or have already passed their original design lifetime. The chosen design lifetime determines the design of a number of components, which are dimensioned such that they are able to perform their intended functions taking into account the expected[82] degradation with age. This degradation may be, for example, due to changes in material physical properties or component condition that results from the ambient conditions (e.g. neutron irradiation) and/or the static or cyclic loading, wear, corrosion, erosion, etc. that the component is subjected to during the assumed lifetime.

The original design assumptions and criteria used in the design of older power plants were often more conservative than those used today. Advances in the understanding of degradation mechanisms, feedback of operating experience during the operating period of these older plants, and improved analytical techniques resulting from advances in technology (including computer-based techniques) has benefitted the design of modern plants, which do not have to incorporate the same level of conservatism in the design of some of their components. This allows the degree of conservatism in the ageing analyses and in the plant operating margins to be better assessed also for existing NPPs. Furthermore, as the structures and components (SCs) approach the end of their original design lifetime, it will often be the case that the ambient stressors and the operational transients to which the SCs have actually been subjected during their operational lifetime will also

[81] Generation III plants, which are currently under construction in some EU Member States are designed for a longer lifetime, typically 60 years.
[82] Evaluated using conservative assumptions.

have been less severe than conservatively assumed at the design stage. Consequently, as the older plants reach the end of their design lifetime, it is likely that the actual condition of critical SCs is better than assumed during the design assessments, which allows plant operation to continue safely beyond the original design lifetime. However, the continuation of the operation of the plant beyond its original design lifetime can only be allowed following a systematic and comprehensive demonstration that it is safe to do so.

From a safety point of view a nuclear power plant could continue in operation beyond its original design lifetime if all components would be kept in adequate condition through maintenance and, when necessary, replacement.

The great majority of structures, systems and components (SSCs) in a nuclear power plant are replaceable. Some may be replaced routinely during normal maintenance procedures. The replacement of others may involve significant investment and extended plant outages. For the purposes of managing the condition of the plant, the replaceable SSCs can be classified as "critical" or "non-critical" for continued safe and efficient operation of the plant.

Non-critical SSCs are those that can be allowed to fail without causing concerns for safety or reliability of the plant. In most cases, they can simply be replaced or repaired when a fault is detected. Critical SSCs, on the other hand, include those that would cause safety or reliability issues if they were to fail. Preventive and predictive maintenance programmes are designed to ensure that such SSCs are replaced or repaired long before there is a significant risk of their failure.

Nevertheless, there are some critical structures and components of a nuclear power plant, the replacement of which cannot reasonably be considered feasible from either a technical or an economic point of view. Therefore, from a safety perspective the lifespan of a nuclear power plant is largely limited by the state of these critical non-replaceable structures and components. Critical SCs that cannot reasonably be replaced generally include the reactor pressure vessel and containment building. They may also include some reactor vessel internal components, parts of the primary coolant circuit and possibly also a part of the electrical cables. The condition of other systems, structures and components can be ensured by proper ageing management (including inspection, monitoring, maintenance and repair or replacement).

A comprehensive programme of activities has to be performed by the licensee in order to prepare for the long-term operation (LTO) of a nuclear power plant. The overriding objective of these activities is to demonstrate that the plant can continue to operate safely, in accordance with its current licensing basis (CLB)[83], for the planned period of the LTO. Consequently, a comprehensive safety evaluation will be a part of this programme of activities. It may well also include the replacement of a number of SSCs, due to ageing or obsolescence of existing SSCs, as well as a number of safety improvements proposed by the licensee or requested by the regulator, in view of the results of the comprehensive safety evaluation.

With regard to comprehensive safety evaluations of existing NPPs, it is noteworthy that such evaluations are not only performed just prior to entering into a period of LTO. In fact the continuous improvement of safety is an established feature of the regulatory and legislative framework in European Union countries as well as in international conventions.

All nuclear regulatory authorities from EU Member States operating nuclear power plants are members of the Western European Nuclear Regulators Association (WENRA).

One of WENRA's stated policy objectives is a commitment to continuous improvement of nuclear safety. This applies to WENRA as an organisation as well as individually to its members, which are 'committed to continuous improvement of nuclear safety in their countries' [3.3.7-12].

WENRA has established a Reactor Harmonisation Working Group (RHWG), the aim of which is to develop a harmonised approach to nuclear safety within the WENRA member countries. To this end, it has produced a set of safety reference levels for existing reactors [3.3.7-13], first published in 2006 and updated regularly, the last update being in 2014. The SRLs reflect the expected practices to be implemented in the WENRA countries with regard to the safety of existing reactors, and the national regulators make a commitment to

[83] Ref. [3.3.7-11] defines the current licensing basis as the "collection of documents or technical criteria that provides the basis upon which the regulatory body issues a licence for the siting, design, construction, commissioning, operation or decommissioning of a nuclear installation valid for the current authorized period". The collection of documents for an operating licence may include the applicable set of regulatory requirements, regulatory orders, licence conditions and exemptions; technical specifications, operating limits and conditions; plant-specific design-basis information as documented in the latest version of the Final Safety Analysis Report (taking into account all plant modifications made during the lifetime of the licence); and other plant documents.

improve and harmonize their national regulatory systems, by implementing the new SRLs in their national regulatory frameworks [3.3.7-12].

Within Issue A: Safety Policy, of the safety reference levels for existing reactors [3.3.7-13], it is specified that the safety policy of licensees shall require continuous improvement of nuclear safety by means of (among others):

- Identifying and analysing any new information with a timeframe commensurate to its safety significance;
- Regular review of the overall safety of the nuclear power plant including the safety demonstration, taking into account operating experience, safety research, and advances in science and technology;
- Timely implementation of the reasonably practicable safety improvements identified.

In this context, 'Regular' is understood as an ongoing activity to review and analyse the plant design and operation and identify opportunities for improvement. Periodic Safety Review (PSR, [3.3.7-14, -15]) is a complementary tool to verify and follow up this activity in a longer perspective.

The PSR is performed by the licence holder under the regulatory control of the competent regulatory authority at least every ten years in EU Member States as a requirement of the nuclear safety directive. The review is intended to confirm the compliance of the plant with its current licensing basis and any identified deviations must be resolved. In addition, the review has to consider any issues that might limit the future life of the facility or its components and plan for their management. The review is also intended to identify and evaluate the safety significance of deviations from applicable current safety standards and requirements, and internationally recognised good practices, taking into account operating experience, safety research, and advances in science and technology. All reasonably practicable improvement measures shall be taken by the licensee as a result of the review [3.3.7-14]. In many EU Member States, the comprehensive safety review performed in the frame of LTO is performed according to the PSR approach.

Through the above processes, licensees should identify and implement reasonably practicable safety improvement measures throughout the lifetime of an NPP. They should also identify any issues that could impact the future lifetime of the facility and implement appropriate compensatory measures where possible.

3.3.7.1.4 NPP decommissioning

Decommissioning is the last part of the lifecycle of nuclear plants. It aims to dismantle the installations no longer used and to dispose of the resulting waste and materials. Decommissioning may include environmental site remediation in case of contamination due to the accidental releases of radioactivity.

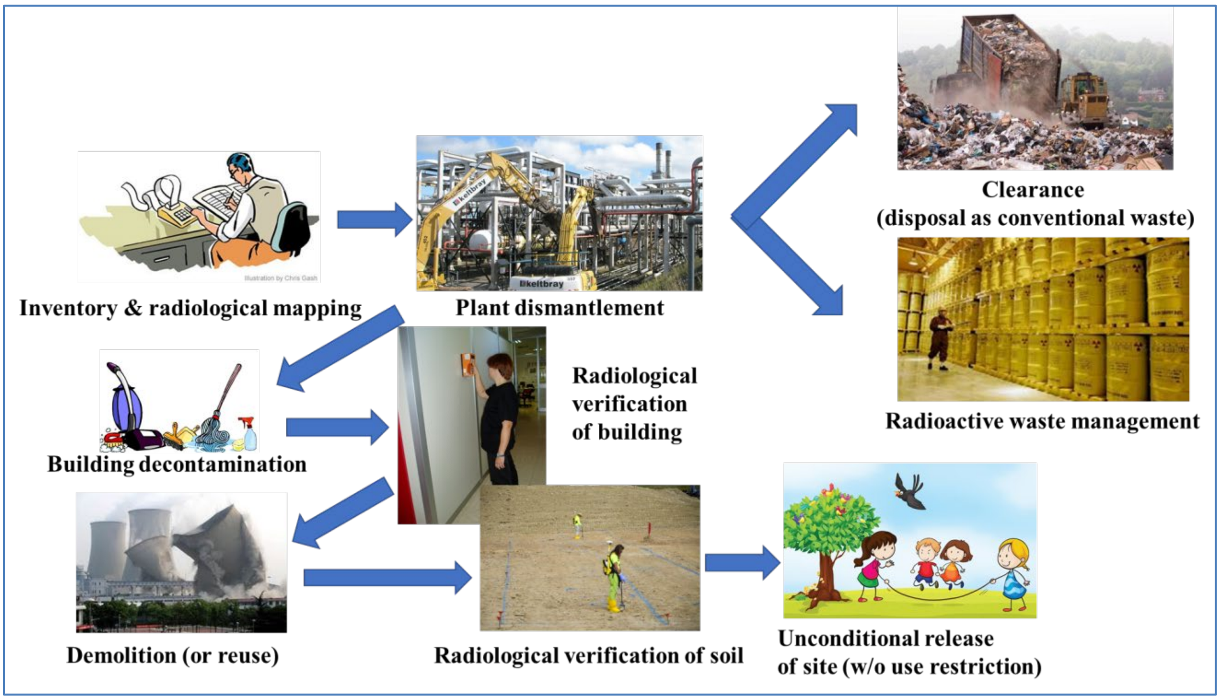
There are three main strategies for decommissioning of nuclear facilities: immediate dismantling, deferred dismantling, also called safe enclosure, and entombment. In the first case, a facility is dismantled right after its shutdown. In the second case, the facility is kept in a state of safe enclosure for several decades followed by dismantling. In the third case, the facility is encapsulated and kept isolated until the radionuclides decay to levels that allow release from nuclear regulatory control. The present trend is in favour of the immediate dismantling.

Decommissioning may have two different end statuses:

- “Green field” (or "unrestricted use"): the site hosting the decommissioned plant is released free of any constraints linked to the past nuclear activity after it has been cleaned from any trace of artificial radioactivity and eventually restored to the previous conditions;
- “Brown field” (or "restricted use"): the plant is dismantled and the site is remediated, but its reuse is limited to restricted types of activity.

The main steps of the decommissioning process (see figure) are:

Figure 3.3.7-4. – Decommissioning process



- Decommissioning planning and licensing;
- Plant characterisation;
- Decontamination;
- Dismantling of components;
- Demolition of buildings;
- Waste and material management;
- Site verification, restoration and monitoring;
- Site release from regulatory control.

Plant characterisation is an essential step at the beginning of the decommissioning process that aims to collect as much data as possible about the physical and radiological inventory of all systems, components, structures and materials present in the plant for designing and planning the decontamination and dismantling phases. Plant characterisation is important for making reliable estimations of the cost and duration of the decommissioning project and of the expected amount and category of waste that will be produced. It is also needed to determine the hazards to which workers and the general public could be exposed.

Decontamination is the removal of surface or bulk contamination from components and structures before or after their dismantling. There is a variety of decontamination techniques depending on the type of material and the nature of contaminant. For surface decontamination of metallic objects multiple options are available (see figure): mechanical dry abrasion, wet cleaning with ultrasound, chemical decontamination with acid or basic liquids, foams or gels. Scarification is used for removing surface contamination from walls and floors of buildings. Contaminated water (for instance from a reactor pond) can be decontaminated by coupling techniques such as filtration, flocculation/precipitation and extraction with ion resins.

Figure 3.3.7-5. – Surface decontamination techniques for metallic items, from left to right: abrasive blasting unit, ultrasound bath and application of gel

Nuclear decommissioning invariably involves dismantling of plant and equipment which has some degree of radioactive contamination. Dismantling methods generally involve size reduction of the components and structures, for example through disassembling, mechanical cutting, thermal cutting or other methods. Measures need to be taken in order to protect the workers and the environment, such as contamination containment, personal protective equipment (PPE) or remote handling techniques (see figure).

Figure 3.3.7-6. – Dismantling in confined space with PPE (left) and with a remote controlled excavator (right)

After removal of all plant components and loose items, only the building structures remain. A complete radiological survey should exclude any trace of residual radioactivity. At this point, the demolition of the buildings can be done, where concrete and masonry are broken down by conventional demolition methods. Alternatively to the demolition, a reuse of the buildings for other purposes can be considered.

Waste and other material from nuclear decommissioning needs to be managed so as to improve safety, and regulations exist to ensure this. Waste and material management involves treatment or conditioning of the wastes into passively safe forms, interim storage and, where waste routes exist, disposal.

After the demolition of buildings and removal of wastes, the site has to undergo the final radiological verification. If residual contamination is found, further site remediation actions have to be taken.

3.3.7.2 Identification of key potential impacts on the environment and human health

The construction, operation and decommissioning of a nuclear power plant is generally subject to an approval by the competent authority of an Environmental Impact Assessment (EIA). In the EU, the directive 2014/52 regulates the contents of such assessments, as well as the provisions needed to ensure adequate participation from relevant stakeholders in the assessments.

The following paragraphs review, for each phase (construction, operation and decommissioning) the main potential environmental impacts that can be expected from NPP projects, and that are usually evaluated for each specific site in its particular EIA.

3.3.7.2.1 NPP construction

Non-radiological impacts

The non-radiological impacts are similar to those encountered by the construction of infrastructure projects of similar scale.

By far the most relevant environmental impacts generated by this phase are those related to the production and transport of the materials involved in the construction of the plant, as well as the impacts related to the electricity and fuel consumption during the construction. Other potential non radiological impacts are: impacts on water systems, impacts on groundwater, road traffic, dust and noise.

(a) Building materials

In terms of weight, concrete and steel represent the majority of the materials needed to build an NPP. The following table provides some estimates of the amounts of steel and concrete required for the three most common reactor technologies.

Table 3.3.7-2. Amounts of materials required for the construction of an NPP

	Steel		**Concrete**		**Reference**
	Metric tons	**kg/kWe**	**m^3**	**m^3/MW**	
PWR	61 161	61	169 200	168	Goesgen NPP (1010 MWe) [3.3.7-7]
PWR	36 000	31	198 580	174	Generic PWR (1144 MWe) [3.3.7-9]
BWR	66 040	54	200 200	164	Leibstadt NPP (1220 MWe) [3.3.7-7]
PHWR (1)	36 000	40	132 500	147	Candu (900 MWe). Calculated from data in [3.3.7-8]
PHWR (2)	31 000	52	113 000	188	Candu (600 MWe). Calculated from data in [3.3.7-8]

We see from the table that the amounts required by MWe installed do not vary much with reactor technology. The differences are probably due to the specificities of the site and to the varying accuracy and completeness of the data used.

Other than steel, copper and aluminium are other important metals required, but in much smaller quantities (< 1.5 t/MWe for copper and < 0.2 t/MWe for aluminium). However their unitary environmental impact is potentially high, particularly in the case of aluminium, due to the large amounts of electricity required for its production. Furthermore, approximately 6 ½ m^3/MWe of wood is required.

(b) Fuel and electricity

In addition to the materials, the fuel and electricity required to build the plant need to be added. This includes electric power and diesel fuel for all machinery required to conduct the civil works, mechanical and electrical systems assembly, temporary buildings etc. During the commissioning phase, all systems and equipment need to be operated during validation tests. Before the in-house nuclear power is available, electricity required relies on offsite power, which is subject to the relevant energy mix. If the share of fossil fuels in that energy mix is significant, commissioning power can be a major contributor to total construction environmental impact. For instance, for the case of Torness NPP [3.3.7-3], electricity consumption during commissioning represented 58% of total CO_2 emissions during construction.

Modern plant-specific life cycle assessments include detailed inventories of materials and energy requirements where the different constituents of materials like concrete or steel (iron, nickel, manganese, etc.) are separately specified, and different contributions from each source (wind, hydro, nuclear,…) to the applicable electricity mix are assessed, see for instance [3.3.7-1].

(c) Impacts on water environment

The plant cooling system often requires the construction of large water intake and discharge infrastructures at the sea coast, lakes or rivers. The construction may require dredging operations, sometimes involving underwater explosions. The dumping of dredging waste may temporarily increase the turbidity of water in the area surrounding the construction and dumping sites. The duration and extension of the area affected depends on the type of marine or river soil, marine currents, etc, as well as on the dredging methods employed. Furthermore, the noise generated by the dredging explosions, if required, may kill, injure or drive away fish present in the area.

(d) Impacts on groundwater

The site excavation may have an impact on the local level of groundwater or on its quality. The construction of foundations and other below ground structures may require lowering the existing groundwater level in order to achieve safe and dry working conditions for the excavations. The duration and extent of this groundwater depression depends on local geological conditions. The quality of the aquifers may also be impacted by contamination with pollutants from the construction works or by mixing with seawater or other water sources, resulting in changes of groundwater salinity.

(e) Road traffic

Construction works generate substantial road traffic, with intensities of several thousand vehicles per day during peak periods, generating emissions to the atmosphere from the combustion of diesel and gasoline engines.

(f) Dust and noise

Dust and noise are mostly associated to the site infrastructure and excavation activities, like rock blasting and other civil works activities, most of them carried out during the first years of construction. Heavy traffic also contributes to dust and noise during the entire phase duration.

Radiological impacts

The fresh nuclear fuel is brought to the site and then loaded into the vessel and irradiated for the first time only during the plant commissioning, immediately before entering commercial operation. Therefore, all radiological impact can meaningfully be allocated to the operation and decommissioning phases of the plant.

However, the mining and processing activities required to manufacture the building materials do generate some radioactive releases, although in very small quantities. The associated occupational collective effective dose per unit of electricity produced due to the construction of nuclear power plants has been estimated at 0.02 manSv/(GWa) [3.3.7-40]. Note that all power generation technologies need building materials, and therefore cause radioactive releases. The value for solar PV is 0.8 manSv/(GWa) and for wind is 0.1 manSv/(GWa) [3.3.7-40]. The main reason for the higher values of renewables is their lower capacity factors and shorter service lifetime.

3.3.7.2.2 NPP operation

Non-radiological impacts

(a) Airborne emissions

Non-radiological airborne emissions from operating facilities are dominated by gases generated by commuting traffic and by the operation of emergency diesel generators (CO_2, CO, NO_x, SO_2 and particles). Diesel generators are normally in standby, but they need to be operated periodically for testing and during emergencies, when normal power sources are unavailable. Requirements on testing duration and frequency vary, but a typical pattern would be a 1 hour test run every month and a 24 hour test run every 1-2 years.

The following table with data from a recent environmental impact assessment in a new 1200 MWe project in the EU provides a quantitative estimate of the emissions involved.

Table 3.3.7-3. Air emissions for a new NPP project estimated during the operation phase

Type of emission	From transportation and commuting traffic (t/year)	From operation of emergency diesel generators (t/year)
Carbon dioxide (CO_2)	1 219	750
Carbon monoxide (CO)	19	---
Nitrogen oxides (NO_x)	4	1.4
Sulphur dioxide (SO_2)	0.01	0.3
Small particles (PM)	0.1	< 1

Source: [3.3.7-39]

(b) Water withdrawal and consumption

The amount of water withdrawn (and then returned with a temperature increase), and the amount of water consumed depend on the type of cooling water system employed by the plant. See Figures 3.3.7-2 and -3 for some typical flows required.

Regarding the water withdrawal, two types of environmental impacts can be distinguished [3.3.7-38]:

— fish and crustaceans killed due to impingement (trapping of larger fish on screens) or entrainment (drawing of smaller fish, eggs and larvae through cooling systems);

— change in ecosystem conditions brought about by the increase in temperature of the discharge water.

However there is no impact due to any chemical or impurity in the water, as cooling water, beyond some minor chlorination, is not polluted by use at the plant.

On the other hand, the water consumption is in itself an environmental impact, particularly in areas where water resource is scarce.

(c) Waste water and other non-radioactive hazardous wastes

Waste water at NPPs is generated from various processes and systems, such as:

— Reject water from the reverse osmosis equipment at the demineralisation plant;

— Filter rinsing;

— Laboratory drain water;

— Floor washing water;

— Laundry waste water;

— Rain water drains;

— Sanitary waste water;

— Flushing water from cooling water intake;

— etc.

All these waste water flows are classified, monitored for radiation (if applicable), treated (either at the site or at offsite water treatment facilities) and disposed according to regulations in place. If not properly managed, the waste water poses a potential risk of soil or groundwater contamination.

Typical sources of hazardous wastes at nuclear plants are batteries, solvents and other chemicals, waste oil and oil contaminated filters and other equipment, fluorescent tubes, electronic components, etc. The total amount generated during the entire operation phase is very small. Similarly to waste water flows, they may pose a threat for the environment if not properly managed.

Radiological impacts

(a) Liquid and airborne emissions

Actinides, activation and fission products are generally contained inside the plant, or removed from the effluent streams by waste management systems, so that they are either absent or present in very low quantities on samples taken around nuclear plants.

The following table shows the radioactivity released by nuclear plants through liquid and gaseous releases during normal operation, broken down by reactor technology. Values were obtained from individual facilities environmental reports that were available in 2002, averaged per reactor type and expressed in activity per unit of electricity produced.

Table 3.3.7-4. Estimated normalised discharges per unit of electricity generated (TBq/(GWa)

	Discharges to atmosphere						**Aquatic discharges**	
Reactor type	Noble gases	Tritium	^{131}I	^{14}C	Particulates	^{35}S	Tritium	Other
PWR	5.8	1.5	$8.0 \cdot 10^{-5}$	0.08	$3.6 \cdot 10^{-5}$	0	18	0.0038
BWR	18	1.3	$4.2 \cdot 10^{-4}$	0.13	$1.8 \cdot 10^{-3}$	0	0.82	0.0021
PHWR	35	200	$2.3 \cdot 10^{-5}$	0.6	$1.7 \cdot 10^{-5}$	0	170	0.031
LWGR	460	26	$9.9 \cdot 10^{-3}$	1.3	$2.7 \cdot 10^{-3}$	0	0.78	0.002
AGR	19	4	$3.2 \cdot 10^{-5}$	1.4	$2.2 \cdot 10^{-5}$	0.066	410	0.81

Source: [3.3.7-40]

As far as the effect on human health is concerned, the impact of these releases is better measured in terms of individual or collective doses, as this effect depends not only on the radioactivity level, but also on the physical and chemical form of the radioisotope involved.

The individual dose to the public, normalised to the unit of electricity produced and considered for a characteristic individual (living at 5 km from the point of discharge and with the typical food and behavioural patterns in the region) has been estimated to be $1.3 \cdot 10^{-3}$ mSv/(GWa) [3.3.7-40] in Europe[84], a value much lower than the reference value for effective dose for public exposure of 1 mSv/year used by the international safety standards (i.e.: IAEA GSR Part 3).

As for collective doses, they are estimated at 0.2 manSv/(GWa) for local/regional level (world-averaged population within 1500 km from an NPP). Some of the radionuclides released have half-lives sufficiently long to continue to expose the population for decades, and therefore are assumed to circulate globally, giving rise to a collective dose over the entire world population, and not only at local or regional level. This additional contribution to the overall globally circulating radionuclides has been estimated, with a value of 1.8 manSv/(GWa) integrated over 100 years. This contribution is entirely dominated by carbon-14. However it must be noted that this global collective dose is distributed across the entire human population, estimated at 10^{10} over the 100-year period considered, which results in doses per capita of 10^{-8} Sv, an extremely low level [3.3.7-40].

It should be noted that ^{14}C and tritium are present in the environment as a result of natural processes as well as nuclear weapon testing, giving rise to an existing background radiation, independent from the operation of commercial power reactors.

Carbon-14 is produced by cosmic radiation in the upper atmosphere by the interaction with nitrogen atoms. Part of it is contained in the atmosphere, but a much larger amount is located in the deep oceans, and exchanges with atmospheric carbon. Additional inventories of ^{14}C were added by nuclear test explosions from 1945 to 1975, especially the atmospheric tests conducted in the early sixties.

84 Differing values across continents reflect the different proportion of reactor types in each region, as well as differences in food habits. Values estimated by [3.3.7-40] for all other areas are lower than for Europe.

Tritium is also produced in the upper atmosphere, converts into water and reaches the surface as rain, leading to a steady state natural inventory. A much larger amount was added by the nuclear weapons tests, although in this case, as the half-life of tritium is about 12 years, most of it has already decayed.

The values for public exposure caused by nuclear power plant operation given above are very low compared to the background radiation (see Chapters 1.3.2.2 and 3.4 and Refs [3.3.7-41, -47]), and similar to doses due to power generation using coal [3.3.7-40].

Liquid waste in nuclear power plants undergoes one or more of the following processes to remove the radioactive substances: filtration, ion exchange, adsorption, reverse osmosis and evaporation. The radioactive impurities filtered or removed from the decontamination of the liquid substances are characterized and conditioned to be shipped as radioactive waste for storage and final disposal as solid waste. The decontaminated liquid is either recycled or released to the environment in a controlled manner, according to the relevant regulations.

(b) Solid radioactive waste

Solid radioactive waste originates from the treatment of liquid and gaseous effluents, from processes that yield contaminated solid materials, or activated solid materials in areas of high neutron flux. Solid radioactive waste from the treatment of liquid and gas effluents include exhausted ion exchange resins and adsorption (charcoal) materials, filter materials, concentrates and sludges. Activated solid materials originate also as a consequence of interventions on structures or equipment in areas of high neutron irradiation. Contaminated solid materials may originate from routine or ad-hoc interventions in the plant, such as maintenance, plant design modifications, decontamination, etc. These activities produce solid radioactive waste such as plastic (protective plastic foils, etc), rubber (gloves), wood (supports), metal (from components, equipment or tools), textile (cloth, protective clothing).

Exhausted ion exchange resins, concentrated waste and sludge are “wet–solid” radioactive waste. Wet solid radioactive waste can be dewatered with different technologies (in phase separation devices in which the solids are allowed to settle, and the water overflows, or in tanks in which water is extracted with pumps) and conditioned in suitable packages. The wet resins, concentrates and sludges are mixed with cement and other additives to obtain a solid monolith with strength and lixiviation resistance properties that can be accepted in disposal facilities. The proportion of cement and additives are dependent on the characteristics of the wet waste.

Exhausted filters are immobilised in drums with concrete walls. Dry radioactive waste is sorted based on the treatment that the solid waste will undergo to reduce its volume. Solid radioactive waste can be classified as compressible, non-compressible, and combustible. Combustible waste can be burned in dedicated facilities. Incineration achieves large volume reductions and concentrates the radioactivity in ashes, which are solidified to be disposed of.

Compressible waste is compacted or super-compacted to remove the air and reduce its volume, and packed.

The operational waste generated in nuclear power plants is treated and pre-conditioned or conditioned on site to make it suitable for shipment either to an external radioactive waste management facility that could carry out additional specific treatments, or directly to a storage or disposal facility.

(c) Spent nuclear fuel

The core of a commercial LWR nuclear power plant with 1 GW_e electric power contains between 80 and 140 t of uranium with an initial ^{235}U enrichment level varying between ~3% and ~5%, depending on the type and design of the reactor and its fuel configuration. The fuel is usually in the form of ceramic uranium dioxide (UO_2) pellets. The pellets have typical diameter of 8-10 mm and a length of ~13 mm. About 300 fuel pellets are stacked in thin ~4 m long tubes. The tubes are made of zirconium alloy, or zircaloy. The filled tubes, sealed at both ends are called fuel rods or pins (Figure 3.3.4-1).

Several fuel rods (from ~90 to ~300, depending on the type and design of the reactor) are then bundled together into a fuel assembly. The assembly is held together by a zircaloy framework that includes, in the case of pressurized water reactors (PWR), control rods and empty channels for controls or instrumentation. A PWR core contains 150-250 fuel assemblies; the boiling water reactor (BWR) core is bigger and contains 400-800 fuel assemblies.

The fuel assemblies are irradiated in the reactor for several years (or cycles). At each reload shutdown (in modern NPPs this occurs every 18-24 months) ~1/4 to ~1/3 of the fuel assemblies in the core are replaced with fresh fuel. The actual fuel consumption, which corresponds to the generation of an equivalent amount of spent fuel, varies with the energy extracted from the unit mass of fuel (this is often called burnup): the higher the burnup, the lower the mass of spent fuel generated. The burnup obtainable from the fuel, in turn, depends on the initial enrichment. For instance, the yearly spent fuel generation in terms of material flow for an LWR with a power of 1000 MWe generating 8.76 TWh of electricity is ~27 t with a fuel burnup of 45 GWd/t (initial ^{235}U enrichment 4%), or 17.5 t with a fuel burnup of 65 GWd/t (initial ^{235}U enrichment 5%).

Table 3.3.7-5 illustrates the yearly spent fuel production in some countries[85].

Table 3.3.7-5. Annual spent fuel arisings in some countries

Country	Unloaded spent fuel (tHM[1]) 2017	Unloaded spent fuel (tHM[1]) 2018
Argentina	86	115
Belgium	96	90
Canada	1 599	1 587
Czech Rep.	60	85
Finland	55	54
Germany	215	309
Japan	240	240
Korea	457	653

Country	Unloaded spent fuel (tHM[1]) 2017	Unloaded spent fuel (tHM[1]) 2018
Netherlands	8	9
Russia	672	729
Slovenia	0	22
Spain	143	128
Sweden	184	215
Switzerland	22	22
UK	541	650
USA	2 184	2 493

(1) HM = heavy metal

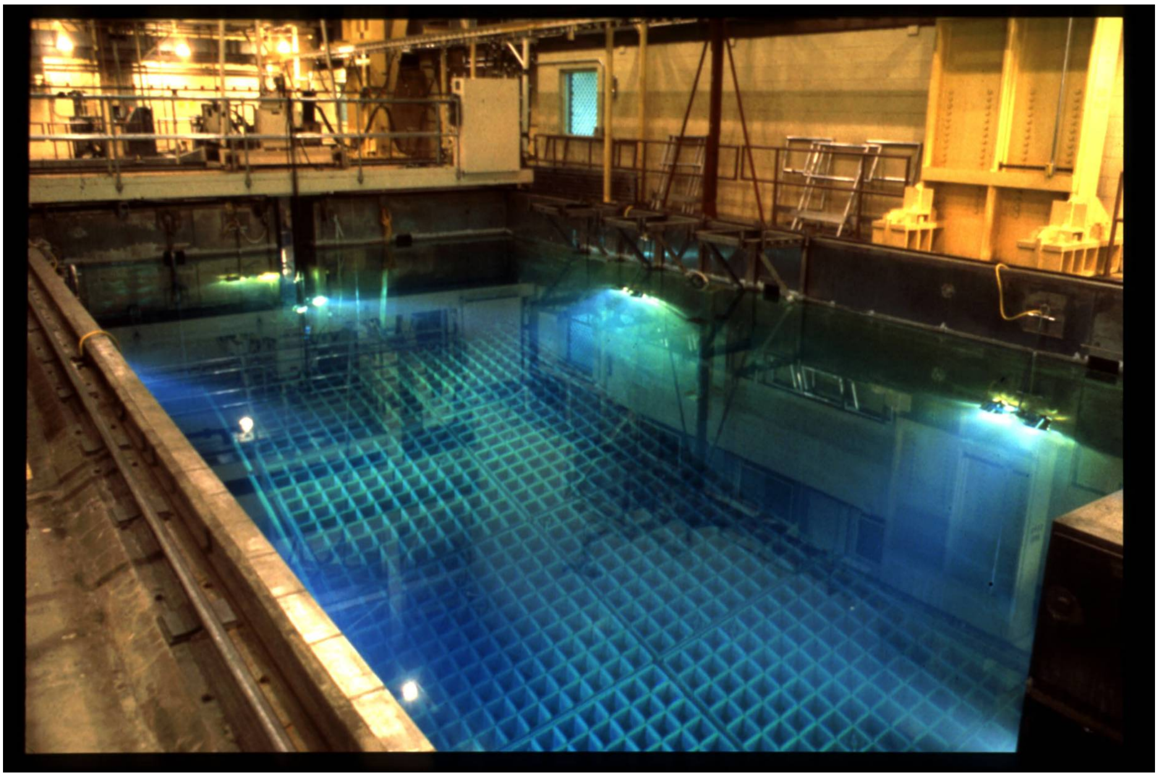
According to the IAEA[86] about 390 000 tHM of spent fuel was discharged from the nuclear power reactors worldwide during the 1954-2016 period. Approximately one third (127 000 tHM) of this amount is reprocessed and the remaining is stored pending either reprocessing or disposal. Some 166 000 tHM of spent fuel was discharged from EU reactors during the same period.

Spent nuclear fuel contains hundreds of radioactive nuclide species and accounts for almost all of the radioactivity generated in nuclear reactors. The overall inventory of the spent fuel evolves over time due to radioactive decay. As a result of the decay reactions, spent fuel generates significant heat after irradiation has ceased, decreasing with time. This must be taken into account in all steps of its management, in particular regarding cooling and shielding.

After its irradiation, fuel is unloaded from the reactor core and is placed under water in the spent fuel pool of the nuclear power station, either within the reactor building or in a spent fuel pool building adjacent to the reactor building. Typically, the spent fuel pool consists of a reinforced concrete structure, its internal wall is covered with a steel liner or coated with a water resistant paint. At the pool bottom special storage racks accommodate spent fuel assemblies. Several metres of water provides for decay heat removal and radiation shielding during storage and fuel handling. Protection against criticality is ensured by adding neutron absorber (such as boric acid) into the water and, in some cases, neutron absorbing components in the storage racks. Several systems connected to the spent fuel pool ensure the adequate conditions of the water. These systems include the spent fuel water cleaning and cooling system, to remove contamination and other impurities, and provide additional cooling for example after the unloading of the entire reactor core, water chemistry control system, to limit the corrosion of the spent fuel assemblies, fuel handling system, leakage detection and collection systems, and instrumentation. The reactor building or the spent fuel pool building ventilation systems are designed to remove eventual airborne contamination that could have been released from the spent fuel pool. Hydrogen generated through radiolysis of the pool water is also vented.

85 OECD Nuclear Energy Data Données sur l'énergie nucléaire Nuclear Energy Data 2019
86 IAEA/EC/NEA Status and Trends in Spent Fuel and Radioactive Waste Management project.

Figure 3.3.7-7. Spent fuel pool



Source: [87]

Depending on the national spent fuel management policy, spent fuel is considered as radioactive waste, or as a valuable resource to be reprocessed and reused in the future. After a few (typically five to ten) years of cooling in spent fuel pools the spent fuel assemblies can be reprocessed or stored in another facility. Spent fuel assemblies that will not be reprocessed are moved to a storage facility specifically built at the reactor site, or away from the reactor awaiting disposal in deep geological formations.

3.3.7.2.3 Lifetime extension or Long-Term Operation (LTO)

From the point of view of lifecycle environmental impacts, LTO of an NPP will have little or no effect on the impacts from the front and back-end of the fuel cycle, as the NPP will continue to require new fuel elements and to produce spent fuel and other operational wastes during the additional years of operation. On the other hand, the lifecycle impacts per GWh of generated electricity originating from the concrete, steel and other building materials used in the construction of the plant, the construction and decommissioning activities and the decommissioning wastes will decrease substantially. This clearly results from the additional energy that will be generated by the plant, while the construction activities and materials, and the decommissioning waste, will increase only marginally, due only to the need to replace some SCs for the life extension period.

3.3.7.2.4 NPP decommissioning

Decommissioning has started to be considered carefully relatively late in the development of nuclear technology. The main reason for that was that the decommissioning approach in the early days of the nuclear era was oriented towards a deferred dismantling (safe enclosure) approach. Since radionuclides decay in time, it was a common belief that keeping an installation in safe enclosure for several decades after the end of its operation would have simplified the decommissioning facing a reduced radiological inventory.

Although the above assumption is certainly true, there are other factors that were later demonstrated to have a negative impact, such as the loss of corporate knowledge and of a competent workforce, large maintenance or refurbishment costs, increasing regulatory requirements and increasing costs of waste disposal over the timescales involved.

[87] https://www.reddit.com/r/submechanophobia/comments/al22b1/power_plant_spent_fuel_pool/

Due to these considerations the attitude has changed in recent decades and now the preferred strategy is immediate decommissioning after shut-down [3.3.7-28]. The major advantage of this historical evolution is that decommissioning has only now started on a large scale, bringing two major benefits:

- It profits from the technological evolution that makes available advanced equipment, procedures and materials;
- It incorporates in the planning of the decommissioning activities an attention to the protection of workers, population and environment that was not considered in the early days of nuclear technology.

Therefore the current common practices in nuclear decommissioning have been developed in an environmentally-friendly mind-set, leading to a minimisation of negative impacts on the environment.

Non-radiological impacts

The non-radioactive impacts of decommissioning can be considered similar to the conventional construction/demolition activities, and in particular they include:

- Release of gaseous and liquid effluents;
- Acoustic emissions;
- Waste production;
- Increase induced in traffic.

In most Member States, the decommissioning of a nuclear plant is subject to an approval by the competent authority of the Environmental Impact Assessment [3.3.7-27]. The Environmental Impact Assessment includes the assessment of the possible impacts and identification of applicable mitigation and preventive measures and necessary monitoring plan.

The decommissioning process generates a large amount of waste from dismantling of the plant and plant components, demolition of buildings, as well as technological waste produced by the decommissioning activities, secondary waste from treatment processes, remediation of contaminated sites and other activities. Nevertheless, large amounts of the materials generated are neither contaminated nor activated above background levels. Such materials can be cleared from any further regulatory control (clearance) and disposed of as a conventional waste, reused or recycled. The dismantling of nuclear installations is probably the most important area of application of the concept of clearance, at least in terms of the volume of materials with a potential for clearance.

While clearance levels may very well be defined generically, the decision whether to apply clearance levels is an individual decision of the competent authorities based on a case-by-case evaluation of the practice giving rise to the contaminated or activated material.

The possibility to clear material from regulatory control has to be pursued at any stage of the nuclear cycle, but it becomes of paramount importance in the decommissioning phase. Various estimations and practical experience show that 90% or more[88,89,90,91,92] of the total material produced when dismantling and demolishing a nuclear installation is potentially clearable, bringing huge savings in terms of material processing, storage and disposal. However, the clearance process differs from country to country. In the absence of a clearance process, amounts of materials to be managed as radioactive waste could significantly increase.

Radiological impacts

Decommissioning of a nuclear installation is the final step in its lifecycle, producing by far the largest amount of radioactive waste as well as inevitably creating hazards associated with radioactive contamination. Measures need to be taken in order to protect both the workers, the public and the environment, and to minimise the creation of additional radioactive waste through the spread of contamination.

[88] https://www.sogin.it/en/sustainability/circulareconomy

[89] https://www.nuklearesicherheit.de/en/science/decommissioning-of-nuclear-facilities/residue-and-waste-management/clearance-during-nuclear-power-plant-decommissioning/

[90] Kuno M., Hamada M. (2017) Radioactive Waste Treatment and Disposal Technique. In: Hamada M., Kuno M. (eds) Earthquake Engineering for Nuclear Facilities. Springer, Singapore.

[91] https://www.eu-japan.eu/sites/default/files/publications/docs/2016-03-nuclear-decommissioning-japan-schmittem-min_0.pdf

[92] Decommissioning of Nuclear Facilities, GRS-S-58, 2nd edition (2017).

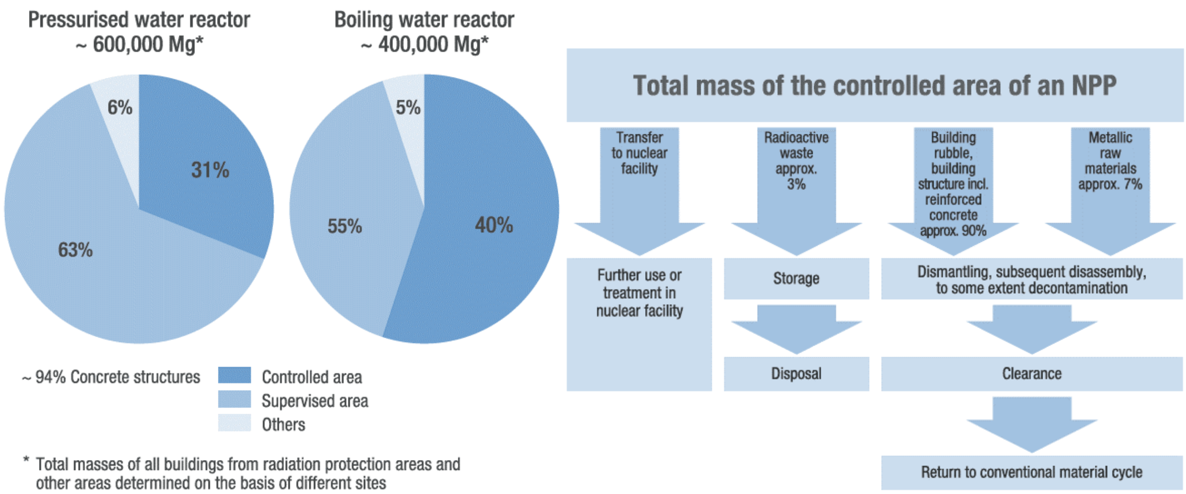
The decommissioning workers are the most exposed group to dose uptake. However, as the vast majority of operations in decommissioning occur in radiation fields close to the natural background, the activities can be performed in normal working conditions and with use of basic conventional PPE (gloves, coveralls, protective eyewear, hard hats, safety footwear, etc.). When there is a possibility of producing dust of activated material, for instance in radioactive component cutting, it is necessary to operate in a confined environment, typically a ventilated temporary containment enclosure, and the operators wear impervious clothing and respiratory protective devices or even pressurised suits. Whenever elements of work involve handling radioactive materials with high dose rate, further protection may be necessary by wearing lead aprons, armoured gloves and boots, constructing temporary shielding or using remotely operated machines deploying dismantling and demolition accessories.

During the decommissioning, it is not expected to have significant release of gaseous radioactive effluents, at least in normal operation and in absence of major accidents. All activities are executed inside the nuclear plant buildings, profiting from the existing containment measures, or inside the modular containment systems for outdoor operations. The negative pressure isolation technique used in nuclear industry generally keeps the inside of the plant in negative air pressure with respect to the surroundings, so any accidental release of airborne contamination would be kept internally. The air removed from the plant is filtered in absolute filters before expulsion and monitored continuously. Any gaseous radioactive releases are monitored and accounted for as they have to be compliant with the established limits.

For what concerns radioactive liquid effluents, these are also monitored and subject to controlled releases respecting the established limits. Discharged liquids must also respect regulations concerning maximum levels of chemicals and other conventional pollutants. Discharge limits for the decommissioning phase are generally reduced to the level of radiological non-relevance (de minimis dose). This means that the sum of authorised discharges (gaseous and liquid) should not produce the effective dose to any individual member of the public superior to 10 μSv in a year (trivial dose that represents a level of risk which is generally accepted as being of no significance to an individual, or in the case of a population, of no significance to society).

The trivial dose criterion is also the radiological basis for establishing clearance levels below which the disposal, recycling or reuse of solid materials coming from decommissioning is released from regulatory control. The dismantling of nuclear installations is probably the most important area of application of the concept of clearance, at least in terms of the volume of materials with a potential for clearance.

Figure 3.3.7-8. Quantities of materials from decommissioning in Germany

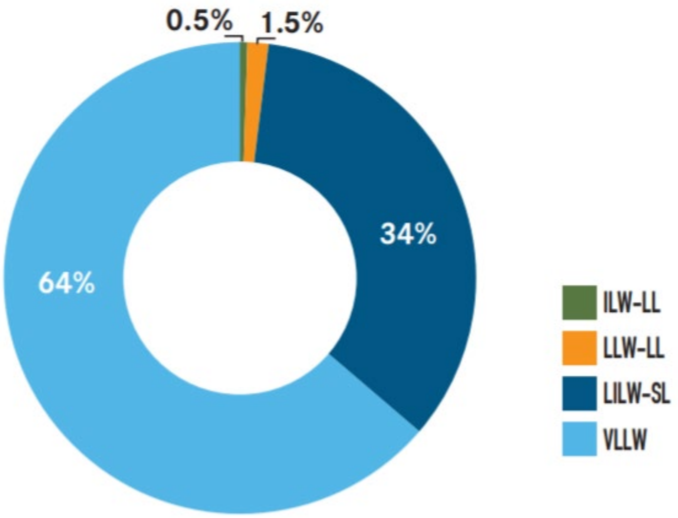


Source: [3.3.7-48]

The radioactive waste that cannot be cleared from the regulatory control still constitutes a significant amount. Most of it comes in the form of concrete or metals. Metallic waste mainly originates from dismantling the reactor, its primary circuit or supporting systems, e.g. reactor pressure vessel and internals, primary coolant pumps, steam generators, tanks, piping, structural supports, etc. Concrete radioactive waste mainly originates from removal of activated as well as surface and subsurface contaminated concrete structures or their parts. Other radioactive wastes, such as insulation, wood or plastics make up only a small part of the overall radioactive waste inventory. In case of graphite moderated reactors (e.g. Magnox or RBMK type reactors), significant amounts of radioactive waste originate from irradiated graphite. A special or hazardous waste that is radioactively contaminated can also be generated during decommissioning, such as contaminated lead, oils or asbestos. Additionally, during the decommissioning process, secondary waste is generated, such as waste from characterisation or decontamination activities, contaminated tools and equipment, protective clothing or shielding material.

Most of the radioactive waste resulting from decommissioning activities is short-lived waste classified as very low or low level waste. Moderate quantities of intermediate level waste might come from the most activated parts of the reactor (such as internals of the vessel and biological shield), whereas generally no HLW is generated in this step, since spent fuel is removed from the plant before starting the decommissioning. A typical distribution of decommissioning waste between different waste categories is illustrated in Figure 3.3.7-9, using the forecast of radioactive waste inventory generated from decommissioning activities in France during the 2017-2040 period. Approximately 98% of the radioactive waste is expected to be short-lived waste (LILW-SL in the French waste classification corresponds to the LLW waste class in GSG-1[93] waste classification) and only 2% of the total waste amount is expected to be classified as ILW (both LLW-LL and ILW-LL in the French waste classification corresponds to the ILW waste class in GSG-1 waste classification).

Figure 3.3.7-9. Waste category distribution of radioactive waste from decommissioning activities in France during 2017-2040 period



Source: [94]

In 2008 IAEA presented a rough estimation which indicates on average 5000 to 6000 metric tons of short-lived radioactive waste and 1000 metric tons of long-lived radioactive waste generated during decommissioning of LWR type reactor per 1 GW_e of installed capacity (excluding radioactive waste originating from site remediation activities)[95]. Similar amounts of radioactive waste could be expected from decommissioning of reprocessing plants but with significantly higher fraction of long-lived radioactive waste. A more recent estimate of the radioactive waste generated during decommissioning activities was done in Germany[96]. Based on experience from the past and current decommissioning projects it is estimated that on average 5000 m^3 of conditioned non-heat generating radioactive waste per LWR will be produced during the decommissioning activities. Actual amounts of radioactive waste will vary depending on multiple factors, such as the type of nuclear installation, application of material clearance, decommissioning strategy (immediate or deferred dismantling), etc.

93 IAEA General Safety Guide No. GSG-1, Classification of Radioactive Waste.
94 National Inventory of radioactive Materials and Waste. Synthesis Report 2018. ANDRA (2018).
95 INTERNATIONAL ATOMIC ENERGY AGENCY, Estimation of Global Inventories of Radioactive Waste and Other Radioactive Materials, IAEA-TECDOC-1591, IAEA, Vienna (2010).
96 ARTEMIS peer-review report - Germany, 2019, https://www.iaea.org/sites/default/files/documents/review-missions/final_artemis_report-germany.pdf

3.3.7.3 Summary of lifecycle analysis results

Although many lifecycle analyses can be found in the literature, a direct comparison of results is usually hampered by their different scope, methodology, assumptions, choice of environmental impact indicators, etc.

Regarding in particular the amounts of materials required to build the facilities, life cycle assessments face a methodological obstacle in their estimates. The most natural approach involves the use of plant design data to obtain the amounts of concrete, steel, etc. required to build the plant. This is the so-called process-based method. The main disadvantage is that data is usually available for some materials, notably the commodities, but not so often for manufactured equipment and processes. An alternative used by some authors has been to use the Economic Input-Output (EIO) approach. In this case, materials amounts are estimated from the monetary flows among the different economic subsectors, available from economic statistics. The benefit of the method is to capture emissions from materials that could have been excluded from the process data tables. However, the EIO sometimes includes emissions which are out of the system boundary, or, what is worse, penalises some technologies. Some authors [3.3.7-6] have found that the EIO method gives 10-20 times higher GHG emissions per kWh for the steel needed to build a nuclear power plant, as compared to the process data approach, while in the case of a solar photovoltaic facility the EIO estimates are only 3 times higher. The reason is that nuclear-grade steel supplies are much more expensive due to higher qualities required and to added value services which cost is embedded in the price of the commodity.

As more and more companies and organisations across all sectors are conducting life cycle assessments, the difficulties in finding suitable process data have been mitigated, and so the latest studies conducted have used the process data approach. Furthermore, current international standards set an upper limit on the maximum emissions that could be neglected due to excluded flows of materials.

However it is still instructive to review the results of recent studies providing detailed data for the NPP phase of the nuclear fuel cycle. Five studies have been selected among the most recent ones available. The selection was guided by the need to cover the main reactor technologies and the four following environmental impact indicators: global warming potential, acidification potential, photochemical ozone creation potential and eutrophication potential. Although many more indicators can be estimated, these four are particularly relevant for the NPP phase of the nuclear fuel cycle, and are amongst the most frequently reported. These are the five sources consulted:

1. A 2019 report from an EU utility company on the environmental impact of a fleet of 4 BWR and 3 PWR reactors [3.3.7-1];
2. A 2017 peer reviewed paper on the environmental impact of Canadian PHWR reactors [3.3.7-2];
3. A 2014 peer reviewed paper comparing the environmental impacts of two fuel cycle options in France (PWR fleet) [3.3.7-36];
4. A 2013 report covering the environmental impact of an AGR 2 reactor unit in the United Kingdom [3.3.7-3];
5. A 2005 peer reviewed paper analysing the GHG emissions of the Japanese fleet [3.3.7-5]. Although the publication date is much earlier than the other studies and its scope is limited to GHG emissions, it has been included for completeness, as it is the only one in this group which has used a mix of process-based and EIO approaches.

Three of the references are peer reviewed articles while the other two are reports from NPP operators, conducted according to established international standards (ISO 14040 and ISO 14044) following specific rules and guidance provided for the power generation sector, and independently reviewed by accredited certification bodies.

The literature review of life cycle assessments for nuclear power including the NPP construction, operation and decommissioning phases has yielded the following impacts per unit product. In all cases, the unit product is defined as 1 kWh net produced at the NPP.

Table 3.3.7-6. Environmental impacts from NPP phases according to different sources

#	NPP service lifetime (yr)	Phase	GWP(1)	AP(2)	POCP(3)	EP(4)
1	44 – 60(5)	Constr/Decomm	0.372	1.85	0.150	0.342
		Operation	0.124	0.925	0.064	0.257
2	n/a	Construction	1.03	4.1	0.33	0.52
		Operation	0.20	1.75	0.09	0.357
		Decommissioning	0.46	17.5	0.69	1.58
3	20 - 50	Constr/Operation/Decomm	2.140	2.89	0.151	0.760
4	35	Construction	2.63	13.4	0.644	4.45
		Operation	0.92	7.08	0.264	2.32
		Decommissioning	0.30	1.87	0.06	0.461
5(6)	30	Construction	2.8	n/a	n/a	n/a
		Operation	3.2	n/a	n/a	n/a
		Decommissioning	0.4	n/a	n/a	n/a

(1) Global Warming Potential, expressed in grams of CO_2-equivalent emitted per kWh produced

(2) Acidification Potential, expressed in milligrams of SO_2-equivalent emitted per kWh produced

(3) Photochemical Ozone Creation Potential, expressed in milligrams of C_2H_4-equivalent emitted per kWh produced

(4) Eutrophication Potential, expressed in milligrams of PO_4-equivalent emitted per kWh produced

(5) 44 years for Ringhals 1/2; 60 years for Forsmark 1/2/3 and Ringhals ¾

(6) Applicable to 1000MW BWR reactors representative of the Japanese fleet. A load factor of 70% has been assumed. For the construction and decommissioning, it includes the construction and dismantling of all fuel cycle facilities required.

It can be seen that all environmental impacts show lower values for light water reactors than for heavy water or gas-cooled reactors. On the other hand, the contribution from the operation phase is lower or much lower than those from construction and decommissioning, for all four indicators.

Some considerations can help understand the differences across the previous estimates:

— As the impact generated by construction and decommissioning is distributed over the lifetime production of the plant, the impact per kWh produced is directly dependent on the service life and load factors assumed. The first of these parameters ranges from 30 to 60 years, introducing a 100% variation on all the estimates due to this factor only. Regarding the load factor, although not all studies clearly identify the value selected, it can be assumed that a range around 70-90% has been used, introducing an additional variation, although less significant.

— Unlike the other cases, the study #5 has used a hybrid process-based and EIO approach, where process data was used for emissions caused by commodities and economic flows for other processes. As discussed earlier, the EIO approach tends to yield higher emissions.

— The electricity consumed during the commissioning phase is a significant contributor to the overall impact of the construction phase. Therefore, the energy mix considered (particularly the share taken by fossil fuels) has an important impact on the results, and energy mixes vary widely across countries, and over time within one country.

— In general, industrial processes have improved their environmental performance during the last decades. Furthermore, as more and more environmental studies are available, more accurate and specific data makes it possible to remove excessive conservatisms embedded in generic data sources. Therefore, recent studies tend to yield lower estimates for emissions, as compared to older ones.

- Beyond methodological differences, the estimates provided in the table cover different reactor technologies built at different sites, so that, to some extent, the variation across the studies correctly reflect the actual variability of the impacts.

3.3.7.4 Legal background and regulations

The most relevant international treaties and agreements for nuclear power plants are the following:

- Convention on Nuclear Safety, adopted in Vienna in 1994, entered into force in 1996, signed by 89 Contracting Parties (current status)
- Espoo Convention – Espoo Convention on Environmental Impact Assessment in a Transboundary Context (February 26, 1991);
- Aarhus Convention – Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters, Aarhus, Denmark, 25 June 1998

At EU level, the following directives are particularly relevant for the construction, operation and decommissioning of nuclear power plants in the EU:

- Nuclear Safety Directive (NSD) – Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014;
- Basic Safety Standards (BSS) – Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation;
- Note that the BSS are in-line with the current ICRP recommendations, which are of global validity (see the document "The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4), 2007");
- Radioactive Waste Directive – Council Directive 2011/70/Euratom establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste;
- Transport Directive – Council Directive 2006/117/Euratom of 20 November 2006 on the supervision and control of shipments of radioactive waste and spent fuel;
- Water Directive – Council Directive 2013/51/Euratom laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption.

The construction, operation and decommissioning of a nuclear power plant "is likely to have significant effects on the environment" and therefore the following environmental EU legislation is relevant for the activities described in this section:

- Environmental Impact Assessment (EIA) Directive – Directive 2014/52/EU of 16 April 2014 amending Directive 2011/92/EU on the assessment of the effects of certain public and private projects on the environment;
- Strategic EIA Directive – Directive 2001/42/EC of the European Parliament and of the Council on the assessment of the effects of certain plans and programmes on the environment;
- Air Quality Directive – Directive 2008/50/EC of the European Parliament and of the Council on ambient air quality and cleaner air for Europe;

In relation to the EIAs, the IAEA has developed specific guidance for the preparation of environmental impact assessments on the nuclear power sector:

- Managing Environmental Impact Assessment for Construction and Operation in New Nuclear Power Programmes, NG-T-3.11 IAEA Nuclear Energy Series, Vienna 2014
- Strategic Environmental Assessment for Nuclear Power Programmes: Guidelines, NG-T-3.17 IAEA Nuclear Energy Series, Vienna 2018

Companies operating in the various areas of civil nuclear energy (e.g. design, construction and operation of nuclear facilities, manufacturing of nuclear materials or components, extraction of raw materials, etc.) are obligated by law to obtain a certification according to internationally recognized quality, environmental, as well as health and safety management standards, such as

- ISO 9001:2015 – Quality Management System;
- ISO 14001:2015 – Environmental Management System;
- ISO 45001:2018 – Health and Safety Management at Work (replacing ISO 18001).

As a rule, the appropriateness of the internal governance of a civil company operating in a specific area of nuclear energy is proven by demonstrating that the company uses internationally recognized management systems to manage nuclear and industrial safety, radiation protection, technological & radioactive waste handling and environmental protection tasks during all phases of the activity concerned. The audit is carried out by an accredited body and repeated periodically.

3.3.7.5 Identification of applicable means to avoid or mitigate the impacts

3.3.7.5.1 NPP construction

(a) Building materials

The constructor of a nuclear power plant has many alternatives to avoid or mitigate the impacts mentioned earlier. All of these measures are common to any infrastructure project of similar size, and are therefore available on many economic sectors. For instance, reference [3.3.7-10] cites the following means, among others:

- Choose concrete, steel and other material providers with low life cycle environmental footprint (using more energy-efficient manufacturing processes, more easily recyclable, with a longer life, etc.), as much as possible
- Use recycled materials as much as possible, such as steelwork and plastics, thus reducing the need to extract raw materials;
- Use materials that have been manufactured from rapidly renewable resources in the design, thus reducing the depletion of finite resources and resources that have a long regeneration cycle;
- Maximise the local procurement of materials to avoid energy use during transport;
- Crush spoil concrete and other demolition or spare material for road sub-base or similar usage;
- Ensure that materials made from composite wood materials do not contain urea-formaldehyde resins;
- Use paints, stains, varnishes, sealants, adhesives with reduced levels of volatile organic compounds;
- Avoid use of tropical hardwoods;

(b) Fuel and electricity

The economic incentive to minimise the cost of fuel and power required during the construction phase will normally help reduce also the environmental impact.

(c) Water environment

Dredging operations can be conducted on the designated areas where suitable environmental studies have been completed, and monitoring of water turbidity can be ensured during dredging.

(d) Groundwater

If groundwater level has to be lowered, hydrological studies can be conducted to identify any risk of mixing the groundwater with other sources, and to verify that surface courses will not be impacted by the lower groundwater level.

(e) Road traffic

Usual conventional measures can be taken to minimise the impact of road traffic. For instance, low speed limits, car-pooling or bus systems for staff, or staggered shift changes.

(f) Dust and noise

Conventional building sector best practices are available to minimise the impacts, for instance:

- Cover open body trucks carrying materials prone to dust generation;
- Cover bare earthy areas with vegetation as soon as possible;

— Plan explosions and highly noisy activities at certain times of the day;

— Install noise insulation barriers at specific locations, if needed.

All these impacts are of course site specific, and must therefore be addressed individually for each project under the framework of an Environmental Impact Assessment process. The IAEA has published guidance specifically addressing the environmental impact assessments for the construction and operation of nuclear power plants [3.3.7-44], [3.3.7-45].

3.3.7.5.2 NPP operation

(a) Airborne emissions

For road traffic emissions, the same mitigation measures already discussed for the construction phase are applicable during operation.

Regarding emergency diesel generators emissions, the critical safety function of this equipment must remain the overriding priority. Although emissions abatement techniques exist in the market, given the short period of operating time, and the potential negative effect of these techniques on safety, the convenience of their implementation is questionable.

(b) Water withdrawal and consumption

The total water needs can be optimised by choosing the appropriate cooling water system design for the site.

The location and detailed design of the intake and discharge areas can minimise the impact of the water withdrawal. Further protection to prevent the entrainment of fish can be provided by using fine mesh nets in the vicinity of the intake, or by using a variety of fish repellents.

Variable speed drives on cooling water pumps can adjust the flow to what is needed at specific operating conditions, reducing water withdrawal needs at stations subject to wide ambient temperature variations, wide tidal variations or partial load following.

Closed cooling plants can use water treatments to decrease the blowdown flow required, thus reducing the total water consumption.

More far-reaching improvements in water management can be achieved by using waste water as primary input for cooling needs. The Palo Verde NPP (nearly 3 937 MWe net), located at the Arizona desert, uses 100% reclaimed water for cooling and other uses (industrial water and potable water), obtained from Phoenix's waste water treatment facilities and conveyed through a 60 km pipeline to the site (by mixed gravity/pumping). Thanks to the cooling towers and evaporating ponds, the plant is operated as a zero-discharge facility [3.3.7-37].

(c) Waste water and other non-radioactive hazardous wastes

All NPPs collect waste water flows from non-controlled premises (no radiation control) into sewer systems through floor drains and other drains and direct them to the waste water treatment plant. Oil separators are used for waste water flows coming from locations where water could be potentially contaminated with oil. Chemicals are stored in containers and labelled accordingly. Storage areas containing oils or chemicals are drained to shielding pools, thus preventing accidental releases of hazardous substances.

(d) Atmospheric and liquid radioactive effluents

Radioactive releases to the atmosphere are subject to legal limits, set in agreement with international guidance so that radiation will not result in any harm for the population or the environment. Utilities continuously monitor the effluents and report the data obtained to the regulatory authority.

In all nuclear power plants, radioactive gases generated at the plant are processed by cleaning systems, where the gases are dried, delayed or filtered using e.g. active carbon filters or HEPA (High Efficiency Particulate Air) filters. Gases are released through vent stacks, and monitoring systems continuously measure the radioactivity levels at each stage of the process.

The total amount of radioactivity released to the marine or fluvial environment, mostly in the form of tritiated water, can be minimised by recycling the contaminated water, or by removing radioactive solids present in the water using mechanical (filters or centrifuges) or chemical processes.

Airborne releases have strongly decreased over time, as shown by [3.3.7-47] for the US fleet for the period 1975-2005. This study attributes the reduction to the improved performance of fuel cladding and waste and effluent control systems, as well as increased holdup times (for the short-lived radionuclides). In the case of liquid emissions (mostly dominated by tritium), the activity released has been overall stable during the period, although with significant variations depending on the specific site.

Many techniques are available to the nuclear power plant operators to minimise the radioactive releases, and a full description would go beyond the scope of this report. See for example [3.3.7-42] for a comprehensive description of best available techniques.

Furthermore, as mentioned for the case of construction, all these impacts must be addressed individually by an Environmental Impact Assessment. The IAEA has published guidance specifically addressing this issue for the operation of nuclear power plants [3.3.7-44, -45].

3.3.7.5.3 NPP decommissioning

The European Commission, the International Atomic Energy Agency (IAEA) and the OECD Nuclear Energy Agency (NEA) have issued a large number of guidelines, recommendations and best practices covering all aspects of decommissioning and waste management.

Several best practice guides are included in the IAEA Safety Standards Series:

- The Basic Safety Standards document [3.3.7-20] is the reference document for radiation protection and has fed the Council Directive 2013/59/Euratom [3.3.7-21];
- Decommissioning of Facilities [3.3.7-22] and its ancillary documents [3.3.7-23, -24, -25] lay down basic principles for managing a programme of decommissioning of nuclear installations;
- Document [3.3.7-26] sets fundamental principles to regulate the controlled discharge of radioactive elements in the environment.

Procedures, methods and best practices that are applicable to avoid or mitigate the identified harmful impacts of decommissioning are also addressed in numerous IAEA technical reports [3.3.7-30] - [3.3.7-32].

Also highly relevant is the guidance related to Environmental Impact Assessment for decommissioning in Ref [3.3.7-27]. Competent authorities of Member States may also benefit from the technical guidance offered by the European Atomic Energy Community, when establishing clearance levels. Guidance on the recycling or reuse of metals [3.3.7-33], guidance on buildings and building rubble [3.3.7-34] and guidance on general clearance levels for practices [3.3.7-35] have been published.

Some examples of mitigation and preventive measures aimed at preventing, eliminating or minimizing potential effects induced by the decommissioning activities on atmosphere and climate, soil and subsoil, groundwater, surface water and biodiversity are:

- Adoption of suitable technical, operational and management measures to contain as much as possible the production of dust during the deconstruction phase;
- Adequate protection of debris accumulation from atmospheric agents in uncovered storage areas and reduction of parking time to the minimum possible;
- Use of additional systems specifically aimed at reducing dust (fog cannon, dust-buster, etc.);
- Monitoring of air quality and climate, according to the Environmental Monitoring Plan;
- Use of operational means and means for transport of materials that ensure reduced emissions of climate-changing gases;
- On-site recovery (after characterisation) of inert materials arising from demolition to be used for the morphological restoration of the site, and in particular for the filling of the cavities originating from the demolition of underground civil structures, so as to limit the production of solid waste;
- Location of all temporary storage areas in already paved zones;
- Limitation of noise emissions by using approved machinery, subject to regular maintenance, or by adoption of screens for the engines in case of noise values above the limits.

3.3.7.6 *Evaluation and summary*

The potential environmental impacts of nuclear power plant construction, operation and decommissioning have been reviewed. During construction, the main issues are related to the mining and processing of the materials required to build the facility (mostly steel and concrete), and to the offsite power required during this phase (particularly during the commissioning tests). Once the plant starts operation, the release of radionuclides to the atmosphere and to the water environment need to be addressed, together with the ecological impact of the withdrawal and consumption of water required to ensure plant cooling. In the case of decommissioning, liquid and airborne effluents are expected to be of very little significance, and other non-radiological impacts and their mitigations are similar to conventional demolition works. As for the solid radioactive waste, its management involves treatment or conditioning into passively safe forms, interim storage and, where waste routes exist, disposal.

All the potential impacts are well known, have been extensively studied and have resulted in comprehensive national and international regulatory frameworks, as described above. The enforcement of regulations and limits contained in these frameworks by the relevant regulatory authorities, together with the environmental management systems put in place by the operators, have resulted on the improvement of the environmental performance of NPPs, in particular regarding the radioactive airborne effluents during plant operation.

A particularly relevant requirement of this regulatory framework is the need to conduct an Environmental Impact Assessment for each nuclear plant project, covering the construction, operation and decommissioning phases. Current legislation regulates the contents of the assessment, and it also contains provisions to ensure a comprehensive participation in the process of all stakeholders involved, including citizens and neighbouring countries. Furthermore, international guidance specific for the preparation of environmental impact assessments of nuclear power plants (from construction to commissioning) has been recently published by different organisations and is currently available.

A literature review of recent available life cycle assessments (LCA) has been conducted, including both peer reviewed papers and certified environmental product declarations conducted by NPP operators. Although the comparison of results among different studies is possible only to some extent, the estimates given by different authors for a number of indicators (global warming potential, acidification potential, photochemical ozone creation potential and eutrophication potential) are generally consistent among each other and with previous estimates.

The results of the LCA surveyed show that the operation of the plants represent a limited fraction of the total environmental impact. For this reason, the life extension of NPPs tends to reduce the environmental load, as the impacts from construction and decommissioning can be distributed over a larger lifetime production, with only marginal increases due to the investments associated to the life extension process.

For all risks identified, the industry has a number of proven control and mitigation measures available to monitor and minimise the impacts. The following table qualitatively ranks the impact indicators (radioactive and non-radioactive) according to their importance in the context of NPP construction, operation and decommissioning, and summarises some of the main mitigation measures available.

Provided that nuclear power plants are built, operated and decommissioned within the limits set by existing regulations, they do not pose a significant harm to any of the TEG objectives.

In the light of the above analysis it can be concluded that NPP operation activities[97] do not represent unavertable harm to human health or to the environment. They do not represent significant harm to any of the TEG objectives, provided that the associated industrial activities satisfy appropriate Technical Screening Criteria.

TSC for the electricity generation from nuclear energy are provided in Chapter 5 and Annex 4 of the present report (*Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy*).

[97] Note that the "NPP operation" lifecycle phase includes the construction, operation and decommissioning of nuclear power plants, as well as the long-term operation of these facilities

Table 3.3.7-7. Importance of NPP operation impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		Prevention or mitigation of potentially harmful impacts	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	++	Selection of materials with low life cycle environmental footprint, recycling materials	Mostly due to materials used in construction (steel and concrete). Therefore LTO tends to reduce the impact.
Water withdrawal	++++	Site-specific design of cooling systems for water use optimisation	The impact arises almost entirely during the operation phase
Water consumption	++++	Site-specific design of cooling systems for water use optimisation	The impact arises almost entirely during the operation phase
Water pollution	++	Site hydrological studies, waste water systems	-
Ecotoxicity	++	Appropriate selection of materials, recycling	Mostly due to materials used in construction (steel and concrete). Therefore LTO tends to reduce the impact.
Human toxicity	++	Appropriate selection of materials, recycling	Mostly due to materials used in construction (steel and concrete). Therefore LTO tends to reduce the impact.
Land use	+	-	-
Atmospheric pollution	++	Conventional building best practices during construction, road traffic optimisation	Mostly due to materials used in construction (steel and concrete). Therefore LTO tends to reduce the impact.
Acidification pot.	++	Appropriate selection of materials, recycling	-
Eutrophication pot.	++	Appropriate selection of materials, recycling	-
Ozone creation pot.	++	Appropriate selection of materials, recycling	-
Production of TW	+++	Selective waste management	-
Depletion of resources	+	-	-
Production of solid RW	++++	Stabilization and capping	
Gaseous RA releases	++++	Filtering and cleaning systems, continuous monitoring of emissions	Mostly ^{14}C, tritium and some noble gases
Liquid RA releases	++++	Recycling of contaminated water, retention before release, filtering and cleaning systems	Mostly tritiated water.

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

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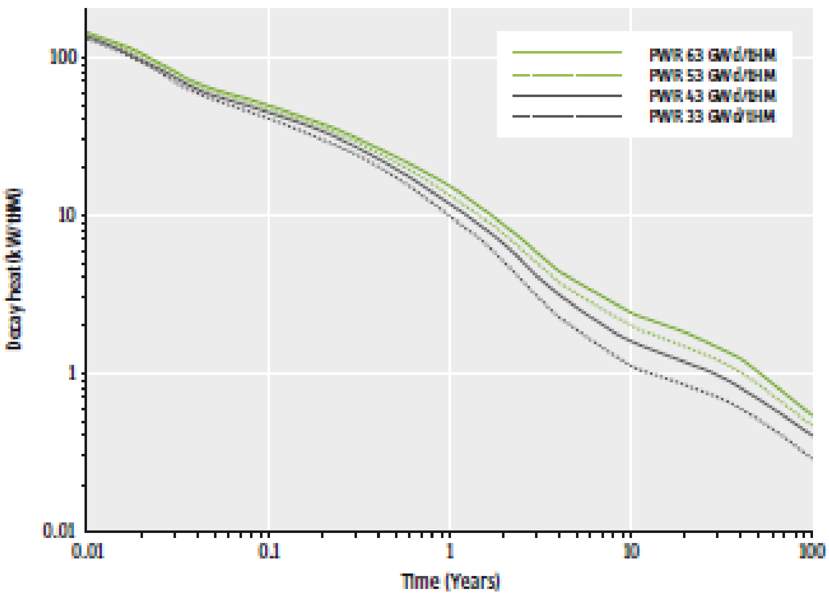
3.3.8 Impact of storage and disposal of radioactive waste, spent nuclear fuel and technological waste

This section is focused on the impact of the facilities dedicated to radioactive waste management as part of the lifecycle assessment of nuclear energy, using the facilities for the management of spent fuel as main reference for description. Part B of the present report provides a more extensive illustration of the typology, characteristics and inventories of radioactive waste generated during the various phases of the nuclear energy lifecycle, and the options and technologies considered for its safe management and disposal.

3.3.8.1 Main stages of spent fuel and high level waste management

At the end of its operating life in the core of the nuclear reactor, irradiated or spent fuel is still "hot" due to the residual power generated by the decay of short-lived radionuclides formed during irradiation. The residual power at shutdown is approximately 7% of the nominal power during reactor irradiation, and decays relatively rapidly (Figure 3.3.8-1).

Figure 3.3.8-1. Decay heat of PWR spent fuel with different burnup



Source: [3.3.8-1]

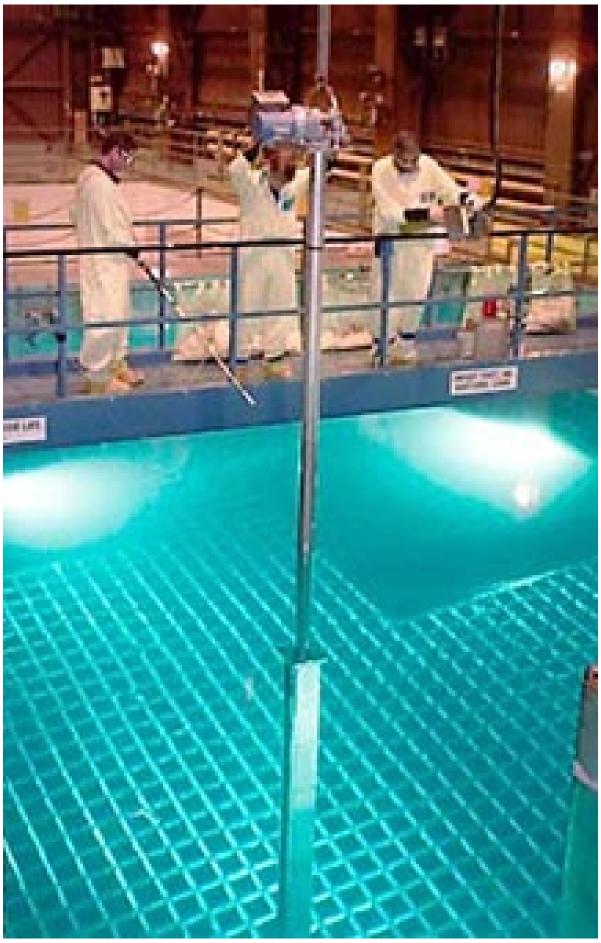
In order to avoid overheating, the fuel assemblies are handled under water (Figure 3.3.8-2). After their use in the reactor core, spent fuel assemblies are typically stored in racks in a decay or cooling pool located in the reactor or in the auxiliary building (see also Chapter 3.3.7.2.2). The water of the pool provides multiple functions:

- Decay heat removal (cooling). The water is cooled and recirculated by pump systems; typically T ≤ 35°C, T_{max} ~50°C.
- Radiation shielding. A few meters of water are enough to minimize both the gamma and neutron radiation from the spent fuel; a water layer ~7 cm thick provides a dose reduction of 50%. Shielding is

normally ensured by providing a minimum of 4 m depth of water above the fuel elements in storage, which is enough to reduce the dose rates to less than 0.01 mGy/hour at the pool surface. Typically 4 m long LWR fuel assemblies are stored at the bottom of decay pools at least 12 m deep.

— Protection against criticality accidents by adding neutron absorbers to the water, e.g. boric acid; depending on the “crowding” of spent fuel assemblies in the pool, solid absorbers (e.g. containing boron-10) can also be inserted in the racks.

Figure 3.3.8-2. Spent fuel management underwater at a cooling pool



Source: U.S. Nuclear Regulatory Commission (NRC)

The water content is controlled: the water is processed by means of ion exchange and filtration to remove eventual contaminants and unwanted impurities that may affect the corrosion resistance of the zircaloy cladding. Hydrogen production by radiolysis is monitored through atmosphere control and managed by ventilation.

Spent fuel must be kept under water until the decay heat is reduced to a level allowing the fuel to be cooled by air, i.e. at least 1 year; typically, it is kept in wet cooling for a few years (e.g. ~5 in Europe, ~10 years in USA). The actual duration of the storage in the decay pool at the NPP can in some cases be extended, depending on the availability of suitable facilities for the subsequent steps of the cycle. The spent fuel assemblies are then loaded into canisters, dried and packaged into casks which provide adequate shielding and isolation, and are suitable for transportation or for transportation and storage (dual purpose casks). Depending on the fuel cycle back-end strategy selected, the loaded containers follow different pathways:

— in case of Pu-U recycling, the transport containers are shipped to the reprocessing facility (in Europe, the Orano facility in La Hague, France or, until recently, the Sellafield facility in the UK), where they are stored in pools awaiting treatment (Figure 3.3.8-3(a));

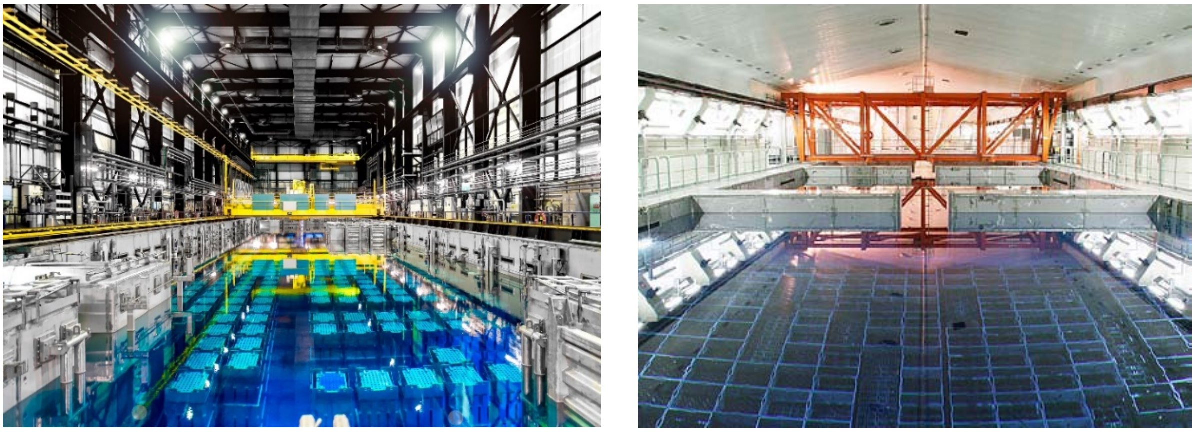
— in case of an open cycle and direct disposal, single or dual purpose containers are brought to interim storage facilities in view of final disposal in a geologic repository.

Interim storage can occur at centralized storage locations or in decentralized facilities (typically, at the nuclear power plant site). It can be "wet storage", in dedicated pools (Figure 3.3.8-3(b)), or "dry storage", in storage or in dual purpose containers. In Europe, interim storage occurs indoors, in dedicated buildings, or vaults (Figures 3.3.8-3 and 3.3.8-4(a)). Outdoor interim storage is limited in Europe (Figure 3.3.8-4 (b)) but is extensively used in the USA.

Figure 3.3.8-3. Spent fuel wet storage

(a) Wet storage at the Orano reprocessing facility in La Hague, France

(b) Interim storage pool at the SKB Clab facility, Sweden



Source: https://www.orano.group/en/nuclear-expertise/orano's-sites-around-the-world/recycling-spent-fuel/la-hague/unique-expertise

Source: [3.3.8-2]

Figure 3.3.8-4. Spent fuel dry storage

(a)

(b)

Source: Ignalina NPP

Source: ensa.es

Similar considerations can be made for the interim storage of vitrified high level waste from spent fuel reprocessing [3.3.8-3]. A comprehensive assessment of the environmental impact associated with the whole life cycle (including waste management and disposal) of nuclear energy in France, comparing the fuel cycle with reprocessing and the open fuel cycle is available in [3.3.8-4].

The main concepts implemented for cooling vitrified waste canisters include air-cooled vaults, pools, and shielded sealed casks. In all cases, the waste is conditioned in standard stainless steel containers. In the case of dry storage vaults, cooling can be by natural convection or forced circulation of air. Natural convection is generally preferred because of its simplicity. However, if the configuration of the vault makes air natural convection difficult, or if extraction filters are to be used, forced ventilation is employed. Essentially, the choice between dry storage vaults and pools depends on the total heat extraction capacity and the thermal power of each HLW canister. Insofar as the cooling time between the unloading of the spent fuel and the vitrification of HLW has steadily increased, air-cooled vaults are preferred; they are currently in operation notably in France, the United Kingdom, Japan, Belgium, and the Netherlands [3.3.8-5].

Figure 3.3.8-5. Vitrified waste canisters and interim storage [3.3.8-5]

(a) Canister for vitrified HLW (H~1.3 m, Ø~0.43 m)

(b) Vitrified waste interim storage room

Source: COGEMA – France

In the case of the open cycle and direct disposal, the purpose of the interim storage stage is to allow the spent fuel to cool down further and to reach a residual power level compatible with the actual geologic disposal conditions. Interim storage of spent fuel, as well as of vitrified high level waste from reprocessing, is thus a necessary step in the management of this type of radioactive waste. Immediate disposal of spent fuel /vitrified waste in a geologic repository is not a viable option.

The typical envisaged duration of interim storage is a few decades. For instance, interim storage of spent fuel dual purpose casks in Germany is licensed for 40 years; the interim storage duration will have to be extended since operation licences of the storage facilities will expire between 2034 and 2047, and the disposal repository will not be available before 2050. In Spain, the interim storage facilities are licensed for 20 years. In France, vitrified HLW packages will require a minimum storage time of 60 to 70 years, depending on the specific decay heat [3.3.8-6].

After the interim storage, the spent fuel containers are transported to a waste encapsulation facility; here the spent fuel assemblies are retrieved from the transportation/storage canisters and repackaged into smaller sealed disposal containers. The disposal containers are transported into the final repository and emplaced in the excavated location.

3.3.8.2 Facilities for spent fuel management

The typology of the facilities associated with the back end of the nuclear fuel cycle includes the interim storage facility, the spent fuel encapsulation plant and the geologic repository for final disposal.

All the above facilities are subject to a licensing process, which includes the assessment of their environmental impact. The applicant must demonstrate that the facility complies with the relevant regulatory requirements set by the national safety authority and that it will not generate any significant environmental or health consequences in the future.

The generic radioprotection criteria that have to be fulfilled by nuclear facilities apply also in this case: safe containment, minimization of radiation exposure, subcriticality and adequate decay heat removal will have to be ensured. In addition, storage facilities must provide long term surveillance, maintenance of the facility and management of the stored objects (e.g. traceability and mapping of the packages, radiological inventory, retrievability, etc.).

Site characterization at the envisaged facility location is also performed to provide the background conditions that the new facilities will impact. This includes radiological measurements, environmental assessment of ecosystems, flora and fauna, cultural description, population density, industrial presence, roads and frequency of transportation [3.3.8-2].

To a large extent, the impact from the construction, operation and decommissioning of the buildings and the installations of these facilities is similar to that described in the preceding sections for other stages of the nuclear energy lifecycle (see in particular Chapter 3.3.7), especially regarding the non-radiologically relevant components (buildings, concrete, roads and infrastructure for transportation, etc.). Moreover, wet interim storage facilities can be considered as standalone long term cooling pools, differing from the cooling pools at the reactor site (see Chapter 3.3.7.2.2) only in terms of size of the facility and age (hence radioactivity level) of the spent fuel stored therein.

In the following, the Environmental Impact Statement for the Swedish interim storage, spent fuel encapsulation and final disposal facilities [3.3.8-2] is frequently used as a reference, as it provides a good example of an integrated assessment exercise. The aspects considered are listed in Table 3.3.8-1.

Table 3.3.8-1. Main components of the SKB Environmental Impact Statement

Impact	Effects and consequences	Risk and safety
Land use	Natural environment	Non-radiological risk
Impact on groundwater level	Outdoor activities and recreation	Radiological risk during operation
Noise and vibration	Cultural environment	Long term safety
Radiation and radionuclide releases	Landscape	
Emission of other substances to air	Residential environment and health	
Light pollution		
Resource consumption		

Source: [3.3.8-2]

3.3.8.3 Impact of the spent fuel and high level waste storage facilities

Direct exposure to gamma and neutron radiation potentially occurs in the immediate vicinity of any storage facility. The presence of adequate shielding and the distance between the waste package and the public ensure that the relevant radiation exposure limits for personnel and the general public are satisfied.

Dry storage

On-site or centralized dry storage facilities do not generate any release of radioactive substances, since the spent fuel is contained in sealed canisters. Discharges to the air or release into water are negligible, due to the leak-tightness criteria for storage casks and the existing rules for surface contamination on the outside of the casks, which do not allow transportation of a surface-contaminated cask outside of the controlled area of the nuclear power plant.

Discharges of any contaminated waste water and/or liquid, which exceeds the maximum allowed activity concentrations specified in the relevant radiation protection regulations, e.g. from routine maintenance or decontamination work on the containers, are not allowed. Such effluents are transferred to dedicated liquid waste treatment facilities for conditioning and disposal.

Wet storage

As in the case of the cooling pools at the reactor sites, there are both gaseous and liquid (water) radioactive releases, occurring under strict monitoring and far below the legal limits. Such emissions are not judged to give rise to any health consequences for nearby residents. Gaseous emissions, e.g. the air from the pool and the controlled area, pass through particle filters, which reduce the radioactive emissions, and leave the facility via the ventilation stack, where monitoring devices continuously measure the radioactivity. The experience from the operation of wet storage facilities shows that releases of airborne activity from the pools are essentially undetectable [3.3.8-2].

Similarly, water from the controlled area is purified by filters and ion exchange resins, and the radioactivity content of the water is checked prior to each discharge.

Solid radioactive waste in the form of protective clothing, ion exchange resins, etc. is collected, characterized, conditioned and packaged for transportation to the disposal repository.

In the case of the Swedish centralized wet storage facility Clab (Figure 3.3.8-6, [3.3.8-2]), in operation since 1985, it is possible to consider actual data from the operation of the facility (e.g. measured activity levels in the pools and in the waste collected annually in the different clean-up systems) and compare it with the limits set in the original operating licence, which were based on calculations from the preliminary safety analysis report. The measured amounts of radioactivity that are collected in the pool and in clean-up systems, are far below the values established in the licensing calculations. During cooling of the transport casks, the actual uptake of ^{60}Co in the filters for different years varies between 0.1 and 1.7 GBq/t of uranium, while the allowed limit is ~500 GBq/t U. The ^{60}Co activity concentration measured in transport casks is ≤ 5 GBq/m^3 water compared to a limit of 145 GBq/m^3 water in the licensing. In the reception pools, ^{60}Co actually collected in the cooling and clean-up system varies in the range 1-6 GBq/t U, compared to a limit of 120 GBq/t U in the licence. Similar considerations apply to the collective dose to the operators: during the period 1985–2009 it varied between 18 and 135 mmanSv / year, well below the allowed limit of 276 mmanSv / year. During the period 2003–2009, an average of about 1 700 cubic metres of purified process water was discharged from Clab. The mean values of the annual radioactive discharges from Clab to the receiving water body during the period 1996–2009 are listed in Table 3.3.8-2.

Table 3.3.8-2. Average yearly radioactive water discharges from Clab during the period 1996–2009

Radionuclide	Average release, Bq/year
Tritium	$2.6 \cdot 10^9$
^{54}Mn	$3.9 \cdot 10^6$
^{58}Co	$1.3 \cdot 10^6$
^{60}Co	$3.8 \cdot 10^8$
^{90}Sr	$2.6 \cdot 10^5$
^{134}Cs	$2.7 \cdot 10^6$
^{137}Cs	$5.5 \cdot 10^7$
^{238}Pu/^{241}Am	$2.9 \cdot 10^4$

Source: [3.3.8-2]

Between 2003 and 2009, Clab gave rise to an average of 37.8 metric tons of radioactive waste per year. The radioactive operational waste is handled and packaged for further transport to the LILW-SL geologic repository SFR. The low level waste is taken to the near-surface repository for low level waste (MLA).

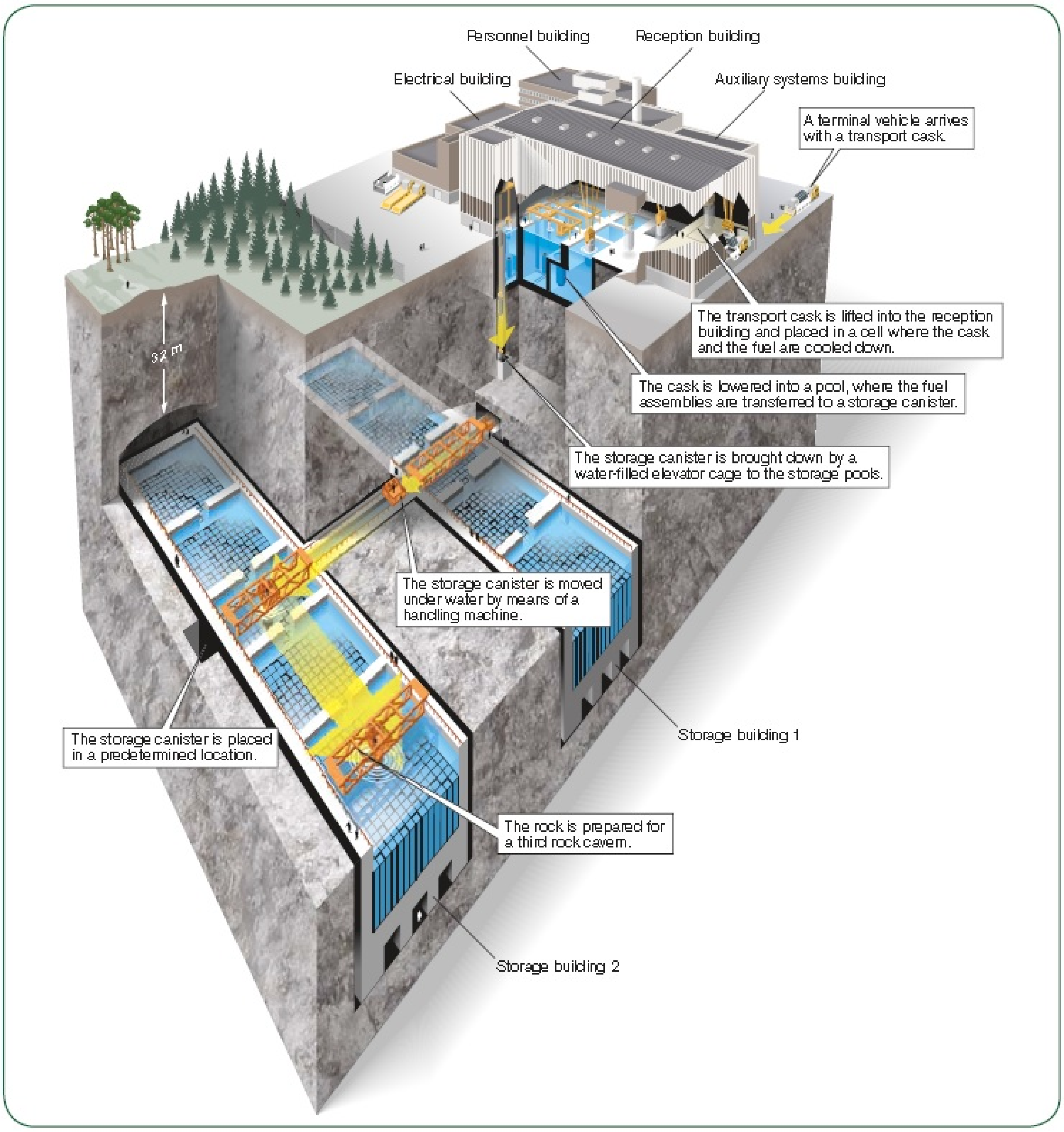
Non-radioactive releases also occur. In the case of Clab (Figure 3.3.8-6), the spent nuclear fuel assemblies and spent core components are transported from the nuclear power plants enclosed in special transport casks, by sea to nearby harbour and overland on specially built vehicles ([3.3.8-2]). Moreover, heated water that has been used to cool the facility is discharged into the sea together with cooling water from the Oskarshamn Nuclear Power Plant (see Chapter 3.3.7.1.2). The water from Clab constitutes ~0.1% of the total discharge.

Since the Clab facility is located underground, the groundwater that flows into the rock caverns hosting the storage pool has to be pumped and discharged into a neighbouring bay. The local drawdown of the groundwater caused by the facility is limited in scope and extent and has not caused any consequences for the natural groundwater levels e.g. in wells. Both the water in the cooling system and inflowing groundwater do not contain radioactive substances [3.3.8-2].

Between 2003 and 2009, Clab typically sent 10.4 metric tons of waste per year for recycling or reuse, while 17.3 metric tons of waste was sent for disposal, incineration or biological treatment. A total of 4.5 metric tons of hazardous waste was sent for recycling, incineration or treatment.

Additional elements included in the environmental impact assessment are the land use, the visual impact and the noise generated by the operations of the facility. In the case of Clab, in operation since 1985, the visual impact and land use are somewhat limited due, respectively, to the surrounding forest and to the fact that the pools are located underground. The operation noise level is low and is not judged to cause any consequences to the local population.

Figure 3.3.8-6. Schematic layout of the spent fuel centralized interim storage facility Clab, Sweden



Source: [3.3.8-2]

Between 2003 and 2009, energy use at Clab was on average 16–17 GWh / year. Between 2005 and 2009, the water consumption of the facility was in average 14 300 m^3 / year.

3.3.8.4 Impact of the spent fuel encapsulation facility

At the end of interim storage the spent fuel will have to be retrieved from storage and packaged, in an encapsulation plant, in the container that will be emplaced in the geologic repository for final disposal. In the Swedish and Finnish cases, the spent fuel assemblies will be encapsulated in welded copper canisters. The encapsulation plant can be located at the site of the final repository, or, especially in case of centralized interim storage, at the storage site.

In the Swedish case, the encapsulation plant will be built at the site of Clab. Despite the handling of large volumes of fuel and of many transport casks, the radioactivity level per fuel assembly will be significantly lower in the encapsulation plant than in Clab. This is because the radioactivity of the fuel will have decreased during storage. Adequate radiation shielding will nevertheless be required during all handling stages. After encapsulation, the spent fuel will no longer be a potential source of airborne radioactivity. The loaded copper canisters will be transported to the repository site by ship.

The radioactive emissions to air and water are expected to be far below the legal limit. Similar to the case of cooling and storage pools, in the encapsulation plant the radioactive species released into water during spent fuel handling in the pools will be collected on filters and ion exchange resins in a water purification system. Moreover, gaseous emissions from the encapsulation plant will be minimized by filtration stages in the ventilation system. Airborne emissions through the chimney will be monitored continuously.

The co-location of Clab and the encapsulation plant will generate interesting synergies: the waste from the encapsulation plant will be managed in the same way as for Clab; the water purification systems will be shared between the facilities; moreover, the cooling water in Clab will provide the heat for heating the encapsulation plant, thus reducing the corresponding emission (however, in the summer the excess thermal energy will be discharged to the sea). The contribution from Clab and the encapsulation plant to the heat discharge in the local fjords will be marginal, compared to the discharge from the nuclear power plant nearby.

Considering other non-radiological impact, the land needed for the new plant and for temporary construction areas is approximately 30 000 m^2. Building some noise barriers will ensure that the construction noise will not exceed the guideline value at the nearest homes. Thanks also to noise suppression measures, noise and vibrations from operations and from shipments to and from the plant will not exceed the limits. However, noise due to road transport during construction of the plant is likely to exceed occasionally the limits for approximately 40 local inhabitants during daytime working hours; peak sound levels will occur during periods when high numbers of heavy vehicles are transiting.

Conventional atmospheric emissions that occur from Clab and the encapsulation plant (including transport emissions) are not expected to generate any significant risk to health or exceed the environmental quality standard for air. Sea transport of fuel-filled canisters to the final repository will be the predominant source of atmospheric emissions.

Approximately 44 000 metric tons of copper will be consumed in the encapsulation of the spent nuclear fuel over a 40–50-year period, which can be compared with the annual global production of copper of 15.5 million metric tons.

When nuclear power will have been phased out and all spent nuclear fuel and other high-level waste will have been transferred to the final repository, both storage and encapsulation facilities will be decommissioned. In the Swedish case, it is estimated that decommissioning can begin around 2070.

3.3.8.5 Impact of the spent fuel, HLW final repository

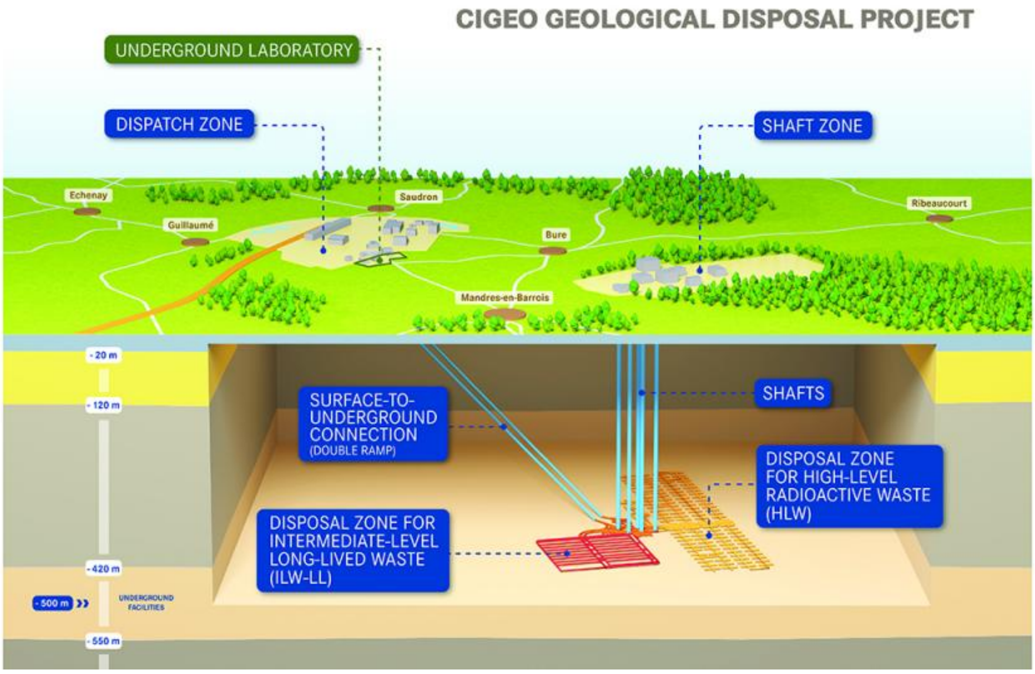
The final disposal of spent fuel and vitrified waste (HLW) will occur in a deep geologic repository. The basic purpose of the repository is to ensure that no harm will be caused to humans and the environment due to radiation from the spent fuel/HLW until the radiotoxicity of the waste has decayed sufficiently. Since the relevant time spans are of the order of some hundred thousand years or more, exceeding human civilization records, the repository is designed to fulfil its safety function without the need for active human monitoring, control and intervention. A more detailed description of the concepts, technological approaches and specific features of the deep geologic repository are provided in part B of the present report.

Here the focus will be to describe schematically the environmental impact and the main requirements associated with constructing and operating a repository.

The most advanced repository concepts in Europe envisage the emplacement of HLW in granite (Finland, Sweden) or clay (France, Switzerland) formations. All repositories will consist of a surface part and an underground excavated part. Figure 3.3.8-7 shows the schematic of the French geological disposal repository CIGEO, to be built at a depth of approximately 500 m for the disposal of HLW and also Long-Lived Intermediate Level Waste (ILW-LL) in clay formation. Figure 3.3.8-8 shows the layout of the Swedish final repository [3.3.8-2] to be constructed in Forsmark and Figure 3.3.8-9 illustrates more in detail the emplacement concept of the Finnish repository under construction at Olkiluoto. Both these repositories are based on the Swedish KBS-3 concepts and present many similarities.

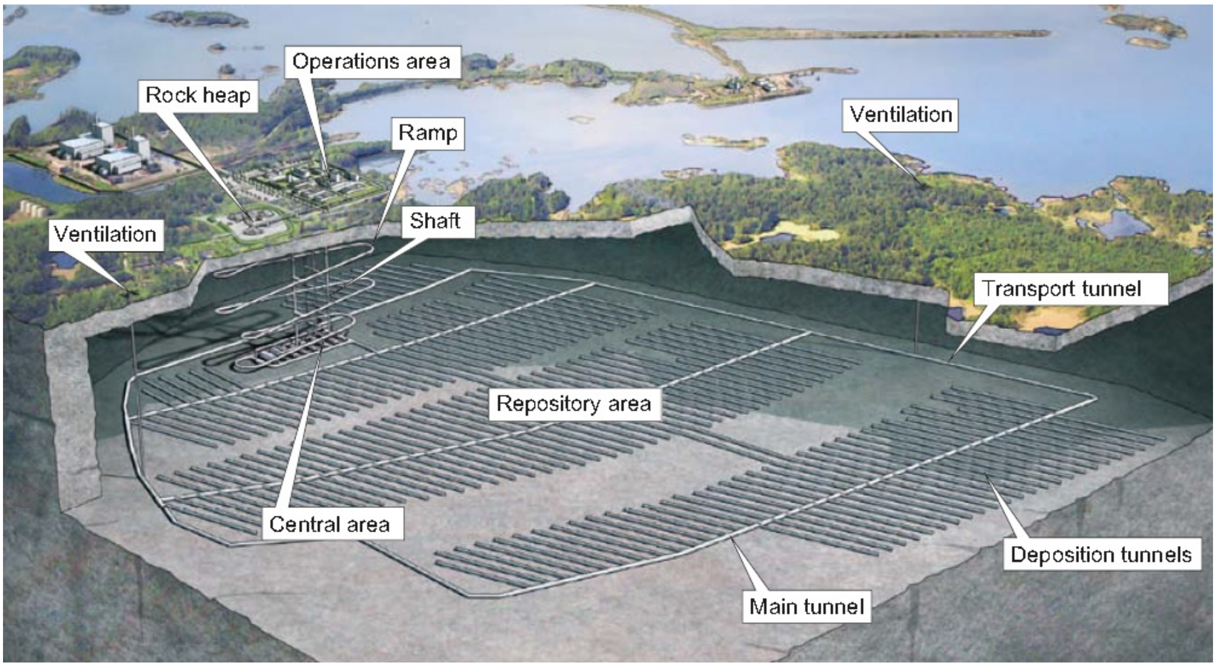
Figure 3.3.8-7. Schematic illustration of the French geological disposal concept in clay CIGEO for HLW and long-lived ILW.

The underground repository area, consisting of main tunnels and deposition tunnels with deposition holes in which the HLW canisters will be emplaced. The surface and underground parts will be connected by ventilation and elevator shafts, plus a ramp for vehicle transport.



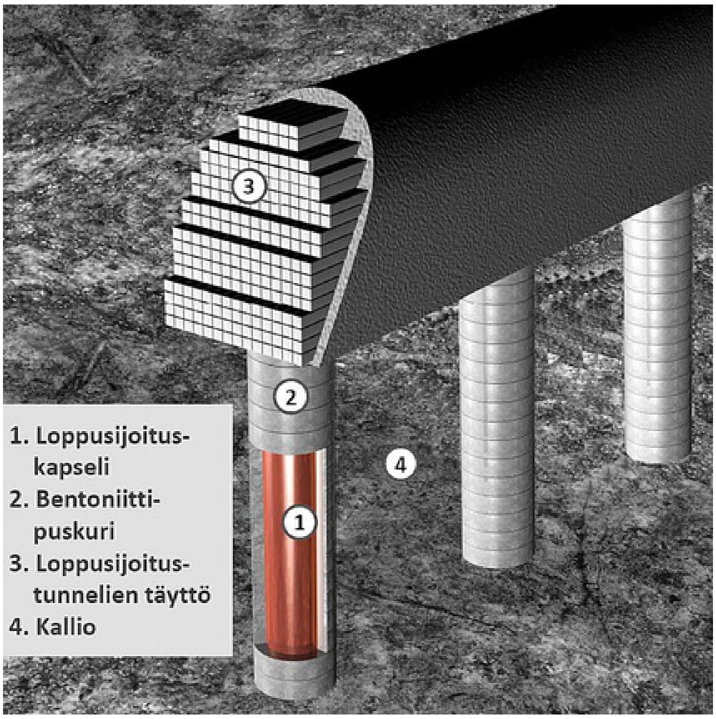
Source: [3.3.8-7]

Figure 3.3.8-8. Layout of the Forsmark Swedish final repository in granite. It is estimated that the repository's tunnels will occupy an area of 3–4 square kilometres at a depth of about 470 metres



Source: [3.3.8-2]

Figure 3.3.8-9. Detail of the multi-barrier disposal concept in the Finnish repository in Olkiluoto



1) Canister emplaced into vertical deposition holes at a depth of 420–430 m, and

2) Surrounded by bentonite clay buffer.

3) Deposition tunnel backfill made of granular bentonite and plugged at the tunnel end with high strength low pH concrete plug.

4) Host rock consisting mainly of mica gneiss and pegmatitic granite

Source: Posiva

In the case of the Swedish repository, the construction of the facility will last 7 years and employ 300–400 workers. Approximately 1.6 million metric tons of rock spoil will be excavated during the construction phase. The rock spoil will be temporarily stored in a rock heap within the industrial area. It is believed that the excess rock material not needed in the project can be sold in the region.

The operating phase of the Forsmark repository will consist of a trial operation and a routine operation sub-phase, which will require a specific licence from the Swedish Radiation Safety Authority (SSM). The routine operation is expected to last ~45 years. The main activities during routine operation are detailed characterization, mining of deposition tunnels, deposition of canisters, and subsequent backfilling and plugging of deposition tunnels. During the operating phase, ~ 6 000 filled canisters will be transported by ship from the encapsulation plant to the final repository and emplaced in the deposition tunnels.

Also in Olkiluoto during the operational period the monitoring of the repository, including both the disposal facility bedrock conditions and the surface environment, will continue on a regular basis, resulting in annual reports that will be submitted to the Finnish Radiation and Nuclear Safety Authority STUK.

In the case of the French repository, the facility will be expanded gradually with the construction of new waste disposal areas as new radioactive waste arrives. It is expected that after 100 years of operation, Cigeo will extend over an area of approximately 15 km^2. The waste will be emplaced in horizontal metal-lined tunnels with a diameter of 0.7 m and a length of 150 m excavated in the clay; the emplacement will be done by robotic systems. Significantly wider and longer tunnels will host the long-lived ILW [3.3.8-7].

No radiologically relevant release or impact to the public is expected during the construction and the operation of the final repository. As long as the sealed canister remains intact, all radioactive substances will be contained. The canister is designed to retain its integrity and tightness during normal operation, disturbances and mishaps. However, adequate radiation shielding will be used to protect the personnel from gamma and neutron radiation. The radiation emitted by the canister will not be noticeable outside of the final repository. During the construction and the operation phases of the repository in Finland and Sweden, radionuclide releases and potential radiation effects will only be caused by the natural radioactivity present in the rock, mainly in the form of radon and radon daughters.

When all canisters have been emplaced in the repository, the facility will be backfilled and closed. After closure of the repository, the local physical-chemical conditions in the repository will be slowly returning to the original state before the start of the construction. In the Finnish case the repository closure will occur after ~100 years of operation: the underground openings in the disposal facility will be backfilled and sealed (Figure 3.3.8-9) to remove/minimize openings that could become water conductive pathways.

Long term post-closure safety will be achieved by means of a system of passive barriers that interact to contain, prevent or retard the dispersal of radioactive substances. The barriers may be engineered or natural (see part B of the present report). The protective function of the final repository against harm caused by radiations is set by relevant regulations. For instance, the time scale for the safety assessment of the Swedish final repository for spent nuclear fuel should cover a period of one million years after closure. The risk criterion set by SSM in Sweden in simplified terms says that people in the vicinity of the repository may not be exposed to greater risks than the equivalent of one-hundredth of the natural background radiation in Sweden today [3.3.8-2]. The Finnish nuclear law [3.3.8-8] states that a final repository under normal operations may not cause a dose to the most exposed member of the public higher than 0.01 mSv/year.

3.3.8.6 Non-radiological impact of final repository: example from the Swedish Environmental Impact Statement [3.3.8-2]

3.3.8.6.1 Land use and visual impact

Most of the sites of national interest in the Forsmark area are deemed not to be harmed by the planned activity. Most of the facility is located in industrial areas; however, a few sites relevant for nature conservation could experience a possible groundwater drawdown, with consequences for rich fens and shallow ponds. Specific measures are thus planned to limit such effects. SKB intends to create new ponds in the surrounding areas to offset the consequences of filling two ponds which are deemed to be of national interest because of the presence of endangered species (the red-listed pool frog).

SKB's land needs are not expected to affect bird protection areas. However, SKB will implement restrictions, training and recommendations for employees who need to get to or move around in areas that are used for nesting by protected species.

The road to the ventilation station to be built ~ 1.5 kilometres east of the operations area will be designed to preserve the rich wetlands that exist in that area.

The visual impact of the final repository surface buildings should be limited due to the presence of the adjacent nuclear power plant and also due the area's industrial character.

3.3.8.6.2 Water pollution and groundwater drawdown

Storm water will be managed locally. Both the construction and the operating phase activities will generate polluted water that will have to be managed. Leachate from the rock heap will be treated to remove oil and particles. Then it will be denitrified to remove explosive residues from the tunnel blasting operations carried out underground: first in a flooding area near the rock heap and finally in a nearby lake.

The drainage water will be treated underground by sedimentation and oil separation, and then discharged. The effects of the discharge are expected to be limited, since the content of nitrogen residues is considered sufficiently low and the receiving body of water is relatively tolerant. The heat content of the drainage water will be recuperated and used to heat the supply air to the underground facility.

Although fractures and fracture zones in the rock underground will be sealed by grouting, grouting cannot render the bedrock watertight and some groundwater inflow into the facility will occur. The in-leaking groundwater will cause surface groundwater drawdown, which can negatively affect water levels in wetlands. If no measures were to be adopted, groundwater drawdown is assessed to entail very significant negative consequences for 2 sites of national interest, significant consequences for 15 sites and noticeable consequences for 8 sites. As a mitigation measure, water supply to the most sensitive and valuable wetland sites are planned.

3.3.8.6.3 Noise and vibration

The noise generated by construction, rock handling and transport activities within the industrial area is deemed to cause small impact. No homes with residents will be affected.

Road traffic to and from the final repository will consist of commuting people, haulage of material and rock spoil. The heaviest traffic burden will occur during the second half of the construction phase, when around 90 rock shipments per day may pass.

The heavy traffic to and from the final repository will lead to an increase in the number of residents exposed to noise levels above the guideline value, at most about 20 persons. Sleep disturbances are not expected, since the heaviest traffic will be in the daytime.

The vibration levels will not significantly increase, but there will be heavier vehicles passing. This may entail a risk of moderate disturbance in a few buildings along national road 76.

3.3.8.6.4 Emissions, air pollution

The final repository and associated transport activities will cause e.g. carbon dioxide, nitrogen oxides and particulates emissions. However, the determined amounts and dispersal of such emissions should not lead to exceeding the legal limits for air quality (environmental quality standards) and should not entail any appreciable consequences for human health or the local environment.

3.3.8.6.5 Energy and resource needs

Ventilation will be a major cause for energy consumption at the facility; therefore, ventilation will be tuned according to the actual operational need.

Approximately 50 000 metric tons of bentonite clay per year will be needed, corresponding to 2.3 million metric tons during the whole operating life of the facility. The total global production of bentonite in 2007 was 15.7 million metric tons. There are no bentonite mines in Sweden, which means that the material will have to be imported. The planned port of entry is about 30 kilometres south of Forsmark.

3.3.8.6.6 Summary of life cycle analysis for the disposal phase.

Life cycle assessment of nuclear energy includes the impact of different stages of the nuclear fuel cycle, including the management and disposal of the radioactive waste. Geological disposal of high level waste and spent fuel contributes between about 2% to about 18% to the overall greenhouse gas emissions of the spent fuel cycle [3.3.8-1], [3.3.8-2], [3.3.8-3] and [3.3.8-5]. The contribution to the production of SOx, NOx, acidification potential, POCP, eutrophication, eco-toxicity and human toxicity are negligible (between a few per thousand to up to two percent), as is the water pollution, water consumption and water withdrawal. Disposal of waste accounts for about 5% of the land use and between 4% and 14% of the production of technological waste when compared with the entire nuclear fuel cycle.

Reference [3.3.8-4] presents an analysis of the environmental impacts of the different deep geological disposal concepts. The results of the analysis show that most of the environmental impacts take place during the operational phase, and are caused by the use of copper in the disposal canisters and bentonite as backfilling material, and between these two, the backfilling material dominates. The impact of the excavation of the deep geological disposal has a rather limited impact, but this impact is local. Greenhouse gas emissions results of the LCA of the disposal concepts of Finland, Sweden and Switzerland are of the same order of magnitude, and are consistent with the figures mentioned above. The analysis also shows that those concepts that do not use copper in their canisters, such as in Switzerland, that plans to use stainless still, or those who mix the bentonite with other materials, such as Finland have a lower environmental impact.

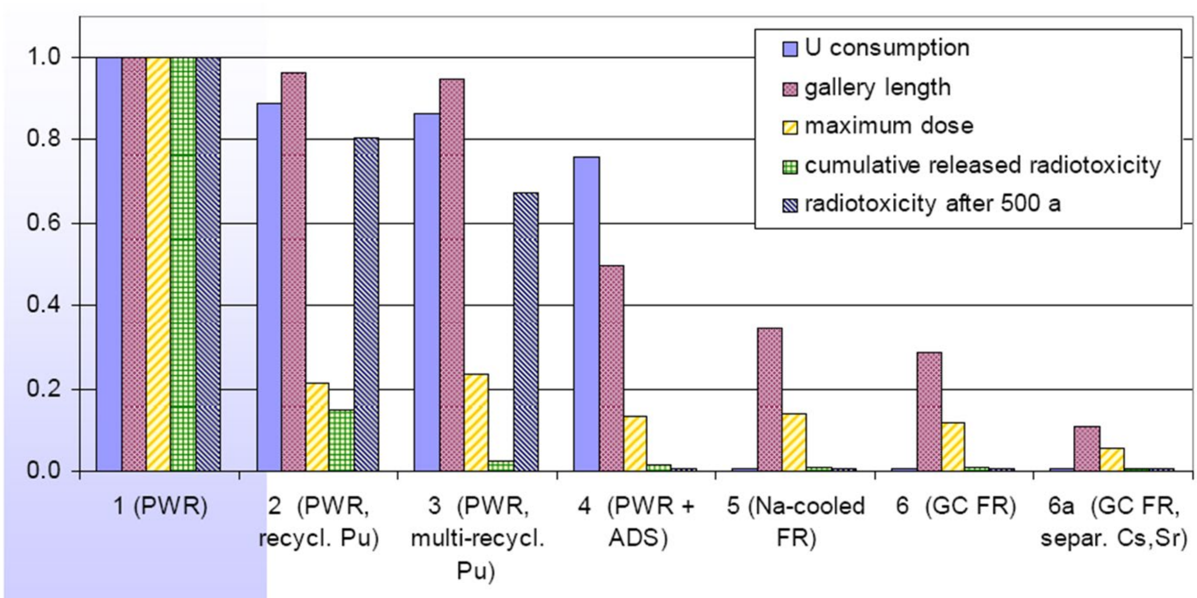
3.3.8.7 Fuel cycle impact on final repository

The geologic repository for final disposal of HLW is a necessary facility in the lifecycle of nuclear energy independently from the fuel cycle implemented. All existing and developing options for the back-end of the cycle generate a certain amount of HLW that requires long term isolation without the need for active human monitoring and control as provided by the deep geologic repository. However, the footprint of the final repository is strongly affected by the fuel cycle considered.

Figure 3.3.8-10 compares relevant quantities corresponding to different fuel cycles. The diagram highlights the effect of introducing increasing levels of recycling on the quantities considered. From the stand point of the final repository, the effect of recycling is to reduce the footprint of the repository and the long term radiotoxicity of the waste.

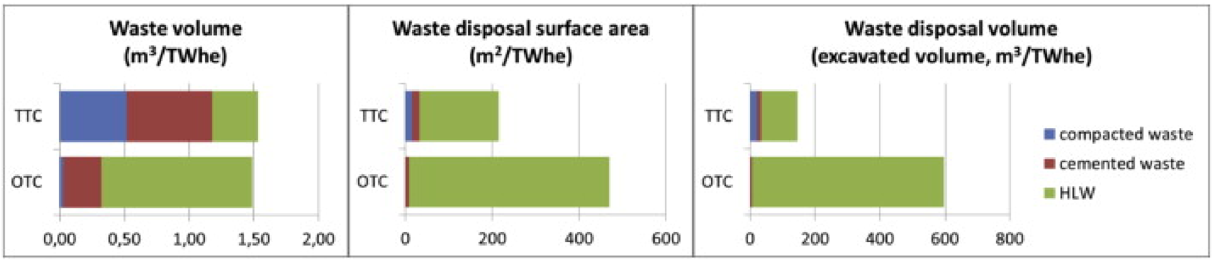
A comprehensive comparison of the fuel cycle with reprocessing (twice-through cycle, or TTC) and without reprocessing (once-through cycle, or OTC) from the French perspective is provided in [3.3.8-4]. In France, the geologic repository footprint is determined by the amount of HLW (mainly vitrified waste) and long-lived ILW to be disposed of. LLW and short-lived ILW are disposed of in surface or sub-surface facilities. Ref [3.3.8-4] provides the following excavated volume values for the different waste packages that will go to the deep repository: 55 m^3/HLW glass canister; 7.3 m^3/ILW-LL compacted waste canister and 21.3 m^3/ILW-LL cemented waste canister. The excavated volume per spent fuel assembly varies from 94 to 111 m^3 depending on the type of spent fuel. A more effective comparison is obtained by normalizing the values to the electricity generated, as shown in Figure 3.3.8-11. The corresponding TTC values normalized per TWh_e are: waste volume 1.53 m^3/TWh_e; total excavated volume 145 m^3/TWh_e; corresponding repository surface area 215 m^2/TWh_e. In the case of OTC, even though the total volume of waste to be disposed of (1.49 m^3/TWh_e) is similar, the excavated volume required is very different: 597 m^3/TWh_e for OTC vs. 145 m^3/TWh_e for TTC. The relatively big difference is reflected also, to a lesser extent, in terms of waste disposal surface area [3.3.8-4].

Figure 3.3.8-10. Comparison of relevant quantities for different fuel cycles normalized to the open cycle based on PWR and direct disposal of spent fuel; options 1 and 2 correspond to existing mature technologies, and the other options to concepts at different development stages



Source: RED-IMPACT[98]

Figure 3.3.8-11. Comparison of the waste volumes, waste disposal surface areas and waste disposal excavated volumes in France for the current twice-through fuel cycle (TTC) with reprocessing and the once-through fuel cycle (OTC)



Source: [3.3.8-4]

3.3.8.8 Disposal of VLLW and short–lived LILW

There is international consensus that very low level waste, low level waste and short-lived intermediate level waste can be safely disposed of in near-surface facilities at a depth of no more than 30 m. The underlying assumption is that the radioactivity of such waste types will decay to background levels within about 300 years, i.e. before institutional control is lost.

[98] Red-Impact: Impact of Partitioning, Transmutation and Waste Reduction Technologies on the Final Nuclear Waste Disposal, EU-funded research project carried out under the Euratom 6th Framework programme

The radioactive waste must be treated and conditioned before being emplaced in the disposal facility. Currently there are different technologies and installations for the treatment, conditioning, storage and disposal of short-lived low and intermediate level waste. For instance, the waste may undergo volume reduction by compaction, or, if it is in liquid form, it can be solidified. Often, the waste is placed in metal or concrete containers and then embedded in concrete. Less robust packaging may be used for very low level waste.

Near surface facilities may be simple trenches or may comprise an array of reinforced concrete cells. Once filled, these trenches or cells may be closed with a concrete slab and then sealed with an impermeable sheath. Finally, the trench or cell can be capped with a layer of clay several metres thick, to ensure the long-term confinement of the waste.

In Europe, repositories of this type exist in France, Hungary, Slovakia, Spain and the United Kingdom. In Finland and Sweden low level waste and short-lived intermediate level waste are disposed of in mined facilities at up to 100m depth.

In addition to these seven countries, other EU Member States, with as well as without nuclear power plants, are at various stages of implementation of low-level waste repositories. [3.3.8-9]).

The availability of active human monitoring and intervention over the timespan required for the waste to decay to background level ensures that the relevant regulations establishing the maximum allowed release of radioactivity are respected and consequently no harm from radiation is caused to the public. The nature of non-radiologic effects on humans and the environment associated with the construction and operation of this kind of facilities, and the related assessment, are similar to the corresponding aspects described for HLW management and disposal facilities.

3.3.8.9 Final conclusions

Disposal of radioactive waste is a necessary step in the lifecycle of a nuclear power plant. Most of the LCA consulted are comprehensive, and include in their results the contribution of the disposal phase to the overall environmental impacts. The disposal contributes slightly to the overall greenhouse emissions, use of land, and generation of technological waste, and does not contribute (the results are zero or negligible) to those indicators representative of the impacts to the Taxonomy Regulation objectives of sustainable use and protection of water and marine resources, pollution prevention and control, and protection and restoration of biodiversity and ecosystems. With regards to the transition to a circular economy, the raw materials used to build the multiple engineered barriers of the disposal facilities (e.g. copper) cannot be recovered, but the amounts needed are small, in particular if compared with the world production and the long timeframes of the disposal. Some materials resulting from the construction of facilities, e.g. part of the rock excavated to construct the tunnels of a crystalline rock repository, can be commercialized.

Although the disposal concepts analysed are rather similar, the magnitude of the impacts (which are mainly due to the operations and reposition) are dominated by the impacts of the activities related to excavating the tunnels and to building the multiple engineered barriers. The environmental impact analysis of the disposal facilities, e.g. those highlighted in this section, includes a description of the measures implemented to mitigate specific effects. Mitigation measures are considered also in the mining of raw materials needed to construct a repository (e.g. metals and bentonite for the engineered barriers) to limit the environmental impact of the disposal phase.

The deep geological disposal facility aims at isolating and containing the radioactive waste until its radioactivity decays to harmless levels. The long term radiological impact of disposal in the post-closure phase of a repository is described in part B of the present report.

Table 3.3.8-3. Importance of radioactive waste disposal phase impacts on the TEG environmental objectives

Non-radioactive and radioactive impact indicators		Prevention or mitigation of potentially harmful impacts	
Indicator	**Importance**	**Appropriate mitigation measures**	**Remarks**
GHG emissions	++	Limiting fossil fuel consumption	Dominated by the extraction of mineral for and construction of the engineered barriers
Water withdrawal	+	Application of best practices and appropriate measures depending on local configuration; relevant also in the metal and bentonite mines.	Dominated by the mining of mineral for and construction of the engineered barriers. Contribution from the excavation of the repository depends on local configuration.
Water pollution	+		
Ecotoxicity	+		
Human toxicity	+		
Land use	++	Disposal sites selected in locations, with limited or no valuable resources.	Land occupancy and visual impact considered.
Water consumption	+	Application of best practices and appropriate measures both locally and in the metal and bentonite mines.	A wet centralized storage facility requires a fraction of the cooling water needed for the operation of a nuclear power plant and generates much smaller thermal loading
Atmospheric pollution	+		-
Acidification pot.	+		-
Ozone creation pot.	+		-
Eutrophication pot.	+		-
Production of TW	++	Decontaminate, reuse and recycle.	
Depletion of resources	++		Use of resources per unit energy produced is very low
Production of solid RW	++	Application of radioactive waste management principles	Produced during operation and decommissioning of the encapsulation plant and auxiliary facilities.
Gaseous RA releases	+	Application of best practices during operation, and functional multiple barriers after closure.	Insignificant releases during the operation phase. Calculated releases during the closure phase well below authorised limits

Liquid RA releases	+	Application of best practices during operation, and functional multiple barriers after closure.	Insignificant releases during the operation phase. Calculated releases during the closure phase well below authorised limits.

Legend

N/A	Not applicable
+	Very low importance
++	Low importance
+++	Normal importance
++++	High importance
+++++	Critical importance

In the light of the above analysis it can be concluded that activities related to the storage & disposal of technological & radioactive waste, as well as spent nuclear fuel do not pose significant harm to human health or to the environment. They do not represent significant harm to any of the TEG objectives, provided that the associated industrial activities satisfy appropriate Technical Screening Criteria.

TSC for the interim storage and final disposal of high-level radioactive waste (including high-level vitrified waste generated during reprocessing) are provided in Chapter 5 and Annex 4 of the present report (*Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy*).

3.3.8.10 References for Chapter 3.3.8

[3.3.8-1] Feiveson, H., Mian, Z., Ramana, M.V. and von Hippel, F. (editors), Managing Spent Fuel from Nuclear Power Reactors: Experience and Lessons from Around the World, International Panel on Fissile Materials, September, 2011, ISBN 978-0-9819275-9-6.

[3.3.8-2] Environmental Impact Statement, Interim storage, encapsulation and final disposal of spent nuclear fuel, SKB, March 2011 – EIS/SKB2011.

[3.3.8-3] Handling and Storage of Conditioned High Level Wastes, IAEA-TECDOC-229, IAEA, 1983.

[3.3.8-4] Poinssot, C., Bourg, S., Ouvrier, N., Combernoux, N., Rostaing, C., Vargas-Gonzalez, M., Bruno, J., Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211.

[3.3.8-5] The Safety of the Nuclear Fuel Cycle, NEA 2005

[3.3.8-6] The Safety of Long-Term Interim Storage Facilities in NEA Member Countries, NEA/CSNI/R(2017)4

[3.3.8-7] ANDRA, https://international.andra.fr/projects/cigeo/cigeos-facilities-and-operation/project-siting-and-facilities-overview.

[3.3.8-8] Nuclear Energy Decree 12.2.1988/161, YEA 161/1988 22 d §. (19.12.2017/1001).

[3.3.8-9] http://www.ensreg.eu/safe-management-spent-fuel-and-radioactive-waste/existing-waste-management-routes.

3.4 Impact of ionizing radiation on human health and the environment

3.4.1 Nuclear energy lifecycle impacts on human health

Human health impacts of different energy-generation technologies were compared in Chapter 3.2.5 (see Figure 3.2-21). Taking into account the impacts on human health from all the different emissions, both radiological and non-radiological[99], from the whole lifecycle of the different technologies, the impact on human health of nuclear energy is seen to be low compared to the fossil fuel chains, and rather similar to the impact from offshore wind.

The total impact on human health from the nuclear energy lifecycle, provided by Hirschberg et al [3.4-1] is 56 mDALY/GWh[100], of which 16.5 mDALY/GWh are due to ionising radiation [3.4-2]. Although using a different methodology[101] and non-identical data and assumptions regarding the nuclear lifecycle, Stamford & Azapagic [3.4-5] calculated the total human health impact from ionising radiation alone at 20.3 mDALY/GWh, which is very similar.

To put the radiation health impacts from nuclear power in context, the authors calculated the health impact from global nuclear electricity generation of 2600 TWh/yr, which amounts to roughly 53 000 DALYs/yr, and compared it with the 597 000 life-years lost as a result of anthropogenic air pollution in the UK alone in 2008. The latter estimate, given by the UK Committee on the Medical Effects of Air Pollutants (COMEAP), refers to premature deaths only and excludes disability induced by the pollution.

The total human health impact from ionising radiation calculated using the methodology adopted in the LCIA studies discussed above results from the integration of very small impacts to individuals over very large populations over very long integration times (up to 100 000 years).

According to Stamford & Azapagic, approximately 90% of the radiological impact is caused by emissions to air of radon-222 from uranium mine tailings over a period of thousands of years, with the remainder being emissions of isotopes like carbon-14 during power plant operation. These results are for the once-through fuel cycle without reprocessing and recycling of plutonium in MOX fuel. However, the authors also performed sensitivity studies, which included reprocessing and recycling up to 8% of MOX fuel and varying the proportions of enrichment performed by centrifuge and gaseous diffusion processes. As enrichment does not contribute significantly to radiological emissions, the latter is not expected to have a significant effect on the human health impact. The maximum calculated human health impact from the sensitivity studies is 31.9 mDALY/GWh, which is about 50% higher than the central estimate provided above.

Poinssot et al [3.4-6] provide data on the environmental footprint, including information on radiological emissions, for the current French reactor fleet and fuel cycle, assuming both once-through and twice-through fuel cycles. The once-through fuel cycle does not involve reprocessing. The twice-through cycle included, at that time, reprocessing and recycling of plutonium in MOX fuel making up one-third of the fuel elements in the core of 22 reactor units representing 31.2% of the installed capacity of the French fleet.

Poinssot et al provide the radiological data in terms of emissions from the facilities (in Bq/kWh), and so they do not include any calculation of the dispersion of the released substances in the environment and their estimated effects on human health that are included in the end-point indicators of [3.4-1] and [3.4-5] discussed above. They show that:

- The total radiological emissions in the once through cycle are dominated by the mining activities (99.98%). The remaining 0.02% is from the reactor operation, while the emissions from U conversion, enrichment and fuel fabrication are negligible.
- The radiological emissions from the twice-through cycle are 53% higher than from the once-through cycle. Emissions from the mining activities are reduced, due to the reduced need for fresh Uranium per unit of electricity produced, and represent 53.4% of the total emissions. The remainder is almost entirely from reprocessing and comprises mainly noble gases (predominantly ^{85}Kr – 44.4%) and liquid tritium (2%). Only 0.23% comes from reactor operation with negligible emissions from the other stages (see Chapter 3.3.5.2.1).

However, the effects of a 1 Bq radioactive release in terms of the resulting dose (in Sv) to members of the public vary considerably for different radionuclides and different release pathways (see Chapter 3.3.5.2.1). The radiotoxicity of ^{85}Kr is very low, and as shown in Chapter 3.3.5.2.1, it contributes less than 15% to the dose to the public from the reprocessing stage of the nuclear lifecycle. The comparatively small amounts of ^{14}C released during reprocessing have a larger impact, being responsible for more than 50% of the public

99 Including the effects of human toxicity, photochemical oxidant formation, particulate matter formation, as well as induced climate change and ionizing radiation.

100 DALY: Disability Adjusted Life Years = years of life lost + years lived with a disability

101 Stamford & Azapagic use CML 2001, while Hirschberg et al use ReCiPe. However, it should be noted that as far as the impact of ionising radiation is concerned, both CML 2001 and ReCiPe use the methodology developed by Frsichknecht et al [3.4-3] (see Goedkoop et al [3.4-4]).

dose from reprocessing. According to UNSCEAR calculations, the major contributor to public radiation doses from the nuclear lifecycle is conventional uranium mining, while the dose contribution from reprocessing is an order of magnitude lower and is slightly less than the contribution from power reactor operation (see Figure 3.3.5-7, Chapter 3.3.5.2.1).

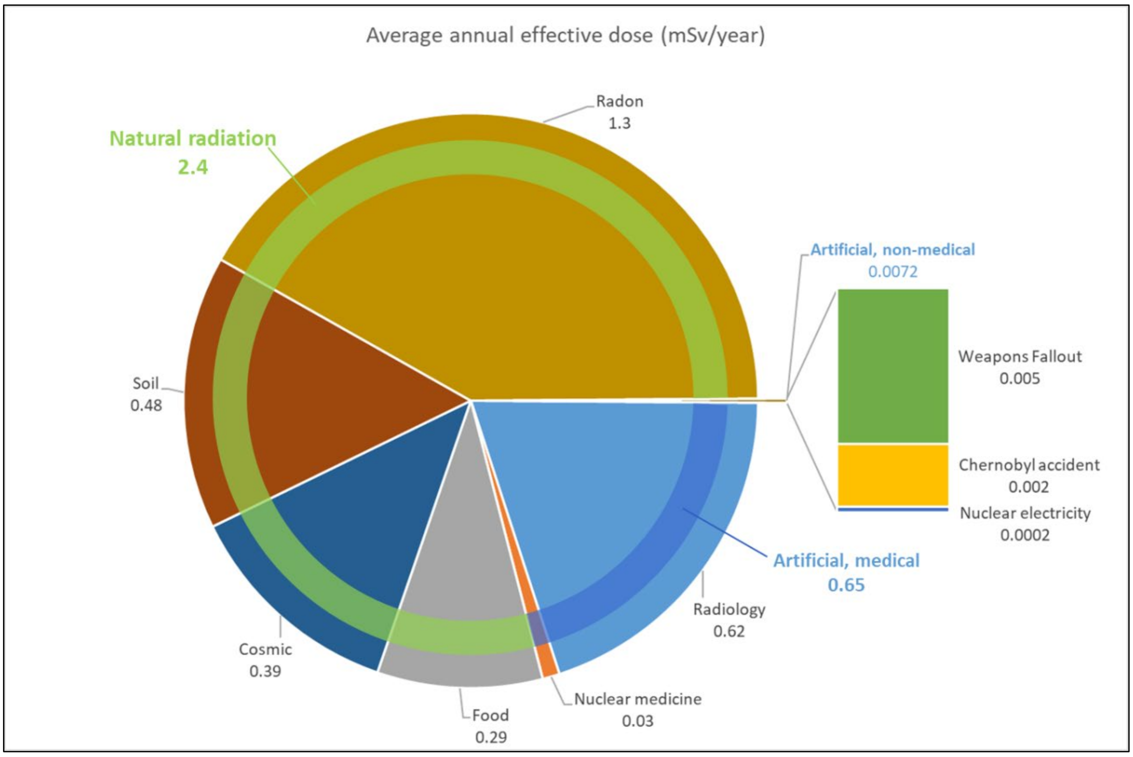
However, as mentioned above, the total impact on human health of these radiological emissions, as well as other, non-radiological emissions from the nuclear energy chain, are comparable with the human health impact from offshore wind energy, according to the LCIA studies referred to above.

That the impacts to individuals from radiation exposure due to the nuclear energy chain are small can be seen from Figure 3.4-1, which shows the worldwide average dose[102] to members of the public from different sources of ionising radiation[103], estimated by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) [3.4-7].

According to the United Nations Environment Programme (UNEP) [3.4-7],

> *"To someone who is reading about the topic for the first time, it may come as a surprise that the sources of radiation causing the greatest exposure of the general public are not necessarily those that attract the most attention. In fact, the greatest exposure is caused by natural sources ever present in the environment, and the major contributor to exposure from artificial sources is the use of radiation in medicine worldwide. Moreover, everyday experience such as air travel and living in well-insulated homes in certain parts of the world can substantially increase exposure to radiation".*

Figure 3.4-1. World average annual per caput public doses due to different sources of ionising radiation



Data from UNEP [3.4-7]

Natural background radiation is responsible for 2.4 mSv/year, or around 78% of the total average annual effective dose to the public of 3.05 mSv/year. The remainder is from artificial sources. Of the dose resulting from artificial sources of ionising radiation, 99% is from medical applications (radiology and nuclear medicine). This receives little public attention, presumably in almost universal recognition that the benefits outweigh the risks involved. The other 1% of the artificial radiation comes largely (about 69%) from the

102 The sievert or Sv is the biological dose unit used to quantify the effect of radiation on humans: the actual doses to which we are exposed are generally expressed in thousandths of a sievert (millisievert or mSv), or in millionths of a Sievert (microsievert or μSv).

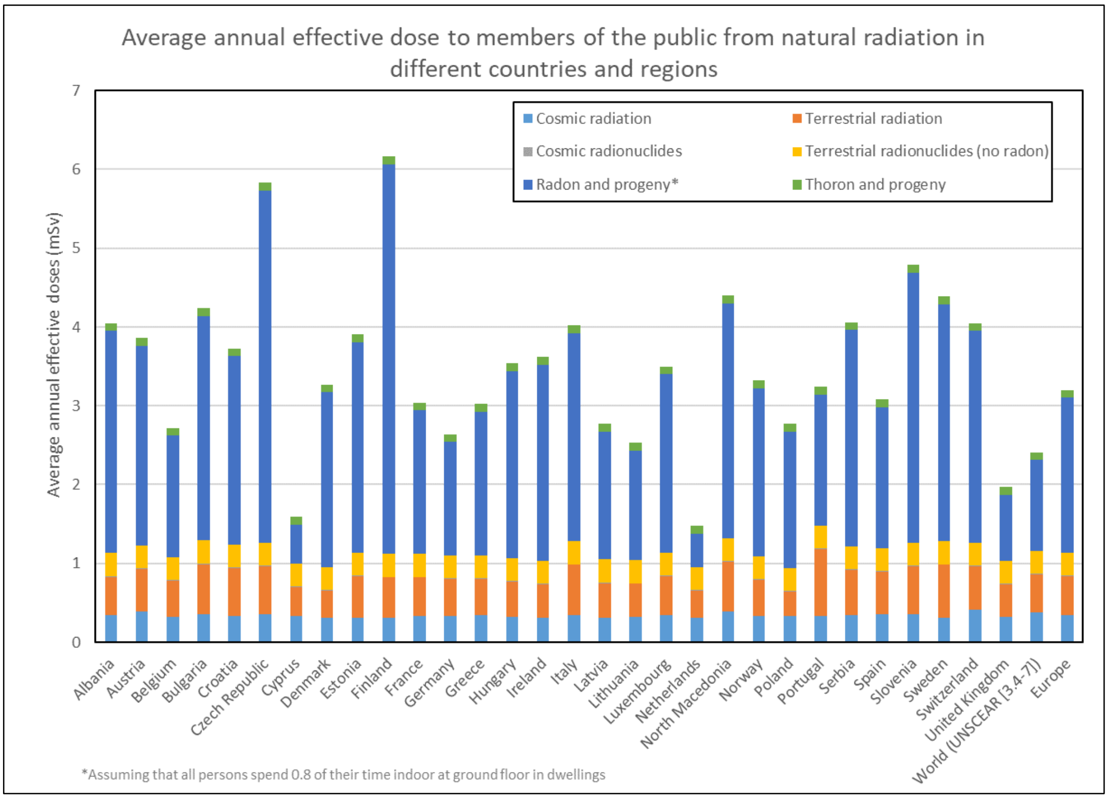
103 Annex 5 provides a detailed description of the various types and effects of ionising radiation and provides definitions of radiation dose, units, biological effects and principles of radiation protection.

fallout remaining from atmospheric nuclear weapons testing carried out up to the early 1960s, as well as from the residual radioactivity of material released during the Chernobyl accident (about 28%)[104]. Radiation resulting from the whole lifecycle of nuclear electricity generation results in an average annual effective dose to the public of only 0.2 μSv/year [3.4-8], which is less than seven thousandths of one percent (0.66 x 10^{-5}), of the total average dose to members of the public from all sources.

For populations living within 100 km of a mine and mill site, UNSCEAR [3.4-8] estimated the population average individual effective dose at 0.025 mSv/a. The corresponding estimate for populations living within 50 km of a reprocessing plant was about 0.002 mSv/a. For nuclear power plants the average effective dose to populations living within 50 km of a plant was 0.1 μSv/a whereas the estimated effective dose to critical groups living within 1 km of the plant was 0.02 mSv/a. These effective doses to regional and local populations all correspond to less than 1% of the dose due to natural background radiation. The global average effective dose to the public from the nuclear power lifecycle also represents only 0.03% of the average dose to the public from all sources of artificial radiation. Artificial radiation dose to individuals is dominated by radiation from medical interventions, mainly radiology. In addition, members of the public are also still exposed to small amounts of radiation resulting from the fallout from nuclear weapons testing and from the Chernobyl nuclear accident. Although doses to the public from these sources are 25 and 10 times greater, respectively, than the dose due to the nuclear power lifecycle, they are still very small compared to natural and medical sources (0.2% of the total dose).

Furthermore, the additional effective doses to members of the public due to the nuclear energy lifecycle are also extremely small when compared to the variations in natural background radiation due to living in different geographic locations. The Joint Research Centre's European Atlas of Natural Radiation [3.4-9] provides detailed information and maps showing the variations in different sources of natural radiation in Europe. Figure 3.4-2, compiled using data from [3.4-9], shows the variation in the average annual doses from natural radiation in most European countries, as well as the Europe and world averages. The national averages range from around 1.5 mSv in The Netherlands, to around 6.2 mSv in Finland, a variation of almost 5 mSv/year.

Figure 3.4-2. Geographic variations in average annual doses from natural radiation

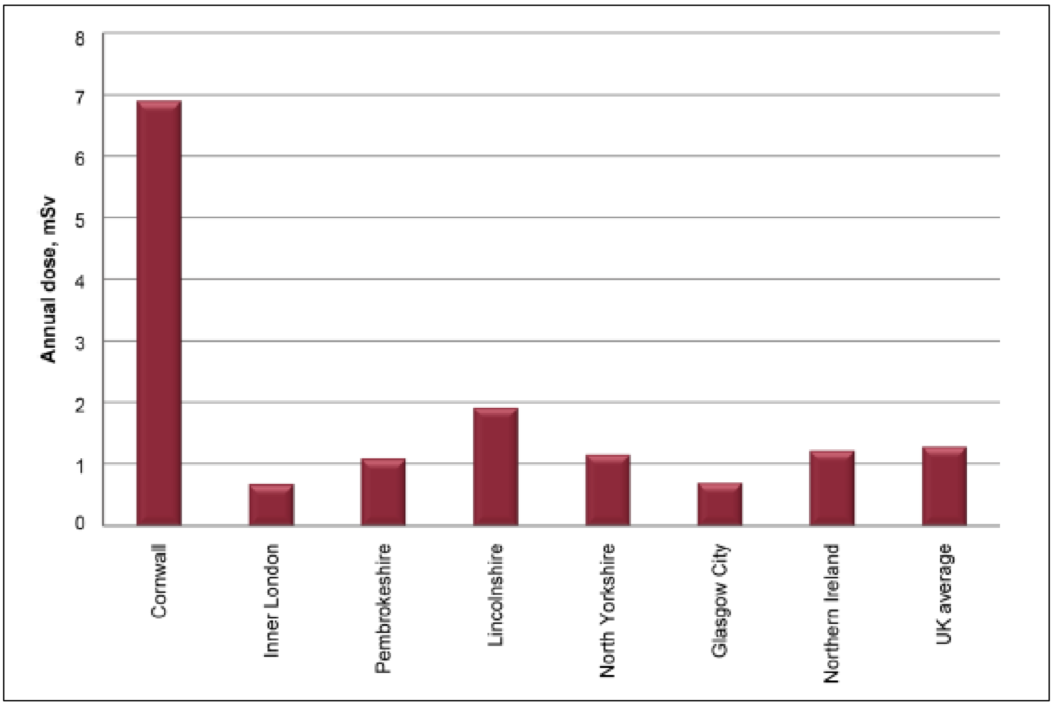


Data from [3.4-9]

104 Figure 3.4-1 shows world average annual radiation doses, but some of the sources mentioned are more heterogeneously distributed geographically than others. The Chernobyl accident obviously had a more important impact in the region surrounding the plant.

Similar variations may be seen within countries, for example in the UK the main contributor to natural radiation doses, which is radon inhalation, varies as shown in Figure 3.4-3. Cornwall has very much higher levels of radiation exposure due to radon gas, almost 6 times the national average. Radon gas is formed by radioactive decay of the small amounts of uranium that is present naturally in rocks and soils. The higher levels in Cornwall are due to the presence of granite in the underlying geology, which naturally contains more uranium than other rocks.

Figure 3.4-3. Illustrative annual doses from inhaling radon in different parts of the UK



Source: [3.4-10]

Worldwide, even larger variations in natural background radiation occur. Ramsar, a northern coastal city in Iran, has areas with some of the highest levels of natural radiation. Inhabitants who live in some houses in this area receive annual doses as high as 132 mSv from external terrestrial sources. The radioactivity of the high background radiation areas of Ramsar is due to ^{226}Ra and its decay products, which have been brought to the surface by the waters of hot springs[105].

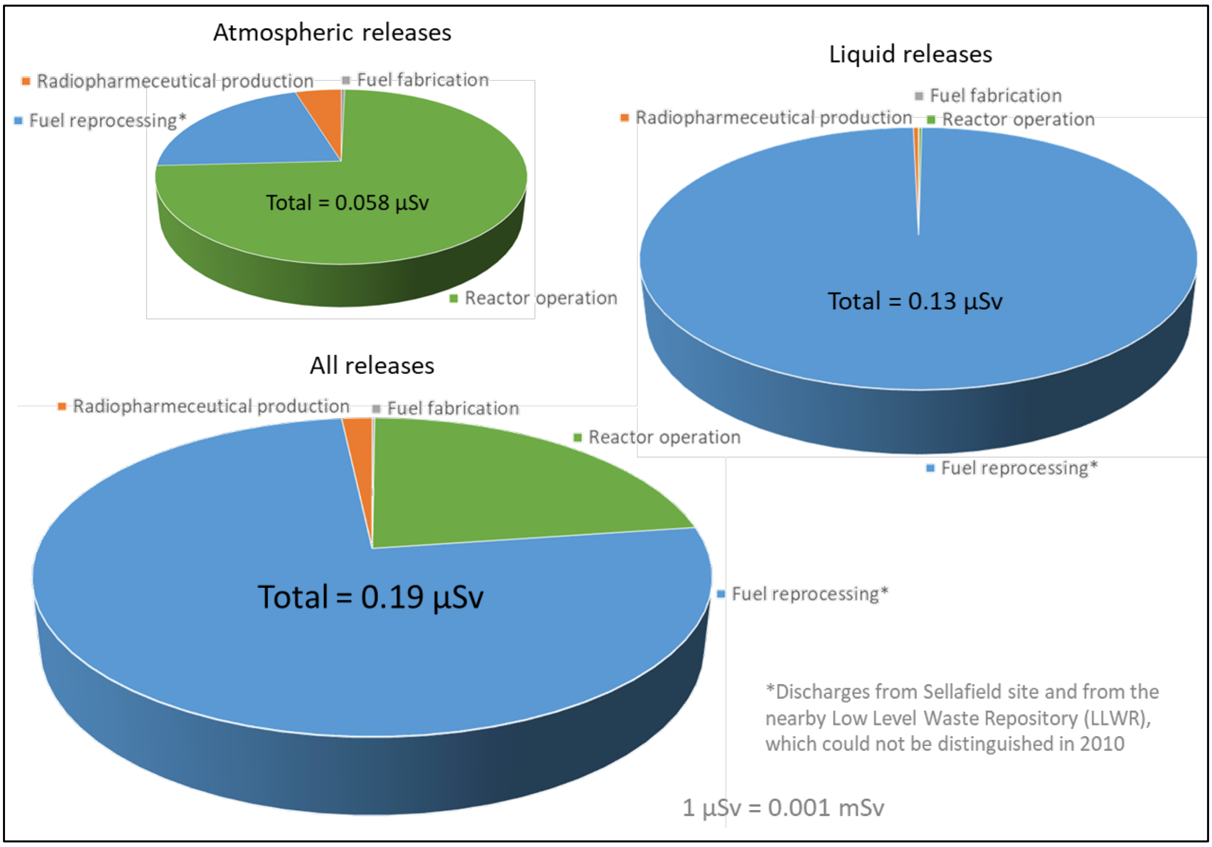
For comparison with the figures given by UNSCEAR and shown in Figure 3.4-1, the exposure of the UK population to ionising radiation from different sources was assessed by Oatway et al [3.4-10]. A summary of per caput doses to the UK population in 2010 from exposure to radionuclides discharged into the environment by UK civil nuclear sites is given in Figure 3.4-4.

The total per caput dose to the UK population in 2010 from exposure to radionuclides released into the environment by the UK civil nuclear industry were estimated to be about 0.0002 mSv, the same as the world average estimated by UNSCEAR, above. The UK has no uranium mining activities, but radiopharmaceutical production facilities were included in the assessment, although these had a small impact on the total dose uptake. Radionuclides discharged to the atmosphere and to the marine environment contributed about 30% and 70%, respectively, to this dose. Nearly the entire dose to the UK population from exposure to radioactivity discharged as a liquid was due to discharges made by the Sellafield site where fuel reprocessing is carried out. The most significant radionuclides were americium-241 (^{241}Am) and plutonium-239 (^{239}Pu) in molluscs and carbon-14 (^{14}C) and caesium-137 (^{137}Cs) in fish. Nuclear power plants were the most significant source of radionuclides released to atmosphere with respect to the UK population dose in 2010. The most significant radionuclides released to atmosphere were ^{14}C, sulphur-35 (^{35}S) and iodine-129 (^{129}I) that had been incorporated in terrestrial foods, particularly milk and grain. It should be noted that Sellafield site is dealing not only with reprocessing for the current UK nuclear energy production. Reprocessing of fuel from overseas

105 http://www.ecolo.org/documents/documents_in_english/ramsar-natural-radioactivity/ramsar.html

customers and from an older generation of UK power reactors is carried out, as well as remediation of legacy installations.

Figure 3.4-4. Per caput effective dose (μSv) to the UK population due to discharges from UK civil nuclear sites



Data from [3.4-10]

To put these annual dose numbers into perspective, they can be compared with the acute effective doses to an individual person from the following sources of ionising radiation:

- Consumption of 100g of Brazil nuts[106]: 0.01 mSv (10 μSv)
- One return flight from London to Cape Town[107]: 0.1 mSv (100 μSv)
- One CT scan of the abdomen and pelvis[108]: 10 mSv (10 000 μSv)

Licensees of nuclear installations have to demonstrate, prior to obtaining a licence, and ensure during operation, that the effective radiological dose to the most affected members of the public are within strict legal limits. These limits correspond to a level of dose below which no significant harm is caused to the population.

With regard to radiological protection and the legal limits for radiation doses to members of the public and workers, most countries follow the recommendations and guidance developed by the International Commission on Radiological Protection (ICRP)[109]. Those recommendations are taken into account in IAEA safety publications, and the Euratom Basic Safety Standards Directive also implements the principles and dose limits recommended by the ICRP. The Directive establishes limits for the effective radiation dose for both workers (occupational exposures) and members of the public. The limit for members of the public is set at 1 mSv/year. Importantly, this dose limit for public exposure shall "... apply to the sum of annual exposures

106 Ref. [3.4-10].
107 Dose due to additional cosmic radiation [3.4-10].
108 Ref. [3.4-11].
109 The ICRP is an independent, international organisation that advances for the public benefit the science of radiological protection (http://www.icrp.org/index.asp)

of a member of the public resulting from all authorised practices". The limit for classified radiation workers is set at 20 mSv per year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv.

It can be seen from the information presented in this Chapter, as well as in Chapter 3.3 and Part B of this report, that the doses to the public from the operations of nuclear energy lifecycle facilities, including radioactive waste management installations, is systematically well within these statutory limits.

As discussed above, uranium mining is the main contributor to the radiological impact of the nuclear energy chain. Other important stages of the lifecycle with regard to radiological impacts are nuclear fuel reprocessing and operation of nuclear power plants. Uranium mining and its radioactive emissions are discussed in Chapter 3.3.1. Nuclear fuel reprocessing is discussed in Chapter 3.3.5 and nuclear power plant construction, operation and decommissioning in Chapter 3.3.7.

3.4.2 Impact of radiation on the environment

Until recently, the prevailing view was that the recommendations, guidelines and statutory limits developed to protect human health from the effects of ionising radiation from artificial sources would be sufficient also to ensure the protection of animals, plants and natural ecosystems. Such a view was supported by the fact that mammals are the most sensitive among the families of plants and animals to the effects of ionising radiation (see Figure 3.4-5). However, the impact of radiation on the environment is beginning to receive more attention than previously.

In the latest publication of the Recommendations of the International Commission on Radiological Protection (ICRP) [3.4-12], the Commission recognised that as a result of the increased interest in the protection of the environment from human activities, there was a growing need for advice and guidance on matters related to the protection of the environment from the effects of radiation, even though such needs have not arisen from any new or specific concerns about radiation effects on the environment.

The Commission confirmed that it subscribes to the global needs and efforts required to maintain biological diversity, to ensure the conservation of species, and to protect the health and status of natural habitats, communities, and ecosystems, and it considers that it is now necessary to provide advice considering a wider range of environmental situations, irrespective of any human connection with them. The Commission therefore believes that the development of a clearer framework is required in order to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis.

By setting out data for some Reference Animals and Plants, in a transparently derived way, and upon which further action may be considered, the Commission intends to offer more practical advice than in the past. The Commission will use this framework to gather and interpret data in order to provide more comprehensive advice in the future, particularly with regard to those aspects or features of different environments that are likely to be of concern under different radiation exposure situations.

While it can be expected that future publications containing recommendations and guidance from the ICRP will contain advice on the protection of plants and animals in the natural environment, it is important to note that the Commission reiterated its continued belief that the standards of environmental control needed to protect the general public are likely to be sufficient to ensure that other species are not put at risk,

UNSCEAR [3.4-7, 13] evaluated the effects of radiation exposure on plants and animals and found that individual responses to radiation exposure varied, mammals being the most sensitive to radiation exposure. The ranges of acute lethal doses, at which 50% of the exposed subjects would be expected to die, for different types of plants and animals are shown in Figure 3.4-5. In general, larger mammals are more radiosensitive than smaller ones, and the same applies also in the case of plants.

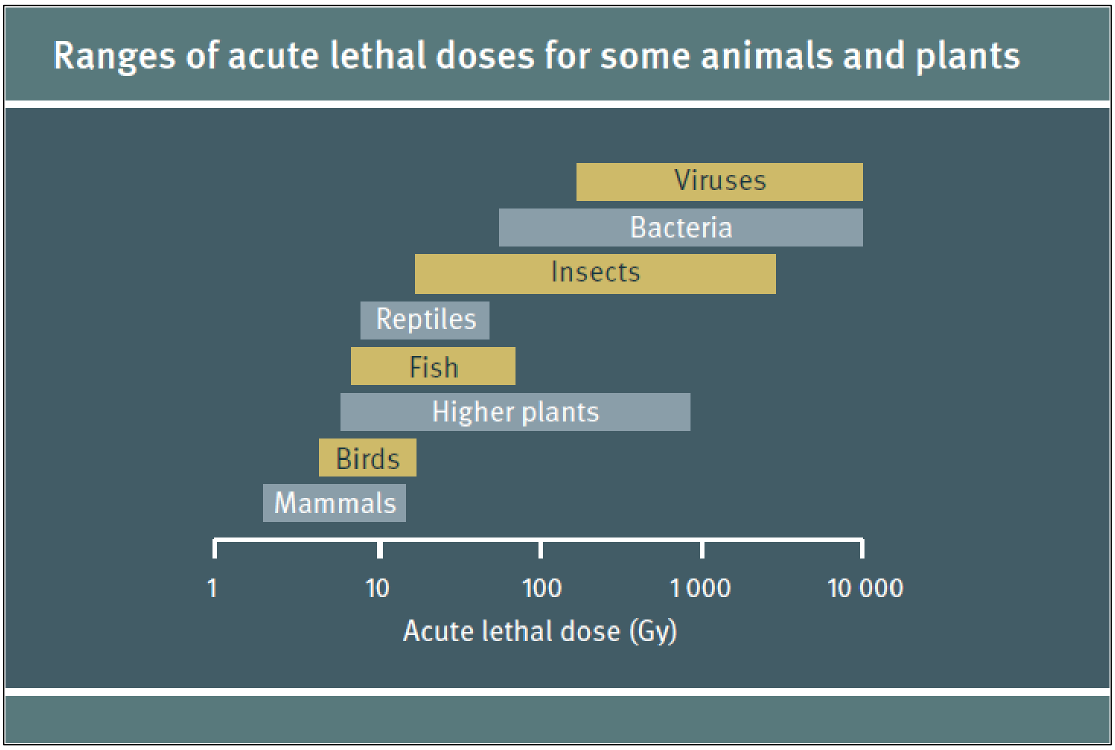
With regard to impacts on populations of plants and animals, reproductive changes are a more sensitive indicator of the effects of radiation exposure than mortality, and mammals are again the most sensitive animal organisms. However, because of the compensation and adjustment possible in animal species, the UNSCEAR considered that it is unlikely that radiation exposures causing only minor effects on the most exposed individual would have significant effects on the population. On this basis, chronic dose rates of less than 100 µGy/h[110] to the most highly exposed individuals would be unlikely to have significant effects on most terrestrial animal communities[111]. Such rates of absorbed dose are equivalent to an effective whole

[110] MicroGray per hour; 1 Gy corresponds to an energy deposition of 1 Joule in 1 kg of target material, see Annex 4.

[111] The corresponding level for communities of aquatic organisms is about 400 µGy/h. These conclusions refer to low linear energy transfer radiation such as gamma and beta radiation. Where a significant part of the incremental radiation exposure comes from

body dose of about 0.9 Sv[112] in one year, which is 900 times higher than the dose limit for members of the public.

Figure 3.4-5. Ranges of acute lethal doses for some animals and plants



Source: [3.4-7]

3.4.3 References for Chapter 3.4

[3.4-1] Hirschberg, S., Bauer, C., Burgherr, P., Cazzoli, E., Heck, T., Spada, M., Treyer, K., Health effects of technologies for power generation: Contributions from normal operation, severe accidents and terrorist threat, Reliability Engineering and System Safety **145** (2016) 373–387.

[3.4-2] Treyer, K., Bauer, C. and Simons, A., Human health impacts in the lifecycle of future European electricity generation, Energy Policy **74** (2014), S31–S44.

[3.4-3] Frischknecht, R., Braunschweig, A., Hofstetter, P., Suter P., Human health damages due to ionising radiation in life cycle impact assessment, Environmental Impact Assessment Review **20** (2000), 159–189.

[3.4-4] Goedkoop, M., Heijungs, R., Huijbregts, M., De Schryver, A., Struijs, J., van Zelm, R., ReCiPe 2008: A life cycle impact assessment method which comprises harmonised category indicators at the midpoint and the endpoint level, Report I: Characterisation, Ministerie van Volkshuisvesting, 6 January 2009.

[3.4-5] Stamford, L. and Azapagic, A., Life cycle sustainability assessment of electricity options for the UK, Int. J. Energy Res. 2012, **36**, 1263-1290, September 2012.

[3.4-6] Poinssot, C., Bourg, S., Ouvrier, N., Combernoux, N., Rostaing, C., Vargas-Gonzalez, M., Bruno, J., Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy **69** (2014) 199-211.

[3.4-7] Radiation Effects and Sources, United Nations Environment Programme, ISBN 978-92-807-3517-8, 2016.

high linear energy transfer radiation (such as alpha particles), the different relative biological effectiveness of the different radiations need to be taken into account.

112 For low linear energy transfer radiation such as gamma and beta radiation.

[3.4-8] Sources and Effects of Ionizing Radiation, United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2008 Report to the General Assembly with Scientific Annexes, VOLUME I.

[3.4-9] Cinelli, G., De Cort, M. & Tollefsen, T. (Eds.), European Atlas of Natural Radiation, Publication Office of the European Union, Luxembourg, 2019.

[3.4-10] Oatway, W. B., Jones, A. L., Holmes, S., Watson, S. and Cabianca, T., Ionising Radiation Exposure of the UK Population: 2010 Review, Public Health England, ISBN 978-0-85951-783-6, April 2016.

[3.4-11] Nuclear Radiation and Health Effects (Updated April 2020), World Nuclear Association, https://www.world-nuclear.org/information-library/safety-and-security/radiation-and-health/nuclear-radiation-and-health-effects.aspx

[3.4-12] Valentin, J. (ed), The 2007 Recommendations of the International Commission on Radiological Protection, Annals of the ICRP, Vol. **34**, Nos. 2-4, Publication 103, ISBN 978-0-7020-3048-2, ISSN 0146-6453, Elsevier, 2007.

[3.4-13] SOURCES AND EFFECTS OF IONIZING RADIATION, United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2008 Report to the General Assembly with Scientific Annexes, VOLUME II, Scientific Annexes C, D and E.

3.5 Impact of severe accidents

Human health impacts of different energy-generation technologies were compared in Chapter 3.2.5 for normal operation situations. In addition to the impacts from normal operation, the possible consequences on the environment and human health of potential severe accidents in the energy sector are not negligible, and it is important to consider these in any comparative assessment.

A significant contribution to the development of a comprehensive methodology for the assessment of accident risks in the energy sector has resulted from the related long-term research activities performed since the early 1990s at the Paul Scherrer Institute in Switzerland (see Hirschberg et al [3.5-1]). As a part of this work, a database of severe accidents[113] that have occurred in the energy sector has been established and is continually updated and extended, and a methodology for evaluating accident risks for different energy generation technologies has been developed. Recognising that accidents may occur in all stages of an energy chain, the database and the assessment methodology cover the whole lifecycle for each energy technology.

The methodological approach to evaluating accident risks differs according to the extent of data available in the database. For fossil energy chains (coal, oil and gas) there is extensive historical accident data available to provide a strong basis for the risk evaluation. For hydropower, limited historical data for OECD countries is supplemented by modelling of hypothetical dam failures. For new renewables, for which historical data is limited, a hybrid approach is adopted, in which available historical data, modelling and expert judgement are used. For nuclear energy, due to the very low number of historical severe nuclear accidents and their significance for risk assessment[114], an approach based on the use of a simplified, site-specific, Level 3 Probabilistic Safety Assessment (PSA[115]) is used to quantify the risks associated with hypothetical severe accidents.

The methodology provides its results in terms of two risk indicators, both based on fatalities. The first is the fatality rate, which is defined as the expected number of fatalities due to severe accidents normalised to the amount of electricity generated in GWh (fatalities/GWh). The second is the maximum credible number of

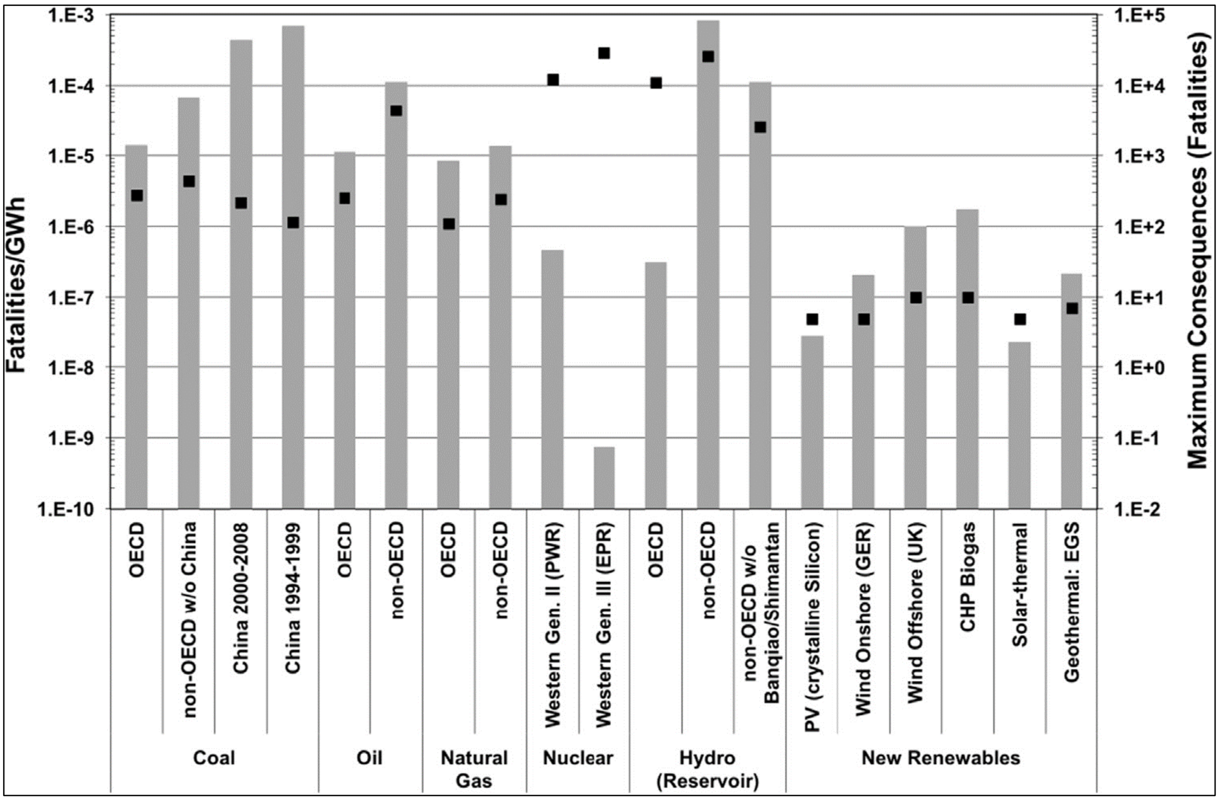
113 The Energy-Related Severe Accident Database (ENSAD).

114 Three core-melt events have occurred to date in nuclear power plants: Three Mile Island (USA, 1979), Chernobyl (Ukraine, 1986), and Fukushima Daiichi (Japan, 2010). The consequences of the TMI accident were relatively low; the total collective effective dose to the public was about 40 person-Sv, which resulted in an estimation of one cancer fatality. The Chernobyl reactor design is not representative of operating plants in OECD countries using different, safer technologies, nor of reactor designs for future deployment globally. The Fukushima accident is not included in the results provided by Hirschberg et al [3.5-1], since a reliable assessment of its consequences were still an open issue at that time.

115 Probabilistic Safety Assessment is a tool for mathematically quantifying the risk associated with a nuclear power plant. Level 1 PSA estimates the probability or frequency of accidents that result in damage to the core of the reactor. The result of a level 1 PSA is referred to as the core damage frequency (CDF). Core damage does not necessarily lead to radiological releases into the environment because the reactor vessel and containment building would both have to fail, or be bypassed, for radiological releases to occur. Level 2 PSA takes the calculation a step further by estimating the frequency of accidents that release significant quantities of radioactivity into the environment. Level 3 PSA provides an end point risk assessment by estimating the frequency of accidents having specific consequences. Those consequences may be, for example, early or latent fatalities resulting from the radiation doses to the population around the plant, or damage to the environment, such as a large area of land contaminated due to deposition of radioactive material released in the accident.

fatalities in a single accident, which provides a measure of risk aversion. The results of applying the methodology to several electricity-generation technologies is shown in Figure 3.5-1 (from Hirschberg et al [3.5-1]).

Figure 3.5-1. Severe accident fatality rates and maximum consequences (black points) assessed for selected electricity supply technologies with the associated energy chains



Source: Hirschberg et al [3.5-1]

With regard to the first metric, fatality rates, the results indicate that current Generation II nuclear power plants have a very low fatality rate compared to all forms of fossil fuel energies and comparable with hydropower in OECD countries and wind power. Only Solar energy has significantly lower fatality rates.

To put some perspective on these results, they can be compared with the health impacts due to normal operation. In order to facilitate such comparison, Hirschberg et al [3.5-1] noted that one premature fatality caused by air pollution roughly corresponds to 10 (chronic) YOLLs. Their normal operation mortality result for nuclear energy under normal operation (see Figure 3.2-19) is 5 mYOLLs/GWh, which is therefore equivalent to 5 x 10^{-4} fatalities/GWh, whereas the fatality rate for accidents is 3 orders of magnitude lower. Hirschberg et al [3.5-1] note that overall, for the different energy technologies, the fatality rates due to normal operation are much higher than the corresponding rates due to severe accidents.

Operating nuclear power plants are subject to continuous improvement. As a result of lessons learned from operating experience, the development of scientific knowledge, or as safety standards are updated, reasonably practicable safety improvements are implemented at existing nuclear power plants. This is a requirement of the EU Nuclear Safety Directive, and is also incorporated in WENRA's[116] safety reference levels for existing reactors [3.5-2]. The result of this continuous improvement is that the calculated frequency of severe accidents in the plant specific PSA reduces over time. This will already be reflected in the fatality rate given in figure 3.5-1. Further reductions may be expected in future, although they may become more marginal as the most important safety improvements have probably been made already, including those following the EU nuclear stress tests.

[116] Western European Nuclear Regulators Association

Generation III nuclear power plants are designed fully in accordance with the latest international safety standards that have been continually updated to take account of advancement in knowledge and of the lessons learned from operating experience, including major events like the accidents at Three Mile Island, Chernobyl and Fukushima. The latest standards include extended requirements related to severe accident prevention and mitigation. The range of postulated initiating events taken into account in the design of the plant has been expanded to include, in a systematic way, multiple equipment failures and other very unlikely events, resulting in a very high level of prevention of accidents leading to melting of the fuel. Despite the high level of prevention of core melt accidents, the design must be such as to ensure the capability to mitigate the consequences of severe degradation of the reactor core. For this, it is necessary to postulate a representative set of core melt accident sequences that will be used to design mitigating features to be implemented in the plant design to ensure the protection of the containment function and avoid large or early radioactive releases into the environment. According to WENRA [3.5-3], the objective is to ensure that even in the worst case, the impact[117] of any radioactive releases to the environment would be limited to within a few km of the site boundary.

These latest requirements are reflected in the very low fatality rate for the Generation III European Pressurised-water Reactor (EPR) given in figure 3.5-1. The fatality rate associated with future nuclear energy are the lowest of all the technologies.

On the other hand, the second metric shown in Figure 3.5-1, maximum consequences, is high for nuclear energy based on both Generation II and III nuclear power plants. It can be seen from Figure 3.5-1 that the numbers for nuclear are comparable with hydro, and accidents in the oil industry can also have very significant maximum credible consequences. For nuclear, the higher figure for EPR, in the region of 30 000 fatalities, reflects the larger radioactive inventory in the higher capacity plant compared to the Generation II PWR. It is dominated by latent fatalities (>95%). This result compares with the upper bound of the estimates of fatalities resulting from the Chernobyl accident, which were also dominated by latent effects[118].

The maximum credible number of fatalities from a hypothetical nuclear accident at a Generation III NPP calculated by Hirschberg et al [3.5-1] is comparable with the corresponding number for hydroelectricity generation, which is in the region of 10 000 fatalities due to hypothetical dam failure. In this case, the fatalities are all or mostly immediate fatalities and are calculated to have a higher frequency of occurrence.

Figure 3.5-2, from the same Hirschberg et al study [3.5-1], compares the frequency-consequence curves for selected full energy chains in OECD and non-OECD countries. The curves for coal, oil, gas and hydro are based on historical data from the period 1970 – 2008. In all cases the data concern immediate fatalities. The curves for nuclear energy are based on a simplified level 3 PSA.

Although extensive historical data is available for the fossil and hydro energy chains, it is nevertheless limited and does not allow extending the frequency-consequence curves below frequencies of about 3 x 10^{-9} to 3 x 10^{-8} fatalities/GWh for the different energy chains. The maximum consequences shown in Figure 3.5-1 correspond to the point of minimum frequency in the respective curves in Figure 3.5-2.

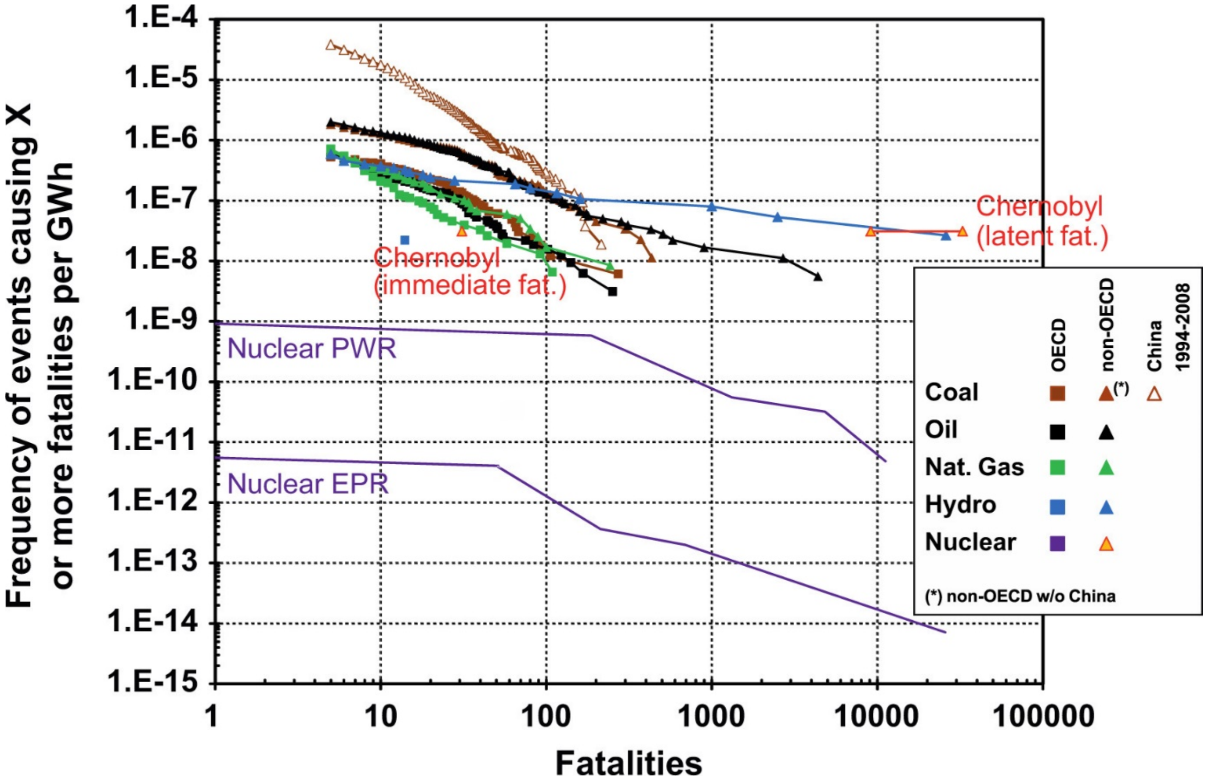
However, the shape of the curves does not indicate that the maximum consequences have been reached, and extrapolation of the curves to lower frequencies would suggest that higher consequences would be likely.

On the other hand, for the nuclear energy chain, the use of a simplified PSA allows extending the calculation to extremely low frequencies, and the maximum consequences shown in Figure 3.5-1 are those corresponding to the lowest frequency from the calculated curves, which are 3 and 6 orders of magnitude lower, for the Generation II PWR and EPR respectively, than the frequencies corresponding to the maximum consequences for the fossil and hydro chains. Moreover, accidents at both Generation II PWR and EPR having frequencies corresponding to the maximum consequences for the fossil and hydro chains (i.e. above 10^{-9}/GWh), would not result in any fatalities.

[117] Impacts include the need for actions to protect the public, such as evacuation, sheltering and iodine prophylaxis, or long-term restriction in consumption of agricultural products from the vicinity of the plant due to land contamination. Any radioactive releases should also be late enough such that the protective actions that are required can be implemented in time.

[118] The Chernobyl accident resulted in 31 immediate fatalities (Burgherr & Hirschberg [3.5-4]).

Figure 3.5-2. Comparison of frequency-consequence curves for full energy chains in OECD and non-OECD countries for the period 1970-2008 (source Hirschberg et al [3.5-1])



Source: Hirschberg et al [3.5-1]

Another important study on the consequences of nuclear reactor severe accidents is the US NRC SOARCA project [3.5-5]. This study took a more deterministic approach to the analysis of severe accident consequences. Analyses were performed for two typical US nuclear power plant units, Surry and Peach Bottom, representing the two main types of reactor, PWR and BWR. PSA was used to identify scenarios to be modelled. The selected scenarios were based on loss of all alternating current (AC) electrical power or "station blackout (SBO)" caused by earthquakes more severe than anticipated in the plant's design. The earthquake scenario presents the most severe challenge to the plant operators as well as offsite emergency responders. Two additional scenarios, in which radioactive material could potentially reach the environment by bypassing containment, were analysed for Surry (PWR).

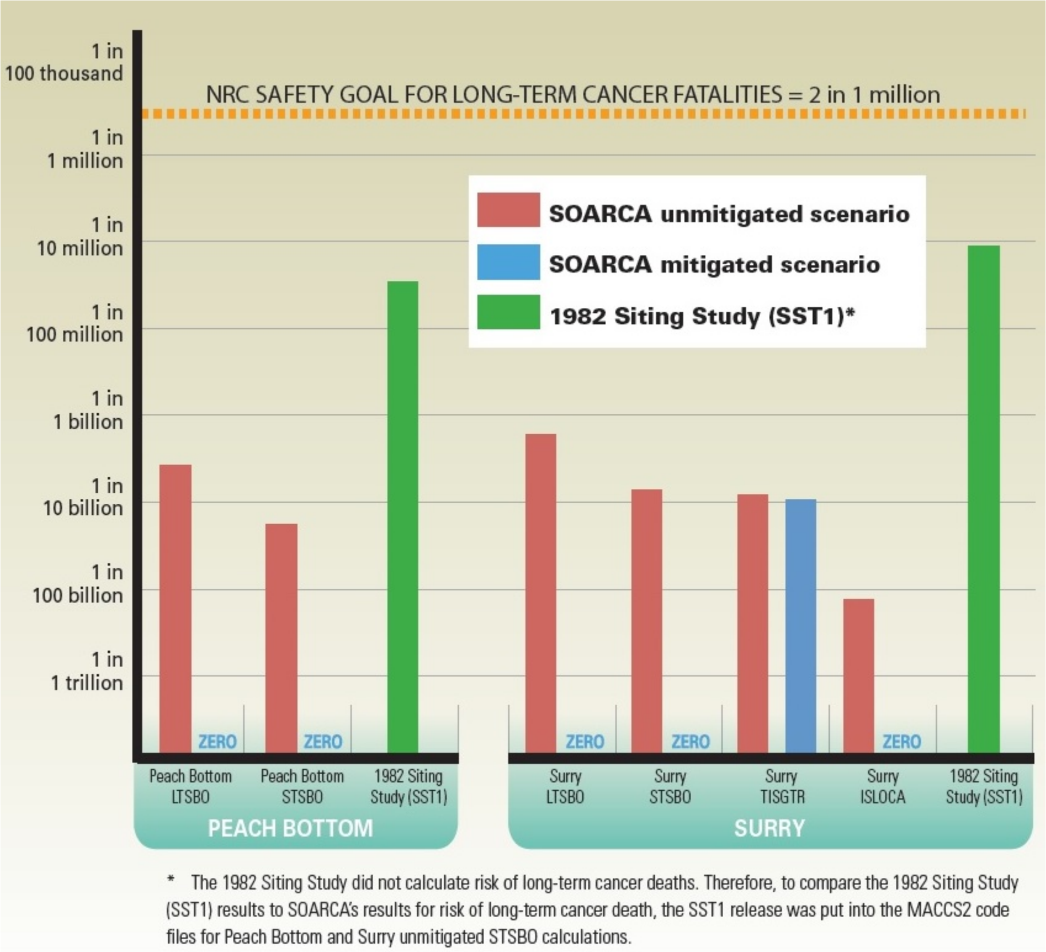
The analysis took into account the effect of off-site emergency response measures designed to protect the public from the effects of the radioactive releases. Results of the analyses are shown in Figure 3.5-3.

The analyses show that even for these severe accidents, the probability of dying from long-term cancer for a member of the public living within 10 miles of the plant is in all cases less than 1 in 1 billion per reactor-year and significantly below the NRC safety goal of 2 in 1 million long-term cancer fatalities per reactor-year.

While the number of human fatalities is an obvious indicator for characterising the maximum severity of accident consequences, and facilitates very well the comparison between technologies, it is important to note that very severe nuclear accidents, as well as non-nuclear severe accidents, can lead to other direct and indirect impacts that might be more difficult to assess. Evaluating the effects of such impacts is not in the scope of the present JRC report, although they can be important for understanding the broader health implications of an accident.

For a comprehensive review of the effects of radiation exposure due to the accident at Fukushima-Daichi nuclear power plant, the reader is referred to the recently published UNSCEAR report [3.5-6]. In this report it is concluded that no adverse health effects among Fukushima residents have been documented that are directly attributable to radiation exposure from the accident and revised estimates suggest that future radiation-associated health effects are unlikely to be discernible. Other effects of the accident on the population and the environment are discussed in the report.

Figure 3.5-3. Scenario-specific risk of dying from long-term cancer for an individual within 10 miles of the plant, per reactor year



Source: [3.5-5][119]

An accident at a Generation III nuclear power plant with the kind of consequences shown in Figure 3.5-1 is a highly improbable event. The calculated frequency of such consequences corresponds to about 10^{-10} per reactor year, or once in ten billion years of operation per reactor. However, such a number of fatalities, even if based on very pessimistic assumptions, has an impact on public perception due to disaster (or risk) aversion.

Disaster aversion refers to an apparent higher importance attached, by some, to a large number of deaths in a single, low-frequency accident compared to an equal number of deaths spread over a larger number of more frequent types of accident. To help put these numbers in perspective, it is useful to compare them with fatality data associated with some other human activities. Compared to a maximum credible number of fatalities of around 30 000 associated with a hypothetical nuclear accident with a frequency of close to 1 in ten billion reactor years of operation, the following are representative of the number of fatalities that occur each and every year due to the mentioned causes:

— Air pollution[120]: In the EU, 400 000 premature deaths per year (burning of fossil fuels contributes significantly to the pollution, so a large number of deaths can be prevented by switching to low-carbon energy sources)

— Tobacco smoke[121]: In the USA, more than 480 000 premature deaths due to smoking; more than 40 000 premature deaths of non-smokers due to second-hand smoke

— Road traffic accidents[122]: In the EU, 22 800 deaths in 2019.

[119] LTSBO – Long-term station blackout; STSBO – Short-term station blackout (battery backup power also lost); ISLOCA - Interfacing Systems Loss-of-Coolant Accident; TISGTR - Thermally Induced Steam Generator Tube Rupture

[120] https://www.eea.europa.eu/highlights/cutting-air-pollution-in-europe

[121] https://www.cdc.gov/tobacco/data_statistics/fact_sheets/health_effects/tobacco_related_mortality/index.htm

3.5.1 References for Chapter 3.5

[3.5-1] Hirschberg, S., Bauer, C., Burgherr, P., Cazzoli, E., Heck, T., Spada, M., Treyer, K., Health effects of technologies for power generation: Contributions from normal operation, severe accidents and terrorist threat, Reliability Engineering and System Safety **145** (2016) 373–387.

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[3.5-4] Burgherr, P. and Hirschberg, S., Comparative risk assessment of severe accidents in the energy sector, Energy Policy 74 (2014) S45–S56.

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[3.5-6] Sources, Effects and Risks of Ionizing Radiation, United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2020 Report, Annex B: Levels and effects of radiation exposure due to the accident at the Fukushima Daiichi Nuclear Power Station: implications of information published since the UNSCEAR 2013 Report (Advance Copy), February 2021.

[122] https://www.europarl.europa.eu/news/en/headlines/society/20190410STO36615/road-fatality-statistics-in-the-eu-infographic

4 Summary DNSH assessment for nuclear energy and recommendations

By using the results and conclusions of the analyses outlined in Chapter 3 (*Summary of results from the state-of-the-art LCA studies on nuclear energy*), the present Chapter 4 provides an overview of the results synthesized and formulates recommendations on the compatibility of nuclear energy with the basic principles and objectives of the Taxonomy.

4.1 Main conclusions of the analyses outlined in Chapter 3.2

Chapter 3.2 provided a detailed comparison of impacts potentially exerted by various electricity generation technologies (e.g. oil, gas, renewables and nuclear energy) on the human health and the environment. The comparison was based on recent LCA studies and utilized science-based evidence only. Note that Chapter 3.2 did not go into the details of potential effects of radioactive materials and radiation on human health and the environment, because these issues were mainly discussed in Chapters 3.3, 3.4 and 3.5.

Main conclusions of the comparison can be summarized as follows:

- Average lifecycle GHG emissions determined for electricity production from nuclear energy are comparable to the values characteristic to **hydropower** and **wind** (see Figure 3.2-6);
- Nuclear energy has very low NO_x (nitrous oxides), SO_2 (sulphur dioxide), PM (particulate matter) and NMVOC (non-methane volatile organic compounds) emissions, the values are comparable to the emissions of **solar PV** and **wind** (see Figure 3.2-8 and -18);
- If other impact categories are considered (e.g. acidification and eutrophication potentials), then nuclear energy is again comparable to **solar PV** and **wind** (see Figure 3.2-10);
- The same is true for freshwater and marine eco-toxicity (see Figure 3.2-11); ozone depletion and POCP (photochemical oxidant creation potential, see Figure 3.2-19);
- Land occupation of nuclear energy is about the same as for an equivalent capacity gas-fired plant, but significantly smaller than wind or solar PV (see Figure 3.2-15).

In addition to the above listed – positive – findings, some areas were identified, where utilization of nuclear energy needs special attention:

- Potential thermal pollution of freshwater bodies

  Large inland nuclear power plants utilizing once-through cooling systems withdraw a large amount of water from the river or lake used as ultimate heat sink for normal plant operation. When the heated-up cooling water is returned to the water body, it represents a significant thermal pollution potential that must be handled adequately. For example, an NPP with 1000 MW_e electric capacity uses about 175 000 – 200 000 m^3/h condenser cooling water, which is warmer than the freshwater body it is taken from by about 10°C, when discharged back to the cooling water outlet channel. In order to avoid harmful thermal pollution effects, the maximum discharge temperature of the condenser cooling water, as well as the maximum temperature of the freshwater body after mixing have to be strictly controlled. Note that for coastal NPPs the thermal pollution of seawater is less of a problem, because the sea represents a practically infinite mixing medium for the warmed-up cooling water if it is discharged into the sea at an appropriate distance from the coast. Water withdrawal options and the avoidance of excessive thermal pollution must be carefully analysed during the site selection process, as well.

- Water consumption

  A general feature of power plants utilizing a specific thermal cycle (e.g. the Rankine cycle) to convert heat to mechanical energy (in our case to the rotation energy of the turbine) is the need for continuous cooling. Chapter 3.2 and 3.3.7 (NPP operations) discuss the various cooling technologies and they highlight that water consumption is very little for the once-through cooling, but technologies using recirculation cooling, evaporative cooling towers or pond cooling usually consume a significant amount of water to compensate for losses due to evaporation. Water consumption characterizing these cooling technologies is comparable to concentrating solar power and coal, for both recirculation and pond cooling (see Figure 3.2-7). During site selection, the available water resources and the potential environmental effects of excessive water consumption must be carefully analysed and an optimal solution must be found, if possible.

Impacts of nuclear energy on the human health and the environment are mostly comparable to hydropower and the renewables, if non-radiological effects are considered.

The analyses outlined in Chapter 3.2 did not reveal any science-based evidence that nuclear energy does more harm to the human health or to the environment than other electricity production technologies already included in the Taxonomy as activities supporting climate change mitigation.

Issues related to water consumption and potential thermal pollution of nuclear energy must be appropriately handled during the site selection, facility design and plant operation phases.

4.2 Main conclusions of the analyses outlined in Chapter 3.3

In Chapter 3.3 the assessments were grouped according to the various lifecycle phases of nuclear energy, in order to obtain a naturally structured picture of the impacts.

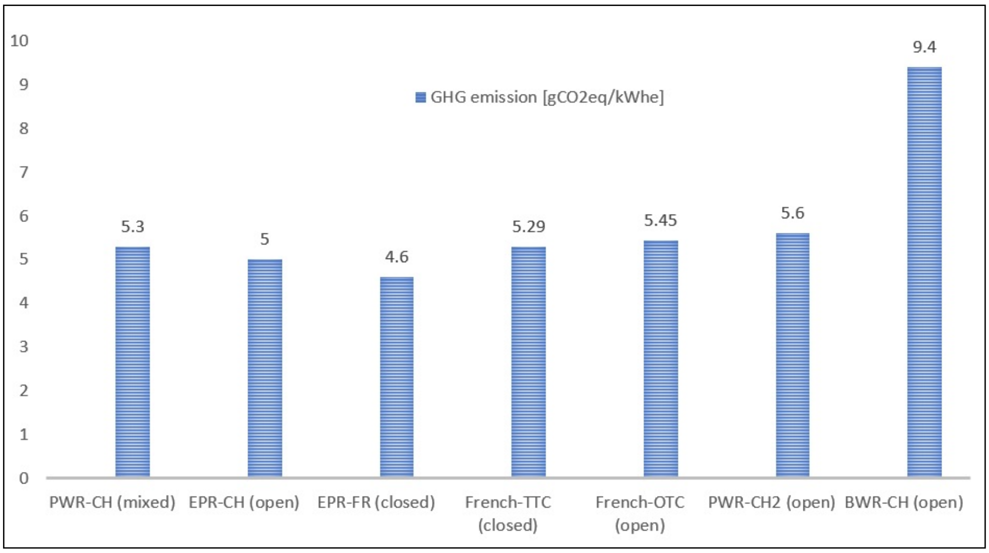
4.2.1 Non-radioactive and potential impact indicators

The assessments have shown that all non-radioactive and potential impact indicators are dominated_by the mining & milling phase, except the GHG emission, where NPP operation gives the largest contribution (see Figure 3.3.1-12] and Tables A.2-1 and A.2-2 in Annex 2). Although NPP operation dominates only GHG emission, it also provides significant contribution to all other impact indicators. This is true for both closed and open fuel cycles.

4.2.1.1 Comparison of GHG emissions from PWRs and a BWR operated in various fuel cycles

Figure 4.2.1-1 compares the results from six PWR LCAs with the result of an LCA carried out for a BWR plant. The different PWR plants were operated in various fuel cycle types (closed, open and mixed), while the BWR used open fuel cycle. It can be seen that the PWR results are rather close to each other, the difference between the highest and lowest value is just around 20%. The BWR shows a significantly higher calculated GHG emission (180% of the PWR average), which is – according to Ref. [4-1] – connected to the fact that in the front-end phase of the analysed BWR plant 50% of the yellowcake production and 50% of the enrichment services were used from the Siberian Chemical Combine (SCC, Russia, Seversk) and these were associated with higher GHG emissions compared to the fuel for the PWRs.

Figure 4.2.1-1. – Comparison of GHG emissions from different NPP types operated in various fuel cycle types

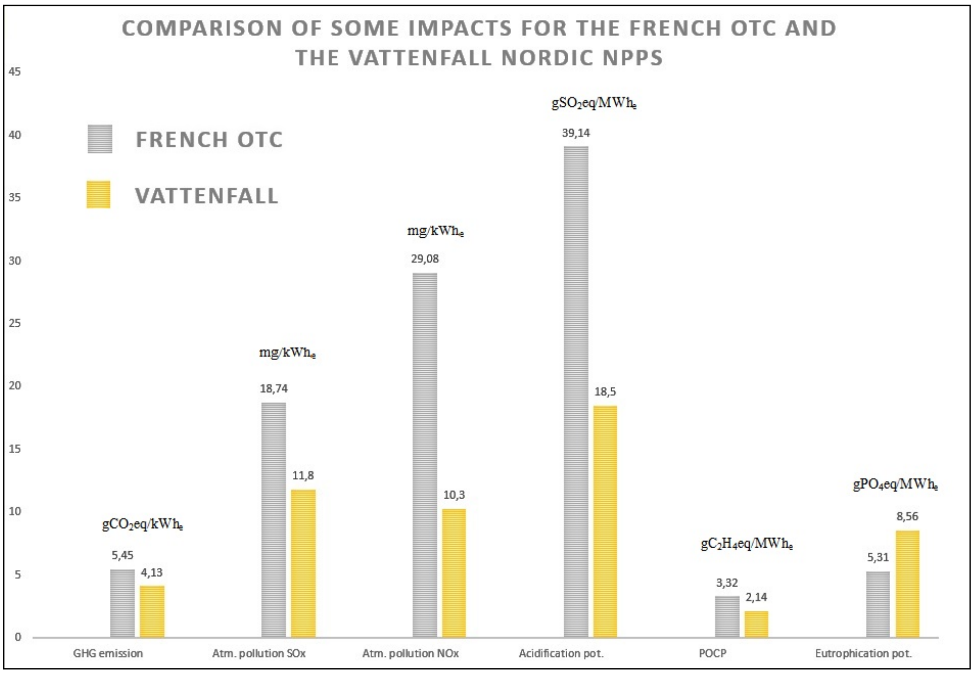


Data from [4-1, -2, -3]

4.2.1.2 *Comparison of impacts from the French and the Vattenfall NPPs*

Reference [4-4] contains the EPD (Environmental Product Declaration) for year 2019 corresponding to the NPP units operated by Vattenfall AB at the Ringhals and Forsmark sites in Sweden. This Vattenfall report is a Type III environmental declaration[123] which was prepared according to the ISO 14025 standard and contains the potential environmental impacts of four BWR and three PWR units. The seven units have a combined generating capacity of 7200 MW_e and they are operated in open fuel cycles.

Figure 4.2.1-2. – Comparison of selected environmental impacts for the French nuclear fleet (with assumed open cycle) and the Vattenfall Nordic NPPs



Data from [4-2 and -4]

Figure 4.2.1-2 shows the comparison of some selected impact indicators[124] for two reactor "fleets", e.g. the French reactors operated in an assumed open cycle and the Vattenfall Nordic NPPs. In most cases the impacts reported by Vattenfall are significantly lower, despite the fact that the EPD considers also the contributions from the transmission grid. The main reason for this was that in the EPD reporting period the contributions from the mining and milling lifecycle phase had been considerably reduced by Vattenfall. The share of the open-pit uranium mines was decreased and about 40% of the uranium supply came from the TENEX company (Russia, Novouralsk), where only reprocessed uranium (RepU) was used for enrichment, thus saving a lot of mining and milling works. The figure also illustrates that one can gain a lot when reducing the emissions in the front-end part of the cycle.

4.2.2 Radioactive impact indicators

For both closed and open cycles, mining & milling is the dominant contributor to the gaseous emissions (due to the radon) and solid VLLW (Very Low Level Waste) production.

In the closed cycle, the NPP operation phase is dominant only in the solid LILW-SL (Short-Lived Low and Intermediate Level Waste) production, but it has significant contribution to the solid ILW-LL (Long-Lived Intermediate Level Waste) production, as well.

In the closed cycle, the reprocessing phase is dominant in the liquid emissions and solid ILW-LL and HLW production. In addition, it has very significant contribution to the gaseous emissions.

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123 Type III declarations are documents prepared according to ISO 14025 and they quantify environmental information on the lifecycle of a product to enable comparisons between products fulfilling the same function.

124 For the sake of comparison only those impact indicators could be selected that were determined in both studies.

In the open cycle – where there is no reprocessing phase – the NPP operation phase is dominant in liquid emissions, plus in the production of solid LILW-SL, ILW-LL and HLW.

Besides these three phases, there is no other nuclear energy lifecycle phase, which provides dominant contribution to any of the impact categories.

If potential impacts on the environment and human health are considered, then the three dominant lifecycle phases of nuclear energy are therefore as follows:

- Uranium mining and uranium ore processing;
- NPP operation (production of electricity by means of nuclear fission reactors)[125];
- Reprocessing of spent nuclear fuel.

The following table shows the lifecycle phases providing dominant contribution to the various impact indicators.

Table 4-1. Impact indicators with the indication of the dominant lifecycle phase

Non-radioactive and radioactive impact indicators	Lifecycle phase with dominant contribution	
	Open fuel cycle	**Closed fuel cycle**
GHG emissions	NPP operation	
Water withdrawal	NPP operation	
Water consumption	NPP operation	
Production of technological waste	NPP operation	
Water pollution	Mining and milling	
Eco-toxicity	Mining and milling	
Human toxicity	Mining and milling	
Land use	Mining and milling	
Atmospheric pollution	Mining and milling	
Acidification potential	Mining and milling	
Eutrophication potential	Mining and milling	
Ozone creation potential	Mining and milling	
Depletion of resources	Mining and milling	
Gaseous radioactive releases	Mining and milling	
Production of solid radioactive waste (VLLW)	Mining and milling	
Production of solid radioactive waste (ILW-SL)	NPP operation	
Liquid radioactive releases	NPP operation	Reprocessing
Production of solid radioactive waste (ILW-LL)	NPP operation	Reprocessing
Production of solid radioactive waste (HLW)	NPP operation	Reprocessing

[125] The "NPP operation" lifecycle phase includes the construction, operation and decommissioning of nuclear power plants, as well as the long-term operation (i.e. service time extension) of these facilities.

Figure 4.2.2-1. Contributions from all lifecycle phases to all impact indicators (closed cycle)

Data from [4-2]

Figure 4.2.2-1 shows the contributions from the individual lifecycle phases to all impact indicators, grouped according to non-radioactive, potential and radioactive impacts. As it is again clearly visible from this combined picture, only the above-mentioned lifecycle phases provide dominant contribution to any of the indicators. Note that the radioactive waste disposal (including the final disposal of high-level waste) does not provide dominant – or even significant – contribution to any of the impact indicators.

Figure 4.2.2-1 plots data corresponding to the closed fuel cycle (TTC), but the picture is very similar if the open fuel cycle is considered, except that reprocessing phase is not present.

4.3 Main conclusions of Chapters 3.4 and 3.5

In addition to the analysis of state-of-the-art lifecycle assessment results, Chapter 3 also discussed the impact of ionizing radiation on human health and the environment (see 3.4) and the potential impact of severe accidents (see 3.5). Conclusions of these sections are as follows:

- The average annual exposure to a member of the public, due to effects attributable to nuclear energy-based electricity production is about 0.2 μSv, which is four orders of magnitude less than the average annual dose due to the natural background radiation[126] (see Figure 3.4-1).
- According to the LCIA (Life Cycle Impact Analysis) studies analysed in Chapter 3.4, the total impact on human health of both the radiological and non-radiological emissions from the nuclear energy chain are comparable with the human health impact from offshore wind energy.
- As far as staff members working at nuclear facilities are concerned, they are protected from the harmful effects of ionizing radiation by strict radioprotection measures monitoring and limiting occupational doses. The ALARA (as low and reasonably achievable) principle is applied also here to optimize plant maintenance works for minimizing worker's radiation doses.
- If health impacts due to normal operation of the various electricity generation technologies are compared, then nuclear energy has the lowest values, both for premature fatalities (caused e.g. by air pollution) and for accident fatalities (e.g. workplace accidents).
- If severe accident fatality rates are compared (see Figure 3.5-1), then the current Western Gen II NPPs have a very low fatality rate ($\approx 5 \cdot 10^{-7}$ fatalities/GWh). This value is much smaller than that characterizing any form of fossil fuel-based electricity production technology and comparable with hydropower in OECD countries and wind power (only solar power has significantly lower fatality rate).
- Severe accidents with core melt did happen in nuclear power plants and the public is well aware of the consequences of the three major accidents, namely Three Mile Island (1979, USA), Chernobyl (1986, Soviet Union) and Fukushima (2011, Japan). The NPPs involved in these accidents were of various types (PWR, RBMK[127] and BWR) and the circumstances leading to these events were also very different. Severe accidents are events with extremely low probability but with potentially serious consequences and they cannot be ruled out with 100% certainty. After the Chernobyl accident, there were focused international and national efforts to develop Gen III nuclear power plants. These plants were designed according to extended requirements related to severe accident prevention and mitigation, for example they ensure the capability to mitigate the consequences of a severe degradation of the reactor core, if such an event ever happens. The main design objective was to ensure that even in the worst case, the impact of any radioactive releases to the environment would be limited to within a few kilometres of the site boundary. The deployment of various Gen III plant designs started in the last 15 years worldwide and now practically only Gen III reactors are constructed and commissioned.
- These latest technology developments are reflected in the very low fatality rate for the Gen III EPR design ($\approx 8 \cdot 10^{-10}$ fatalities/GWh, see Figure 3.5-1). The fatality rates characterizing state-of-the art Gen III NPPs are the lowest of all the electricity generation technologies.
- In addition to the "fatality rate per GWh" metric, severe accidents potentially occurring in the electricity generation industry are characterized by another metric, called maximum consequences. Conservatively estimated values of this metric are rather high for both Gen II and Gen III plants, comparable to the hydropower in non-OECD countries (see Figure 3.5-1). For the EPR design, the quoted reference study predicts 30 000 fatalities as upper bound.

[126] Global average of per capita radiation dose due to natural background is 2400 μSv per annum

[127] The RBMK reactor is a special and differing NPP design and it was constructed in the former Soviet Union only

Note that in Figure 3.5-1 the "maximum consequences" data for the non-nuclear electricity production technologies are real historical data reflecting the officially registered number of casualties (e.g. after a major hydropower-dam accident). Contrary to this, for nuclear energy the "maximum consequences" values correspond to calculated data which were derived by using highly conservative assumptions (e.g. application of a simplified Level 3 PSA model, dense population in the 100 km region around the plant, no off-site mitigation measures, see Ref. [4-5] for more details). In addition, more than 95% of the calculated fatalities can be attributed to latent (i.e. long-term cancer) fatalities, which are strongly influenced by site-specific population data and model-specific assumptions.

The consequence analysis outlined in Ref. [4-6] was prepared in the US NRC SOARCA project and it takes into account the effect of on-site and off-site severe accident mitigation measures, as well. Some related results of the SOARCA project are shown in Figure 3.5-3.

Note that the data plotted in Figure 3.5-3 take into account the effect of mitigation measures and therefore provide more realistic estimates.

4.4 Evaluation and conclusions

The analyses outlined in Chapter 3 revealed several potentially harmful impacts of nuclear energy on human health and the environment. The majority of these impacts can be prevented by careful site selection, appropriate facility design and construction, as well as by rigorous operation and waste management practices. However, some impacts potentially exerted by activities belonging to the three "dominant" lifecycle phases need special attention and management, as follows.

— Uranium mining and ore processing
  - Safe management of rock waste dumps and tailings
  - Prevention of radon emanation (prevention of gaseous radioactive releases)
  - Prevention of dusting and dispersal of solid radioactive substances
  - Ensuring adequate radioprotection of workers and the public
  - Protection of water bodies (avoidance of surface- and groundwater pollution)
  - Minimisation of solid radioactive waste production (VLLW only)
  - Ensuring adequate remediation of closed or abandoned mining sites.

— NPP operation
  - Prevention of thermal pollution related to water withdrawal
  - Limitation of water consumption
  - Limitation of conventional releases with focus on toxic materials
  - Ensuring adequate radioprotection of workers and the public
  - Limitation of gaseous and liquid releases
  - Limitation of solid radioactive waste production (mainly VLLW and LILW-SL)

— Reprocessing of spent nuclear fuel
  - Ensuring nuclear safety during operations (in particular sub-criticality, cooling and containment of radioactive materials)
  - Ensuring adequate radioprotection of workers and the public
  - Limitation of conventional releases with focus on toxic materials
  - Limitation of gaseous and liquid releases
  - Limitation of solid radioactive waste production (mainly ILW-LL)
  - Limitation of solid radioactive waste production (HLW)

— Interim storage and final disposal of HLW (including high-level vitrified waste generated during reprocessing

  - Ensuring nuclear safety during operations (in particular sub-criticality, cooling and containment of radioactive materials)
  - Ensuring adequate radioprotection of workers and the public
  - Limitation of conventional releases with focus on toxic materials
  - Limitation of gaseous and liquid releases
  - Limitation of conventional (technological) waste production

The above challenges can be duly averted, as there exist appropriate measures to prevent the occurrence of the potentially harmful impacts or mitigate their consequences. The prevention can be achieved by using existing technology at reasonable costs.

It can therefore be concluded that all potentially harmful impacts of the various nuclear energy lifecycle phases on human health and the environment can be duly prevented or avoided. The nuclear energy-based electricity production and the associated activities in the whole nuclear fuel cycle (e.g. uranium mining, nuclear fuel fabrication, etc.) do not represent significant harm to any of the TEG objectives, provided that all specific industrial activities involved fulfil the related Technical Screening Criteria.

The requirements ensuring the fulfilment of the necessary limitations and prevention measures are described in the associated TSC tables (see Chapter 5 and Annex 4 for details).

4.5 Recommendations

Motivated by the above conclusions it is now justified to proceed with the development of appropriate Technical Screening Criteria (TSC) for the nuclear energy-based electricity generation according to the approach practiced by the TEG in their work (see [4-7] and [4-8]).

In the TSC development process the nuclear energy-based electricity generation can be considered as an activity significantly contributing to the climate change mitigation objective. Other associated industrial activities in the nuclear fuel cycle (uranium mining & milling, fabrication of nuclear fuel, reprocessing of spent nuclear fuel, final disposal of high-level radioactive waste, etc.) can be treated as activities enabling the safe and sustainable utilization of nuclear energy.

The process for developing the relevant TSC tables is outlined in Chapter 5.

4.6 References for Chapter 4

[4-1] X. Zhang, C. Bauer: Life cycle assessment (LCA) of nuclear power in Switzerland, Final Report, Paul Scherrer Institute (PSI), 16 July 2018

[4-2] Ch. Poinssot et al.: Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211

[4-3] A. Simons and C. Bauer: Life cycle assessment of the European pressurized reactor and the influence of different fuel cycle strategies, Proc. of the Institution of Mechanical Engineers, Part A: Journal of Power and Energy, April 2012 (http://www.sagepublications.com)

[4-4] Certified Environmental Product Declaration EPD of Electricity from Vattenfall Nordic Nuclear Power Plants, UNCPC Code 17, Group 171 – Electrical energy, S-P 00923, 2019-12-31, Vattenfall AB, Sweden

[4-5] S. Hirschberg et al.: Comparative assessment of severe accidents in the Chinese energy sector, PSI Report N° 03-04 (ISSN 1019-0643), Paul Scherrer Institute, March 2003

[4-6] Modelling potential reactor accident consequences, State-of-the-art reactor consequence analyses (SOARCA): Using decades of research and experience to model accident progression, mitigation, emergency response, and health effects, US NRC, NUREG/BR-0359, Revision 1, December 2012

[4-7] Financing a Sustainable European Economy, Technical Report, Taxonomy: Final report of the Technical Expert Group on Sustainable Finance, March 2020

[4-8] Taxonomy Report, Technical Annex, Updated methodology & Updated Technical Screening Criteria, March 2020

5 Illustrative Technical Screening Criteria for selected lifecycle phases of nuclear energy

5.1 Background and general considerations

Chapter 5 provides the description of the approach for developing illustrative Technical Screening Criteria (TSC) for selected lifecycle phases of nuclear energy. The actual TSC tables corresponding to the selected lifecycle phases are given in Annex 4, here only the method for their development is described.

The DNSH sections in the corresponding TSC tables were filled in by using the results of the DNSH analyses performed in Chapter 3.3 for the various lifecycle phases of nuclear energy. If there was a comparable TSC table published in the Technical Annex of the Taxonomy Report (see Ref. [5-2]) or in the recent revision of TSC tables (see Ref. [5-3]) then this table was adapted for the specificities of nuclear energy. If there was no table presented for a similar activity in [5-2] or in [5-3], then a TSC table had to be developed completely new.

Draft TSC tables were developed for the following selected lifecycle phases of nuclear energy:

— uranium mining and ore processing ("mining and milling");

— NPP operation (production of electricity by means of nuclear fission reactors)*;

— reprocessing of spent nuclear fuel;

— interim storage and final disposal of high-level radioactive waste (including high-level vitrified waste generated during reprocessing);

** Note that the "NPP operation" phase includes the construction, operation and decommissioning of nuclear power plants, as well as the long-term operation (i.e. service time extension) of these facilities.*

The selection of the first three LC phases in the above list is justified by the results of lifecycle analysis: all non-radioactive and potential impact indicators are <u>dominated </u>by the mining & milling phase, except the GHG emission, where NPP operation gives the largest contribution (see Figure 3.3.1-12 and Tables A.2-1 and A.2-2 in Annex 2). Although NPP operation dominates only GHG emission, it also provides significant contribution to other impact indicators. This is true for both closed and open fuel cycles (see also Chapter 4).

If radioactive impact indicators are considered, then mining & milling is the dominant contributor to the <u>gaseous emissions</u> and <u>solid VLLW production</u> (for both the closed and open cycles).

In the closed cycle, the NPP operation phase is dominant only in the <u>solid LILW-SL production</u>, but has significant contribution to the <u>solid ILW-LL production</u>, as well.

In the closed cycle, the <u>reprocessing phase</u> is dominant in the <u>liquid emissions</u> and <u>solid ILW-LL and HLW production</u>; but it has very significant contribution to the <u>gaseous emissions</u>, as well.

In the open cycle – where there is no reprocessing phase – the <u>NPP operation phase</u> is dominant in <u>liquid emissions</u>, plus in the production of <u>solid LILW-SL, ILW-LL and HLW</u>.

Finally, the "interim storage and final disposal of high-level radioactive waste" phase (including high-level vitrified waste generated during reprocessing) was selected to address some concerns expressed by the TEG in connection with the long-term safety and potential environmental impacts of final disposal of HLW.

In order to be in line with the TSC listed in the TEG reports [5-1] and [5-2], kept rather qualitative than quantitative, usually it was not necessary to give precise limit values in the TSC tables developed for the selected LC phases of nuclear energy. Therefore, often fulfilment of regulatory requirements and/or regulatory limits are provided as proof of not doing harm to the environment. It was supposed that this approach is accepted also in the case of nuclear energy.

According to [5-1], the TSC tables corresponding to climate change <u>mitigation</u> must address the following environmental objectives of TEG:

(2) Adaptation = climate change adaptation;

(3) Water = protection of water and marine resources;

(4) Circular Economy = transition to a circular economy;

(5) Pollution = pollution prevention and control;

(6) Ecosystems = protection and restoration of biodiversity and ecosystems.

The fulfilment of the first environmental objective ((1) Mitigation = climate change mitigation) is discussed in the LCA section (see Chapter 3.3) in the frame of GHG emission analysis, but anyhow it is the overall condition to include a specific activity into the Taxonomy as a contributor to climate change mitigation.

Note that no TSC were prepared for the climate change adaptation objective, because the nuclear energy was primarily considered as a potential contributor to climate change mitigation.

The fulfilment of the first environmental objective ((1) = Climate change mitigation) is determined from the magnitude of the associated GHG emissions and the Taxonomy uses it to decide whether a specific electricity generation technology can be included in the Taxonomy or not.

The final TEG report ([5-1]) states that "*Any electricity generation technology can be included in the Taxonomy if it can be demonstrated, using an ISO 14067 or a GHG Protocol Product Lifecycle Standard compliant Product Carbon Footprint (PCF) assessment, that the life cycle impacts for producing 1 kWh of electricity are below the declining threshold*". (The threshold is currently set to 100g CO_{2e}/kWh). Note that the ISO 14067 standard is focusing on the determination of the carbon footprint of a product, and it is fully consistent with the ISO 14040 and ISO 14044 international standards on life cycle assessment (LCA).

5.2 Correspondence to the NACE codes

NACE codes[128] corresponding to the main lifecycle phases are listed in Annex 3 (see Table A.3-1).

If we consider the various lifecycle phases of nuclear energy production, then the involved activities depend on whether an "open" or a "closed" fuel cycle is used.

— The **front-end** of the nuclear fuel cycle can be covered by the following NACE codes:

— **B.07.21** (mining & milling + yellowcake preparation) + **C.20.13** (conversion of yellowcake to UF_6 + enrichment) + **C.24.46** (manufacture of nuclear fuel elements and fuel assemblies);

— The construction phase consists of **C.25.30** (manufacture of nuclear reactors) and **F.42.22** (construction of utility projects for electricity and telecommunications);

— The operation phase corresponds to **D.35.11** (electricity generation) + **C33.11** (repair and maintenance of nuclear reactors) or – in case of a nuclear cogeneration plant - **D.35.30** (steam and air conditioning supply) + **C33.11**;

— The decommissioning and site remediation phase is covered by **E.39.00** (remediation activities and other waste management activities);

— Activities in the **back-end** of the nuclear fuel cycle depend on the type of the cycle applied (open or closed):

- Open fuel cycle - the back-end here involves the management and final disposal of spent nuclear fuel only, and it is covered by activities **E.38.12** (collection of hazardous waste) + **E.38.22** (treatment and disposal of hazardous waste);

- Closed fuel cycle – The back-end here starts with the activity **C.20.13** (manufacture of other inorganic basic chemicals = reprocessing of nuclear fuel). If no MOX fuel is produced then the closed cycle is finished by activities **E.38.12** + **E.38.22 (**management and disposal of high level vitrified waste generated during reprocessing). If MOX fuel is to be fabricated, then activity **C.24.46** (manufacture of nuclear fuel elements and fuel assemblies) is also present in the back-end. If RepU (reprocessed uranium) is also used for nuclear fuel production, then it is re-enriched (activity C.20.13) and the enriched uranium is used to fabricate UO_2 fuel in activity C.24.46.

The NACE codes used in the TSC tables were determined according to the above considerations.

[128] NACE = Statistical classification of economic activities in the EC. The abbreviation comes from its French name (Nomenclature statistique des Activités économiques dans la Communauté Européenne). The current version (Revision 2) is defined in Regulation (EC) N° 1893/2006.

5.3 Development of Technical Screening Criteria

Our approach to define TSC for the selected nuclear energy lifecycle phases followed the process as described in the TEG reports (see Refs. [5-1], [5-2] and [5-3]). Potentially harmful impacts of nuclear energy were identified by using results from relevant lifecycle analysis studies and by analysing the underlying technological processes. In order to properly characterize environmental and human health impacts, studies using internationally acknowledged and commonly used impact indicators were utilized. The analysis also included a detailed study of relevant legal aspects and regulations, focusing on EU Directives and industry-specific standards or recommendations. After identifying key potential impacts on environment and human health, applicable means to eliminate or mitigate these impacts were enumerated. The results were outlined in Chapter 3.3 for each phase in the nuclear lifecycle and for both closed and open fuel cycles. Chapter 3.3 constitutes the "Do No Significant Harm" (DNSH) analysis section of our study, where the potentially harmful impacts of a specific lifecycle phase are summarized in the "Importance of impacts on the TEG environmental objectives" tables. These tables were then used as starting points to define appropriate TSC.

By using the results and conclusions of the above analyses, one can derive and synthesize data and other information (e.g. applicable standards or relevant best available techniques) required to fill in the corresponding DNSH sections in the TSC tables corresponding to the various lifecycle phases of nuclear energy.

During the development of the TSC the relevant non-nuclear criteria were adjusted to specific nuclear lifecycle conditions and were complemented by criteria accounting for radiation protection and radioactive emission control aspects of nuclear energy.

The relevant EU directives and regulations – together with the national laws and regulations in effect – are considered as legal obligations to be compulsorily satisfied in the EU and their fulfilment is a minimum condition for eligibility.

In the following subchapters the main features of the TSC developed for the selected four nuclear energy lifecycle phases are discussed.

5.4 Development of TSC for the NPP operation phase

5.4.1 Introduction

As TSC table examples, the TSCs corresponding to the "Electricity generation from geothermal energy" and "Electricity generation from gaseous and liquid fuels" (see Annex 1 in Ref. [5-3]) were utilized as starting point for the "NPP operation" phase. TSC related to other electricity generation technologies (e.g. hydropower and bioenergy) were also considered and utilized, if feasible.

Note that the fourth environmental objective (Transition to a circular economy) is not filled in [5-3] for several electricity generation activities, but it could be filled in with meaningful content for the NPP operation (e.g. minimized conventional and radioactive waste production).

As discussed in detail in subchapter 1.3.2, a nuclear power plant is a special electricity generating facility utilizing the controlled nuclear fission to produce heat, which is then converted to electricity by means of appropriate technological processes. An NPP can be abstracted as a large conventional thermal power plant, where the "boiler" part used for combusting gas, oil, coal, biomass, etc. has been replaced by a nuclear reactor, accommodated in specially constructed reinforced buildings forming the so called nuclear island of an NPP. Outside of the nuclear island, the applied equipment and characteristic technological processes do not essentially differ from those used in conventional power plants and one can make use of certain analogies when developing TSC for these plant sections.

The fuel extraction/production and waste treatment processes for an NPP and for a conventional thermal power plant radically differ and here no analogies can be used to develop appropriate TSC. The front-end and back-end of the nuclear fuel cycle are entirely nuclear-specific activities, which must be handled separately, because they have an authentic TSC set.

The pivotal difference between a nuclear and a conventional thermal power plant is the presence of radioactive materials in the NPP during its operation and decommissioning phases. Radiation levels above certain thresholds are definitely harmful and therefore in an NPP adequate measures must be taken to protect the operating personnel, the public and the environment from the harmful effects of radioactive materials. The obligation to introduce and practice these adequate protection measures is reflected in the

TSC, together with the liability for the consistent application of the ALARA (as low as reasonably achievable) principle for radiation protection and to limit environmental impacts.

5.4.2 Nuclear safety criteria

The TSC table for electricity generation from nuclear energy (see Annex 4) considers two basic cases:

1. Extension of the service time of existing nuclear power plants and
2. Construction and operation of new nuclear power plants.

Considering nuclear safety requirements for existing NPPs, the compliance with the WENRA Safety Reference Levels (RLs) for Existing Reactors (see Ref. [5-4]) is required as a minimum.

The Western European Nuclear Regulators' Association (WENRA) develops a harmonized approach to nuclear safety since 2006, when the first set of RLs for operating NPPs was published. The RLs reflect expected practices to be implemented in the WENRA countries and they primarily focus on safety of the reactor core and spent nuclear fuel. The RLs are regularly revised when new knowledge and experience are available, for example, the current version of the RLs takes into account the lessons learned from the Fukushima accident and the insights from the EU stress tests.

Compliance with the Euratom Nuclear Safety Directive (NSD) [5-5] is also required. The nuclear safety objective specified in article 8a of the NSD requires that nuclear installations are designed, sited, constructed, commissioned and operated with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:

- Early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;
- Large radioactive releases that would require protective measures that could not be limited in area or time.

This objective applies to new nuclear installations, but for existing nuclear installations (those having been granted a construction licence for the first time on or before 14 August 2014), it must be used as a reference for the timely implementation of reasonably practicable safety improvements, including in the framework of the periodic safety review. Similar requirements are specified in the WENRA Safety Objectives for New Nuclear Power Plants [5-6] (see below). In the EU, the continuous improvement of nuclear safety of existing reactors is a general requirement; it is reflected in the NSD and in the WENRA RLs. This gradual improvement process is efficiently assisted by the periodic safety reviews (further details are provided later in this chapter).

New nuclear power plants must at least meet the WENRA Safety Objectives for New Nuclear Power Plants [5-6]. WENRA expects new NPPs to be designed, sited, constructed, commissioned and operated in line with these objectives. The [5-6] objectives promote the defence-in-depth approach at all levels of plant protection and require that multiple failure events and core melt accidents should be considered in the design of new NPPs. WENRA requires that accidents with core melt which would lead to early or large radioactive releases have to be practically eliminated. For those accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public and the environment, and that sufficient time is available to implement these measures (see [5-6] for more details). These requirements are meant to ensure that even accidents with core melt have limited consequences on the public, even in the vicinity of the NPP.

Although the WENRA safety objectives outlined in [5-6] are meant for new NPPs only, these objectives should also be used as a reference to help identify reasonably practicable safety improvements for existing plants during periodic safety reviews.

For new reactors, full compliance with the article 8a nuclear safety objective of the Euratom Nuclear Safety Directive [5-5] is required, if their construction licence was granted for the first time after 14 August 2014.

Among other important provisions, the nuclear safety objective of the NSD requires that nuclear installations are designed, sited, constructed, commissioned and operated with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:

- Early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;

— Large radioactive releases that would require protective measures that could not be limited in area or time.

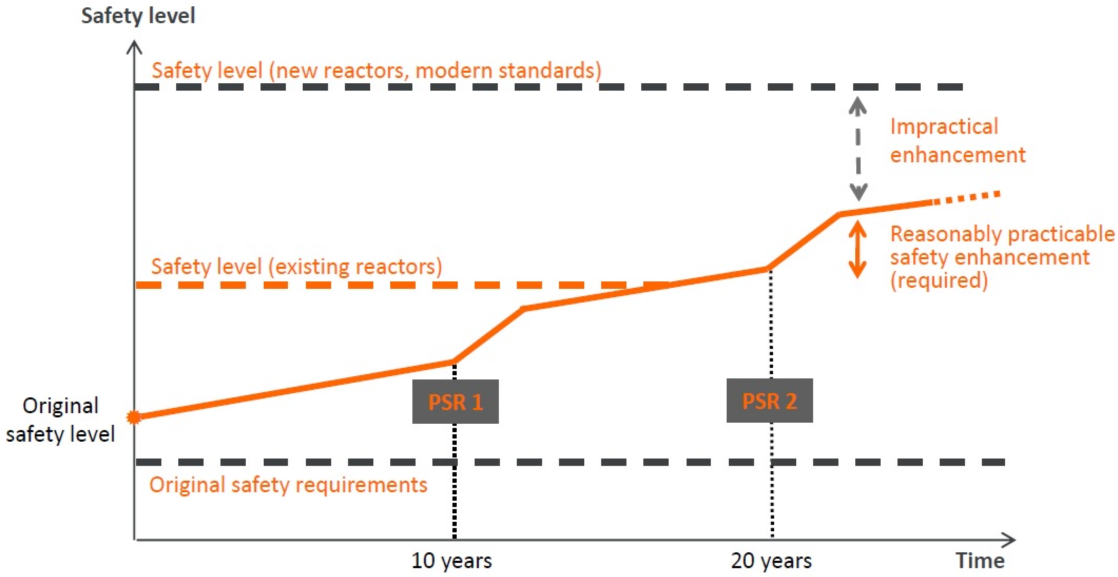
The NSD establishes the principle that the off-site measures required to protect the public and the environment in case of a severe accident must be limited both in terms of the area and the timescale over which they need to be implemented. Note that these provisions are fully in line with the WENRA safety objectives for new NPPs.

The service time extension activity (it is also often referred to as LTO = long-term operation or LTE = lifetime extension) obviously concerns existing reactors only. Instead of shutting down an operating NPP when reaching its initial design service time (usually 30 or 40 years), the plant's Operator may decide to continue the electricity production for an additional 10, 15 or 20 years. The main condition for LTO is that the nuclear safety of the facility can be maintained at a sufficiently high level during the continued operation and the environmental impacts are acceptable during the extended operation.

From the point of view of climate change mitigation, the benefits of LTO are apparent:

1. A sizeable, ready-to-deliver, low-carbon and dispatchable electricity generating capacity is gained immediately;
2. The environmental burden caused by the construction of an equivalent replacement capacity[129] can be saved, as usually LTO does not involve large construction activities[130].

Figure 5-1. Basic concept of WENRA to achieve the continuous improvement of nuclear safety for existing nuclear power plants



Source: [5-7]

Some 30 or 40 years ago, the currently operating NPPs were constructed according to standards that were considerably different from those used today. Meanwhile the regulatory requirements evolved considerably, together with the scientific and engineering background of safety demonstration. In the EU, the basic means to continuously improve the nuclear safety of existing reactors is the system of periodic safety reviews, supplemented by a continuous activity to review and analyse the plant design and operation and identify opportunities for improvement taking into account operating experience, safety research, and advances in science and technology. The role of the periodic safety reviews and the basic concept of WENRA for achieving the continuous improvement of nuclear safety for existing NPPs is illustrated in Figure 5-1.

129 It does not make crucial difference whether the replacement capacity is established by deploying renewables or by constructing a new large thermal power plant.

130 When the LTO is accompanied by a plant power uprating and/or equipment modernization program, then there are associated construction works, but the scope of these activities represents only a small fraction of those related to the construction of a new NPP.

5.4.3 Other criteria

Radioprotection provisions must comply with the Euratom Basic Safety Standards [5-8] or – for activities outside the EU – with the latest recommendations of the ICRP (International Commission on Radiological Protection) [5-9]. The above two sets of requirements represent today the state-of-the-art of radioprotection measures and protocols.

As the summary DNSH assessment in Chapter 4 states, during NPP operation the issues related to cooling water consumption and potential thermal pollution of freshwater bodies must be addressed adequately. TSC addressing the thermal pollution issue are based on the BREF ICS [5-10] document applying the water quality limitations laid down in Directive 78/659/EEC [5-11]. Directive 78/659/EEC was repealed by the WFD (Water Framework Directive) [5-12] in 2013. Note that after the introduction of WFD several regulations previously included in the Directive 78/659/EEC (e.g. those related to thermal pollution of freshwater bodies) are covered by national regulations or by the environmental permit of the licence-obligated industrial facility.

Non-radioactive emissions are limited by the application of ranges corresponding to best available techniques as outlined in the [5-13] document for large combustion plants. Analogously, for NPPs with thermal power between 1 MW and 50 MW, the limitations set by the [5-14] document for medium size combustion plants should be adopted.

Limitations related to radioactive discharges to air, water bodies and ground (soil) must comply with the corresponding EU regulations (with BSS [5-8] and the Drinking Water Directive [5-15]). For activities performed outside the EU, the ICRP recommendations [5-9] are relevant.

The draft TSC for electricity generation from nuclear energy can be found in Annex 4.

5.5 Development of Technical Screening Criteria for the uranium mining and milling phase

Note that in its work TEG considered aspects of the NACE Macro-Sector B (*Mining and quarrying*) to the extent these supported activities in Macro-Sector C (*Manufacturing*). However, TEG did not undertake a full evaluation of the mining and quarrying sector, therefore related TSC tables were not produced and they are not available in Ref. [5-2] or [5-3]. Therefore, we could not use any TSC table developed by the TEG for NACE Macro-Sector B.

Instead, we used as starting point for the uranium mining and processing activity a relevant TSC table from NACE Macro-Sector C, namely "3.5 Manufacture of other low-carbon technologies". This is a TSC table for an enabling activity and in our approach the "uranium mining and milling" was considered as an enabling activity, in accordance with Article 10(1), point (i) of Regulation (EU) 2020/852 (see [5-16]).

Article 10(1), point (i) of Regulation (EU) 2020/852 refers to activities enabling any of the activities listed in points (a) to (h) of paragraph 10(1) in accordance with Article 16. Article 10(1) point (a) refers to "generating, transmitting, storing, distributing or using renewable energy in line with Directive (EU) 2018/2001[131]" and in this sense concerns only activities related to renewable energy. If nuclear energy can be included in the Taxonomy as an activity significantly contributing to the climate change mitigation objective, then all related raw material mining, processing and fuel-manufacturing activities should be included as activities enabling nuclear energy based electricity production.

Where feasible, the DNSH assessment was split across three uranium-mining technologies:

— Open pit mines;

— Underground mines;

— In-situ leaching (ISL) mining.

The reason for this separation was the large difference between the applied technologies and consequently the differences in the potentially harmful effects to human health and environment.

Criteria defined in the TSC table focus on ensuring the fulfilment of the following requirements:

— Safe management of rock waste dumps and tailings;

[131] Directive (EU) 2018/2001 of the European Parliament and of the Council of 11 December 2018 on the promotion of the use of energy from renewable sources

— Prevention of radon emanation (limitation of gaseous radioactive releases and dusting);
— Protection of water bodies (avoidance of surface- and groundwater pollution);
— Minimisation of the production of solid radioactive waste (VLLW only);
— Ensuring adequate remediation of closed or abandoned mining sites.

The proposed TSC table for the uranium mining & milling lifecycle phase is provided in Annex 4.

5.6 Development of TSC for the reprocessing of spent nuclear fuel

The NACE code for this activity is C.20.13 = Manufacture of other inorganic basic chemicals (C.20 = Manufacture of chemicals and chemical products). Note that this code also includes enrichment of uranium and thorium ores and nuclear fuel reprocessing.

In document [5-3] (Annex II) under this NACE code TSC tables are provided for three activities: manufacture of disodium carbonate, chlorine and black carbon. These activities are included in Annex II because they provide substantial contribution to climate change adaptation.

In our approach, the reprocessing of spent nuclear fuel is an enabling activity, because it contributes to manufacturing of nuclear fuel and slightly decreases the volume & radiotoxicity of high-level radioactive waste to be disposed of. Therefore we could not utilize the above three TSC tables as example, because they were related to climate change adaptation. Instead, we again used as starting point for the TSC for reprocessing the TSC table "3.5 Manufacture of other low-carbon technologies" provided in Annex I of Reference [5-3].

Criteria defined in the TSC table focus on ensuring the fulfilment of the following requirements:

— Ensuring nuclear safety during the operations (in particular sub-criticality, cooling and containment of radioactive materials);
— Ensuring adequate radioprotection of workers;
— Limitation of conventional releases with focus on toxic materials;
— Limitation of gaseous and liquid radioactive releases;
— Limitation of radioactive waste generation;

The proposed TSC table for the reprocessing lifecycle phase is provided in Annex 4.

5.7 Development of TSC for the interim storage and final disposal of spent fuel and high-level radioactive waste

Activities related to the interim storage and final disposal of spent fuel and high-level radioactive waste (including high-level vitrified waste generated during reprocessing) were considered as activities belonging to NACE codes **E.38.12** (collection of hazardous waste) and **E.38.22** (treatment and disposal of hazardous waste).

During its work, the TEG considered the NACE codes for water, sewerage, waste and remediation and identified nine economic activities that offer a substantial contribution for climate mitigation, see Chapter 5 of Ref. [5-2]. However, the TEG concluded that NACE Codes **E38.12**, **E38.22** and E38.31 (dismantling of wrecks) were of less relevance from a climate mitigation perspective and no TSC tables were developed for them at this stage. They will be considered at a later stage, as these activities are important enablers for subsequent material recovery, reuse and recycling activities.

Therefore, no example TSC tables were available for use as starting point for our work, but the TEG developed some TSC for the non-hazardous waste (e.g. E38.11 – collection and transport of non-hazardous waste in source-segregated fractions; E38.21 & F42.99 – anaerobic digestion of bio-waste; E38.21 & F42.99 – composting of bio-waste, both in Annex I and Annex II of [5-3]).

However, the TSC tables listed in Annex I (i.e. for climate change mitigation) are rather simple and provide very limited help to develop the TSC for final disposal of HLW. Therefore we used them only as "skeletons" and developed the required TSC from scratch. Formally the "disposal of HLW" was considered as enabling activity, because the safe management and adequate final disposal of radioactive waste – among other conditions – contribute to the long-term sustainability of nuclear energy.

Potentially harmful effects of the following related activities shall be regulated by the requirements entered into the TSC table:

- interim storage of spent nuclear fuel;
- interim storage of vitrified waste generated during reprocessing;
- construction, operation and safe closure of deep-geological repositories (DGRs) for the final disposal of high-level radioactive waste;
- post-closure period of the repository;

Criteria defined in the TSC table focus on ensuring the fulfilment of the following requirements:

- Ensuring nuclear safety during the operations, including transport manoeuvres (in particular sub-criticality, cooling and containment of radioactive materials);
- Ensuring adequate radioprotection of workers during the operations;
- Limitation of gaseous and liquid releases during operation and post-closure;
- Limitation of conventional waste generation (e.g. during excavation, manufacturing, decommissioning and dismantling of encapsulation plant and auxiliary facilities);
- Limitation of migration of radionuclides from the repository to the accessible biosphere;
- Ensuring the fulfilment of Taxonomy-specific requirements.

Disposal of low level and short-lived intermediate level waste is less challenging than disposal of high level waste, and thus it is considered that the Technical Screening Criteria as developed for interim storage and disposal of high level waste and spent fuel cover as well the disposal of low level and short-lived intermediate level waste.

The proposed TSC table for the interim storage and final disposal of high-level radioactive waste lifecycle phase is provided in Annex 4.

5.8 References for Chapter 5

[5-1] Financing a Sustainable European Economy, Technical Report, Taxonomy: Final report of the Technical Expert Group on Sustainable Finance, March 2020

[5-2] Taxonomy Report, Technical Annex, Updated methodology & Updated Technical Screening Criteria, March 2020

[5-3] Annex to the Commission Delegated Regulation (EU) xxx/xxx supplementing Regulation (EU) 2020/852 of the European Parliament and of the Council by establishing the technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change mitigation or climate change adaptation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives; ARES (2020)6979284 - 20/11/2020, EC, 20 November 2020

ANNEX I – Technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change mitigation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives

ANNEX II – Technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change adaptation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives

[5-4] WENRA Safety Reference Levels for Existing Reactors, Update in relation to lessons learned from TEPCO Fukushima Dai-Ichi accident, Report of the WENRA RHWG, 24th September 2014

[5-5] Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014

[5-6] Safety of new NPP designs, Study by the WENRA RHWG, March 2013

[5-7] Wanner, H., Safety Requirements for Long Term Operation or Ageing Aspects and for Design, Construction and Operation of New Nuclear Power Plants, ENSREG Conference, Brussels, 29 June 2015 (http://www.ensreg.eu/sites/default/files/1.1-Wanner.pdf)

[5-8] Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

[5-9] 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

[5-10] BREF ICS – Integrated Pollution Prevention and Control (IPPC) Reference Document on the application of Best Available Techniques to Industrial Cooling Systems, EC, December 2001

[5-11] Council Directive of 18 July 1978 on the quality of fresh waters needing protection or improvement in order to support fish life (78/659/EEC)

[5-12] Directive 2000/60/EC of the European Parliament and of the Council of 23 October 2000 establishing a framework for Community action in the field of water policy

[5-13] T. Lecomte et al.: Best Available Techniques (BAT) Reference Document for Large Combustion Plants; Report EUR 28836 EN, JRC Science for Policy Report, 2017

[5-14] Directive (EU) 2015/2193 on the limitation of emissions of certain pollutants into the air from medium combustion plants

[5-15] Council Directive 98/83/EC of 3 November 1998 on the quality of water intended for human consumption

[5-16] Taxonomy Regulation: Regulation (EU) 2020/852 of the European Parliament and of the Council of 18 June 2020 on the establishment of a framework to facilitate sustainable investment, and amending Regulation (EU) 2019/2088

PART B

Specific assessment on the current status and perspectives of long-term management and disposal of radioactive waste

1 Radioactive waste management: main principles and legal framework

1.1 Main principles of radioactive waste management

The fundamental safety objective applicable to all facilities and activities handling radioactive materials is to protect the people and the environment from the harmful effects of ionizing radiation [1-7]. Thus, the basic and foremost goal of radioactive waste management is to ensure that the radioactive waste materials are contained and sequestered from the biosphere throughout all stages of waste management.

During the operational lifecycle of nuclear energy, including decommissioning, interim storage and emplacement of waste in the final repository, the waste containment and sequestration objectives have to be implemented for discrete, somewhat limited time lengths. Direct monitoring and intervention by the operators ensures maintaining the safe functions of all the shielding and containment barriers isolating the radioactive waste. No radionuclides are released from the waste and no radiological pollution and/or harm to the biodiversity and ecosystems (including marine environment) occur during the operational lifecycle stages.

For the final disposal of radioactive waste, the objective has to be fulfilled until the radiotoxicity level of the waste has decayed sufficiently to ensure that the maximum allowed dose contribution set by the relevant regulation is not exceeded. For the waste containing long-lived radionuclides, in particular for the spent nuclear fuel and the vitrified waste from reprocessing (High-Level Waste, or HLW), which are characterized by high concentration of long-lived radionuclides and the most intense radioactivity level, the decay time required to reduce the radiotoxicity down to the relevant threshold can be of the order of a hundred thousand years. Ensuring the safe containment and isolation of the waste for very long timespans cannot rely upon active human monitoring and intervention.

The solution to this scientific and technological challenge is the disposal of spent fuel and HLW in a remote (deep) and stable geological formation (Deep Geological Repository, or DGR). After filling, sealing and closing the repository, fulfilling the basic objective does not require long-term active monitoring. Multiple barriers, both man-made and natural (provided by the natural configuration of the repository), are interposed between the waste and the human environment ensuring the long term containment.

There is broad consensus in the scientific, technological and regulatory fields that final disposal in DGR is the most effective, safest and feasible solution for the long term management of spent fuel and HLW, which ensures that no significant harm is caused by the waste to human life and the environment [1-25]. This is also acknowledged by Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste [1-8]. The safety case of the deep geological repository must satisfy strict regulatory criteria and requirements. It is based on an extensive, wide-ranging, multi-disciplinary body of research, typically performed over several decades, which addresses specific and combined mechanisms and physical-chemical properties affecting the performance of the individual barriers and of the integral system. The safety demonstration addresses both the “normal” long-term behaviour of the repository and the behaviour in case of “perturbations”, such as glaciation, seismic events, and human intrusion, and it is assessed independently by the competent regulatory authority

After having obtained the required licences and approvals, a deep geologic repository for spent fuel and HLW is under construction in Finland [1-21]. Spent fuel and HLW repositories are at an advanced licensing stage (construction licence applied for) in Sweden [1-22] and France [1-23]. Repositories for short-lived intermediate and low level radioactive waste are in operation in several Member States [1-24].

The different types of waste resulting from the various stages of the nuclear energy lifecycle, including operation of nuclear power plants, and from applications of other nuclear technologies are characterized by different levels of radioactivity and radiotoxicity, which decrease with time (see Annex 6).

The radioactivity dose to the population due to radiation of natural and medical origin varies considerably from person to person, depending on the place of residence, on the lifestyle and on the medical treatments received. Nuclear and public health laws and regulations (see Chapter 1.2 and Annexes 5 and 6) set strict limits on the radioactivity dose to the population not due to natural radioactivity and medical treatments. Such limits include the radioactivity dose contributions resulting from all nuclear technology applications. These limits also specify the maximum allowed radionuclides concentration and radioactivity levels in the environment and affecting human life (e.g. in food, water, buildings, etc.). The typical maximum permissible dose to the public of 1 mSv[132] per year represents approximately 42% of the average natural exposure in the world (United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and 31% of the average natural dose in Europe [1-1]. The actual average dose received by the public from non-natural and non-medical nuclear applications is in fact significantly lower than the maximum permissible level; in particular, the average exposure to a member of the public resulting from the nuclear energy lifecycle is estimated to be 0.0002 mSv/y (see Chapters 1.2 and 3.4 of part A of the report, and Annexes 5 and 6).

The above-mentioned regulations set the maximum allowed levels of radioactivity and dose, below which no significant harm is caused to the human population and to the environment (biosphere) also for the radioactive waste management activities. For instance, the nuclear law and regulations in Finland [1-3], [1-27] states that a final repository for spent nuclear fuel under normal operations may not cause a dose to the most exposed member of the public higher than 0.1 mSv/y. In Sweden, the maximum allowed dose contribution due to the final repository for a person that would live in its vicinity is 0.014 mSv/y [1-4], [1-5], and [1-6]. These limits are very low.

The basic and foremost objective of radioactive waste management is to ensure that the radioactive waste materials are contained and sequestered from the biosphere throughout all stages of waste management.

Objectives and principles of radioactive waste and spent fuel management

There is wide international consensus on the principles ruling the activities and facilities for radioactive waste and spent fuel management, as acknowledged in Council Directive 2011/70/Euratom [1-8] establishing a framework for the safe and responsible management of radioactive waste and spent fuel, and in the Joint Convention for the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management [1-9], to which 83 IAEA Member States are Contracting Parties, including all EU Member States.

Radioactive waste and spent fuel management stages, especially the final disposal of long-lived radioactive waste, extends over very long time, encompassing many generations. A basic ethical requirement informing this domain is the principle that the activities of today shall not cause negative impacts and shall not impose undue burdens on future generations (namely, no burden heavier than those that we allow to our generation should be imposed). The protection of the future generations and their environment against the harmful effects of ionizing radiation originating from radioactive waste and spent fuel disposal facilities after their closure must be guaranteed by the means put in place today. This ethical principle explains for instance why permanent spent fuel and high level waste storage under active human control is not an acceptable solution for long-term management, as maintaining the safety of the storage facility would constitute an undue burden imposed to future generations.

The fundamental safety objective to protect the people and the environment from the harmful effects of ionizing radiation is achieved by means of suitable measures to

— control the radiation exposure of people and the release of radioactive material into the environment;
— restrict the likelihood of events that might lead to a loss of control over any source of radiation;
— mitigate the consequences of such events, should they occur.

In the specific case of radioactive waste and spent fuel disposal, the waste must be contained and isolated from the accessible biosphere for as long as the waste remains hazardous: the required timeframe can extend, depending on the characteristics of the waste, from a few decades to hundreds of thousand years.

These measures include a combination of technical solutions and an appropriate framework that establishes the rights and responsibilities of the different stakeholders (operators, waste management agencies, regulatory authorities, and the State), the requirements applicable to carrying out the waste and spent fuel management activities (licence, skills and competences, human and financial resources); the requirements applicable to building and operating facilities (safety assessment, licence, etc), as well as the mechanisms for independent control, verification and enforcement.

An **integral strategy** for the safe and responsible management of radioactive waste and spent fuel includes, inter-alia,

[132] The sievert or Sv is the unit used to measure the biological doses which are the most significant for a living being: the actual doses are generally expressed in thousandth of a sievert (millisievert or mSv). The annual effective dose due to natural sources to which people in Europe are exposed is of 3.2 mSv per person on average; additional dose is received from medical treatments; a very small fraction is due to other human activities and nuclear energy (see Section 3.4 of part A, Annex 5 and Annex 6).

- a **national policy** enshrining the objectives and principles of radioactive waste and spent fuel management;
- a **national framework** (legal, regulatory and organisational) establishing the responsibilities, obligations and requirements for all the entities involved, such as licence holders and regulatory authorities and the relations among them; and
- the adoption and implementation of **concepts, plans and technical solutions** to manage the radioactive waste or spent fuel. The measures must cover all stages of management, from generation to final disposal; moreover, they must adequately address other potential non-radioactive hazards of the waste, such as chemical and biological.

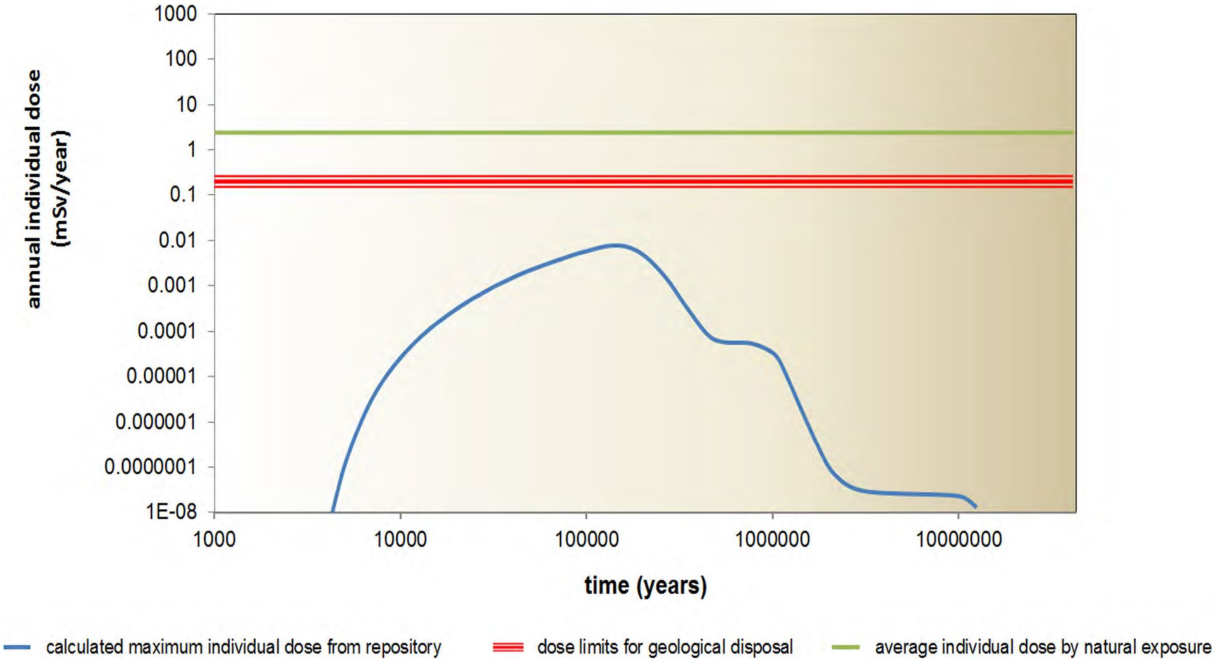
The generation of radioactive waste shall be kept to the minimum that is reasonably practical. This principle does not refer necessarily to the minimisation of the volume of radioactive waste that has already been generated, but aims at preventing the generation of additional waste to the extent possible, through adequate selection and design of the processes, and through implementation of decontamination, clearance, reuse and recycling. The aim is reducing as much as feasible the amount of material that must be eventually managed as radioactive waste.

Interdependencies between the different steps of waste management shall be taken into account. Management of radioactive waste must be considered as an integral activity that involves successive stages in which the waste is subject to different treatments. The subsequent treatment stages may be carried out in different facilities and under the responsibility of different operators. Taking interdependencies into account ensures that the waste conditions at the end of a given management step are fully compliant with the requirements of the following management steps. In this way, the management of the waste in the different stages is done consistently, smoothly and efficiently, avoiding extra interventions or steps back in the process. A typical example is a waste package treated and conditioned in a radioactive waste management facility that needs to be shipped to the disposal facility. The properties of the waste package (e.g. shape, size, weight, radioactivity content, doses, etc) must be compatible with the transport means, with the handling equipment of the receiving facility and with the waste acceptance criteria of the receiving facility.

Measures shall follow a graded approach. The severity and stringency of the waste management measures must be commensurate with the risk of the waste to be managed. Small amounts of low level radioactive waste require less stringent measures than heat generating high level waste. This principle is well reflected in the disposal technologies used for each category of radioactive waste: low and very-low level waste is typically disposed of in surface or near-surface disposal facilities, up to a few tens of meters below ground level for up to a few hundred years, whereas high level heat generating radioactive waste and spent fuel will be disposed of in engineered facilities embedded in deep geological formations hundreds of meters below ground level for several hundred thousand years.

Spent fuel and radioactive waste shall be contained adopting passive safety features, especially for the long term final disposal stage. Although the safety of disposal facilities during operation might include and/or be based on active measures, long-term safety of radioactive waste e.g. after closure of the disposal facility must not depend on any institutional control and must be based on inherent passive features. Passive features include engineered and natural barriers that do not require continuous supplies to active systems (e.g. electricity), periodic maintenance, replacement of parts, or permanent surveillance. In the case of a deep geological repository for final disposal of spent fuel and high level waste, the structures of the facility and the natural media must perform their containment functions without external interventions for as long as necessary. The individual and combined actions of the barriers strongly delay and minimize the corrosion of the waste, the release of radionuclides and their transport through the retarding and impermeable media and through the geological formations interposed between the waste and the biosphere. Figure 1-1 shows an example of the expected dose caused by the geological repository to the most exposed individual living in its vicinity as a function of time. The actual dose will be two orders of magnitude lower than the maximum level allowed by the regulation, which, in turn, is one order of magnitude lower than the dose from natural sources.

Figure 1-1. Example of modelling the performance of the geologic repository for high level waste (Belgian case): expected radiation dose to most exposed individual, compared to maximum allowed dose and to natural radiation dose



Source: [1-10]

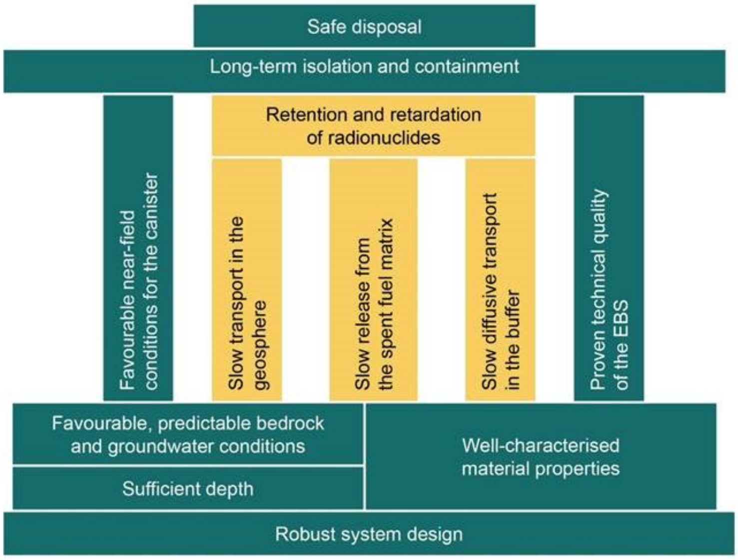
The cost of management of radioactive waste and spent fuel must be borne by those who produced the waste, or "polluter pays" principle. The prime responsibility for the management of radioactive waste lies with the producer of such waste. Notwithstanding, appropriate mechanisms have to be put in place to ensure that the ultimate responsibility for the management of radioactive waste lies with the State. This is particularly relevant in those cases in which the prime responsibility cannot be established, or the waste producer cannot fulfil its responsibilities.

A **stepwise evidence-based, documented decision making process is applied to all stages of management.** This principle aims at ensuring that all the important safety related issues have been taken into account in each and every stage of management, counting not only the technological developments and operational experience, but also considering social aspects and the involvement of the public. As the management of radioactive waste and spent fuel is a long-term process, a stepwise evidence-based, documented decision-making process allows for the selection of the best option at each stage, thus developing the best management route taking into account the state-of-the-art, and incorporating scientific and technical advances as well as societal considerations. A sound radioactive waste and spent fuel management programme will identify appropriate milestones to evaluate and make the relevant decisions for the future steps.

As mentioned above, there is worldwide scientific consensus that disposal of spent fuel and HLW in stable geological formations including multiple engineered and natural barriers containing the radioactive waste is the most effective solution to achieve the required long term isolation of radiotoxic substances [1-25]. The consensus among the experts extends to the conclusion that disposal in a deep geologic repository is technically feasible and that sufficient confidence in the overall safety of geological disposal of spent fuel and HLW has been reached to begin implementation [1-26] (see Chapter 5.2 of part B of this report). Figure 1-2 depicts in graphic form the main components, which inform the Finnish safety concept for the final disposal of spent nuclear fuel in a deep geologic repository. The procedure to authorize and implement the final disposal of spent fuel/HLW is complex and includes several steps and iterations (see Chapter 5.2 of part B of this report). In addition to extensive scientific and technological investigations, subjected to several levels of reviewing and questioning, the most advanced approaches include public consultation and involvement of the local communities in the approval process.

Figure 1-2. Schematic illustration of Posiva's safety concept for final disposal of spent fuel in Finland.

The green boxes describe the primary safety features and the yellow boxes the secondary safety features. They ensure that over a very long period of time (up to 1 million years) the disposal system will be affected only by a slow radionuclide diffusion process. EBS = engineered barrier system.



Source: Posiva (Finland).

DNSH and very long term management of radioactive waste

The application of the DNSH criteria to the various stages of radioactive waste management is affected by the specific nature of the waste. The impact associated with the construction and the operation of the facilities associated with waste handling, transportation, storage, and disposal is essentially of conventional, non-radiologic nature, and is described in Chapter 3.3.8 of Part A of the present report. The long term potential impacts of radioactive waste relevant to the DNSH criteria, in particular in the post-closure phase of the final disposal facility, are of radiological nature. Due to the high radiological hazard potential of radioactive waste forms, especially in the case of spent fuel and HLW, and as required by the relevant regulations, all steps of radioactive waste management fulfil the requirements and are designed to ensure that the waste remains fully contained and isolated from the environment at all times. Depending on the type of waste, this function is provided first by having a stable and corrosion-resistant waste form in which the radioactive waste is immobilized in a very stable low solubility matrix, e.g. the waste glass incorporating actinides and fission products from reprocessing of spent fuel. In the case of spent fuel, this function is provided by the integrity and tightness of the fuel rod cladding and/or other encapsulating containment. The waste forms are packaged in leak tight containers and/or within barriers that provide shielding for radiation and avoid any release and dispersion of radionuclides potentially affecting the population and the ecosystems.

The final disposal of spent fuel and HLW in a deep geological repository foresees its emplacement in a multi-barrier (engineered and natural) system in a stable geologic formation several hundred metres below ground level. The multi-barrier configuration of the repository prevents radioactive species from reaching the biosphere over the time span required to fulfil the strict dose limits imposed by the relevant regulations. The individual properties and the combined behaviour of the barrier materials and of the repository environment contribute to delay, block and minimize the release of radionuclides from the waste package, to delay the transport across the engineered barriers, and eventually to reduce and further delay the migration through the geological media (natural barriers). Therefore, all stages of radioactive waste management, including final disposal, do not cause radiological pollution and do not degrade healthy ecosystems, including water and marine environments. The avoidance of significant harm to humans and to the environment is ultimately ensured by the compliance with the regulatory limits set for the radioactivity dose contribution to the non-professionally exposed population, which is a pre-condition for the authorization and licensing of any radioactive waste management facility. This compliance must be ensured and demonstrated for all the steps subjected to active monitoring by the operators and also for the very long term duration associated with the final disposal of long lived and high level waste and spent fuel (post-closure phase).

Radioactive waste management strategies, especially innovative concepts currently under development aiming at "closing the fuel cycle", include aspects associated with the recycling and recuperation of materials present in irradiated nuclear fuel and valuable either as fuel or for other industrial applications

The following sections describe relevant aspects of radioactive waste management, with particular attention for the long term management of high level waste and spent nuclear fuel, along the lines envisaged by the Terms of Reference of the present Report.

1.2 Legal framework for long-term management of radioactive waste and spent fuel

The radiological hazard of radioactive wastes decreases with time due to radioactive decay. However, in case of long-lived radionuclides, it could take thousands of years to reach safe levels and during this time, containment and isolation from humans and the living environment must be ensured. The management of long-lived radioactive wastes, such as high level waste and spent fuel, is the primary focus of this chapter. Under long term we consider the time needed for the construction, operation and closure of a disposal facility, including post-closure period. This chapter highlights the legal instruments and provisions specifically addressing long term management of spent fuel and radioactive waste. The main legal instruments addressing the management of spent fuel and radioactive waste in the long term are the Joint Convention [1-9] and Council Directive 2011/70/Euratom [1-8].

Main provisions and principles addressing long term management

The safe and responsible long term management of spent fuel and radioactive waste is achieved not only by means of appropriate technical solutions. Administrative, legal and regulatory framework plays an essential role in this process. It is crucial to clearly establish the duties, responsibilities and obligations for each of the stakeholders (operators, radioactive waste management organisations, authorities), the requirements for the activity (e.g. licence, safety assessments, sufficient skills and resources, availability of sufficient funds, etc.) as well as the different mechanisms for control, verification and enforcement (e.g. regulatory approval of documents, inspections, etc.).

As the long term management concerns many generations to come, besides the requirements to set up an integrated national framework the main provisions in the international legislation focus on:

— avoiding imposing undue burdens on future generations;

— ensuring safety of disposal facilities in the long term;

— ensuring responsibilities in the long term.

Ensuring safe and responsible management of spent fuel and radioactive waste is the national responsibility of each state and it is the fundamental principle on which nuclear safety legislation is based at the international level. The ultimate responsibility of Member States for the safety of spent fuel and radioactive waste management is reaffirmed by the Joint Convention (preamble vi) and embedded in Council Directive 2011/70/Euratom (Art. 4(1)).

Long term management in the Joint Convention

At the international level, the main legally binding instrument directly addressing spent fuel and radioactive waste management is the IAEA Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Its provisions cover all management stages, including long term management. The Joint Convention not only requires protecting individuals, society and the environment from radiological hazards, but it also requires avoiding actions that impose reasonably predictable impacts on future generations greater than those permitted for the current generation (Art. 4(vi) and 11(vi)) as well as **avoiding imposing undue burdens on future generations** (Art. 4(vii) and 11(vii)). Moreover, articles 13 to 15 set out requirements concerning the safety of waste management facilities during the siting, design, construction, and operation phases. Some of those requirements relate to the long term safety, e.g. the requirement to evaluate all relevant site-related factors on the safety of a disposal facility after closure, and the requirement to evaluate the safety impact of a facility on individuals, society and the environment taking into account possible evolution of the disposal facility's site conditions after closure.

Article 17 of the Joint Convention lists **institutional measures that have to be ensured after closure of the disposal facility**. They include measures such as preservation of records (location, design and inventory), active or passive institutional controls (monitoring or access restrictions, if required), etc.

Additionally, article 22(iii) requires **financial provisions** that enable the appropriate institutional controls and monitoring arrangements **to be continued for the period deemed necessary following the closure of a disposal facility**. More information on the Joint Convention is provided in Chapter 2.2.2 of Annex 1.

In the past, some states used to dump radioactive wastes into the marine environment (i.e. sea disposal) as a long term solution. This practice continued from 1946 to 1993 [1-11], when the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention) and its amendment (London Protocol) effectively introduced the prohibition of the dumping of any radioactive waste into the marine environment.

Long term management in international recommendations

As provided in Chapter 2.4 of Annex 1, complementary to the treaties that are legally binding on its State Parties, there are international recommendations that are not legally binding but that are internationally accepted. For example, similar provisions on record keeping can be found in the **IAEA Code of Conduct on the Safety and Security of Radioactive Sources** [1-12] (e.g. article 22(c) requires keeping records of the transfer and disposal of the radioactive sources on termination of the authorizations).

Documents published as part of the IAEA Safety Standards describe detailed safety measures that can help State Parties implement the conventions they ratified and some of those documents include provisions concerning long term management of radioactive waste. At the highest level the **IAEA Safety Fundamentals** [1-7] provide that responsibilities for safety by licensees and regulators must be fulfilled not only in relation to present operations but also to future operations (para 3.7). Moreover, **protection of future generations** is among the main safety principles and it is stated that subsequent generations have to be adequately protected without any need for them to take significant protective actions (para 3.27). The IAEA Safety Fundamentals also stresses the importance of avoiding imposing an undue burden on future generations. The waste producers should not only keep generation of the radioactive waste to the minimum but also seek and apply safe, practicable and environmentally acceptable solutions for its long term management (para 3.29).

The **IAEA's Safety Requirements** document GSR-1 (Rev. 1) on the Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety [1-13] complements the Safety Standards on the adequate protection of future generations. It requires the government to make provision for the **appropriate research and development** programmes in relation to the disposal of radioactive waste, in particular programmes for verifying safety in the long term (para 2.32).

The **IAEA specific safety requirements** SSR-5 [1-14] on disposal of radioactive waste establish requirements applicable to all types of radioactive waste disposal facilities. It is linked to the fundamental safety principles for each disposal option and establishes a set of strategic requirements that must be in place before facilities are developed. Consideration is also given to the safety of existing facilities developed prior to the establishment of present day standards.

In addition to the SSR-5, IAEA has published a general safety guide on the management system for the disposal of radioactive waste GS-G-3.4 [1-15].

IAEA has issued several specific safety guides concerning different types of disposal:

- SSG-23 Safety case and safety assessment for disposal of radioactive waste [1-16];
- SSG-14 Geological disposal facilities for radioactive waste [1-17];
- SSG-1 Borehole disposal facilities for radioactive waste [1-18];
- SSG-29 Near surface disposal facilities for radioactive waste [1-19];
- SSG-31 Monitoring and surveillance of radioactive waste disposal facilities [1-20]

More information on international recommendations is provided in Chapter 2.4 of Annex 1.

Long term management in Council Directive 2011/70/Euratom

The management of radioactive waste and spent fuel is regulated at the European Union level through various legal instruments adopted under the Euratom Treaty. First of all, the Euratom Treaty requires each Member State to inform the Commission on any plan for the disposal of radioactive waste in whatever form and to provide data to make it possible to determine whether the implementation of such plan is likely to result in the radioactive contamination of the water, soil or airspace of another Member State (article 37). The Commission shall deliver its opinion within six months, after consulting the group of experts referred to in article 31 of the Euratom Treaty.

Council Directive 2011/70/Euratom, establishing a Community framework for the responsible and safe management of radioactive waste and spent fuel, is the central legislation in this field. The Directive aims at ensuring the responsible and safe management of radioactive waste and spent fuel to avoid undue burdens on future generations. It is based on the International Atomic Energy Agency (IAEA) Safety Standards and reaffirms the principles of prime responsibility of licence holders, under the supervision of the national competent regulatory authority, and the ultimate responsibility of Member States for the management of the radioactive waste and spent fuel generated in them. As in the case of the Joint Convention – the Directive affirms ethical obligation of each Member State to **avoid any undue burden on future generations** (recital 24, article 1(1)). Moreover, it stresses that the storage of radioactive waste, including long term storage, is only an interim solution, but not an alternative to disposal (recital 21).

Member States are required by the Directive to establish and maintain an appropriate national policy and a national programme for its implementation, as well as a legislative, regulatory and organisational framework, which amongst others provides for the coordination between national bodies. Article 4(3) defines the principles for the national policy. One of them specifically addresses long term management by stating **that spent fuel and radioactive waste shall be safely managed, including in the long term with passive safety features**. As, due to the long timeframes, it is impossible to rely on institutional controls and active safety systems, passive safety features are crucial in ensuring long term safety. Article 5(1)(e) indicates that the national framework, among other elements, shall provide for **appropriate measures for the post-closure periods** of disposal facilities. In their national programmes Member States have to indicate concepts or plans for the post-closure period of a disposal facility's lifetime, including the period during which appropriate controls are retained and the means to be employed to **preserve knowledge of that facility in the longer term** (Article 12(1)(e)). As part of the licensing of the facility, the safety demonstration shall cover the operation or closure of a disposal facility as well as the post- closure phase of a disposal facility (Article 7(3)).

In principle, it is required to dispose of radioactive waste in the Member State in which it was generated (Article 4(4)). However, there's an exception if at the time of shipment an agreement, taking into account the criteria established by the Commission in accordance with Article 16(2) of Directive 2006/117/Euratom, has entered into force between the Member State concerned and another Member State or a third country to use a disposal facility in one of them. Commission Recommendation 2008/956/Euratom of 4 December 2008 lists the criteria for the export of radioactive waste and spent fuel to third countries (as stated in Article 16(2) of Directive 2006/117/Euratom).

More information on **Council Directive 2011/70/Euratom** is provided in Chapter 3.2.3 of Annex 1.

1.3 Structure of Part B

Part B describes relevant aspects of the management of radioactive waste, with particular attention on the long term management of spent fuel and high level waste, along the lines envisaged by the Terms of Reference of the present Report.

Chapter 1 presents the objectives, main principles and a summary of the legal framework of the management of radioactive waste and spent fuel.

Chapter 2 highlights the typologies and the classification of radioactive waste generated during the various steps of the nuclear fuel cycle described in part A, and summarizes the current global and EU radioactive waste and spent fuel inventories.

Chapter 3 presents the strategies and technologies available for the management of radioactive waste, focusing especially on the processes rather than in the details of the technologies.

Chapter 4 presents the different aspects of interim storage of radioactive waste and spent fuel as a necessary step prior to disposal.

Chapter 5 is dedicated to the final disposal of radioactive waste and spent fuel. It addresses the surface and near-surface disposal of low level short lived radioactive waste and provides a schematic description of the main geologic disposal concepts for HLW and spent fuel in Europe. The rationale and conceptual approach, the tools and criteria informing the validation and the implementation of deep geological repositories are described, together with specific safety criteria, and features associated with the safety case and long-term performance assessment.

Chapter 6 describes the strong contribution of R&D to the development and the implementation of the long-term solutions for the management of radioactive waste, including a historical perspective, the main scope of current research efforts, how research is organised in the EU, main actors, tools, trends, and future perspectives.

Relevant for part B, Annex 1 describes the legal framework applicable to the nuclear fuel cycle, and thus also for the radioactive waste management, both from the nuclear and environmental perspectives.

Annex 6 provides some background notions about long-term radioactivity and radiotoxicity of radioactive waste.

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2 Inventory of radioactive waste and spent fuel in the European Union

2.1 Generation of radioactive waste and spent fuel

Radioactive waste is generated in almost every country, even though the quantities in non-nuclear power countries are very small compared to the ones operating nuclear power plants.

The largest fraction of the radioactive waste comes from operation and decommissioning of nuclear power plants and associated nuclear fuel cycle activities (see Chapter 3.3. of part A of this report). Table 2.1-1 shows the typical annual waste generation per unit energy broken down to the different stages of the nuclear fuel cycle. The table does not include waste from uranium mining and milling activities. In terms of activity, most of the radioactive waste is generated during operation of nuclear power plants, in particular by the fuel irradiation. However, in terms of volume, most of the waste comes from decommissioning of nuclear power plants and other nuclear facilities at the end of their operational lifetime, mainly as low level waste.

Table 2.1-1. Typical annual radioactive waste generation rates from the nuclear fuel cycle (excluding mining and milling)

Stage / Activity	Waste physical state (Solid, Liquid, Gas)	Amount in m^3/GW.year
FRONT END		
Conversion	L/S	50
Enrichment	G/L/S	25
Fabrication	L/S	
UO_2		75
MOX		5.64
Nuclear Power Plant Operation		
Evaporator concentrates	L	50
Filter sludges	L	10
Ion exchange resins	S	2
Decontamination conc.	L/S	10
Absorber rods, neutron sources, etc	S	0.1
Others	S	260
BACK-END		
Reprocessing		
Hulls/hardware	S	15
Feed sludge	S	0.02
Tritium bearing effluents	L	70
HLW	L	28
ILW	L	25
LLW	L/S	80
Direct spent fuel disposal		
Spent fuel assemblies (tHM)		30
Decommissioning of nuclear fuel cycle facilities		
Conversion	S	0.5 – 1
Enrichment	S	5
Fabrication	S	1 - 2
Power plant	S	375
Reprocessing	S	5

Source: IAEA (2017) [2-1]

Radioactive waste is generated not only as a result of nuclear fuel cycle activities. There is a broad range of other activities that use radioactive materials and produce radioactive waste, such as research and education, medicine, industrial and agricultural activities. Thus, even countries without nuclear power plants generate radioactive waste and have to ensure that it is managed responsibly and safely.

Operation and decommissioning of research facilities, such as laboratories and research reactors, are significant radioactive waste generators among non-power generating activities. Depending on the facility type, a wide range of radioactive waste (ranging from VLLW to ILW) may be generated, such as items contaminated during handling of radioactive materials, containers for production of radioisotopes and irradiation of samples for neutron activation analysis, components of installations exposed to neutron beams, ion exchange resins, irradiated components of the reactor monitoring equipment, control rods, etc. [2-2]. Spent fuel from research reactors has completely different characteristics compared to the fuel from power reactors and as such it requires dedicated management procedures. Originally, most of the research reactors were operating using highly enriched uranium (HEU) fuel. Due to nuclear proliferation concerns US and Russian Federation have offered HEU spent fuel take-back programmes to minimize the HEU inventories [2-3]. Thus, instead of establishing their own programme for spent research reactor fuel management most of the countries operating research reactors have opted for an agreement with the fuel suppliers to send back the spent fuel.

In medicine and industrial activities, radioactive material usually comes in the form of sealed (encapsulated) and unsealed (non-encapsulated) radioactive sources. Both types of radioactive sources are used in diagnostic as well as in treatment procedures: unsealed sources are used in applications such as measurement levels of drugs or hormones in biomedical samples, use of radionuclides as tracers in monitoring body functions, lung ventilation imaging, treatment of thyrotoxicosis, ablation of the thyroid tissue during cancer treatment, etc; sealed sources are used for calibration and reference standards, bone densitometry, anatomical marking, brachytherapy, teletherapy, blood irradiation and other purposes [2-4].

In industrial and agricultural activities radioactive waste may be generated from the manufacture or use of radioactive materials during calibration and quality control of equipment, non-destructive testing, construction, food irradiation (preservation), insect sterilisation, geological exploration, tracing fertiliser uptake by plants and other activities [2-5]. Installation of certain products, such as lightning rods or ionisation chamber smoke detectors containing sealed radioactive sources, has been prohibited in most countries and previously installed devices are systematically being replaced. The products that are still in use have to be collected and managed safely as radioactive waste. Detailed description of consumer products containing radioactive materials can be found in [2-6].

Radioactive waste from non-power generating activities is usually managed in the same way as in the case of waste with similar properties resulting from nuclear power generation. Countries managing radioactive waste from their nuclear fuel cycle facilities usually benefit from having well established radioactive waste management processes and facilities for management of radioactive waste originating from outside of the nuclear fuel cycle activities. Conversely, countries without nuclear fuel cycle facilities even if having relatively small radioactive waste inventory must establish radioactive waste management processes and construct all the necessary waste management facilities to ensure safe and responsible radioactive waste management from generation to disposal.

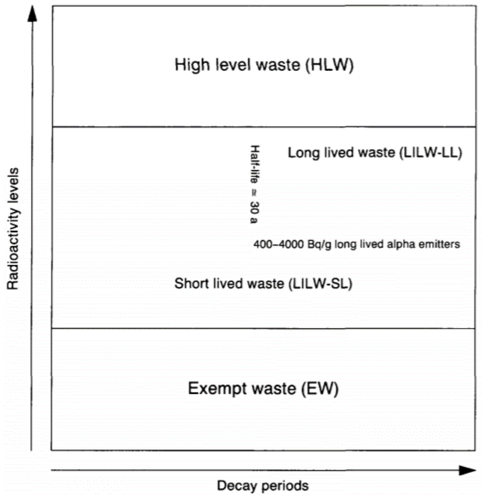
Most of the radioactive waste resulting from non-power generating activities consists of short-lived radionuclides. Management of this type of waste usually involves storage for a time span allowing radioactive decay and subsequent disposal as conventional waste. However, radioactive waste containing relevant amount of long-lived radionuclides requires a greater degree of containment and long term isolation from the environment, e.g. deep geological disposal. For small countries having only a small inventory of long-lived radioactive waste development of deep geological repository could be financially challenging. Thus, most of them are currently exploring alternative disposal options, such as the use of shared disposal solutions or implementation of borehole disposal concepts.

2.2 Classification of radioactive waste

Radioactive waste generated in various stages of the nuclear fuel cycle could significantly differ in terms of physical, chemical and radiological properties. This diversity results in different potential hazards to human health and the environment requiring different levels of protection measures. Depending on the specific properties of the waste, there are multiple options and technological solutions available for management of the radioactive waste from generation to disposal. To facilitate waste management processes and their regulations the radioactive waste is classified based on properties such as radioactivity level, decay time, physical state, etc.

There were multiple waste classification schemes used around the world and this made information management and communication, both at national and international level, difficult. An attempt to standardize its classification at international level was made by IAEA in 1970 by proposing to use high level waste, intermediate level waste and low level waste classes. [2-7]. In 1981 this classification evolved by adding differentiation between short-lived and long-lived waste for intermediate level and low level waste classes [2-8]. In 1994 IAEA has revised and updated its recommendation on the radioactive waste classification [2-9]. This classification recommendation included the exempt waste class.

Figure 2.2-1. Radioactive waste classification system (IAEA 1994)



Source: IAEA (1994) [2-9]

In 1999 the Commission of the European Communities recommended to the Member States and their nuclear industry to adopt a common classification system of radioactive waste for national and international communication purposes as well as to facilitate information management in this field [2-10]. The proposed classification system was expected also to be used for providing information concerning solid radioactive waste to the public, the national and international institutions and the non-governmental organizations. It was not expected to replace technical criteria where required for specific safety considerations such as licensing of facilities or other operations. The proposed classification system was based on the IAEA (1994) classification scheme with some changes to take into account the views and practical experiences of European national experts - for instance the IAEA recommended limit of heat generation in LILW radioactive waste (2 kW/m^3), was not retained. The EC recommendations consisted of the following waste classes:

1. **Transition radioactive waste**: radioactive waste (mainly from medical origin) which will decay within the period of temporary storage and may then be suitable for management outside of the regulatory control system subject to compliance with clearance levels.
2. **Short-lived low and intermediate level waste** (LILW-SL): radioactive waste with nuclides whose half-life is less than or equal to those of ^{137}Cs and ^{90}Sr (around 30 years) with a limited alpha long-lived radionuclide concentration (restriction of long-lived alpha emitting radio-nuclides to 4 000 Bq/g in individual waste packages and to an overall average of 400 Bq/g in the total waste volume).
3. **Long-lived low and intermediate level waste** (LILW-LL): radioactive waste with long-lived radionuclides and alpha emitters whose concentration exceeds the limits for short-lived waste.
4. **High level waste**: radioactive waste with such a concentration of radionuclides that generation of thermal power shall be considered during its storage and disposal (the thermal power generation level is site-specific, this waste is mainly forthcoming from treatment/conditioning of spent nuclear fuel).

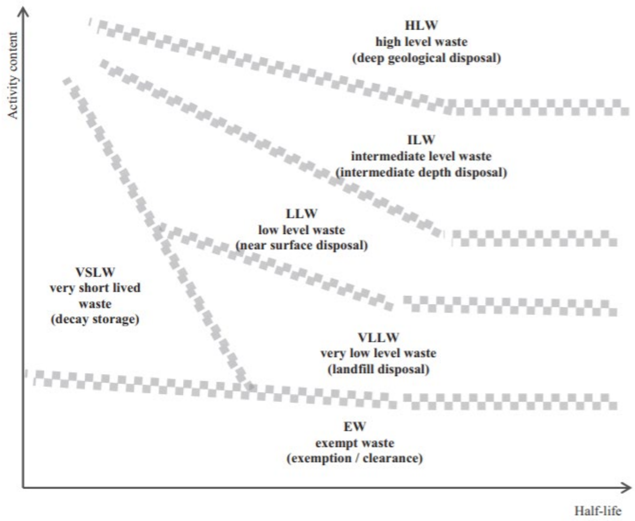
In 2009 IAEA published a General Safety Guide on classification of radioactive waste [2-11]. The proposed system is based on the minimal disposal requirements necessary to ensure that the radioactive waste will be sufficiently isolated from the environment in the long-term (i.e. different waste classes were linked to the disposal options). It defined six waste categories:

1. **Exempt waste** - concentrations of radionuclides small enough not to require provisions for radiation protection. Such material can be cleared from regulatory control and does not require any further consideration from a regulatory control perspective.
2. **Very short-lived waste** - containing only very short half-life radionuclides, thus such waste can be stored until the activity has fallen beneath the levels of clearance, allowing for the cleared waste to be managed as conventional waste.

3. **Very Low Level Waste** (VLLW) - does not need a high level of containment and isolation and, therefore, is suitable for disposal in near-surface, landfill-type facilities with limited regulatory control.
4. **Low Level Waste** (LLW) - is above clearance levels, but with limited amounts of long-lived radionuclides. It requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near-surface facilities.
5. **Intermediate Level Waste** (ILW) - because of its content, particularly of long-lived radionuclides, requires a greater degree of containment and isolation than that provided by near surface disposal. However, ILW needs no provision, or only limited provision, for heat dissipation during its storage and disposal. ILW may contain long-lived radionuclides, in particular, alpha emitting radionuclides that will not decay to a level of activity concentration acceptable for near surface disposal during the time for which institutional controls can be relied upon. Therefore, waste in this class requires disposal at greater depths, of the order of tens of metres to a few hundred metres.
6. **High Level Waste** (HLW) - waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or waste with large amounts of long-lived radionuclides that need to be considered in the design of a disposal facility for such waste. Disposal in deep, stable geological formations, usually several hundred metres or more, below the surface is the generally recognized option for disposal of HLW.

The first two waste categories do not require long-term management or disposal as radioactive waste due to their short-lifetime and/or levels allowing the exemption or clearance from regulatory control.

Figure 2.2-2. Radioactive waste classification system (IAEA, 2009)



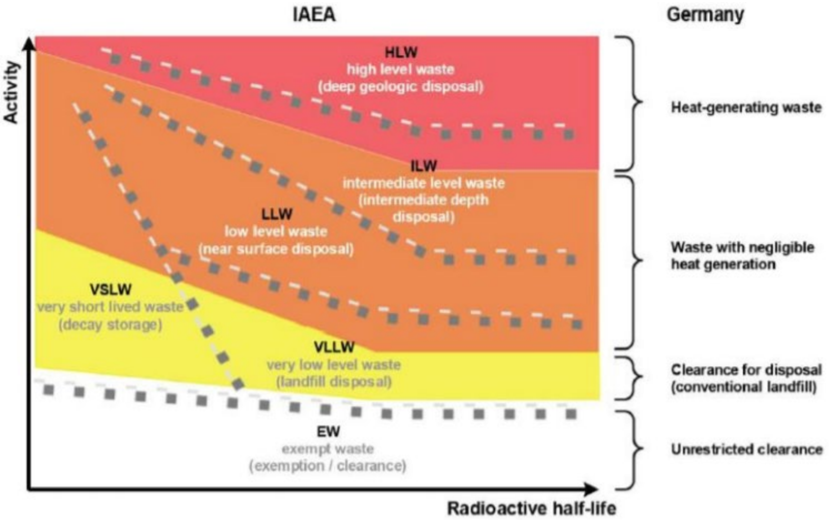
Source: IAEA (2009) [2-11]

The above mentioned waste classification schemes were primarily aimed at assisting countries in the development of their waste classification systems and/or using it to communicate their inventory data at national or international level. Internally, countries are free to define and use their own radioactive waste classification systems according to the national needs. As illustrated below with a few examples, due to similar terminologies used in different waste classification schemes it is important not only to provide the waste class when reporting radioactive waste inventory, but also to clearly identify the waste classification scheme it refers to.

In the EU multiple waste classification schemes are in use, but most of them are aligned either with the EU (1999) or IAEA (2009) recommendations. For illustration purposes we will provide a few examples of radioactive waste classification systems used by EU Member States.

Germany. The main driver behind the radioactive waste classification system used in Germany is a national policy to dispose of all radioactive waste in a deep geological disposal facility. As a result of this there are only two radioactive waste classes defined – heat-generating waste and waste with negligible heat generation. In comparison with the IAEA GSG-1 recommendation, heat-generating waste class corresponds to HLW and some ILW, while waste with negligible heat generation corresponds to LLW and to the major part of ILW (see Figure 2.2-3).

Figure 2.2-3. Comparison of waste classification system used in Germany with the IAEA GSG-1

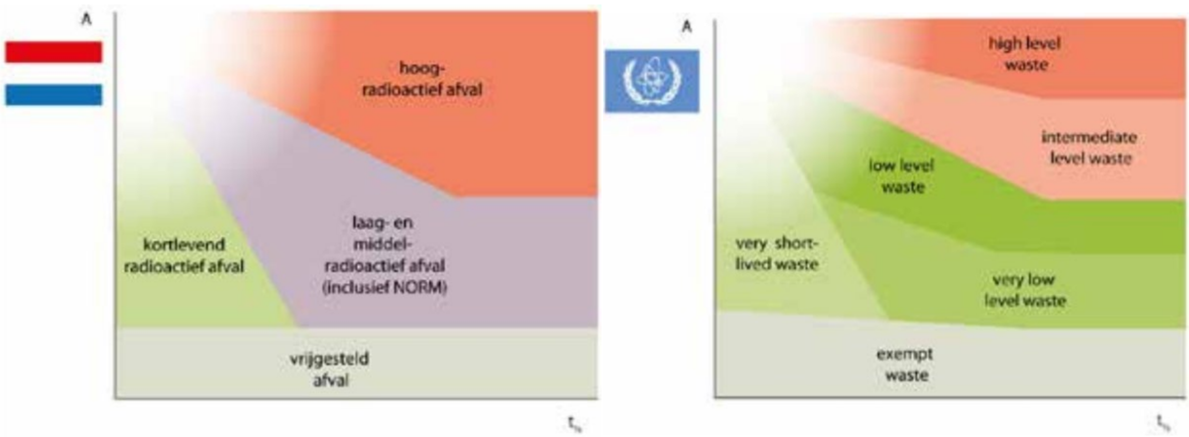


Source: IAEA, 2019. [2-12]

The Netherlands. In the Netherlands, radioactive waste is divided into four categories: high level radioactive waste, low level and intermediate level radioactive waste (including NORM waste), short-lived waste and exempt waste. These categories are based on activity and half-life [2-13].

The Dutch HLW ('hoog radioactief afval') mostly originates from reprocessing of spent fuel and this class covers HLW and ILW waste classes of the IAEA GSG-1. Within the HLW class distinction is made between heat-producing HLW and non-heat-producing HLW. The Dutch LILW ('laag- en middelradioactief afval') class covers VLLW and LLW waste classes of the IAEA GSG-1. A comparison of the Dutch radioactive waste classification system against the IAEA GSG-1 scheme is provided in Figure 2.2-4.

Figure 2.2-4. Waste classification system in the Netherlands



Source: The Netherlands, 2016. [2-13]

Spain. There are four waste classes defined in Spain, namely VLLW, LILW, Special Waste or SW and HLW. The classification scheme of Spain is comparable to the classification of the IAEA GSG-1 (2009) as shown in Table 2.2-1.

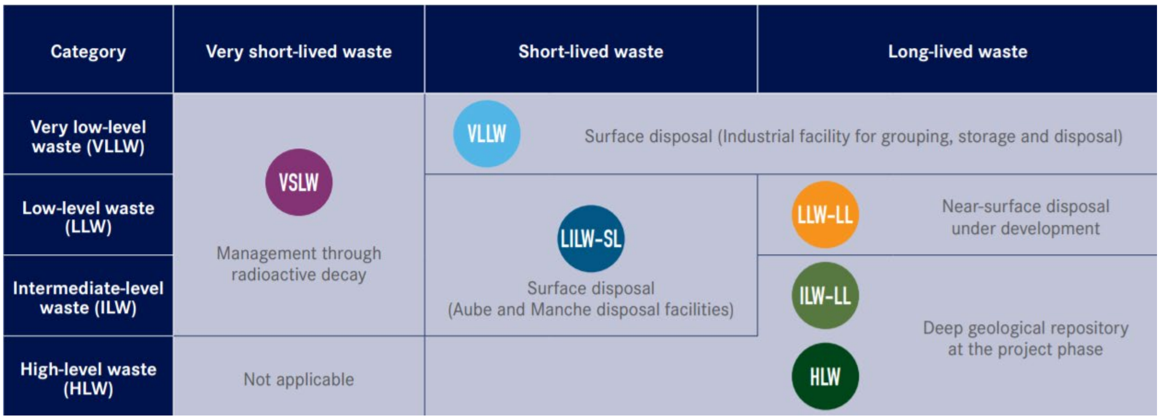
Table 2.2-1. Correspondence between Spanish classification system and IAEA GSG-1 (2009).

Classification system in Spain		**IAEA GSG-1 (2009)**
VLLW	Residuos de muy baja actividad (RBBA)	VLLW
LILW	Residuos de baja y media actividad	LLW
SW	Residuos especiales	ILW
HLW	Residuos de alta actividad	HLW

Source: Spain, 2018. [2-14]

France. The radioactive waste classification system in France is based on the level of radioactivity and the half-life. There are six radioactive waste classes and the classification is very close to the one of EU (1999) (see Figure 2.2-5). Table 2.2.-2 shows the correspondence of the French classification system to the IAEA GSG-1

Figure 2.2-5. Waste classification system in France



Source: ANDRA (2018) [2-15].

Table 2.2-2. Correspondence between French classification system and IAEA GSG-1 (2009)

Classification system in France		**IAEA GSG-1 (2009)**
VLLW	Les déchets de très faible activité (TFA)	VLLW
LILW-SL	Les déchets de faible et moyenne activité à vie courte (FMA-VC)	LLW
LLW-LL	Les déchets de faible activité à vie longue (FA-VL)	ILW
ILW-LL	Les déchets de moyenne activité à vie longue (MA-VL)	ILW
HLW	Les déchets de haute activité (HA)	HLW

Source: IAEA (2018) [2-16].

These examples of national classification systems clearly illustrate how the use of radioactive waste classes could create confusions and misunderstandings if it is not specified to which classification system they belong.

A summary of the waste classification systems used by various EU Member States can be found in Annex II of the Commission Staff Working Document SWD(2019) 435 final (17/12/2019). [2-21]

2.3 Amounts of radioactive waste and spent fuel in nuclear fuel cycle

Inventory information is one of the cornerstone elements in the development of national radioactive waste and spent fuel management programmes as national decisions on the management (and, in particular, the disposal) methods are based primarily on the amounts of current and estimated future radioactive wastes.

Inventory data are collected and managed at a national level, usually by a radioactive waste management organization or regulatory authority. Most of the countries are publishing their national inventory data on a regular basis (usually the data are published every 3 years). At the same time, all EU Member States have inventory reporting obligations at international and/or regional levels.

Council Directive 2011/70/Euratom [2-22] establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste (the Radioactive Waste Directive) requires EU Member States to notify their current inventory data together with future estimates to the European Commission on a regular basis (by 23 August 2015 for the first time and then every 3 years).

According to Article 14(2)(b) of the Council Directive 2011/70/Euratom, the European Commission is required to submit to the European Parliament and the Council an inventory of radioactive waste and spent fuel present in the Community's territory and the future prospects, based on the Member States' reports on the implementation of the Directive. The latest Commission report was adopted on 17/12/2019 [2-17] and it was accompanied by the Staff Working Document presenting an overview of spent fuel and radioactive waste inventory in the EU as of the end of 2016 including the future prospects [2-21].

Since 1992 the European Commission has published a series of "Situation Reports"[133] which were developed in order to analyse the situation of spent fuel and radioactive waste management in the EU and inform the stakeholders about it [2-18]. Since the Directive entered into the force, situation reports have been discontinued and replaced by the Commission report and the linked staff working documents on the implementation of the Directive.

The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [2-19] is another legal instrument requiring its signatories to report radioactive waste and spent fuel inventories (all EU Member States are signatories to the Joint Convention). However, unlike in the case of the Council Directive 2011/70/Euratom, there's no further processing of national inventories and corresponding analysis of global inventories within the framework of the Joint Convention. As most of the Joint Convention national reports are made publicly available, they are often used by other projects and initiatives (e.g. the world nuclear waste report[134]) as input source to compile or analyse inventories at global or regional levels.

There are projects relying on a voluntary inventory data submission, such as a common project by EC/IAEA/NEA Status & Trends in radioactive waste and spent fuel management. The first project's report was published in 2018 (IAEA, NW-T-1.14) [2-20] and it presented inventory data as of end 2013. The second report is under preparation and it will present global inventory data as of end 2016.

Current radioactive waste inventory in the European Union

The latest European Union radioactive waste inventory data (inventory dated end 2016[135]) are provided in the second Commission report on the implementation of the Council Directive 2011/70/Euratom.

The estimated total inventory of radioactive waste in the EU territory at the end of 2016 was 3 466 000 m^3. It is important to note, that 71.6% of this volume has been already disposed of. The amount of radioactive waste in storage was 983 000 m^3. The distribution of radioactive waste amounts per class (IAEA GSG-1) is shown in Table 2.3-1.

The inventory data includes all radioactive waste present on the EU territory originating from various civil activities. Most of the radioactive waste originates from nuclear power plants and associated nuclear fuel cycle activities. Smaller volumes of radioactive waste are generated as a result of non-power applications of

[133] The last one of the series was "Commission staff working paper, Seventh situation report, radioactive waste and spent fuel management in the European Union; SEC(2011) 1007 final, 22.8.2011".

[134] https://worldnuclearwastereport.org/

[135] Inventory data from 28 EU member states

radioactive materials, such as the manufacturing of radioisotopes for medical and industrial applications, or research facilities such as laboratories and research reactors.

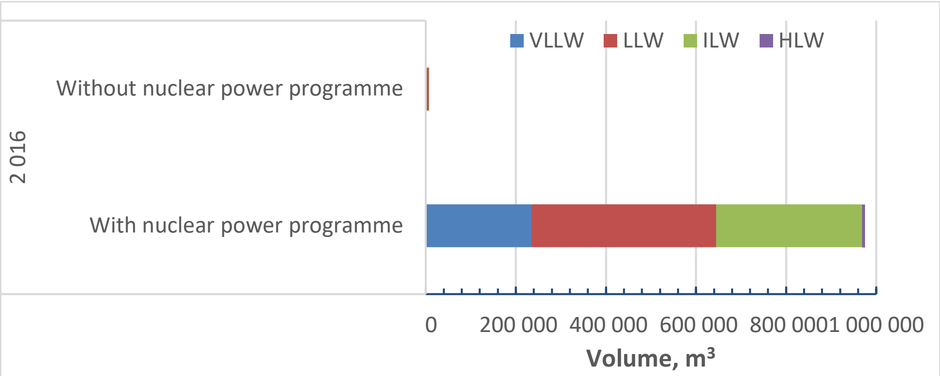
Table 2.3-1. Volumes and status of radioactive waste in the European Union, end of 2016.

Waste class (IAEA GSG-1)	Amounts (m^3)		
	Stored	Disposed of	Total
VLLW	234 000	369 000	603 000
LLW	417 000	2 102 000	2 519 000
ILW	326 000	12 000	338 000
HLW	6 000	0	6 000
TOTAL	983 000	2 483 000	3 466 000

Source: [2-21]

By comparing amounts of radioactive waste in EU Member States with and without nuclear power programmes, it can be seen that Member States without nuclear power programme have less than 1% of the overall radioactive waste amounts in storage.

Figure 2.3-1. Volumes of stored radioactive waste by class in Member States with and without nuclear power programme, end of 2016.



Source: [2-21]

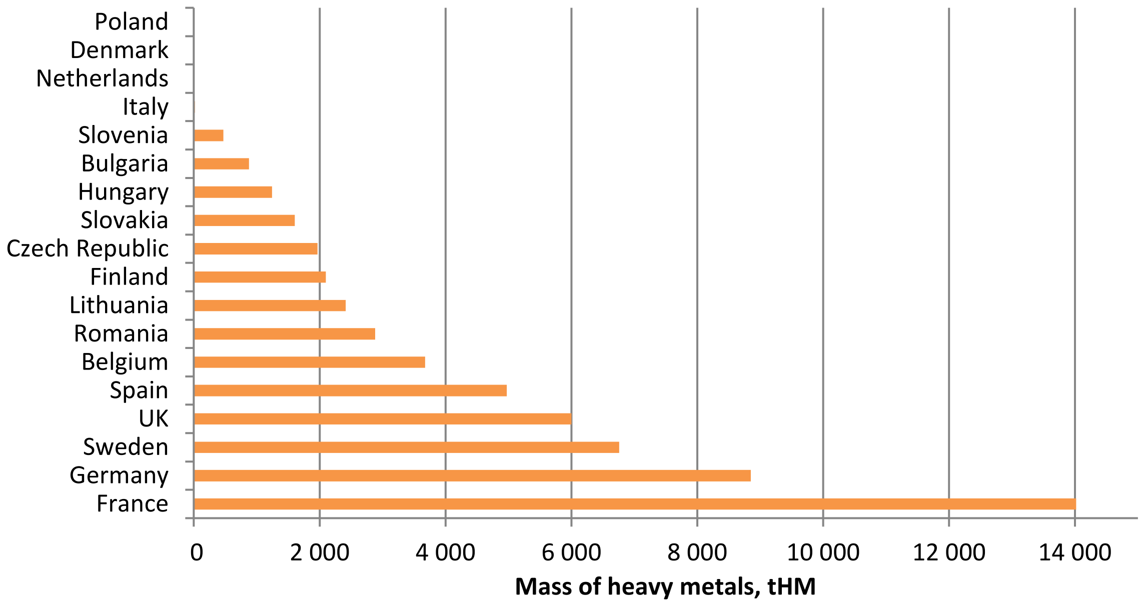
Current spent fuel inventory in the European Union

EU Member States use different strategies with regards to the management of spent fuel. Some Member States have chosen to reprocess spent fuel; some Member States have chosen the once-through fuel cycle option by which spent fuel will be directly disposed of in deep geological disposal. A few Member States applied both approaches – part of their spent fuel is reprocessed and the remaining spent fuel will be directly disposed. During reprocessing, uranium and plutonium are recovered and separated from fission products, which are radioactive waste (mainly HLW and ILW). Spent fuel is currently stored pending reprocessing or until disposal facilities become available. The first deep geological repository for spent fuel disposal will start its operation within the present decade in Finland. Corresponding repositories are in advanced licensing stages in Sweden and France as well.

At the end of 2016 approximately 58 000 tHM of spent fuel was stored in the EU and around 900 tHM of spent fuel (about 1.5 %) was sent for reprocessing outside the EU with the resulting radioactive waste from reprocessing expected to return as specified in the relevant agreements. These amounts include spent fuel coming from both power and non-power (e.g. research, isotope production) reactors.

The most recent reported amounts of spent fuel stored in individual Member States are shown in Figure 2.3-2. Some Member States have smaller inventories of spent fuel (or none) in storage than what was generated by the nuclear power plants because all or part of it has been reprocessed. On the other end, countries with neither past, nor current reprocessing, have comparably high spent fuel inventories.

Figure 2.3-2. Spent fuel in storage (end of 2016)



Source: [2-21]

Given that today there is no disposal route in operation for spent fuel (the first disposal facilities will become operational in 2024-2035) and that not all Member States have their spent fuel reprocessed, the amount of spent fuel in storage is increasing.

The majority of Member States operating nuclear power plants intends to directly dispose of their spent fuel in deep geological facilities without reprocessing, although two Member States are considering future reprocessing abroad. With the termination of reprocessing activities in 2018 at THORP and the ceasing of reprocessing of spent fuel in the UK by 2020, France is the only Member State with an ongoing industrial policy on domestically reprocessing spent fuel.

Future inventory in the European Union

Decommissioning of nuclear power plants will become an increasingly important activity for the European nuclear industry in the coming years due to the ageing of the reactor fleet. This will have an important impact on the amounts of radioactive waste generated, especially VLLW and LLW, and should thus be taken into account when planning disposal and storage facilities. As can be seen from Table 2.3-2, it is expected that the total VLLW volume will more than double by 2030 compared to the current amounts, and that a significant increase of LLW by approximately one third compared to the current amounts will occur as well. The main contributors to this increase are the Member States with the largest nuclear programmes.

It is expected that ILW will increase by approximately 35% by 2030. The biggest part of this increase will come from decommissioning activities. An increase of about 50% in vitrified HLW will result from reprocessing of spent fuel (mostly in France).

With regards to spent fuel, an increase from the present 58 000 tHM to 76 000 tHM in 2030 is estimated. It has to be noted that the majority of Member States have not reported inventories from planned newly built nuclear power plants. It is expected that by 2030 the spent fuel inventory will increase further by approximately 10%. As some Member States proceed with spent fuel reprocessing, the actual increase does not represent the actual amount of spent fuel discharged from the reactors. Part of the spent fuel is sent for reprocessing outside the EU and it is expected that around 1 100 m^3 of radioactive waste from spent fuel reprocessing will be returned by 2030.

The EU inventory data provided in Table 2.3-2 include radioactive waste and spent fuel inventory from the UK. The current and future EU inventory information was prepared before 31 January 2020 when UK was still an EU Member State

Table 2.3-2. Current and estimated future amounts of radioactive waste in the EU MSs for 2030

	VLLW (m^3)	LLW (m^3)	ILW (m^3)	HLW (m^3)	SF (tHM)
2016	603 000	2 519 000	338 000	6 000	58 000
2030	1 360 000	3 322 000	455 000	9 000	76 000

Source: [2-21]

To illustrate the significance of the UK inventory with respect to the total EU inventory, an updated EU27 inventory estimate was prepared (Table 2.3-3).

Table 2.3-3. Current and estimated future amounts of radioactive waste in the EU MSs (without UK)

	VLLW (m^3)	LLW (m^3)	ILW (m^3)	HLW (m^3)	SF (tHM)
2016	601 000	1 420 000	190 000	5 000	52 000
2030	*1 209 000*	*1 970 000*	*264 000*	*8 000*	*68 000*

Source: JRC, 2021.

Global radioactive waste and spent fuel inventory

Global radioactive waste and spent fuel data is collected and analysed in the frame of the common IAEA/EC/NEA project "Status and trends in radioactive waste and spent fuel management". The most recent data are collected and will be published in the second report which will present global inventory data as of the end of 2016. This sub-section is based on the information from this project.

In total there are approximately 38 million m^3 of solid radioactive waste worldwide. More than 80% of it is already disposed of and less than 20% is currently in storage. Most countries already have disposal options available for the VLLW and LLW and, similarly to the EU case, the most significant part of the already generated VLLW and LLW amounts have already been disposed of (85%). The amounts and status of the global radioactive waste inventory are provided in Table 2.3-4.

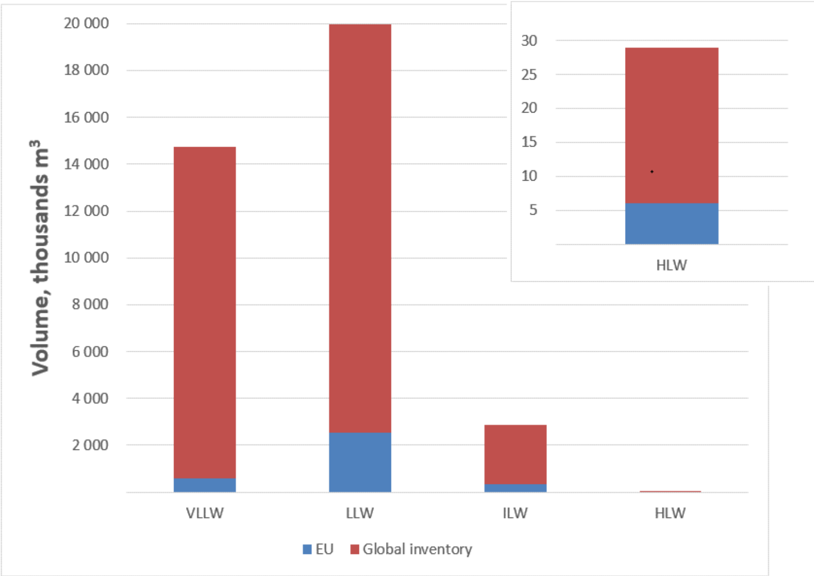
Table 2.3-4. Volume and status of global radioactive waste inventory, end of 2016

Waste class (IAEA, GSG-1)	Amounts (m^3)		
	Stored	**Disposed of**	**Total**
VLLW	2 918 000	11 842 000	14 760 000
LLW	1 471 000	18 499 000	19 970 000
ILW	2 740 000	133 000	2 873 000
HLW	29 000	0	29 000
Total	**7 158 000**	**30 474 000**	**37 632 000**

Source: IAEA/EC/NEA status and trends joint project

The EU inventory constitutes less than 10% of the global inventory. The amounts of EU inventory compared to the global inventory per waste class are shown in Figure 2.3-3.

Figure 2.3-3. Quantities of global and EU radioactive waste inventory



Source: IAEA/EC/NEA status and trends joint project

In addition to the solid radioactive wastes some countries (i.e. the United States of America and the Russian Federation) have accumulated large volumes of liquid wastes (around 62 million m^3) that will require specific management approaches. As most of this waste originates from defence activities it is not discussed in this report.

During the 1954-2016 period, about 390 000 tHM of spent fuel was discharged from nuclear power reactors worldwide. Approximately one third (127 000 tHM) of this amount is reprocessed and the remaining is stored pending either reprocessing or disposal. Some 166 000 tHM of spent fuel was discharged from EU reactors during the same period - approximately 108 000 tHM of spent fuel was reprocessed and 58 000 tHM is currently in storage and pending further management steps (reprocessing or disposal).

2.4 Main radionuclides affecting the properties of high level waste

The presence of fissile (or fissionable) species in the waste may present potential criticality risks (see also Chapter 4.2 of part B of this report). A criticality safety assessment may be relevant in the case of spent fuel or waste forms containing significant amounts of plutonium and/or other fissile nuclides (actinides). Appropriate spacing among waste units and/or the inclusion of neutron absorbing materials in the waste package or in the layout of the storage or repository facility can address such concerns [2-24].

The radioactivity and the residual power of HLW are governed by different families of radionuclides during different time intervals (Annex 6 and [2-25]). During the first century after discharge, the main contributors to the radioactivity level and the heat load of spent fuel and vitrified waste are short-lived fission products (namely ^{90}Sr - ^{90}Y, ^{137}Cs – ^{137m}Ba) together with short-lived actinides.

In the case of spent fuel, after some 70 years the heat production due to actinides becomes equivalent to that from fission products; during the subsequent 100 000 years it is governed by Pu isotopes. ^{241}Am contributes significantly to the heat load for about the first 1 000 years. In the case of vitrified HLW, the residual power is a limiting factor for the loading fraction of the short-lived fission products into waste glass [2-26] (see Table A6-2 in Annex 6). In the case of HLW not containing Pu and minor actinides, which would result e.g. from closed fuel cycles of possible future implementation involving full recycling and burning of actinides, the heat dissipation issue would not have long-term relevance due to the relatively fast decay rate of the short-lived fission products [2-27], [2-28] and [2-29].

Similar considerations can be made concerning the radioactivity level of the HLW. If present in the waste form, the contribution to the radioactivity by long-lived actinides (^{239}Pu, ^{240}Pu, ^{241}Am) becomes predominant after a few hundred years. In particular ^{239}Pu dominates the activity up to ~100 000 y. After very long times, up to 1 million years, actinides such as ^{237}U, ^{237}Np, ^{242}Pu will constitute most of the radioactivity in spent fuel. Very long-lived fission products, present after ≥10^5 years, include e.g. ^{107}Pd, ^{126}Sn and its daughter ^{126m}Sb.

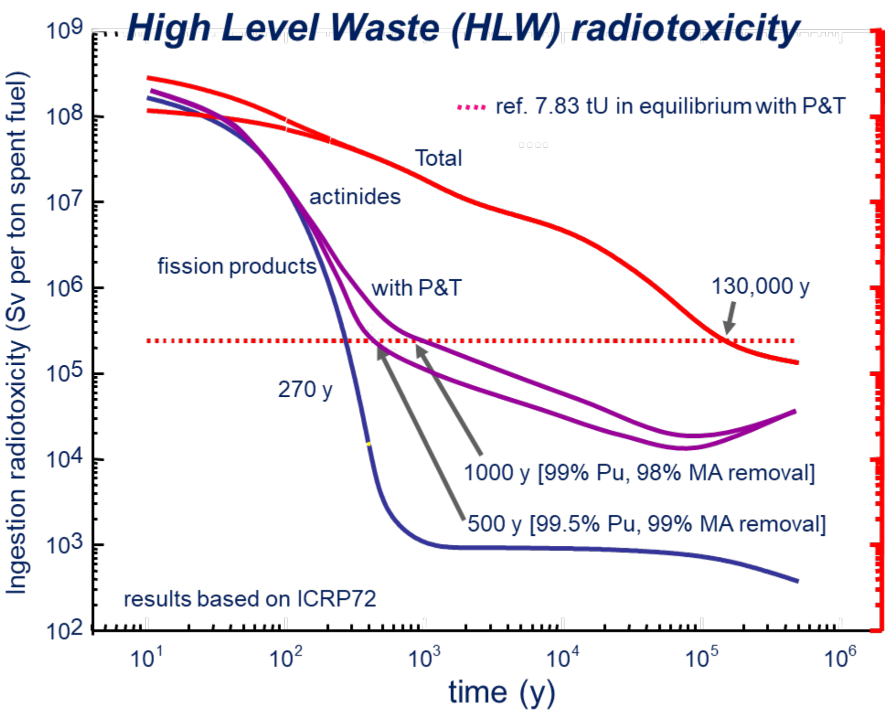
In addition to the above mentioned actinides, long-lived fission products such as ^{99}Tc, ^{129}I, ^{79}Se and ^{135}Cs will be contributors to the long-term radiotoxicity of HLW and spent fuel. These radionuclides will govern the long-term radiotoxicity in the case of HLW not containing actinides. Figure 2.4-1 shows the radiotoxicity as a function of time for spent fuel, HLW from today's reprocessing (recycling of U and Pu), and HLW resulting from closed fuel cycle concepts. The curves on the diagram are normalized to a natural uranium ore and illustrate the potential variation of the time necessary to reduce the radiotoxicity to a level corresponding to "natural" conditions associated with different fuel cycle back-end strategies.

Neutron activation products generated in structural components (e.g. grids, spacers, etc.) or from impurities in the reactor core contribute to a small extent to the radioactivity and the heat production in the HLW. In the long-term perspective (more than 100 000 years), ^{59}Ni, and, in smaller amount, ^{94}Nb will be present, together with ^{93}Zr and ^{93m}Nb, which are both activation and fission products.

^{14}C and ^{36}Cl are also relevant radionuclides, due to their relatively high radiotoxicity, and, in the case of ^{36}Cl, the long half-life.

Figure 2.4-1. Radiotoxicity as a function of time of HLW incorporating different radionuclide families, corresponding to different fuel cycle back-end strategies.

The dotted horizontal line represents the radiotoxicity of the natural minerals from which the uranium was extracted.



Source: www.nucleonica.net.

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3 Strategies and technologies for radioactive waste management.

The primary goal of radioactive waste management is to reduce the risks posed by the radioactive waste to as low as practicable and justifiable and various strategies are used to achieve that [3-1]. Different countries might opt to define their radioactive waste management strategy in different ways. Some countries might define the strategy at national level and entrust an entity or organisation responsible for implementation, or establish the requirements and impose the obligation of waste management to the waste generators.

The preferred strategy for management of radioactive waste is its concentration and isolation from the biosphere in a suitable disposal facility for as long as the waste remains hazardous. For certain types of waste with a low concentration of activity, typically gaseous and liquid effluents the management strategy is its dilution and release to the environment. This is carried out under regulatory control following strict procedures ensuring that releases are below authorised limits, and it is outside the scope of this section.

Like all waste management, radioactive waste management is based on the simple premises: that waste must be managed safely and responsibly, and that it is better to avoid its generation than having to treat or dispose of it, the latter of which must be considered only as a last resort. Management options thus include, in increasing order of environmental impact and hence decreasing order of strategic preference:

— Prevention of radioactive waste generation,

— Minimisation of waste generation, both in terms of quantity and radioactivity content;

— Reuse or recycling of materials that must eventually be treated as waste; and

— Waste disposal.

Minimising the volume of waste reduces the requirements of the waste management system, and consequently the associated costs. Prevention and minimisation of waste generation can be achieved, for example, by designing optimised procedures and practices in the different stages of the nuclear fuel cycle; by removing all material with radionuclide concentration below the clearance level from regulatory control, if allowed in the national framework; and by the application of decontamination techniques to concentrate the radioactivity in a smaller amount of waste and release the rest for reuse and recycle if possible.

However, the generation of radioactive waste is inevitable, and this waste shall then be managed in such a way that people and the environment are protected at all times from the harmful effects of ionizing radiation. Waste management strategies and technologies aim at minimising the amount of waste that needs to be disposed of, and preparing the waste in a form suitable for the handling and processing in the different management stages, including storage, and finally achieving a suitable form for disposal. Although storage and disposal are crucial steps in the management of radioactive waste, they are dealt with separately in Chapters 4 and 5 of part B of this report. This section focuses mainly in the pre-treatment, treatment and conditioning of radioactive waste.

A simplified outline of the radioactive waste management cycle is shown in Figure 3-1.

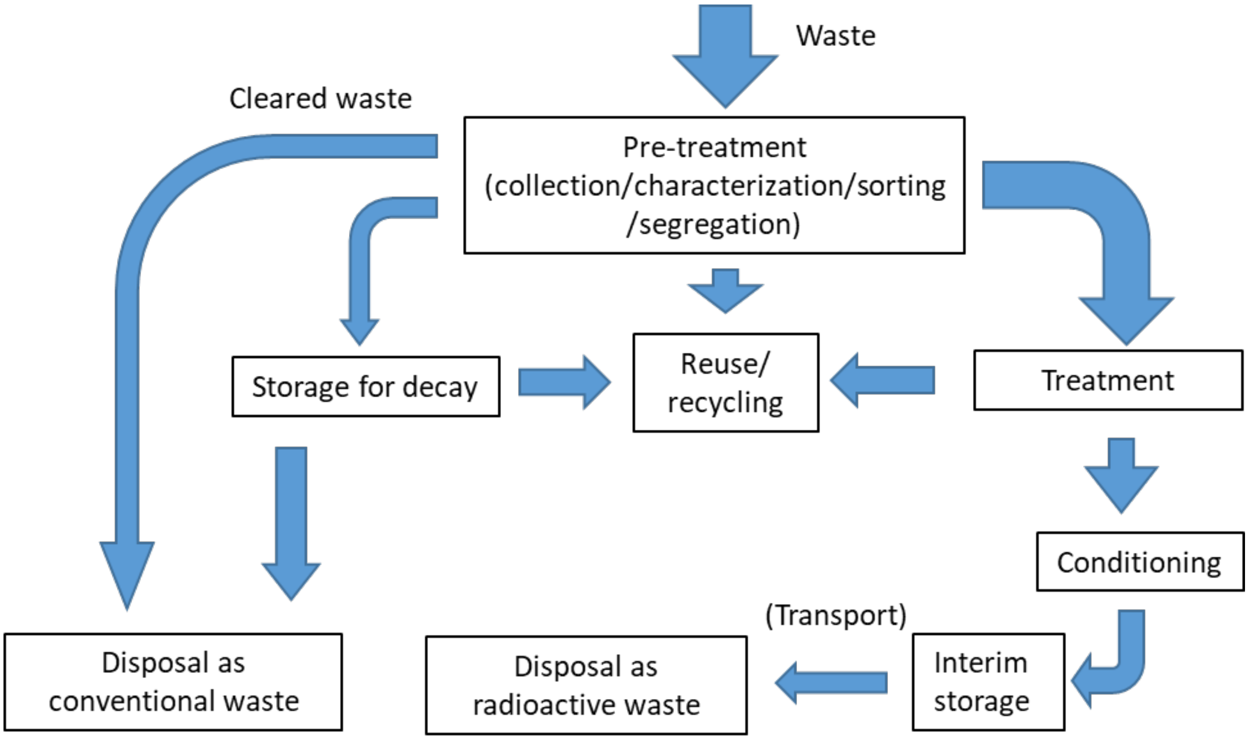
Radioactive waste management starts with the collection, characterisation, sorting and segregation in categories depending on the properties of the waste and the management route. Waste is collected at origin in appropriate containers (for solid waste) or tanks (for liquid and gaseous waste), and prepared for pre-treatment.

Characterisation consist in determining the properties and composition of the materials in view of sorting and segregating them depending on the treatment routes. Characterization starts at the time of waste generation and continues across all phases of the waste management cycle. In general, in-situ measurements are better suited to those radionuclides emitting penetrating radiation (typically gamma rays and neutrons). For alpha and beta emitters and for low concentrations of radioactivity, samples from the waste streams are collected for laboratory physicochemical and radiological analysis.

Nuclides that do not emit penetrating radiation are considered hard to measure radionuclides. Methodologies combining laboratory measurements with analytical modelling allow quantifying the presence of these hard to measure nuclides in waste, enabling adequate characterisation.

Waste is sorted and segregated based on the radioactivity content, state (liquid, solid, gas), physical, chemical and biological properties (combustible or non-combustible, compactable or non-compactable, heat emitting, etc). These characteristics also determine the most adequate treatment and conditioning options.

Figure 3-1. Flow diagram for radioactive waste management



Source: JRC, 2021.

If allowed by the national legal and regulatory framework, materials can be removed from regulatory control through a clearance process that would require the demonstration that the radioactivity content of the material is sufficiently low to ensure that the potential radiological impacts it may cause are negligible. In the case of materials with very short lived radionuclides, the clearance levels are fulfilled after a short time of storage for decay. Cleared materials can then be reused, recycled, or managed as conventional waste. Some materials or equipment that cannot be removed from regulatory control can anyhow be authorised to be reused or recycled maintaining the regulatory control

Waste **treatment** includes all the operations intended to prepare the waste for conditioning, by changing the characteristics of the waste. Depending on the waste nature, basic treatment objectives encompass volume reduction, removal of contamination from the waste, and change of waste composition. Different treatment techniques can be applied depending on the nature and characteristics of the waste. The following paragraphs present briefly the most common ones.

Decontamination consist in the removal of radioactivity from the surface of solids, typically metal and concrete. Decontamination is based on physical techniques such as water jets, blasting, scarification, or on chemical techniques such as washing, or application of acid and basic solutions. Decontamination allows for significant radioactive waste volume reductions, as it removes the portion of material that is contaminated from the rest achieving very high decontamination factors.

Compaction and super-compaction, used for soft material (typically technological waste such as cloths, gloves, plastic, etc), consists in a strong volume reduction by compressing drums filled with compactable waste in high-pressure compactors;

Metal melting is used to concentrate the radioactive content of metal waste in a small amount of secondary waste, making the decontaminated bulk of the metal available in ingots for recycling; metal melting facilities include systems to trap radioactive airborne nuclides present in the fumes or gases produced during melting.

Incineration (with a plasma torch or with conventional incinerators) can be applied to concentrate the contamination of combustible waste in a much smaller volume of ashes; incineration facilities include systems (filters or scrubbers) to retain airborne nuclides present in the fumes and gases produced during combustion. Ashes need to be immobilised by grouting.

Filtration consists in the separation of non-soluble particles in suspension in a liquid or in a gas by circulating it through a porous media. The porous media withholds the suspended solids that are afterwards managed as solid waste. Separation can be enhanced by chemically induced precipitation, or adsorption;

Ion exchange consists in the separation of soluble radioactive species by circulating it through a bed of ion-exchange resins in a demineralizer. The dissolved ions in the liquid stream are exchanged by ions of the same charge from the resins. Radioactive ions carried by the liquid stream are thus withheld in the resin bed. Spent resins are immobilised and further managed as solid radioactive waste.

Reverse osmosis consist in forcing the flow of liquid radioactive waste through a semi-permeable membrane. A solution with a high concentration of impurities is retained upstream of the membrane, while pure water is driven downstream of the membrane;

Evaporation consist in concentrating an aqueous solution by removing the water by boiling. The steam is condensed and recycled for reuse, or release. The concentrate retains the radioactive materials and is normally solidified with cement or dried and immobilised and further managed as solid radioactive waste;

Waste treatment techniques yield a large stream of material with a lower concentration of radioactivity and a small stream of material that concentrates most of the radioactivity of the original material coming from the process itself or from the different systems to retain radionuclides. This second stream is also known as secondary waste, and must be managed appropriately. The characteristics, nature and treatment and conditioning requirements of the secondary waste generated are a fundamental factor in the decision to use a particular radioactive waste treatment technology. The principles of radioactive waste management are also applicable to secondary waste, and treatment technologies and practices aim at limiting the generation of secondary radioactive waste as much as feasible.

Waste **conditioning** involves all those operations that produce a final waste package suitable for handling, transport, storage and disposal. For the majority of the radioactive waste, conditioning yields a waste package that can be disposed of directly, and that constitutes one of the engineered barriers against the potential release of radioactivity to the biosphere. For some waste, notably high and intermediate level waste, and spent fuel, conditioning yields waste packages suitable for the long-term interim storage necessary until deep geological disposal facilities for these types of waste are available. Usually, this waste needs to be reconditioned or reencapsulated before being disposed of. Conditioning is also required for transport of radioactive waste from one place to another (e.g. from a radioactive waste treatment plant to a disposal facility). In this case, the waste conditioned for transport needs to be reconditioned or reencapsulated for the interim storage or disposal. Typical conditioning measures include immobilisation of the waste, encapsulation in suitable containers, and overpacking for further protection and dose reduction if necessary.

Immobilisation consist in converting the waste into a solid form, by, for example solidifying liquid waste with cement or other additives to obtain a solid monolith, or by vitrification of liquid waste to obtain a glass matrix. The waste is integrated with the matrix that contains it at molecular level.

Encapsulation of waste consist in enclosing radioactive waste in a suitable container that protects and shields the waste. The waste in the containers can be further immobilised by adding cement or concrete, as is normally done with solid waste. In the particular case of spent fuel and high level waste, the container shall ensure the dissipation of the residual heat, and provide appropriate protection against spent fuel criticality by passive means. If necessary, waste packages are overpacked, or placed into a secondary containment for additional protection or shielding, or to fulfil transport requirements.

The objective of immobilisation is to limit as much as possible the mobility of the radionuclides, and prevent their release out of the waste form, and their eventual migration across the engineered and natural barriers to the accessible biosphere. Several materials have been used as matrices for waste immobilization, such as glass (vitrification of high level waste from the reprocessing of spent fuel), cement (immobilisation of liquid and solid low and intermediate level waste), and bitumen (immobilisation of solid radioactive waste), among others.

Conditioning technologies are selected depending on the properties of the waste and the acceptance criteria, aiming at optimising the amount of waste that can be contained per unit volume, at long durability and stability, and chemical compatibility with the waste and with the disposal environment, etc.

Transport involves the transfer of radioactive waste between successive management stages, and it is heavily dependent on how the waste management is organised in the country, for instance on the number of centralised treatment, conditioning, storage or disposal facilities. Waste must be conditioned and packed to protect the public and the environment including in the case of accidents. Internationally accepted regulations for the transport of radioactive material contain the necessary requirements to ensure the protection of the public and the environment during the transport.

3.1 References for Chapter 3

[3-1] Policies and Strategies for Radioactive Waste Management. NW-G-1.1. International Atomic Energy Agency. Vienna, 2009.

[3-2] Selection of Technical Solutions for the Management of Radioactive Waste. TECDOC-1817. International Atomic Energy Agency. Vienna, 2017.

[3-3] INTERNATIONAL ATOMIC ENERGY AGENCY, Disposal of Radioactive Waste, IAEA Safety Standards Series No. SSR-5, IAEA, Vienna (2011).

4 Storage of radioactive waste.

It is important to highlight the difference between storage and disposal of radioactive waste. Both are necessary stages of radioactive waste management.

Storage can be considered the placement of waste in a nuclear facility where isolation, containment, and human and environmental protection are provided under active human control. Generally it is intended that the waste will be retrieved from storage at a later time for further treatment. Storage is a necessary intermediate stage, which cannot replace the need for disposal of long-lived waste forms (including HLW) in a geologic repository. Thus after storage long-lived waste and HLW will have to be retrieved and transported to the disposal facility.

Disposal is the final stage of radioactive waste management and consists of placing the waste form in a configuration that ensures that no significant harm, as defined in the relevant regulations, is caused to human life and the environment. The potential hazard of radioactive waste is associated with the decay of the radionuclides present therein. Therefore, in the case of long-lived waste forms, in particular HLW, the long-term containment and protection functions have to be ensured by a system of multiple redundant barriers that will operate in a passive mode and will not require active human control and intervention (see Chapter 1; the disposal of radioactive waste is described in Chapter 5). No undue burden shall be posed on future generations.

No matter what fuel cycle solution is chosen, there will always be the need for a geologic repository for final disposal of HLW.

4.1 Storage of low and intermediate wastes

From generation to the final disposal, radioactive waste goes through multiple waste management processes and storage may take place at a number of management stages. Type and design of storage facilities will differ depending on the purpose and duration of the storage, type of the waste to be stored and on the radioactive waste management stage. The most common reasons to store radioactive waste are [4-1]: to allow for the decay of short-lived radionuclides; to collect and accumulate a sufficient amount of radioactive waste for treatment, conditioning or prior to its disposal; to provide long-term storage when there's no suitable disposal facility.

Figure 4.1-1. Decay storage of large components in the storage facility Zwischenlager Nord (Germany)

Source: www.ewn-gmbh.de

Radioactive wastes are generated in various physical forms and having different levels of radioactivity. For radioactive wastes contaminated only with short-lived radionuclides, storage for a specific period of time could reduce concentration of radionuclides of concern by radioactive decay. If concentration of radionuclides is reduced below the established clearance levels, the resulting materials could be removed from regulatory control and managed as a conventional (non-radioactive) waste. For higher activity radioactive wastes the time needed for decay below the clearance levels could be too high. However, temporary storage for decay could reduce the radioactivity to lower levels and potentially reduce risks and costs of further radioactive waste management. Example of radioactive components stored for decay are shown in Figure 4.1-1.

In radioactive waste management a temporary storage could be necessary due to operational reasons. Typically, limited amounts of radioactive waste are temporarily accumulated and stored awaiting treatment and/or conditioning. However, some of the radioactive wastes could be accumulated and stored for longer periods if there's no suitable treatment or conditioning process in place, until such process becomes available. In such cases, duration of radioactive waste storage could extend for several years or even longer.

After treatment and/or conditioning into passively safe forms, radioactive waste is sent to the interim storage where it is stored awaiting further management steps (e.g. radioactive waste disposal or clearance from regulatory control if during storage period radioactivity decays below set clearance levels). The duration of the radioactive waste storage in most cases depends on whether the next management steps are available.

In case the final disposal solution is already available and operational, radioactive waste is usually sent directly to the disposal facility. However, it could be stored for a short period in an interim storage on the radioactive waste generator's site. The storage time depends mainly on acceptance and transport arrangements. Short-term storage is typical for short-lived radioactive waste classified as VLLW or LLW, as many countries already have operational near-surface or surface disposal facilities.

Radioactive waste containing significant amounts of long-lived radionuclides is classified as ILW and it should be disposed of in geological disposal as it requires a higher degree of isolation from the environment. As geological disposal facilities are not yet operational, long-lived radioactive waste is stored in a long-term storage awaiting implementation of the disposal solution. The storage time in such cases could be several decades or even more, as planning and implementation of geological disposal is a lengthy process. Depending on the radioactive waste inventory size and the size of the radioactive waste management programme, long-term storage could established on the radioactive waste generator's site or implemented in a centralized manner as a centralised storage facility where all radioactive waste generators deliver their radioactive waste.

Figure 4.1-2. ILW storage at the Trawsfynydd (UK)

Source: https://www.powermag.com/

The interim storage facilities are designed to protect waste packages from major natural (e.g. earthquakes, flooding, tornados, etc.) as well as man-made hazards (e.g. fire, malevolent act, in some cases also aircraft impact) and to ensure that operators and the general public are protected from any radiological hazards. Due account shall be taken of the expected period of storage, and, to the extent possible, passive safety features shall be applied. The importance of the final waste package is that its characteristics are taken into account with those of the storage facility in the safety assessments. As radioactive waste is stored for long time periods, it is important to ensure that stored waste packages can be easily inspected and monitored, and, if necessary, retrieved and preserved in suitable conditions. For long-term storage in particular, measures shall be taken to prevent degradation of the waste packaging.

The complexity of an interim storage facility depends on whether the waste to store is in conditioned or non-conditioned form. Storing only conditioned waste complying with the requirements for transportation and disposal in the final repository would reduce the requisites of an interim storage facility, making it similar to a conventional building; this means that some of the required safety features are imposed on the packaging. On the other hand, the characteristics of an interim storage facility qualified to store also raw waste would be more demanding, but would offer greater opportunities in decommissioning strategies and timing. An assessment for the selection should take into account existing regulations, technical requirements, infrastructure aspects, radiological conditions and economical aspects.

Some of the radioactive waste packaging examples used for storage at interim storage facilities are shown below.

For storage of short-lived radioactive waste different kinds of packaging can be used. VLLW and certain LLW can be stored without conditioning matrix. The most common container used is the carbon steel drum in versions of different capacity, the 220-litre drum shown in figure 4.1-3 being the most frequent. Unconditioned VLLW can be subject to specific treatments to reduce, as much as possible, inner voids by means of “light” grouting.

Figure 4.1-3. Drums for VLLW/LLW

Source: nks

LLW conditioned with matrix are grouted generally into steel containers such as the one shown in figure 4.1-4.

Figure 4.1-4. Interim low and intermediate level waste storage (Federal Interim Storage Facility, Switzerland)

Source: ENSI

Figure 4.1-5. Containers for ILW

Source: NDA.

In addition to the ordinary function of containerization, packaging for ILW may need to provide enhanced radiation shielding, for beta-gamma contaminated waste, or airtightness, for alpha contaminated waste, or both. Generally, they are prismatic or cylindrical steel containers. Some models are shown in figure 4.1-5.

4.2 Storage of spent fuel and high level waste

Interim storage of spent fuel is a necessary stage of spent fuel management. Based on the IAEA/EC/NEA Status and Trends in Spent Fuel and Radioactive Waste Management there are approximately 260 000 tHM of spent fuel in storage worldwide. In the open cycle scenario, its primary, technically driven purpose is to let spent fuel cool down to a level which is compatible with the disposal in the final repository; hence there is a minimum storage time determined by the residual heat of the spent fuel and by the acceptance criteria of the final repository (see also Chapter 3.3.8 in part A of this report). In more general terms, interim storage bridges the gap until a deep geological facility is available to dispose of spent fuel or high level radioactive waste. The evolution of the national strategies for the back-end of the nuclear fuel cycle, including changes of the overall strategy (e.g. moving from reprocessing to open cycle), development of the reprocessing capacity (for those countries opting for a partially closed cycle), and the evolution of plans and timeframe for the construction of deep geological disposal facilities, have resulted in increased reliance and attention on interim storage. In several countries where the decision to build the geologic repository is postponed, the duration of the interim storage will increase; timeframes of the order of one century are considered.

Detailed design and operation of spent fuel and other waste storage are described in [4-1] and [4-2]. The timeframes associated with the different stages of the fuel cycle back-end were discussed in [4-3]. A recent overview of the current situation in several countries is available in [4-4] and [4-5].

Spent fuel pool at the nuclear power plant (see also Chapter 3.3.8 in part A of this report)

The capacity of the spent fuel pool at the nuclear power plant is a limiting factor to implement storage operations in-situ. The design capacity of the spent fuel pool of the nuclear power plants commissioned in the past century is based on the need to cool down spent fuel assemblies until they could be transported during the operational life of the reactor and entails only limited applications for relatively long storage.

In addition to the above-mentioned factors, the need for additional spent fuel storage capacity is associated also with the implementation of long-term operation of the nuclear power plant.

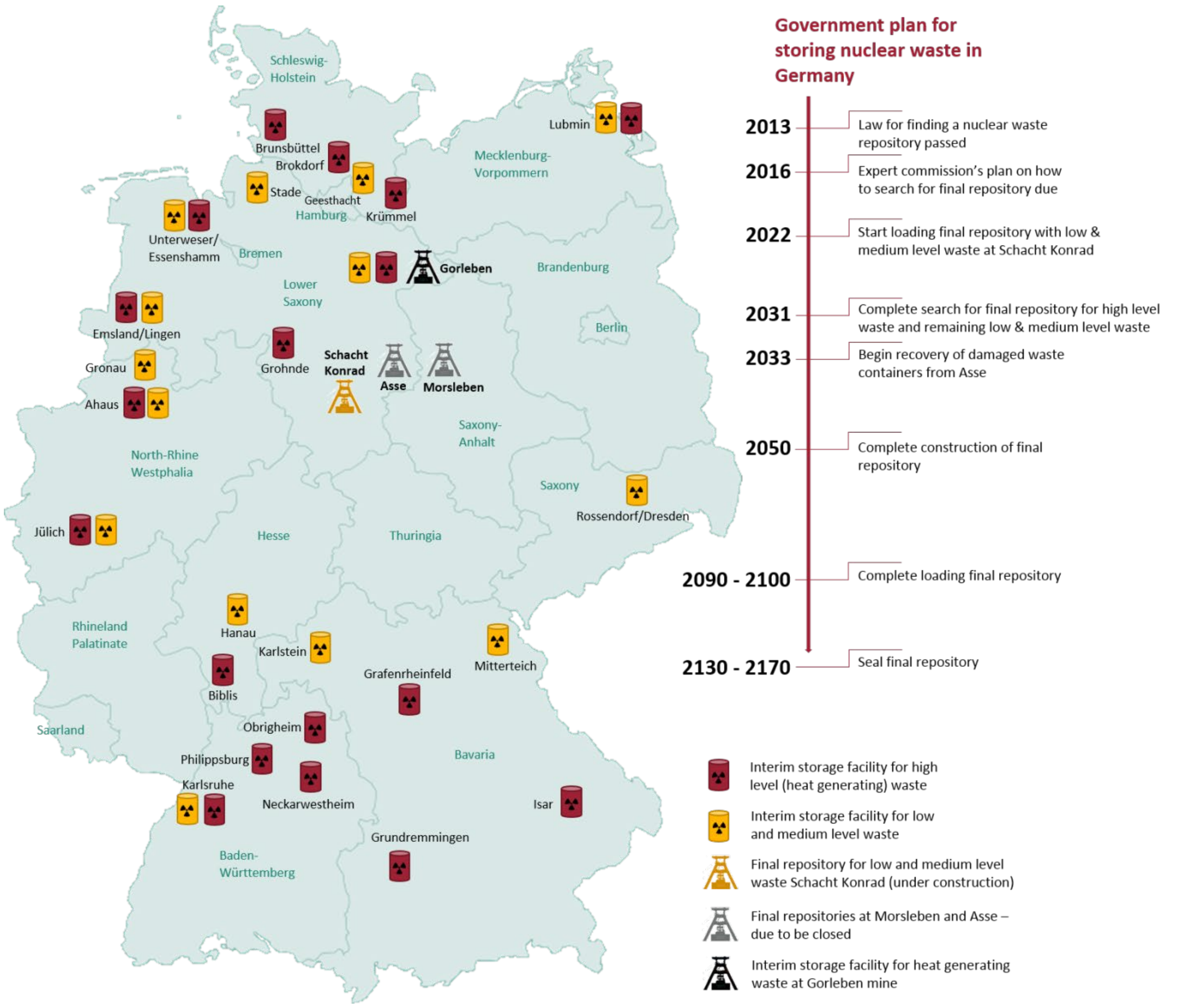
The capacity of the pools was increased by increasing the density of spent fuel assemblies in the pool. Reducing the distance between fuel assemblies required in most cases a modification of the storage racks by incorporating neutron absorbers in their frame. Re-racking alone cannot be a solution to address the scarcity of space. Extending use and capacity of independent spent fuel storage facilities (i.e. not depending on the operation of a nuclear power plant) provides the necessary solution.

Independent storage facilities

Dry or wet storage facilities (see also Chapter 3.3.8 of part A of this report) at the reactor site or in centralized locations away from the reactor are in operation in several European countries. Figure 4.2-1 shows a map of locations hosting radioactive waste in Germany, including spent fuel storage facilities.

Independent spent fuel storage facilities can be built at the reactor site, or away from the reactor site, and typically are based on dry storage technology although there are examples of independent spent fuel storage facilities using wet technology. National strategies might consider on-site storage as a first step prior to storage in a centralized facility whose aim is to store spent fuel until a disposal facility is available. Centralized facilities store the spent fuel of several nuclear power plants or research reactors, and are based on dry or wet storage technology. Their expected lifetime tends to be longer, for instance, in the case of continuing nuclear energy production (through long-term operation and/or construction of new nuclear power plants). Some countries, such as Spain and the Netherlands have or plan to have centralized storage facilities with a foreseen design life of 100 years (that in the case of the Netherlands could be extended for an additional period of 100 years if needed). During this time, research and assessments of the different final disposal options available will be performed, including new technologies or approaches emerging in the next 100 years. This will provide the basis for the ultimate decision to be taken.

Figure 4.2-1. Radioactive waste locations in Germany, including spent fuel storage facilities.



Source: www.cleanenergywire.org

The operation of storage facilities is licensed for a certain duration [4-4]) and is subject to periodic safety review (typically, every 10 years). Licence extensions can be granted through a specific process [4-5]. The requirements set by the regulations are based on a defence-in-depth approach including multiple layers of protection.

In order to guarantee compliance with the storage safety functions defined by the regulations, the design of an interim storage facility should adequately address the following factors [4-5]:

— passive safety;

— multiple barrier containment;

— robust storage facilities with adequate storage capacity;

— appropriately established waste acceptance criteria for storage;

— effective storage facility maintenance, inspection and retrieval;

— record management.

Irrespective of its duration, the interim storage is a step of the radioactive waste management which is characterized by various levels of direct monitoring, inspections, periodic assessments, and allows direct human intervention.

Figure 4.2-2. HABOG centralized HLW dry storage facility in the Netherlands.

Source: COVRA

Dry storage

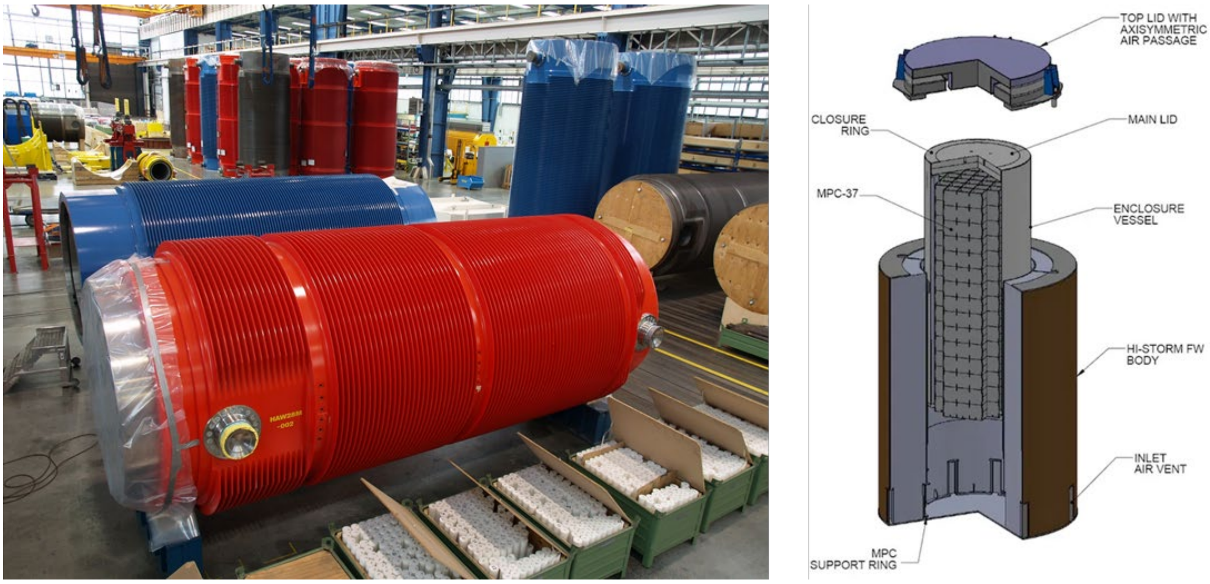
Several technologies are available for dry storage. Spent fuel assemblies are placed in baskets inside canisters. Damaged fuel assemblies are encapsulated in special capsules (quivers) and placed in specific locations of the basket. The canisters are placed inside containers, which provide additional shielding, and protective functions. After the loading operations (typically occurring in the pool), the containers are drained, dried, pressurized with an inert gas and their lid is welded or bolted with a gasket configuration to ensure airtightness. The containers (or canisters) are maintained airtight with bolts and gaskets or welds to ensure confinement of the radioactive substances inside. The inert gas filler limits the corrosion of the fuel assemblies' structure and at the same time enhances the heat transfer and subsequent removal of decay heat by natural convention of air. Thus the different components of the interim dry storage designs ensure fulfilment of the safety functions: the basket maintains the geometrical constrains of the fuel assemblies, and contains neutron absorbers in its structure to prevent criticality; the containers also provide for biological shielding for neutron and gamma radiation by using adequate materials (e.g. lead, resin, concrete, etc); In addition to the presence of shielding material in the container, full radiation shielding is ensured by placing the container inside an overpacking or by inserting the canister in vaults, depending on the design of the storage facility. Retrieval of spent fuel if necessary is possible during the storage period.

Some container designs are dual purpose: they are suitable for the storage of spent fuel and for transport. Figures. 4.2-3 and 4.2-4 show the schematic layout of spent fuel and vitrified HLW in use in Germany. The evolution of the design of dry storage containers has been characterized by an increasing capacity in terms of spent fuel assemblies. Typical containers capacity today can be in excess of 30 PWR or 50 BWR assemblies. The maximum capacity is affected by the heat load generated by the spent fuel and by overall mass and handling considerations. In case of dry storage and direct disposal of mixed uranium-plutonium oxide spent fuel (MOX), the higher heat generation rate compared with uranium oxide spent fuel must be factored in and may impose additional requirements for the interim storage facility, and, eventually, the final repository.

Figure 4.2-3. CASTOR® spent fuel storage casks. (a)V/19 (PWR), (b)V/52 (BWR), (c) HAW28M (HLW).

Source: VGB PowerTech 5 2015 [4-6]

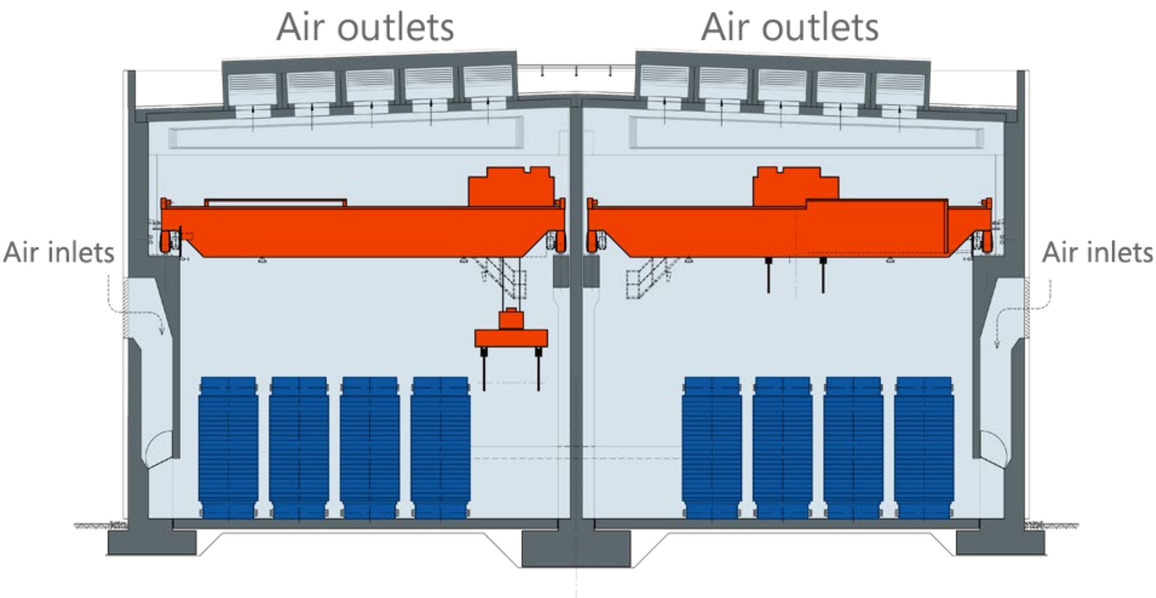
Figure 4.2-4. CASTOR® spent fuel storage casks.



Source: GNS and HOLTEC International

In case of dry storage on site, the containers are stored in a reinforced concrete building that ensures physical protection and enhances the natural convection of air for residual heat removal. Figure 4.2-5 shows an example of a dry storage vault facility on site.

Figure 4.2-5. Schematic layout of an on-site dry storage facility in vault.



Source: BGZ, Germany; https://bgz.de/en/interim-storage/

In an alternative design, spent fuel is placed in neutron absorbing baskets inside airtight steel canisters, which are contained inside an overpack made up by concentric steel cylinders alternating with high-density concrete, and placed vertically on a seismically designed reinforced concrete slab The overpacks are stored outdoors (Figure 4.2-6). Other outdoor dry storage options include placing the canisters into horizontal concrete storage modules.

Figure 4.2-6. Outdoors spent fuel dry storage facility, José Cabrera nuclear power plant, Spain.

Source: www.latribunadecuidadreal.es

Centralized spent fuel dry storage facilities (Figures 4.2-2 and 4.2-7) typically consist of a concrete structure (building) for physical protection and radiation shielding with a cask reception and handling area, and a storage area. Leak-tight canisters are stored in the storage area or can be placed in concrete wells that are plugged when filled.

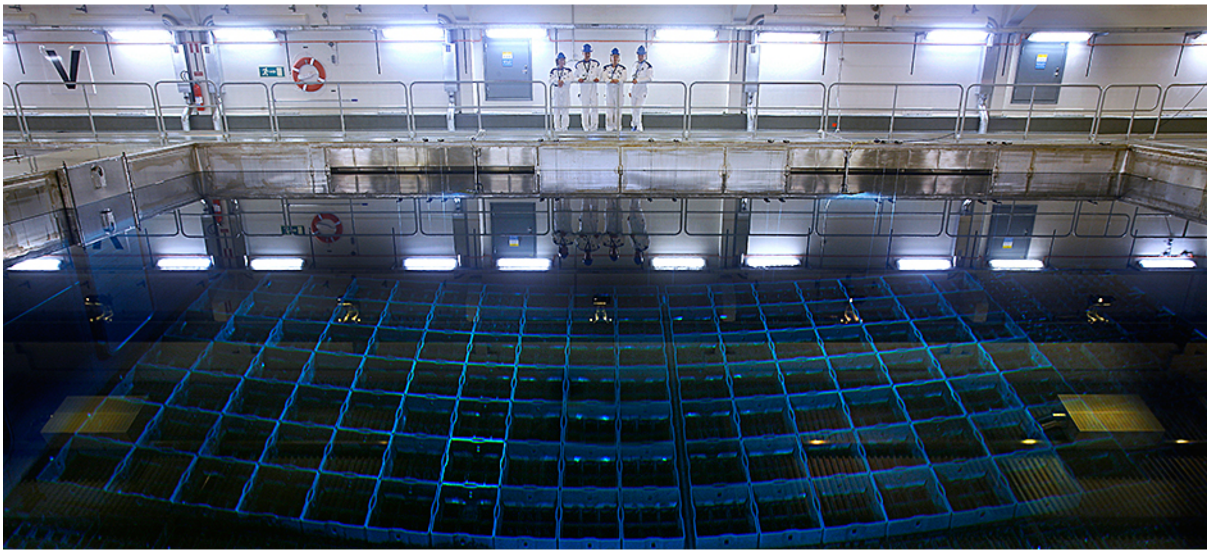
Figure 4.2-7. Centralized spent fuel dry storage facility in Gorleben, Germany.

Source: GNS

Wet storage

The main alternative to centralized dry storage (e.g. as pursued by Sweden and Finland) is to use wet spent fuel storage installations (Figure 4.2-8). These facilities are rather similar to the spent fuel pools of the nuclear power plants and include improvements and features to take into consideration its independence from power plant systems. Underwater storage allows for a more efficient cooling, maintaining the spent fuel at lower temperatures, as well as easier radiation shielding. However, it requires the operation of active systems, produces more waste (e.g. by the water purification system), and needs continuous operation and maintenance (see Chapter 3.3.8 of part A of this report).

Figure 4.2-8. CLAB centralized spent fuel wet storage facility, Sweden.



Source: SKB

Sweden operates a centralized spent fuel wet storage facility (CLAB, see Chapter 3.3.8 of part A of this report) consisting of two large pools excavated about 30 m deep in the bedrock to store under water all the spent nuclear fuel produced in the country before it is disposed of in a deep geological facility. Currently, the spent fuel cooling time envisaged before loading the spent fuel assemblies into the disposal canisters is approximately 30 years. The facility includes cooling and cleaning systems to maintain adequate conditions of the water. In a connected building above ground, handling and ancillary systems are in place for reception of casks and storage of spent fuel assemblies. In this scenario, spent fuel undergoes two drying processes: one when loading into the dry cask used for transportation to CLAB, and a second when loading into the dry cask for transportation to the encapsulation facility. The spent fuel assemblies are subjected to an intermediate quenching process when unloaded from the dry transportation cask and stored under water in CLAB.

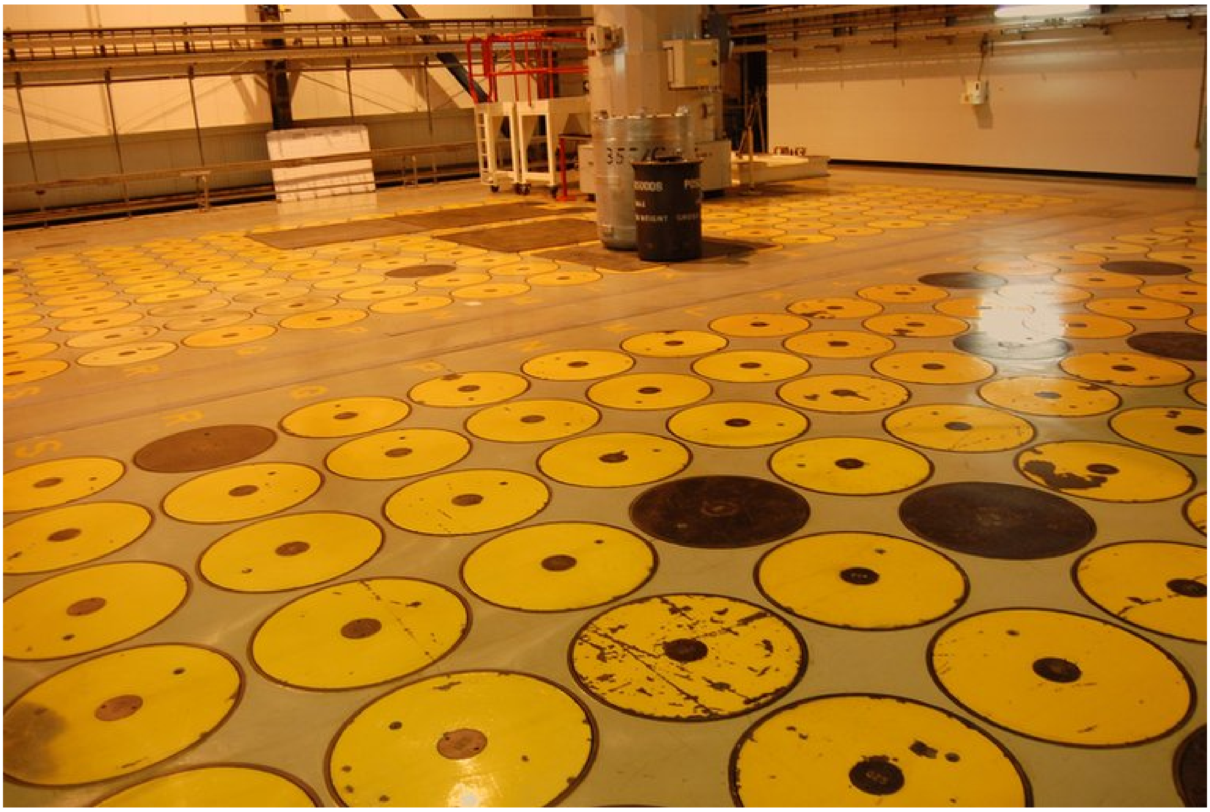
Reprocessing facilities require wet storage, and typically include large buffer storage capacity for spent fuel awaiting reprocessing. In France, after 1-2 years of cooling in the pool at a reactor site, the spent fuel assemblies are transported in dry casks to the wet storage at the La Hague reprocessing facility. The spent fuel must cool for an additional 5 years before being suitable for reprocessing; the typical residence time in the La Hague pool before reprocessing is around 7 years [4-4]. Currently, spent mixed oxide fuel (MOX) is not reprocessed in La Hague as its fissile content is not suited for multiple recycling in light water reactor. The irradiated MOX remains stored at the wet storage facilities, awaiting the introduction of advanced reactor systems which will allow implementing a closed fuel cycle with multiple recycling of uranium and plutonium and transmutation of long lived actinide nuclides.

High level radioactive waste from reprocessing of spent fuel is vitrified in steel containers (Figure 4.2-9). These containers provide only the confinement of radioactive material. Therefore, the waste is stored in suitable containment and shielding structures, e.g. in concrete vaults at the reprocessing facilities (Figure 4.2-10), or in dry storage containers for HLW (Figure 4.2-3 c).

Figure 4.2-9. Stainless steel container for vitrified HLW.

Source: Sciencedirect.com

Figure 4.2-10. Storage of high level waste in vault at Sellafield, UK.



Source: BBC

Implications of extended storage

Generally, spent fuel storage containers are designed for storage and are not suitable for disposal. Multi-purpose containers suitable for transportation, storage and disposal have been considered. The main advantage would be the elimination of spent fuel handling after having placed the spent fuel assembly in the container. However, there would be several disadvantages. In particular, it is not guaranteed that at the time of disposal there will be no need to re-open the package for technical or regulatory reasons. Moreover, optimization criteria for dry storage push towards increasing the number of spent fuel assemblies loaded in the storage canisters; this is in conflict with the limitations in mass, volume and heat load of the spent fuel containers to be emplaced in the deep geologic repository. Therefore, at the end of the interim storage stage spent fuel needs to be retrieved and encapsulated in a different (smaller) container suitable for disposal. As the storage of spent fuel is expected to last much longer than initially foreseen, the effects of the extended storage conditions on the conditions and behaviour of the spent fuel assemblies after such long storage periods are currently the subject of systematic research programmes [4-4].

Both the wet and dry storage technologies currently implemented guarantee storage conditions in which corrosion and other negative ageing effects do not compromise the safety function and performance during subsequent management steps. Extending the safety assessment to cover very long storage timespans requires the characterization and full understanding of potential long term ageing mechanisms (e.g. the effect of thermal cycles/history on spent fuel rods during the different steps of spent fuel management, effects of auto-irradiation) and their potential effect on the relevant properties of the spent fuel assemblies and of the container system (e.g. mechanical integrity, resistance against corrosion, tightness). The goal is to confirm that spent fuel assemblies and containers will retain their integrity and functionality, allowing repackaging and transportation after extended storage in excess of one century, and/or to define preventing or mitigating measures potentially necessary to cope with significant degradation of any containment system (cladding, canister, cask, welds/sealing, etc.). The support of R&D with EU funded projects such as DEMO and EURAD, national research projects such as ESCP (USA), GRS and BAM (Germany), and international research projects, such as IAEA BEFAST I, II, III and SPAR I, II, III, IV, is needed to provide relevant evidence covering the timescale of extended interim storage (see also Chapter 6 of part B of this report)

4.3 References for Chapter 4

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[4-2] WS-G-6.1 Storage of Radioactive Waste (IAEA, 2006)).

[4-3] JRC-EASAC2014: Management of spent nuclear fuel and its waste, JRC, EASAC 2014, ISBN 978-92-79-33885-4).

[4-4] The Safety of Long-Term Interim Storage Facilities in NEA Member Countries, NEA/CSNI/R(2017)4

[4-5] Storage of radioactive waste and spent fuel, NEA 7406, 2020

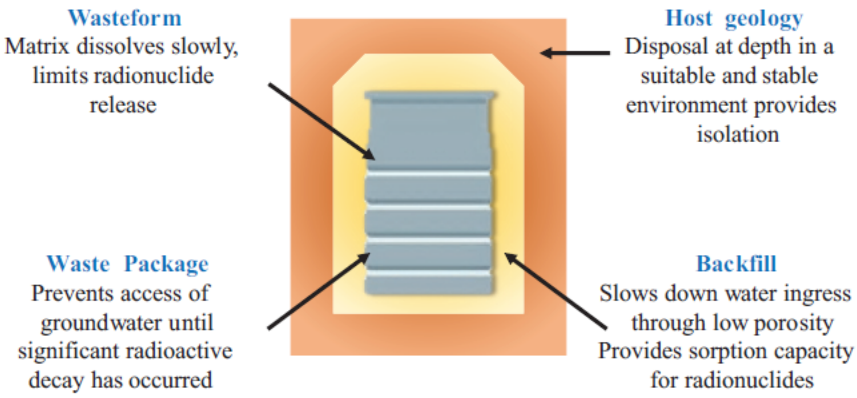
[4-6] H. Wimmer et al., CASTOR® and CONSTOR® A well established system for the dry storage of spent fuel and high level waste, VGB PowerTech 5 2015

5 Disposal of radioactive waste

To be sound and credible, the radioactive waste and spent fuel management strategy must consider a final disposal stage as the end-point. Even when taking into account the developments of technologies and practices for minimising radioactive waste generation currently under study, there will be always a finite amount of radioactive waste that needs to be safely disposed of. Disposal is the last step in the process of radioactive waste management, and consists of the emplacement of radioactive waste in an appropriate facility without the intention of retrieval. Disposal facilities are designed to contain the radioactive waste and to isolate it from the accessible biosphere[136] and from the public for as long as its radioactivity remains hazardous. More specifically, the disposal facilities aim at reducing the likelihood (and consequences) of human intrusion, and at inhibiting, reducing and delaying the migration of radionuclides from the waste to the accessible biosphere; in case radionuclides are released and eventually reach the biosphere their amounts are sufficiently low that the potential radiological consequences are negligible. Strict dose or risk limits to the public, well below the levels that ensure no harm is caused to the public and the environment, are established by relevant regulations. International standards require that the calculated dose or risk to the representative person who might be exposed in the future must not exceed 0,3 mSv in a year, corresponding to a risk (probability of fatal cancer or serious hereditary effect) of 10^{-5} per year [5-1]. The actual dose limit to the representative person associated with a waste repository set by national regulations is generally well below 0.3 mSv/y. Thus the radionuclides of the radioactive waste must be contained in a disposal facility designed so that they will not reach the accessible biosphere in significant amounts, and will never exceed the limit below which they can cause no harm.

This goal is technically achieved by interposing a series of barriers between the waste and the environment. Figure 5-1 schematically illustrates the multi-barrier concept. Some of the barriers are engineered and some are provided by the natural properties of the host rock of the repository.

Figure 5-1. Multi-barrier concept.



Source: Corkhill, Claire & Hyatt, Neil. (2018) [5-2]

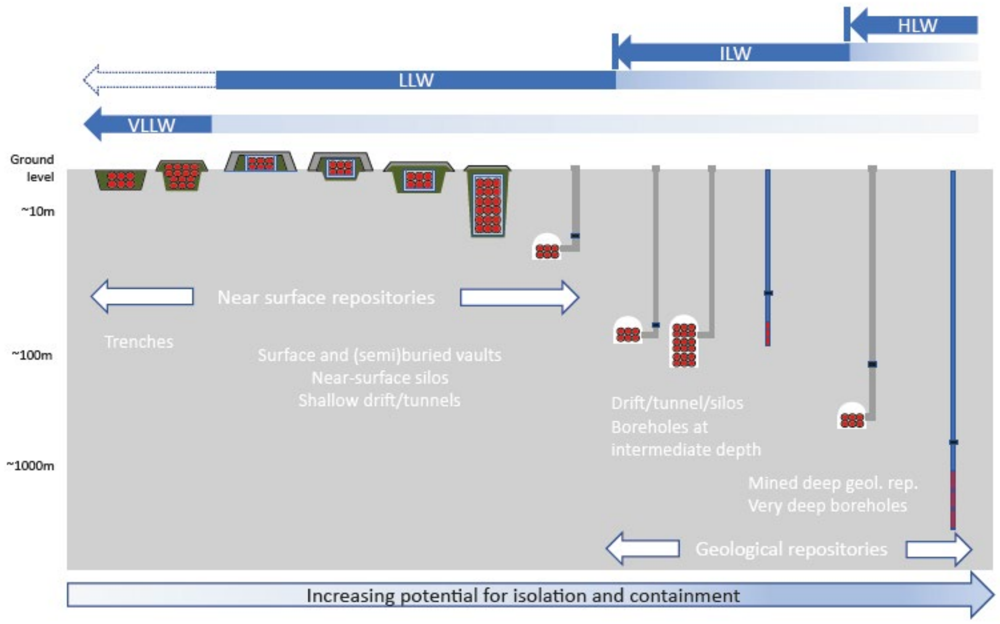
The disposal routes and technologies used to dispose of radioactive waste depend on the characteristics of the radioactive waste: physical and chemical forms, specific activity, heat generation, half-life of the radionuclides and others. The depth of the disposal facility is one of the elements which, combined in the multi-barrier system, contribute to the containment and isolation of the waste. The multi-barrier concept includes engineered or man-made structures (the waste form, containers, concrete structures, buffer material, etc), and natural barriers, namely a geologically stable formation which characterizes the repository (host rock). Engineered barriers are designed as a function of the host geological formation and aim at containing and delaying the release of the radionuclides of the waste to the surrounding geological formation. The natural barriers aim at providing stable conditions in order to maximize the performance of the engineered barriers, and at limiting and slowing down the migration of potentially released radionuclides to the accessible biosphere.

136 The accessible biosphere is the part of the environment that is normally inhabited by living organisms, in this framework, the accessible biosphere encompasses those elements of the environment, including groundwater, surface water and marine resources, that are used by or accessible to the public

The isolation and containment measures are commensurate with the potential harm, which is linked to these characteristics and to the classification of radioactive waste presented in Chapter 2.2. of part B of this report. Therefore, exempt waste which has very low levels of radioactivity does not need to be managed as radioactive, and can be incorporated in conventional waste streams. Very short-lived radioactive waste can be stored to decay for a relatively short, well defined period of time, and then cleared and managed as conventional waste.

Very low level waste is normally disposed of in landfills, with no or minimum engineered barriers and limited regulatory control. A few decades after closure, institutional control can be phased out. Waste originating from mining and milling activities is generated in large amounts and contains long-lived radioisotopes, although it possesses very low specific activity[137]. This waste is stabilised on-site and covered with layers of rocks and soil. Low level waste is disposed of in surface or near-surface facilities (up to a depth of a few tens of meters – typically up to 30 m) with passive engineered and natural barriers aimed to last a few hundred years. The same route can be used for intermediate level waste with half-lives below ~30 years. Longer-lived intermediate level waste is disposed of at facilities a few tens, to a few hundred meters deep – typically up to 300 m. Spent fuel and high level waste, which in addition to high levels of radioactivity also generate non-negligible decay heat, are to be disposed of in deeper geological disposal facilities several hundred – typically more than 300 - meters below ground level, with engineered barriers and embedded in stable geological formations whose characteristics and evolution in the long term are predictable. Figure 5-2 shows the typical depths and concepts associated with the final repositories for different types of radioactive waste. Simpler concepts and designs such as boreholes or shaft facilities for disposal of small amounts of institutional radioactive waste or disused sources (not generated in the nuclear fuel cycle) are outside the scope of this report.

Figure 5-2. Disposal facilities depth depending on the radiological hazards of the waste.



Source: IAEA, (2020) [5-3]

The lifecycle of a radioactive waste disposal facility comprises a pre-operational phase, which includes the siting, design and construction, an operational phase that covers the period in which radioactive waste is emplaced in the facility until and including its physical closure, and a post-operational phase that encompasses monitoring, surveillance and institutional control. The pre-operation and operation of the disposal facility is subject to nuclear safety regulations in a similar way to other nuclear facilities, with periodic safety evaluations, controls, verifications and inspections. While the facility is in operation, the means available ensure the safety and the response to potential radioactivity releases in a similar way than any other nuclear facility in operation. The peculiar characteristic of disposal facilities is that their safety function

137 Activity per unit mass or per unit volume.

must be ensured beyond closure and beyond the end of any post-closure control. This is achieved by a combination of technical solutions, a well-established legal, administrative and regulatory framework, institutional control, and a robust process for long-term safety demonstration. The latter is based on the implementation of the multiple barriers scheme described above. Post-closure institutional controls are limited in time and depend on the nature of the disposal facility. As an example, institutional monitoring and control is limited to a few decades for very low level disposal, and a few hundred years (typically 300) for surface or near-surface facilities for low level waste disposal. Once closed, intermediate and deep geological disposal facilities remain isolated from the accessible biosphere, and institutional control is not needed.

The timeframes for the development of disposal facilities, from their conception until post-closure and phase out of institutional control encompass several decades, in excess of a century. The disposal of radioactive waste is implemented through a stepwise approach. Each step is taken based on a documented decision-making process, in which all relevant, scientific and technical advances, operational experience, social aspects and updates in the legal and regulatory framework can be incorporated. This process allows making decisions that are flexible and do not oblige sticking to a rigid roadmap for the entire lifecycle of the facility, and that involve all the relevant stakeholders in the process. This makes it possible to incorporate new knowledge, decide among different options that are available, or go back to a previous step if necessary. For instance, retrievability, or the capacity to retrieve the radioactive waste from its emplacement, is a required option during the operational phase up to closure of the facility. Once closed, however, the potential impact of incorporating provisions for retrievability on the safety of the disposal should be carefully assessed, as this could interfere e,g, with the protection of the facility against intrusion.

The most important challenges to demonstrate and verify safety stem from the very long timeframes during which the radioactive waste remains hazardous. With the partial exception of the so-called natural analogues (i.e. sites where natural nuclear reactors occurred billions of years ago (see Chapter 6.4.3 of part B of this report), there is no empirical evidence generated by a radioactive waste disposal facility that has gone through all the three stages (pre-operational, operational, and post-closure) for the entire timeframe foreseen (up to a hundred thousand years for a deep geological repository). Although in the world there are many near surface and some intermediate level waste disposal facilities that have been in operation for several decades, and a few of them are already in the post-closure period, none has completed its entire lifecycle.

The long timeframes of the disposal of spent fuel and high level waste also raise concerns about how the conditions of the site might evolve in the remote future, including the impact of the facility and the waste emplaced therein on the surrounding media (e.g. due to heat generation), and how society and human behaviour would be tens or hundreds of thousands of years from now.

For this reason the safety of disposal during the post-closure phase is demonstrated by a robust and reliable process which confirms that dose or risk to the public are kept under all circumstances below the required limits. The safety demonstration includes a description of the site and features of the disposal facility, the characteristics and amount of waste that can be emplaced (waste acceptance criteria), and a description of a relevant series of scenarios including potential and extreme events that could lead to the release of radionuclides from the waste and to subsequent exposure of the public to radiation. The safety demonstration includes calculations and models of the behaviour of the engineered barriers under different circumstances, of the migration of the radioisotopes through the natural barriers, of the effects of climate events, hydrogeological, seismic and other phenomena, and of the impacts and consequences of potential releases of radionuclides from the waste to the public and/or to the environment.

The models and calculations represent the international scientific consensus on the state of the art of the knowledge generated by many decades of study and research of the phenomena that govern the transport and migration of the radionuclides across the entire disposal system. The analysis is reliant on scientific evidence supported by vast amounts of data from studies involving natural systems and the waste forms to be disposed of (as described in Chapter 5.2 of part B of this report), which provide an accurate characterisation of the host bedrock, and detailed knowledge on the behaviour of the engineered barriers. The analysis, in turn, is underpinned by the application of the natural laws that govern the long term behaviour of the geological bedrock and the evolution of the relevant external factors (e.g. the climate). The safety demonstration includes comprehensive sensitivity and uncertainty analysis to support the robustness of and the confidence in the results. Additionally, the safety demonstration is thoroughly reviewed independently and critically by the regulatory authority.

The following sections provide more insights on the near-surface and deep geological disposal of radioactive waste.

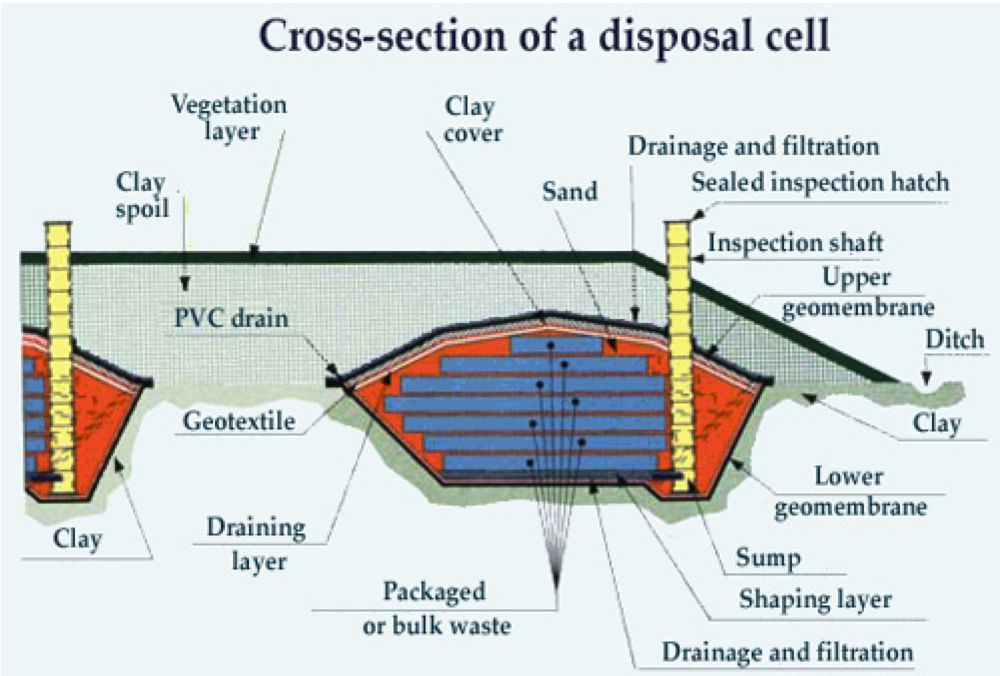
5.1 Disposal of low level waste

The objective of a near surface disposal facility is the isolation of the low level radioactive waste from the accessible biosphere and the public for a period of a few hundred years, typically 300. It is considered that after that period of time there are no more radioactivity hazards. On such a timescale, the behaviour of the materials that constitute the engineered barriers is well known and predictable, and the barriers are considered sufficiently reliable. Therefore, there is no need for deep geological repositories for the disposal of low level waste. Although the waste acceptance criteria are specific for each facility, near surface disposal facilities establish radionuclide content limits associated with half-lives and specific activities: higher concentrations are allowed for beta/gamma emitters with half-lives shorter than some 30 years; and lower concentrations are accepted for alpha emitters and other longer-lived nuclides.

Near surface disposal facilities encompass a variety of designs for the emplacement of solid radioactive waste: earthen trenches, above ground engineered structures, engineered structures just below the ground surface, and rock caverns, silos and tunnels excavated at depths of up to a few tens of metres underground.

Very low level waste is disposed of in landfills with no or minimum engineered barriers, while low level radioactive waste in solid form is disposed of in surface or near surface facilities with multiple engineered and natural barriers (Figure 5.1-1). In the past, the occurrence of disposal of poorly conditioned low level waste in trenches without engineered barriers led in a few cases to the release of radionuclides, which prompted IAEA to recommend a systematic reassessment and, if needed, upgrade of those disposal facilities.

Figure 5.1-1. Schematic layout of a near surface disposal facility for low level waste. The actual depth of the emplacement depends on the radiological features of the waste.

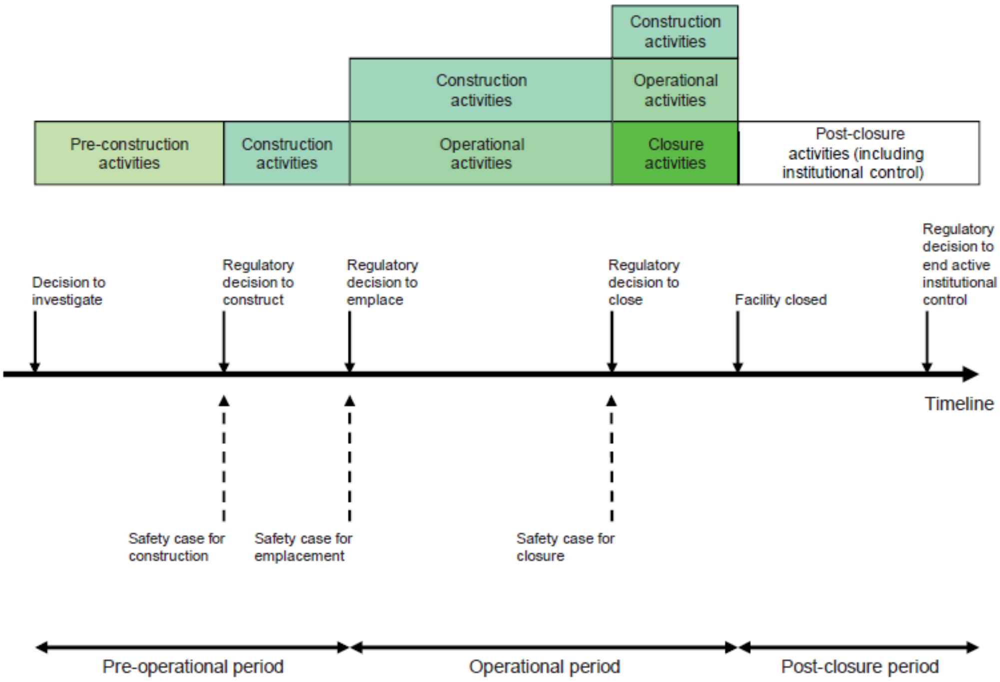


Source: ANDRA

Although there is no need for deep geological repositories for the disposal of low level waste, some countries such as Sweden and Finland are disposing their low level radioactive waste in low and intermediate level waste disposal facilities located between 60 and 100 m below ground level, and Germany will use mines at depths of several hundred meters to dispose of radioactive waste regardless of its classification. There are other countries that operate disposal facilities at different depths.

The lifecycle of a near surface disposal facility expands over several decades, and can be divided into pre-operational, operational (including closure), and post-closure phases (or periods). The pre-operation phase includes siting, design and construction of the facility. During the operational phase the facility receives and emplaces the waste and finalises the closure. The post-operational phase comprises all monitoring and surveillance activities under institutional control after closure. The post-operational phase is relatively limited in time, usually to a maximum of a few hundred years (typically 300). Once it is concluded, the site can be released for free use. Figure 5.1-2 shows the typical stages in the lifecycle of a near surface disposal facility. These stages define decision making points compatible with the stepwise approach defined in Chapter 1 of part B of this report.

Figure 5.1-2. Typical stages in the lifecycle of a near surface disposal facility



Source: IAEA [5-6]

During the operational phase, facilities and buildings are available, in addition to the disposal area for the waste handling and emplacement activities. These include, among others: reception, inspection and buffer storage of the waste packages; waste conditioning facilities; laboratories for quality assurance, radioactivity monitoring, and tests; security; medical service; warehouses; administration; radiation protection; heating, ventilation and air conditioning; water treatment plant; electricity supply; diesel generator.

In the disposal area, the containment and isolation of the waste from the accessible biosphere is based on a multi-barrier concept. Barriers are natural or engineered structures that aim at containing the radionuclides of the radioactive waste and delay and block their migration, should they be released. Defence in depth in near surface disposal facilities is achieved ensuring that safety functions are not dependent on a single feature or a single element.

Surface or near surface disposal facilities typically include three barriers[138].

The first barrier is constituted by the waste package itself, in particular the matrix that retains the radionuclides and the additional layers of materials that overpack them in a form suitable for handling and disposal. The waste package is designed to prevent the release of radionuclides beyond a certain threshold. Waste package specifications or waste acceptance criteria are requirements (limits) to radiological, physical and chemical properties of the waste that can be disposed of in a specific facility. The waste acceptance criteria are defined in such a way that, in combination with the performance of the other barriers, the release and migration of the radionuclides to the biosphere in the conditions of disposal will be strongly limited, ensuring the compliance with the regulatory limits.

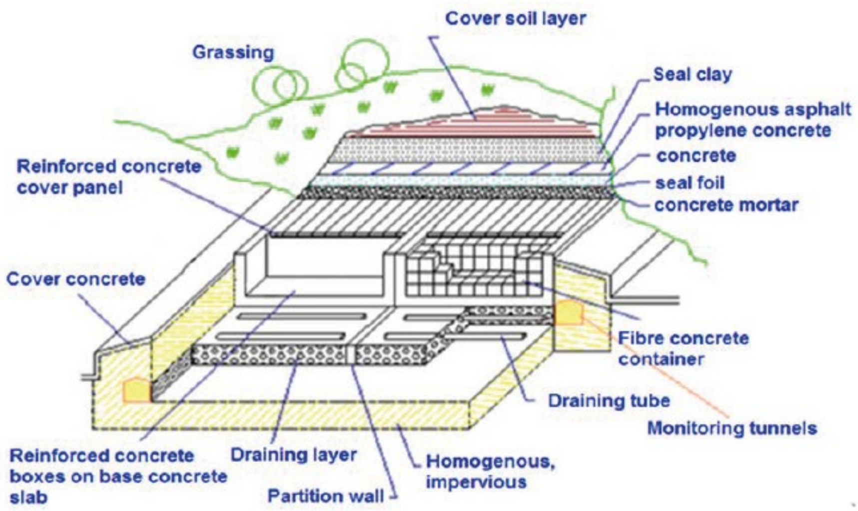
The second barrier is an engineered barrier designed to limit the amount of water that can access the waste packages, as water is the main vehicle for the mobilization and transport of radionuclides from the waste (by washing or lixiviation) to the biosphere. This barrier is constituted by the concrete disposal vaults or cells, the

[138] The details of the description correspond to disposal facilities similar to El Cabril in Spain, L'Aube in France or Mochovce in Slovakia

grouts and backfills, the leakage collection and monitoring system, and the final cover, or, depending on the type of facility, the sealing of the tunnels, shafts, galleries and access routes. The final cover is only installed as part of the closure of the facility, which takes place at the end of the operational phase, up to several decades after the reception of the first waste. Some modular designs account for the partial installation of covers over disposal cells that have been already filled with waste packages. The design and characteristics of the cover must consider the erosion due to inclement weather, and prevent or strongly hamper human intrusion.

The third barrier is a natural barrier constituted by the geological medium in which the installation is sited. Its purpose is to delay the migration of the radioisotopes that might have been released from the disposal facility site. Figure 5.1-3 describes the layout and the barriers of the Slovak Mochovce disposal facility.

Figure 5.1-3. Design concept of the Mochovce (SK) disposal facility.



Source: IAEA [5-3]

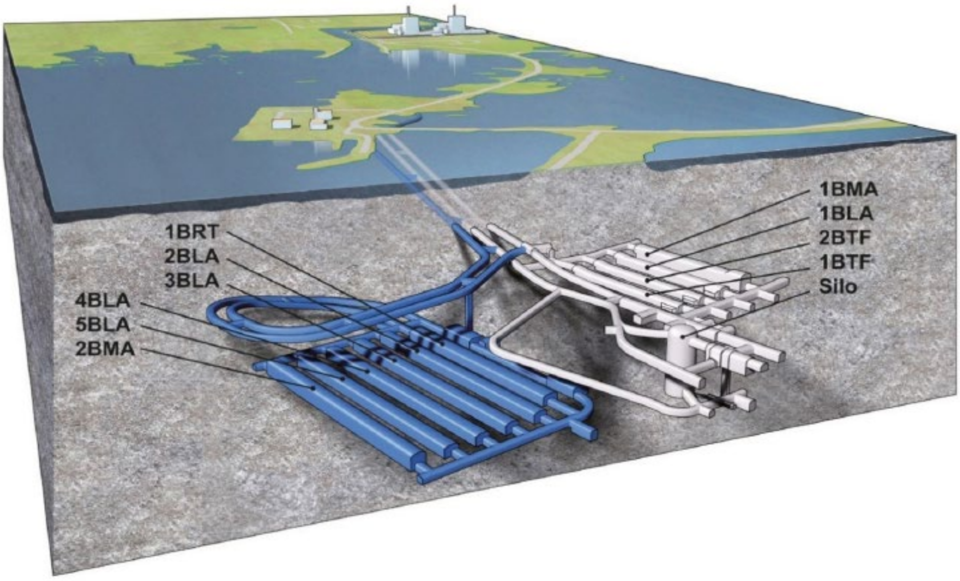
During operation, radioactive waste packages are received in the disposal facility, inspected and stored in the buffer storage. The waste packages are then placed and immobilised (grouted) in larger concrete containers. The concrete blocks that constitute the conditioned waste are piled up in the disposal cells. Once complete, the gaps are filled with gravel, and the cell is covered by a concrete lid manufactured on-site. Once all of the cells are filled and covered, the final cover made of alternating high and low permeability layers that protect the cells and the containers is installed. A temporary waterproof structure protects the open cell from weather-related occurrences during the filling operations. The temporary structure is mounted on rails, and once the final lid is placed it is moved to the next cell.

Safety and radiation protection during the operation of the facility are ensured by active and passive means in a similar way to other facilities that handle radioactive waste. During operation, the means available allow for a continuous radiological monitoring and surveillance of the performance of the disposal facility, allowing prompt response to any relevant occurrence, including potential radioactivity releases.

After the facility closure, a period of institutional control begins. Institutional control includes an active phase for knowledge preservation, prevention of human intrusion, and monitoring and surveillance to detect any potential degradation of the engineered barriers. During this phase implementing corrective measures, up to and including retrieval of the radioactive waste if necessary, is possible. A passive phase of institutional control is also implemented: it includes the archiving of the relevant information, and the installation of durable markings to try and prevent human intrusion. Institutional control monitoring and surveillance proactively supports the confidence in the effectiveness of the disposal facility to contain the waste and isolate it from the biosphere.

Safety after closure is achieved considering the reliability of the engineered barriers to contain the radioactive waste and isolate it from the biosphere. During the first 300 years, it must be proven that under any foreseeable circumstances, doses or risks are kept below established limits. In the case of near surface disposal facilities, the main routes that may cause radioactivity dose contributions to individuals or the public are the release of radionuclides from the waste form and their transport in water or as gases, the inadvertent exposure of individuals caused by intrusion, and potential disruptive natural or man-caused events. Figure 5.1-4 illustrates the layout of the low level waste underground repository in Sweden.

Figure 5.1-4. Low level waste underground disposal facility in Sweden.



Source: IAEA [5-3]

Specifically, the transport of radionuclides by water and release to the environment depends on the stability of the barriers and their resistance to water corrosion and penetration. Understanding the hydrogeology (for underground facilities) and climate (rain) effects at surface or near surface sites is a necessary requirement to implement a disposal facility. The safety case and analysis of potential accident scenarios must include thorough characterization and understanding of the following features of the disposal facility:

— chemical compatibility and interaction among the different materials present: waste and waste packages, immobilisation cement or concrete, engineered barriers, cover, and geological formation in which the disposal facility is sited;

— release rate of the radionuclide from the waste due to lixiviation;

— rate of transfer through the engineered and natural barriers;

— transport and accumulation in the environment, e.g. through the food chain,; the transport and dispersion of radionuclides by air must consider dust caused by human activity (e.g. civil works) at or near the disposal facility.

Human intrusion is postulated only after the institutional control period has ended. Inadvertent human intrusion scenarios consider for example excavation for natural resources or water, civil works foundations, agriculture or archaeology. Its likelihood can be reduced by installing adequate physical barriers and maintaining proper knowledge and records.

Disruptive events must consider extreme natural events such as earthquakes, floods, etc, or man-caused events, including malevolent ones.

After 300 years, it is assumed that the engineered barriers have fulfilled their containment function and are not required any longer. At this time, the radioactivity level of the short-lived species in the waste (half-lives shorter than about 30 years) has decayed to harmless levels. Safety is ensured in this respect also by the limitation on the amount of long-lived waste allowed in the disposal facility set by the waste acceptance criteria.

The safety demonstration includes all the assessments and evidence that proves that the near surface disposal facility will contain the radionuclides in the waste and isolate them from the accessible biosphere for as long as they remain hazardous. The doses or risks to the exposed individual will remain below the established limits ensuring that no significant harm is caused to humans and the environment.

The safety demonstration covers the operational and post-closure phases, and includes a description of the disposal system, identification of features, events and processes that may affect the performance of the facility in the longer term, identification and description of the scenarios (combination of future more or less plausible events) of the evolution of the facility, and the calculations and models used. The possible routes and rates for the migration of the radionuclides from the waste to the accessible biosphere, and their evolution with time are analysed in depth.

The definition of the scenarios includes more probable normal evolution scenarios, which consist in the extrapolation of current conditions assuming smooth changes on those conditions, as well as less probable scenarios in which disruptive events that can substantially change the "baseline" conditions occur. The safety demonstration must duly justify which scenarios are considered, and which ones are disregarded, based on realistic assumptions and avoiding purely speculative extreme scenarios that could unnecessarily penalize the safety case, and limit the amount of radioactive waste that can be disposed of in a given facility.

The future evolution and the behaviour of the disposal facility are determined by means of physical/chemical models and calculations that reflect the current state of knowledge on the behaviour of the materials that constitute the engineered barriers and of the natural media in which the disposal facility is sited.

The models consider the conditions for the mobilization and the release behaviour of the radionuclides within the waste form, the behaviour and the durability of the materials (concrete, cement, others) that constitute the engineered barriers, the transport and migration of released radioisotopes across the engineered and the natural barriers to the accessible biosphere, and the radiological consequences of such occurrences. Models also determine the effect of different abnormal scenarios on the behaviour of the radionuclides, considering for instance phenomena that might accelerate the release and migration of radionuclides to the environment (e.g. anomalous degradation of barriers, extreme weather conditions, seismic/volcanic activity, human intrusion, etc). The models are based on scientific evidence and have been validated using research results and data on the actual behaviour of operating and closed disposal facilities.

Uncertainties as well as sensitivity analysis contribute to the confidence in the results, and to the robustness of the safety demonstration. To this end, sensitivity analysis must demonstrate that small changes in the parameters of the models do not cause large impacts in safety.

Disposal of low level waste in near surface facilities is an industrial reality. Disposal facilities for radioactive waste generated in the nuclear fuel cycle have been constructed and have been operating for many years in many countries such as (list is not exhaustive) Bulgaria, Czech Republic, Finland, France, Germany, Hungary, Japan, Norway, Russian Federation, Romania, Slovakia (see figure 5.1-3), Spain (see Figure 5.1-5), Sweden, the USA and the UK. Some facilities in these countries have been in operation for several decades, and a few, the majority of which are used to dispose of institutional waste, have been closed and entered the institutional control phase.

Figure 5.1-5. El Cabril near surface disposal facility, Spain.

Source: IAEA [5-3]

Safety of operating facilities is re-evaluated periodically, typically every 10 years, taking into account accumulated domestic and international operating experience, current conditions, new developments in technology, outcomes from research and development, new regulations and social aspects. The outcomes of the periodic safety review are used to further improve safety and optimize operations.

In general, the mechanisms and processes put in place for the disposal of radioactive waste in near surface facilities is robust and allows in practice for the identification of non-safe situations and provides for the improvement of the safety of the disposal.

There are some cases in which the safety (re)assessment of disposal facilities indicated challenging conditions and resulted in the decision to recondition part of their radioactive waste and dispose of it in the same or in another facility. An example of this is the Asse II, a rock salt mine in Germany that was used to dispose of low and intermediate level waste between 1967 and 1978. Since the mine revealed safety issues, it was decided to retrieve the waste and dispose of it in a different facility [5-5].

In other cases, actions have to be undertaken in response to evolving requirements for the disposal of radioactive waste, in particular becoming more stringent. The low and intermediate level waste disposal Centre de la Manche in France was in operation between 1969 and 1994, and entered the institutional control period in 2003. Waste acceptance criteria evolved over this period, and some of the waste had to be retrieved, reconditioned and immobilised in concrete cells.

More information and other examples of upgrading near surface disposal facilities can be found in [5-7]

5.2 Deep geological disposal of spent fuel and high level waste

5.2.1 Basic principles for geological disposal

The very long-term management of spent nuclear fuel and other high level waste forms should be based on two ethical principles [5-16]: inter- and intragenerational equity. Intergenerational equity is the responsibility of the current generation to minimize risk and burdens for future generations. Intragenerational equity concerns the balance of resource allocations and the involvement of various sections of the society in a fair and open decision-making process.

In concrete terms, the intergenerational equity entails:

- choosing technologies and strategies which minimise the resource requirements, cost and risk burdens passed on to future generations;
- not unduly restricting the freedom of choice of future generations.

Intragenerational equity entails:

— implementation through an incremental process deployed over several decades, factoring in the results of scientific and technological progress;

— public involvement in the decision-making process, in particular local communities directly affected.

There is consensus in the scientific and regulatory communities that geological disposal is the preferred solution for the long term management of spent nuclear fuel and other high-level long-lived radioactive waste forms, including high-level waste resulting from closed fuel cycle scenarios [5-14, 5-15, 5-17 5-22] The objective of disposal is to isolate the waste from the biosphere for extremely long periods of time, and ensure that the dose to the public caused by any residual radioactive substance reaching the biosphere at any time will be below the maximum level set by the relevant regulations (typically at concentrations that are orders of magnitude lower than, for example, the natural background levels of radioactivity). Fulfilling this requirement includes providing reasonable assurance that any risk from inadvertent human intrusion would be very small [5-16].

5.2.2 Deep geological repository design principles

Disposal facilities are designed to ensure both operational safety and post-closure safety. The operational safety of geological disposal facilities is provided by means of engineered systems and operational controls; the post-closure safety is provided by means of multiple engineered and natural barriers. Disposal facilities are designed to be passively safe after closure, even though some monitoring and institutional control might continue [5-66].

The International Atomic Energy Agency (IAEA) Radioactive Waste Management Glossary [5-67] defines a barrier as “A physical obstruction that prevents or delays the movement (e.g. migration) of radionuclides or other material between components in a system, e.g. a waste repository. In general a barrier can be an engineered barrier which is constructed or a natural barrier which is inherent to the environment of the repository.” A definition of the multi-barrier concept is provided e.g. by the Swiss Federal Nuclear Safety Inspectorate, as: “A series of different engineered and natural barriers which will prevent and delay the movement of radionuclides contained in the waste, in order to ensure the safety of a repository” [5-68]. Each barrier (including e.g. waste form, canister, buffer, geologic media) contributes to safety providing some degree of redundancy with respect to isolation and/or retention of radionuclides [5-69].

The systems, referred to as Deep Geological Repositories (DGR), are always based on the multi-barrier principle including a combination of engineered and natural barriers. Although a high degree of independence and redundancy among the different barriers is envisaged when designing the repository, repository barriers are not fully independent and redundant, as in the case of the “defence-in-depth” approach used in nuclear reactor safety, but rather act in a complementary manner. Chemical and mechanical interactions between natural and engineered barriers will occur [5-69].

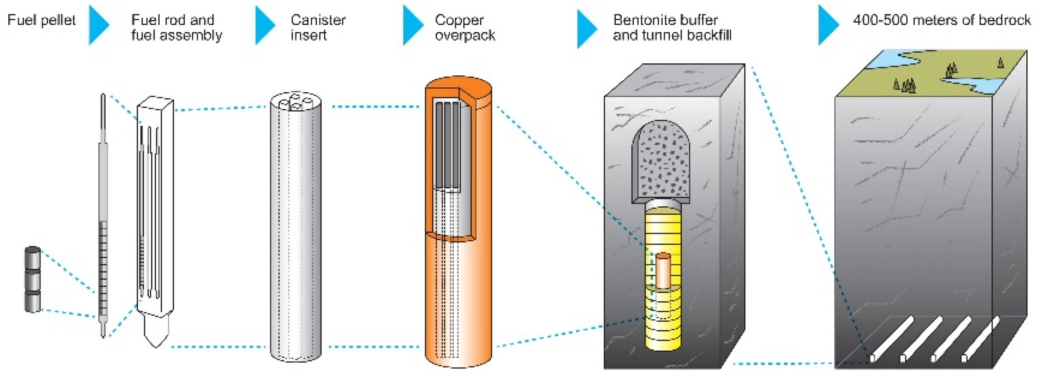
In concrete terms the layout of a DGR consists of properly designed waste packages emplaced in an Engineered Barrier System (EBS), within excavated or drilled openings, located at a depth of some hundred metres, in a stable geological environment [5-14, 5-17]. The time-scales to consider are beyond what is considered for any other engineered structure. For instance, it takes several hundred thousand years until the radioactivity of spent nuclear fuel has decayed to the same level as natural uranium [5-37].

In addition to engineered and natural barriers, the safety assessment of DGR (see Chapter *5.2.3.2*) also refers to “near-field” and “far-field”. The near-field is the excavated area of a disposal facility near or in contact with the waste packages, including the filling or sealing materials, and those parts of the host medium/rock whose characteristics have been or could be altered by the disposal facility or its contents.

The far-field is the geosphere outside a disposal facility, comprising the surrounding geological strata, at a distance from the disposal facility such that, for modelling purposes, the disposal facility may be considered a single entity and the effects of individual waste packages are not distinguished [5-12].

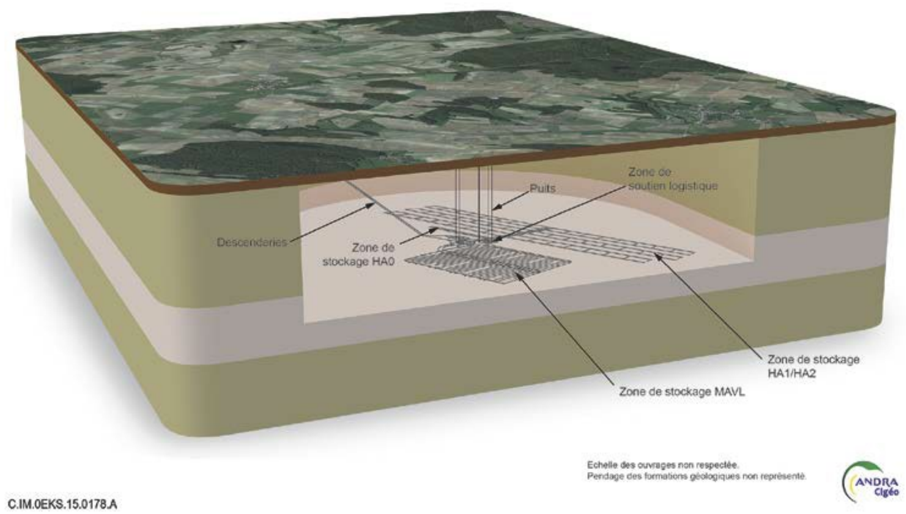
The availability of a suitable host rock with long-term geological stability is the basis for any DGR. The two most common types of host rocks are crystalline rocks, e.g. granite and gneiss, or argillaceous formations, i.e. strongly consolidated clays. Other host rocks considered include rock salt and volcanic tuffs [5-14]. Since the different barriers are complementary, the host rock also determines the required properties of the engineered part of the DGR. Argillaceous formations have a particular thickness and are only available at a specific depth, which stipulates the emplacement depth. Crystalline rocks are generally available from the surface to very significant depths, and in this case, the waste is emplaced at a sufficient depth to ensure low groundwater content and flow and very stable geochemical and hydro-chemical conditions. Crystalline rocks contain, on a very large scale, fracture zones that may form conduits for water flow and solute transport by convection. The waste packages and the other engineered barriers therefore constitute key components against release of radionuclides, and the hard rock retards the migration of radionuclides released from the engineered barrier system. In clay there are no fractures and the diffusivity is extremely low, so the consolidated clay itself is the main barrier and there is somewhat less emphasis on the waste package. Figures 5.2.2-1 and 5.2.2-2 illustrate the KBS-3 concept which is developed in Sweden and Finland for crystalline rock and the French DGR concept for argillaceous rock, respectively. Figure 5.2.2-3 shows the layout of the Finnish DGR for spent fuel under construction in Onkalo.

Figure 5.2.2-1. KBS-3 system with spent fuel assemblies in a copper/cast iron canister, surrounded by bentonite buffer and backfill in crystalline rock



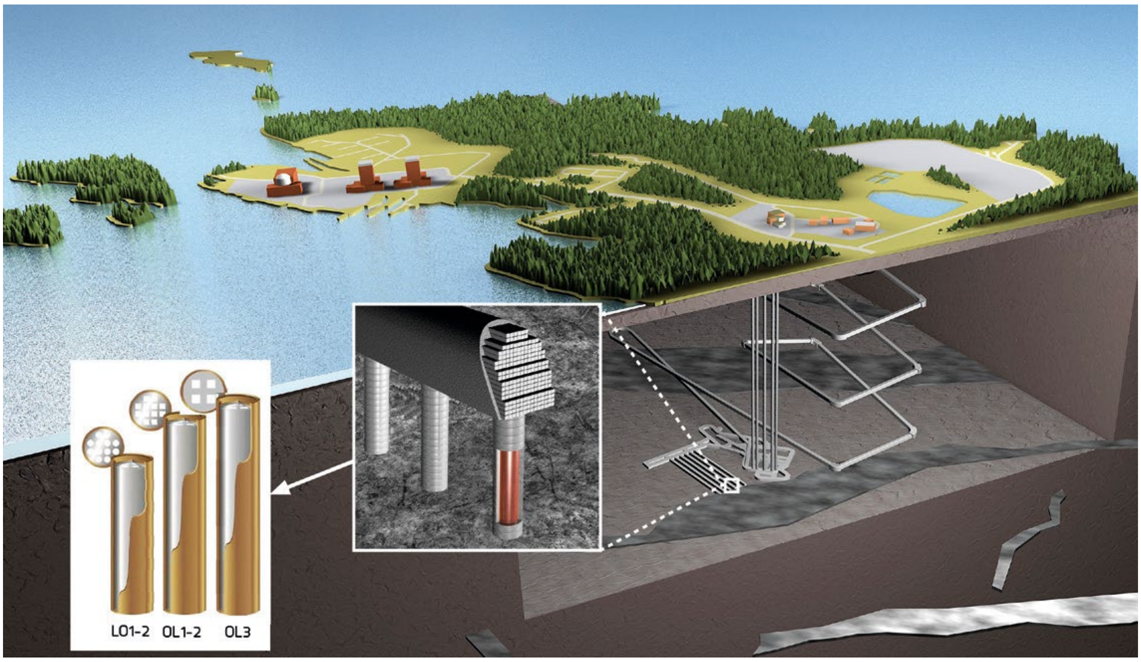
Source: SKB

Figure 5.2.2-2. Schematic layout of Cigéo, the French DGR repository for high-level and long-term intermediate level waste located in argillaceous rock.



Source: ANDRA

Figure 5.2.2-3. The Onkalo spent nuclear fuel repository (Finland)

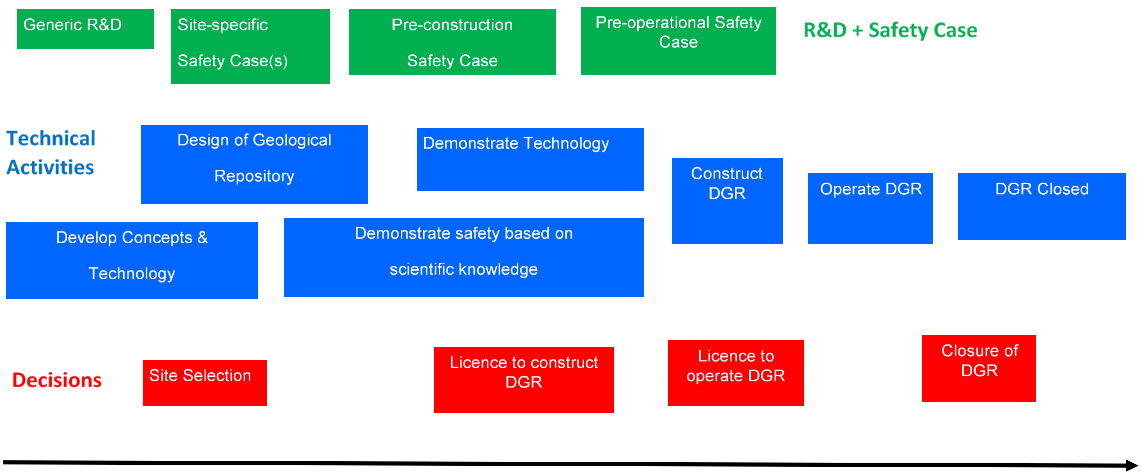


Source: Posiva

5.2.3 The national deep geological repository projects

Figure 5.2.3-1 outlines the different phases of a national DGR project in chronological order from the first planning to its closure. R&D and the build-up of the safety case, technical activities, and decision stages are illustrated in green, blue and red respectively. The process in all phases also involves interaction between the waste management organizations, the regulators and the general public, in line with the intragenerational equity principles, and as stipulated in the Radioactive Waste Directive [5-13]. The specific phases and requirements may differ in different countries as they are subject to licensing in accordance with the national legislation. Figure 5.2.3-2 shows an overview of the state of advancement for the development, design and construction of Deep Geological Repositories in Europe [5-17]. International collaboration through OECD/NEA [5-22] to [5-33], European collaboration through Euratom projects, the technology platform IGD-TP (https://igdtp.eu/), [5-17] and to a somewhat lesser extent IAEA, and more recently the joint programme EURAD (https://www.ejp-eurad.eu) [5-19] and [5-20] have been very important agents for accelerating the development through sharing information, consensus building and joint projects (see also Chapter 6.3).

Figure 5.2.3-1. Main phases in the implementation of geological disposal.

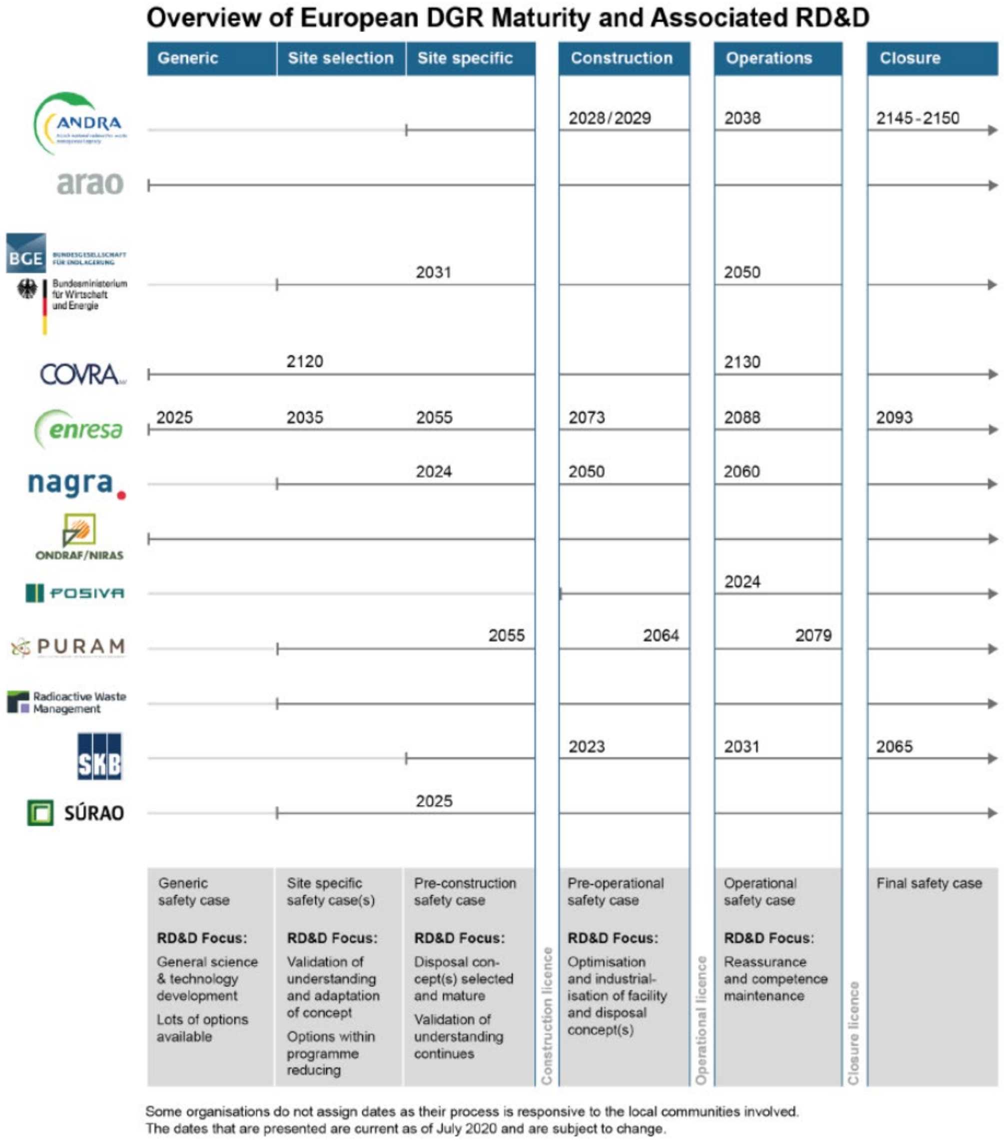


Source: JRC (2021)

The role of the different stakeholders will be described first. This will be followed by a description of concepts central to the implementation phases of a DGR and then by a general overview of the implementation phases outlined in Figure 5.2.3-1. The three most mature projects in Europe, namely Sweden and Finland, based on crystalline rock, and France, based on argillaceous formations, will be described in some more detail.

Figure 5.2.3-2. Deep geological repository milestone table indicating the approximate current and future stage of facility implementation for selected European WMOs.

The stages are indicative and it is likely that there will be a degree of overlap between activities in each stage, particularly for construction and operations (which will be undertaken in parallel).



Source: ANDRA/France ; ARAO/Slovenia ; BGE/Germany; COVRA/Netherlands ; ENRESA/Spain; NAGRA/Switzerland; ONDRAF/NIRAS/Belgium; POSIVA/Finland; PURAM/Hungary; RWM/UK; SúRAO/Czechia; SKB/Sweden

5.2.3.1 Role of the stakeholders

The stakeholder is defined as any actor - institution, group or individual - who has an interest or role to play in the radioactive waste management processes [5-12], [5-15], [5-27], [5-31], and [5-32].

Key stakeholders, and their roles include [5-31]:

- Policy-makers: define the policy options and assess their consequences; set the ground for the decision making process; inform and consult the stakeholders in the policy decision process.
- Waste generators: provide financing and (depending on the national configuration of the nuclear programmes) establish a Waste Management Organization to implement radioactive waste management solutions.
- Waste Management Organizations: propose safety options and radioactive waste management solutions and investigate them under different assumptions; commission and acquire necessary R&D and technological knowledge to develop and implement solutions; interact with local communities and regions; interact with policy-makers and regulators.
- Regulators: define regulatory requirements and guidance; define a regulatory process and make choices regarding regulatory options; review the implementer's safety options and design and ask for possible complements or modifications; verify the compliance of operation with relevant criteria and guidelines; communicate the bases of regulatory decisions; serve as a source of information and expert views for other stakeholders.
- Potential host communities: accept or reject the proposed facility; interact with regulators and policy-makers; negotiate with waste management organizations for the benefit of the local community.
- Scientific experts and research organizations: carry out scientific/technical investigations subject to peer review process; advise institutional bodies such as regulators and waste management organizations; act as technical intermediators providing scientific evidence for discussion between general public and decision-makers; provide balanced and qualified input for all stakeholders.

The role and interactions of the different stakeholders must be clearly defined, in particular between policy-makers, regulator and waste management organization. All the stakeholders need to be involved during the entire process, which must be characterized by transparency, trust and confidence building through open dialogue among the stakeholders, and in particular with the general public and the decision-makers. The documentation, observations and the field and laboratory studies that support a repository safety case are likely to be both massive and unintelligible to a non-expert stakeholder, and to the average member of the public. Yet it is only through a broad consensus of all stakeholders and the public that proposed repository will be accepted. The challenge is to communicate the case for safety in plain language, which accurately reflects the outcome of the scientific and technical studies, analyses and calculations [5-29].

5.2.3.2 Key concepts

The design and operation of deep geological repositories constitutes an unequalled scientific and technological challenge due to the geological time-scales and the complexity of processes that control the safety functions, which infers a number of uncertainties, "known unknowns" as well as "unknown unknowns". To this end, a number of approaches and key concepts form the basis and are absolutely crucial. There may be some variation in the definition of the concepts by different sources. In this report we adopt definitions primarily from the IAEA Safety Glossary 2018 [5-12], ICRP [5-15] and OECD-NEA [5-27].

Safety functions

A Deep Geological Repository needs to be designed for a number of fundamental, but complementary, safety functions at different times [5-12], [5-14], [5-17], and [5-25]

- Early containment: complete containment of short-lived and highly active radionuclides for some hundreds up to thousands of years, primarily within the engineered system, in particular, the waste packages ensure containment for these "early" stages after closure of the repository.
- Limitation of releases: limiting and retarding the rate and concentration of radionuclides that may be released over time from the engineered barrier system (EBS) to the geological media (far field). This is achieved by a combination of physical and chemical processes for which the most important task is to limit the access of groundwater to the wastes, and the transport of radionuclides released from the waste form from the repository through the geologic media to the biosphere. Clay formations, as the natural geological medium or as engineered buffer around the waste packages in crystalline rock, strongly contribute to fulfil this function. Release may also be minimized by inhibiting or limiting the corrosion of the waste form (e.g. by designing the near field configuration to achieve a beneficial combination of materials and local physicochemical properties), and by facilitating sorption or precipitation of released radionuclides onto surfaces in the EBS or in the geologic media.

— Dispersion of the flow of radionuclides in groundwater: for instance by three-dimensional fractures in crystalline rocks where the migration rate is very slow, and by overall dilution, to the extent that radionuclide concentration eventually reaching the biosphere is extremely low.

The degree to which the repository relies on each safety function depends largely on the host rock. Clay formations are characterized by extremely low diffusion, which limits water access, release and flow of radionuclides. In crystalline rock more emphasis is given to long-term containment provided by the design properties of the waste packages; the waste packages are surrounded by an engineered clay buffer to limit transport of radionuclides in case of release from the waste packages; additionally, the crystalline rock provides a very stable environment for retardation, dispersion and dilution.

Safety Case, Safety Assessment and Performance Assessment

The Safety Case (Figure 5.2.3-3) in the IAEA Glossary is defined concisely as "A collection of arguments and evidence in support of the safety of a facility or activity" [5-12], whereas the ICRP has the following definition: "A safety case is a structured set of arguments and evidence demonstrating the safety of a system. More specifically, a safety case aims to show that specific targets and criteria are met" [5-15]. OECD-NEA adopts a somewhat broader definition: "an integration of arguments and evidence that describe, quantify and substantiate the safety of the geological disposal facility and the associated level of confidence". A Safety Case also includes the compilation of underlying evidence, models, designs and methods that give confidence in the quality of the scientific and institutional processes as well as the resulting information and analyses that support safety [5-25].

The Safety Assessment is a systematic analysis of the hazards associated with a geological disposal facility and the ability of the site and designs to provide the safety functions and meet technical requirements. The task involves scenario analysis, model representation and developing an understanding of how, and under what circumstances, radionuclides might be released from a repository, how likely such releases are, and what would be the consequences of such releases to humans and the environment.

The results of the safety assessment – i.e. the calculated numerical results for safety indicators – are supplemented through the safety case by a broader range of evidence that gives context to the conclusions or provides complementary safety arguments, either quantitative or qualitative. The safety case and safety assessment can be described using flow charts, as illustrated in Figure 5.2.3-3, with the safety assessment as an integral part of the safety case. Such a flow chart typically includes: system description; modelling (process-level, system-level, data gathering); safety assessment (safety and performance indicators, timescales); treatment of uncertainties (classification, strategy, mathematical models); regulatory issues.

Performance assessment is closely related to the safety assessment, but it differs from safety assessment in that it can be applied to parts of an authorized facility (and its surroundings) and does not necessarily require the assessment of radiological impacts [5-12].

A key activity in the development of a repository safety analysis is the comprehensive identification of the potentially relevant factors, often termed "features, events and processes" (FEPs). The International FEP Database by OECD/NEA [5-28] provides a comprehensive collation of FEP information from performance assessments and scenario development studies from national and international projects.

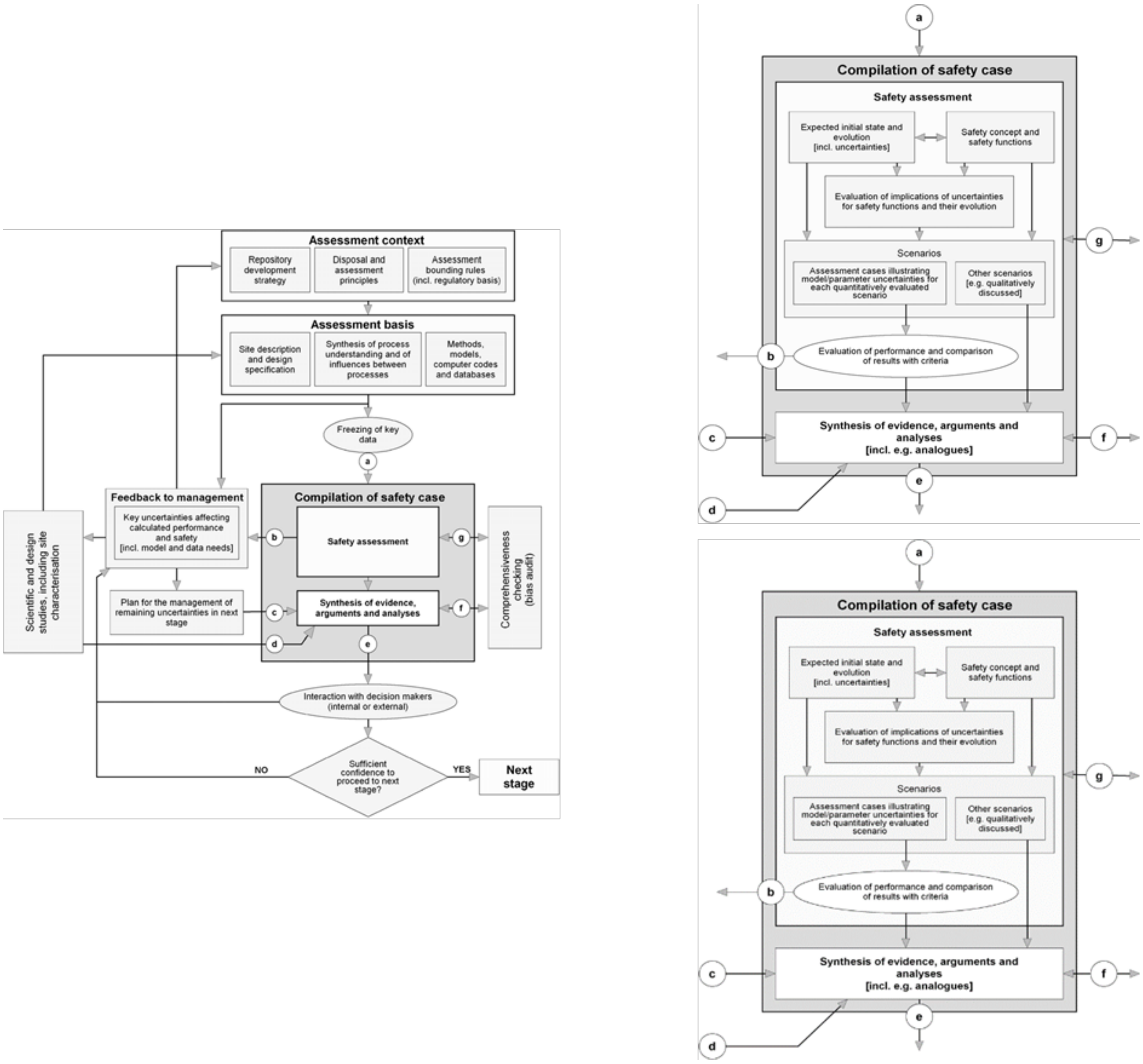
The safety case and associated safety assessment are established at the early phases and evolve as the implementation phases progress (Figure 5.2.3-1). The safety case and assessment are prerequisites for the licensing stages.

Treatment of uncertainty

Uncertainties are inherent in the safety assessment of deep geological repositories due the very long time frames and the complexity of the processes. The treatment of uncertainties therefore constitutes a main challenge to demonstrate the safety case. The uncertainties can be classified into three main categories [5-25]: 1) scenario uncertainties; 2) model uncertainties and 3) data and parameter uncertainties. Another classification of uncertainties is epistemic (lack of knowledge) or random. Typically model uncertainties are epistemic, while scenario and parameter uncertainty may be both, epistemic and random.

Scenario uncertainties, which need to be addressed, refer to changes that a DGR can be exposed to via scenarios that challenge the safety case. This includes for instance climate changes, seismic/volcanic events as well as societal changes and possible human intrusion.

Figure 5.2.3-3. Left: Example of a high-level generic safety case flowchart, showing the key elements and linkages. Right: Detailed generic flowchart of the safety assessment component, which is included in the compilation of a safety case of the upper level generic flowchart.



Source: [5-25]

Model uncertainties result from incomplete knowledge and understanding of the processes that control the safety functions of the DGR and how well the assumptions and assessment models in safety or performance assessment describe the actual processes.

Data and parameter uncertainties are linked to the model parameters that may be incomplete, not measured accurately, random or simply not available.

There are different approaches for addressing these uncertainties:

— selecting host rocks with demonstrated stabilities for the time-scales needed to isolate the waste and technical solutions that require no monitoring and further actions in the post – closure phase;

— investigating and assessing natural analogues, typically for long-term scenarios for which there is not data — examples are retention of fission products and actinides in geological clay formations at natural nuclear reactor sites, e.g. in Gabon and Cigar lake [5-34]. An important additional contribution to long-term safety assessments by natural analogue studies is to provide qualitative information on which processes and features to include in the assessments;

— adopting conservative assumptions;

- demonstrating robustness by confirming compliance with the safety requirements in *what-if* scenarios, which may include extreme cases, e.g. assuming all waste packages failed at once;
- applying mathematical models such as probabilistic or sensitivity analysis, typically addressing data, scenario and model uncertainties; desirably adopting a full probabilistic approach;
- demonstrating that uncertain processes can be ruled out on the basis of extremely low probability, e.g. seismic events in stable regions;
- demonstrating that a specific uncertainty is irrelevant for the safety assessment.

Events that are likely at a very long-term scale are included as the reference scenario. This includes for instance future ice-ages for Sweden and Finland: in particular, the weight and associated pressure due to the ice and the change in the geological formations resulting from the ice-age, such as shear movements, are taken into account in the assessment [5-37], [5-38], and [5-47].

Indicators and acceptance criteria

Assessing the safety performance of a DGR requires quantifiable indicators with associated acceptance values. Most national regulations stipulate safety criteria in terms of dose and/or risk which must be evaluated for all relevant scenarios affecting the specific repository. Dose and risk are not straightforward to quantify: for instance, dose determination requires, in addition to data on radionuclide migration and concentrations in the biosphere, assumptions on the behaviour of the individuals in a far future (see e.g. European Atlas of Natural Radiation, JRC [5-70]). The description and presentation of the indicators are also very important when communicating with non-technical audiences.

An adequate set of indicators is needed to assess the overall system performance, reflecting the multiple lines of evidence used in the safety case and the associated safety and performance assessments. To this end three groups of indicators can be identified: *safety indicators*; *performance indicators* and *safety function indicators* [5-25].

Safety indicators reflect mainly the radiological aspects and include: annual effective dose [Sv/a]; radiotoxicity concentration in the biosphere water [Sv/m^3]; relevant radiotoxicity flow through the geosphere into the biosphere [Sv/a]. The radiotoxicity transfer from the geosphere to the biosphere groundwater, and the resulting additional radiological harm from drinking water should remain significantly below natural radiotoxicity levels; the annual dose for an individual should remain far below the limits set by the regulations.

Performance indicators are useful for assessing the level of safety of the total system and how such level is attained. Performance indicators are quantities that can be calculated and describe the performance of system components and in particular their interaction. Examples are concentrations or flows of radionuclides within or between components, needed for instance to compute the release through waste package, buffer, backfill of the repository tunnels in the near field, and, in the far field, the migration of radionuclides through the geologic media to the biosphere for selected scenarios.

The *safety function indicators* refer to the specific role of the safety function and the specific component of the repository system. A safety function indicator is defined by SKB as a measurable or calculable quantity that quantitatively characterises the extent to which the safety function under consideration is fulfilled [5-35]. The purpose of the safety function indicators is to assign quantitative limits to specific safety functions that can be measured or calculated. Examples are peak temperatures in the buffer or swelling pressure affecting the integrity of the copper canister in the KBS-3 system.

5.2.3.3 Implementation phases

The development and implementation of a national programme for a deep geological repository as outlined in Figure 5.2.3-1 is likely to take several decades from the start until operation (see also JRC-EASAC report, [5-18]); the operational stage may entail a duration ranging from decades to more than a century as indicated in Figure 5.2.3-2 for the different European national programmes. The specific project depends on the waste inventory, the available host rock, the national regulations; it can also be affected by the experience shared by the "front runners" in case of countries with longer implementation timeframes. A common denominator for any European national programme is that it complies with the international conventions [5-10] and the Radioactive Waste Directive [5-13], it is adaptive and stepwise, and it includes public engagement. In particular, the following stages can be identified.

Stage 0: typology and inventory of the waste to be disposed of (see Chapter 2.3 of part B of this report). This depends on the size of the nuclear programme and whether a closed or open fuel cycle has been adopted. This affects the size of the DGR.

Stage 1: Identification of available host rocks and decision on DGR design. Construction of underground research laboratories in representative host rocks [5-33]. The research and development is generic and different options are open.

Stage 2: Selection of site; the research and development and data collection become site specific and design options are reduced. Note that site selection is not only based on the best geological formation; local acceptance and absence of highly valuable natural resources, which would e.g. increase the risk for human intrusion, or affect the environmental impact of the new facility, also need to be taken into account (see Chapter 3.3.8 of part A of this report). A site specific safety case is then developed in preparation for licence application to construct the DGR and auxiliary facilities such as encapsulation plants. The emphasis is on demonstrating safety based on the scientific knowledge and on the technical feasibility. The implementer should also present alternative approaches (see e.g. [5-39]).

Stage 3: Submission of licence application by waste management organization to the regulator based on a pre-construction safety case, and in compliance with national regulations. Construction licence awarded by the government. This may require revision of the safety case and design options depending on feedback from the regulator, expert peer reviews and other stakeholders.

Stage 4: Start construction of the repository and auxiliary facilities. Preparation of application to operate the facility based on pre-operational safety case and technology demonstration based on national regulation. Review of the application by the regulator and expert peer review. Post-operational safety must also be demonstrated. Licence to operate awarded by the government after all requirements are fulfilled.

Stage 5: Decision to start the disposal. Operational practices adjusted to account for experience feedback and the development of scientific knowledge and technology. Preparation for closure and post operational phase.

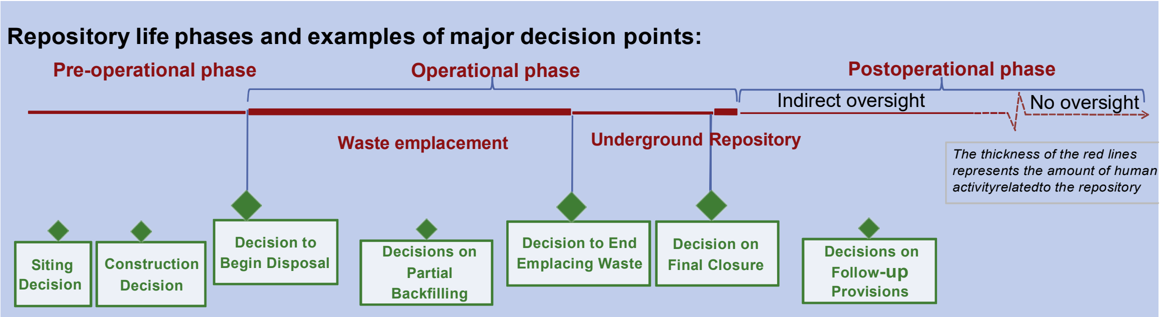
Stage 6: Decision on final closure and implementation of closure.

Stage 7: Post closure phase where monitoring and oversight is gradually phased out.

Following these stages should ensure inter- and intra-generational equity, and should achieve the main objective that the final disposal does not result in any significant harm for present or future generations. Figure 5.2.3-4, complementary to Figure 5.2.3-1, highlights the pre-operational (stage 0 – 4), operational (stage 5) and post-operational phases (stage 5-6). For the moment Finland is at Stage 4 whereas Sweden and France are at Stage 3. The process for these three countries will be summarized in the Chapter 5.2.4.

It should be emphasized that the process should be stepwise and reversible (the reversibility of the stages of the process establishing the DGR should not be confused with the retrievability of the waste emplaced in the repository, described below). At the early stage, when basic options (such as host rock selection) are kept open, reversibility could only mean revisit preliminary concepts and/or introduce adjustments factoring in cost updates, new data, new scientific knowledge or technical capabilities. As an example, horizontal rather than vertical emplacement of copper canisters in the repository is now being considered for the KBS-3 system [5-49]. Moreover, a DGR is a scientifically based engineered structure and should follow the "BAT" — Best Available Technology — and "ALARA" — As Low As Reasonably Achievable — principles with respect to radiological effects. Both the scientific understanding and the technology will evolve during the long process to construct a DGR and the national RD&D plans and the safety case must therefore be updated based on the development of the scientific knowledge and technological know-how, the increasing amount of data and the feedback from the experience and the maturity of the national project. The waste management organizations continuously leverage the research front, but as the project progresses more emphasis is given to the technical feasibility and concrete engineering tasks associated with implementation. As stated above the involvement of the different stakeholders, in particular the independent regulator, is important; the local community and region must also be supportive.

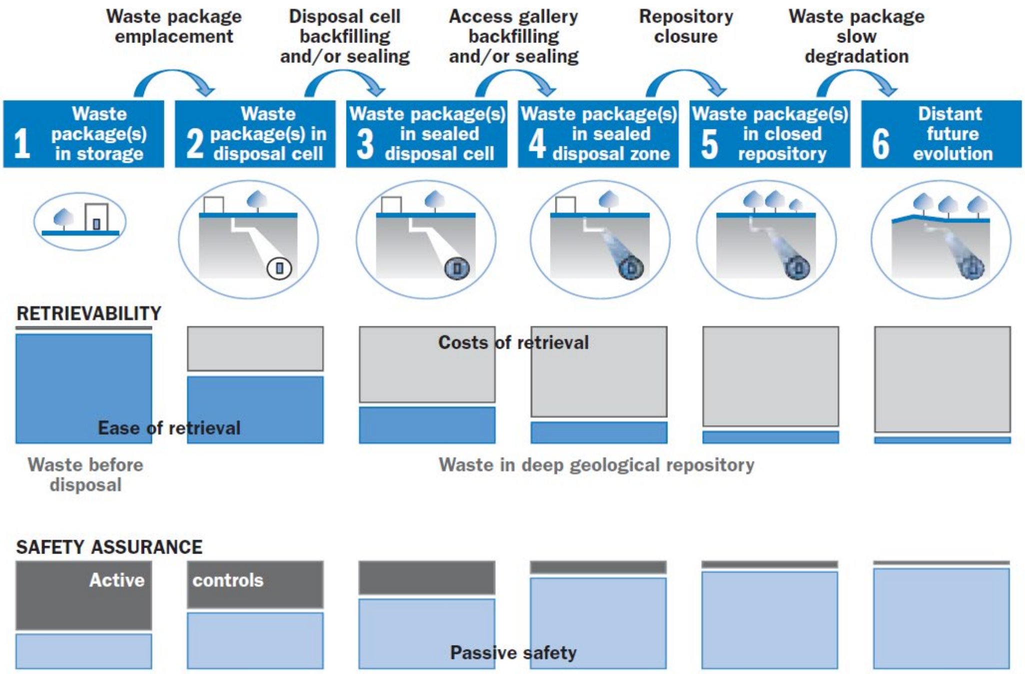
Figure 5.2.3-4. Disposal facility phases and relevant oversight periods.



Source: [5-15]

Once the repository is in its operational phase, the system can be designed for possibilities to retrieve waste packages that have already been emplaced in the repository. There are several reasons for considering waste retrievability options as the project advances. One reason for retrievability is that there could be superior solutions for waste management becoming available in the future due to science and technology advances. Another reason is that high level waste, in particular spent fuel, could be "burnt" to reduce the long-term toxicity of the final waste by transmutation [5-49]. Retrievability is a requirement in France [5-62] whereas in other countries, e.g. Sweden, feasibility of retrieval has been demonstrated, but it is not a requirement [5-37]. Postponing the closure of the repository to allow for retrievability introduces some additional cost and generates necessarily some safety compromises, in particular if the system is designed for no monitoring. Moreover, over time the feasibility of retrieval is reduced, in particular after closure and with phasing out of monitoring, as illustrated in Figure 5.2.3-5.

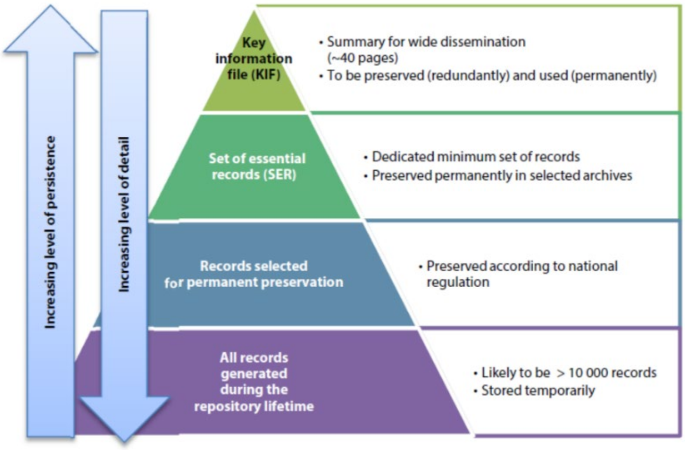
Figure 5.2.3-5. Evolution of the retrievability cost and "ease" during the operational and closure phases of a French DGR



Source: [5-31]

The radioactive waste needs to be kept isolated from the biosphere for very long times. Pursuant the intergenerational equity principle it must be a requirement that Records, Knowledge and Memory (RK&M) prepared as part of the DGR project are maintained after its closure in order to allow future generations to make informed decisions regarding the repository and its content, including to prevent inadvertent human intrusion [5-30]. The solutions depend on the time-frame and become more challenging the further we look into the future. In the short term, detailed records can be preserved essentially using today's technology, whereas for much longer time frames less detailed information can be kept, but using very stable methods [5-30]. Figure 5.2.3-6 illustrates such a hierarchical RK&M system.

Figure 5.2.3-6. Strategy RK&M across generations



Source: [5-30]

5.2.4 Implementation of national projects

As mentioned above the geological disposal must follow the same basic principles in all EU Member States. The technical solutions differ primarily due to the availability of host rock, but also to the waste inventory and the national regulations. The ultimate goal is always to demonstrate that the geological disposal does not impose any significant harm or risk to present and future generations.

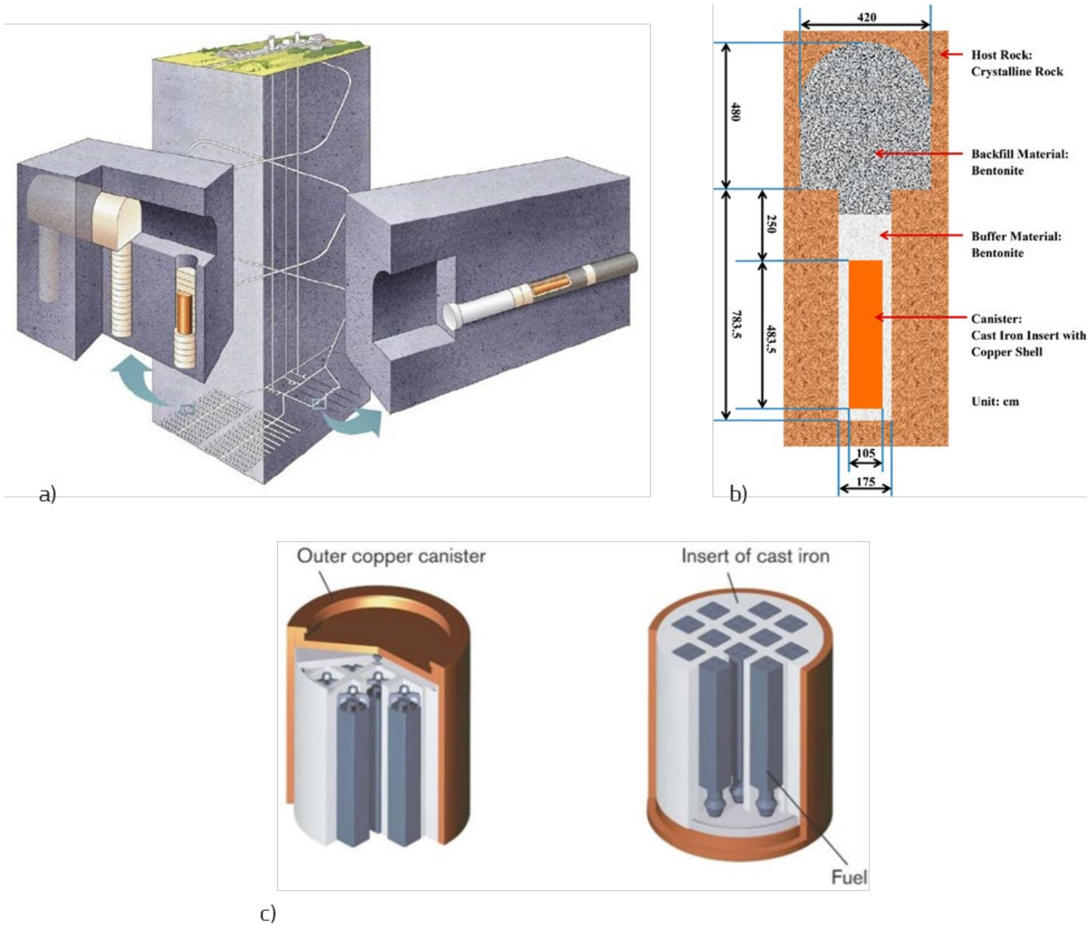
Crystalline Rock: KBS-3 project in Finland [5-42] to [5-54] & Sweden [5-35] to [5-41]

As seen in Figure 5.2.3-2 the two most advanced project for construction and operation of deep geological repository are in Sweden and Finland. Both are based on the KBS-3 concept that has been developed since the late seventies by the Swedish waste management organization SKB and the Finnish counterpart POSIVA. It was initiated in Sweden, but it is now largely a joint project [5-46]. The concept is illustrated in Figure 5.2.4-1. It is based on the multi- barrier concept and involves encapsulating spent nuclear fuel in copper canisters with a cast iron insert. The canisters are deposited, surrounded by a buffer of bentonite clay, in deposition holes in a tunnel system at a depth of about 500 metres in a crystalline bedrock. The bedrock should be geologically very stable; water conducting fractures and the groundwater flow at the deposition depth should be low. The long-term isolation of the spent nuclear fuel inside the canister is based on the cast iron to provide the strength and on the copper shell for the corrosion resistance. The buffer consists of tightly compact sodium bentonite clay that hinders groundwater access, holds the canister in place, provides damping and greatly retards radionuclide migration. The backfill material limits the advective transport in deposition tunnels and keeps the buffer in place. The bedrock itself should provide chemically favourable conditions, favourable hydrogeological and transport conditions, mechanically stable environment and favourable thermal conditions.

The decision process and implementation phases in Finland and Sweden are similar. Tables 5.2.4-1 and 5.2.4-2 summarize the timeline for the KBS-3 project in Finland and Sweden, respectively.

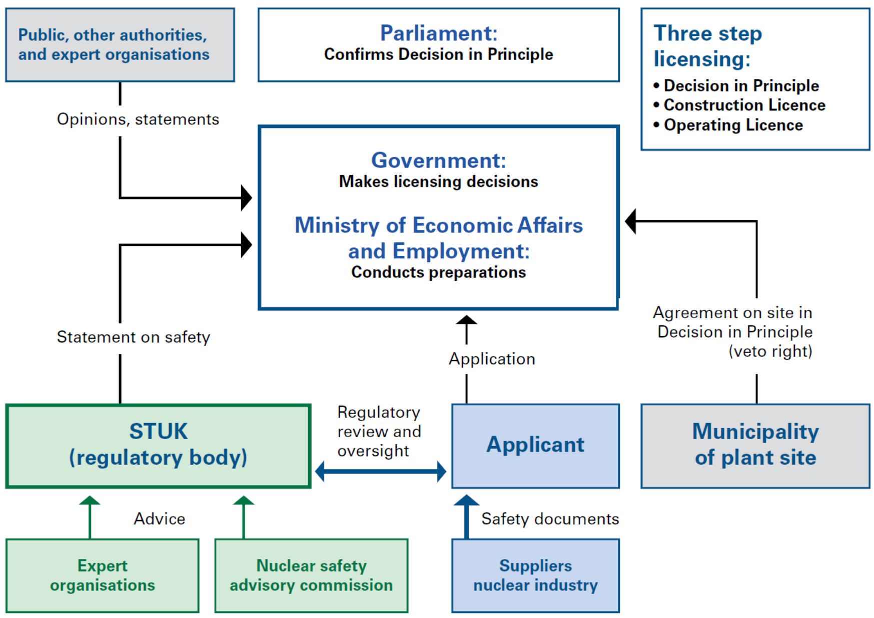
The legal basis for licensing the construction and operation of a DGR in both countries is regulated by nuclear activities acts and environment impact acts with the nuclear regulator SSM and Environmental Board (MMB) in Sweden [5-35], [5-39] and regulator STUK and Land and Ministry of Employment and the Economy in Finland [5-41], [5-52], [5-54]. The Finnish decision process and the role of different stakeholders are depicted in Figure 5.2.4-2.

Figure 5.2.4-1. Schematic illustration of the KBS-3 concept a) overall repository with vertical and horizontal emplacement b) Engineered Barrier System with canister, bentonite buffer and backfill material c) schematic layout of the copper canister and the cast iron insert with the spent fuel assemblies.



Source: SKB

Figure 5.2.4-2. Flowchart of the decision process in Finland illustrating the role of stakeholders



Source: [5-42]

Table 5.2.4-1. Chronology of DGR development and approval in Finland

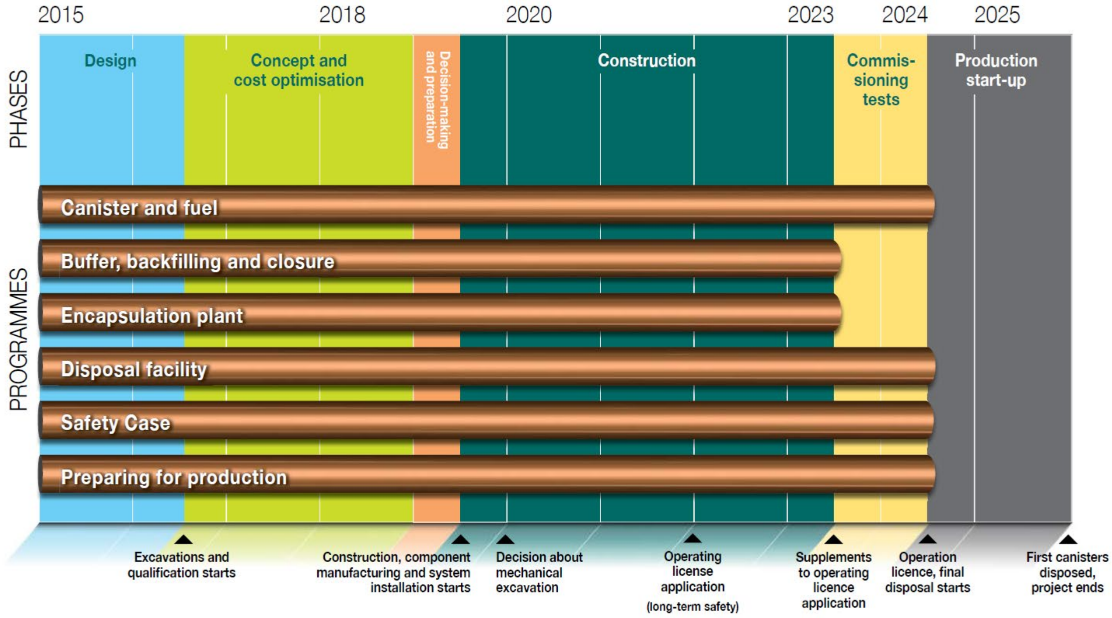
Year	Event
1983	Government Policy Decision for geological disposal
1987 – 1999	Site characterization
1999	Olkiluoto proposed and accepted (municipality & STUK)
2000- 2001	Decision-In Principle by the Government, Ratification by Parliament for SF disposal in Olkiluoto
2004	Start of construction of an underground rock characterization facility (ONKALO) in Olkiluoto in 2004
2008	Environmental Impact assessment submitted by POSIVA [5-52]
2012	Licence application construction disposal facility and encapsulation plan submitted by POSIVA [5-40] to [5-45]
2015	Safety Assessment by STUK [5-53] and Construction licence for the Encapsulation and Disposal Facility granted by Government in November 2015
2016	Licence application construction approved by STUK. Construction work of disposal facility started in December 2016
2019	Construction of the encapsulation plant started
2020	Excavation work started at DGR site
2021	Licence application for operation expected to be submitted
2024	Expected start of operation
2024-2120	Operational phase

Finland completed the site selection before Sweden and obtained the licence to construct the repository in 2016 whereas in Sweden the licence is expected to be granted in 2021. One reason for the faster process in Finland could be the approach to siting.

An advantage with the crystalline hard rock is that suitable geological conditions are widespread. Sweden therefore initially considered a large number of potential sites, finally reduced to two: Oskarshamn and Östhammar, which both host nuclear power plants. Both municipalities supported the DGR, and the decision in favour of Östhammar/Forsmark was based on more favourable geological conditions [5-39]. As a compensation, the encapsulation plant will be located in Oskarshamn.

Finland focused on the Olkiluoto site from the beginning. Detailed safety requirements on the management of spent nuclear fuel and radioactive waste resulting from the production of nuclear energy in Finland are provided in the YVL Guides [5-44], [5-45]. The licence application to operate the facility is expected to be submitted in 2021. This will require that all pending issues are adequately addressed; the application will have more focus on operation and post-closure and fully demonstrate feasibility. The focus of STUK's regulatory control will change from the overall safety case development to the demonstration of the disposal system processes and, in particular, the emplacement of the disposal canisters. The experiences from the review and assessment are available in [5-44]. Figure 5.2.4-3 shows the timeline for the various phases from 2015 until 2025 when the repository is expected to be fully operational.

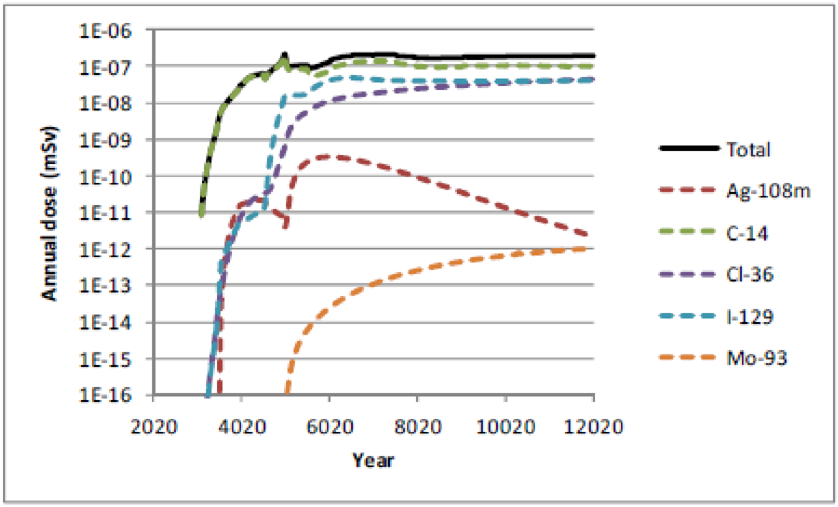
Figure 5.2.4-3. Timeline for the construction and operational phases of the DGR in Finland



Source: [5-42]

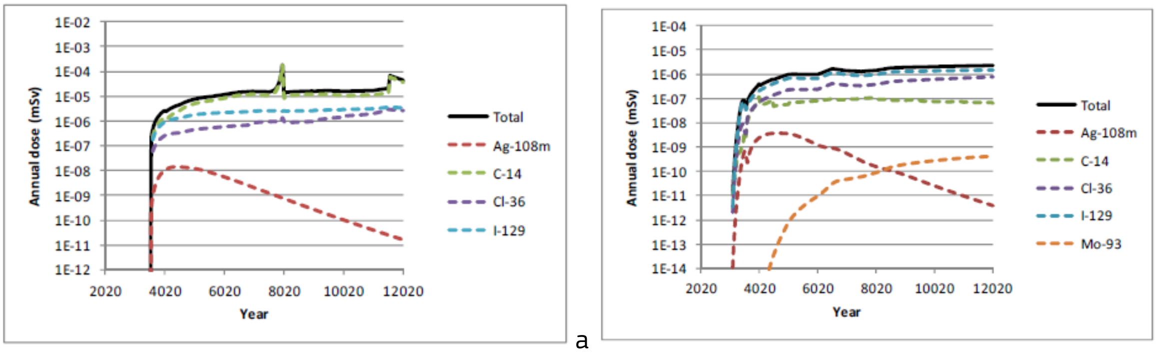
The results of the safety assessment can be expressed in terms of dose to humans (either most directly exposed or in other groups) as a function of time. In the Finnish case [5-72], the calculations extend to cover a time range up to 10 000 years. The Biosphere Safety Assessment - Reference Case (BSA-RC) is describing the base scenario; additional biosphere calculation cases are associated with variant scenarios and provide sensitivity assessment. The Reference Case and the sensitivity case calculations are used to demonstrate compliance with regulatory requirements (the maximum allowed annual dose may not exceed 0.1 mSv in the Finnish case). In particular, the variant scenarios aim also at investigating individual or combined uncertainties, in order to demonstrate the robustness of the compliance with the regulatory requirements. Disturbance scenarios are also considered. They are used to demonstrate robustness up to extreme situations, or *what-if* cases (either in the biosphere or in the repository). Due to their very low likelihood, the *what-if* cases do not necessarily have to show compliance with regulatory dose limits. Figure 5.2.4-4 shows the results of the reference case computation. Figures 5.2.4-5 and 5.2.4-6 illustrate sensitivity case calculations and *what-if* scenarios, respectively.

Figure 5.2.4-4. Yearly dose contribution to the most exposed group for the Reference Case of the Safety Assessment of the Finnish DGR



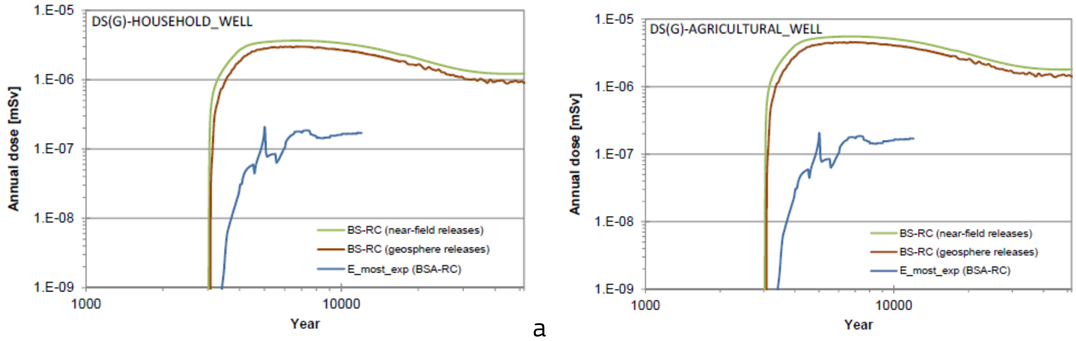
Source: [5-72]

Figure 5.2.4-5. Yearly dose contribution to the most exposed group from the Safety Assessment of the Finnish DGR: (a) variant case assuming container failure at a specific location in the repository; (b) case combining several "variances"



Source: [5-72]

Figure 5.2.4-6. Yearly dose contribution to the most exposed group from the Safety Assessment of the Finnish DGR. The dose contribution for the reference case is compared to disturbance scenarios: (a) *what-if* case assuming that a deep well for family drinking use is dug; (b) *what-if* case assuming that a deep well for multiple uses is dug.



Source: [5-72]

In Sweden the implementation follows an approach similar to the Finnish case, as summarized in Table 5.2.4-2. Since 1986 SKB has produced 12 R&D plans, the most recent one in 2019 [5-38], which has been implemented in a large number of scientific and technical studies to demonstrate the safety of the KBS-3 concepts as well as looking at alternatives in line with the regulatory requirements. For instance the first complete safety report (SR) for the SKB SR-Can project was in 2006 [5-36] and the more complete version of the SR-Site project was published in 2011 [5-37], in support of the licence application of 2011 for the construction of a repository in Forsmark and an encapsulation plant in Oskarshamn.

Figure 5.2.4-7 shows the flowchart used by SKB for SR-Site.

Figures 5.2.4-8 and 5.2.4-9 show the computed far field effective annual dose rate as a function of time caused by the repository for the reference scenario and for *what-if* scenarios, respectively.

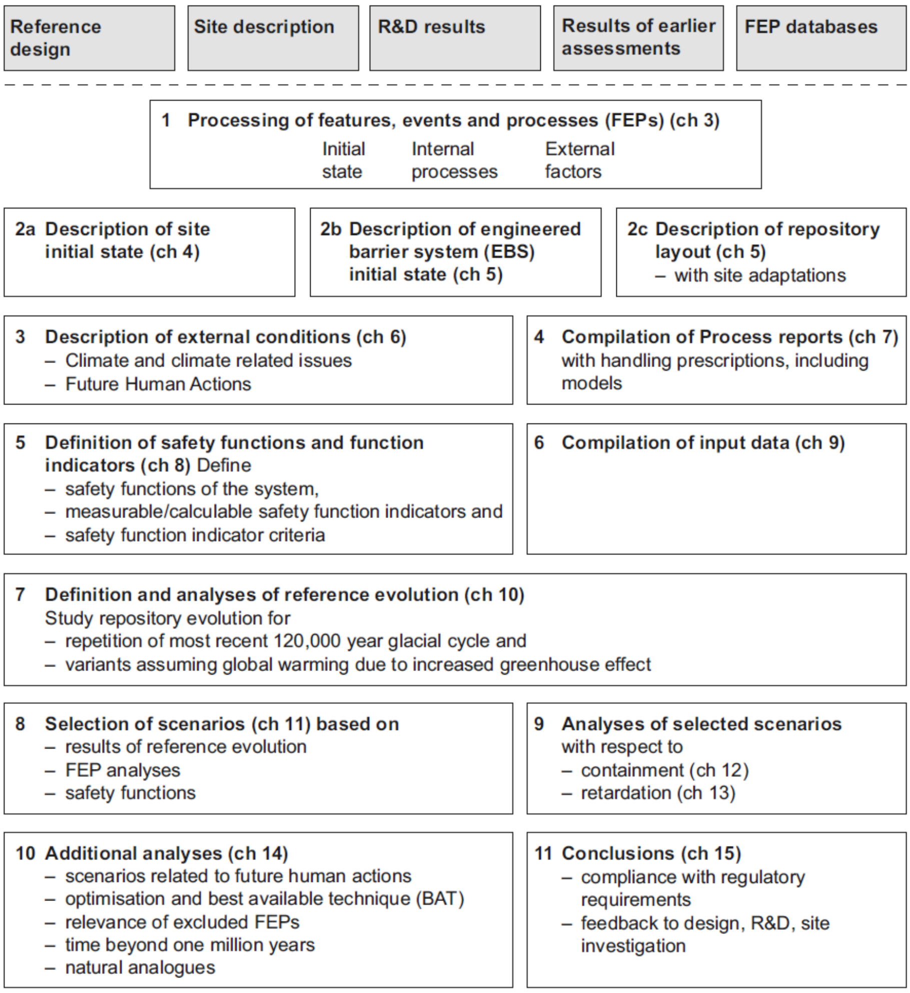
The dose values plotted on the diagrams can be compared with the maximum allowed dose to the general public due to artificial radioactivity of 1 mSv/year (the annual radiation dose an average Swede is exposed to due to natural background is ~4mSv/year), and with the maximum allowed effective dose for the general public caused by the repository of 0.014 mSv/year, which is approximately 2 orders of magnitude lower than the maximum allowed dose due to artificial radioactivity. The diagrams show that the dose caused by the repository will be well below the maximum allowed limit and hence will cause no significant harm to humans. Significant (and quite unrealistic) deviations from the repository design have to be postulated in order to obtain a dose to the exposed people exceeding the maximum allowed limit (Figure 5.2.4-9).

As an example, the licensing iterations in Sweden are mentioned here. The applications and supporting documents have been subjected to international peer review [5-24], thorough regulatory review by SSM and examination by MMB. SSM has requested and received a substantial amount of supplementary information on various topics. Most of the review was completed in 2015 and SSM submitted a positive statement in 2016. In 2018 SSM and the court (MMB) submitted their final reviews to SKB's licence application. SSM recommended approval of the licence request whereas the court required that SKB provides further evidence that long-term corrosion that could jeopardize the integrity of copper canisters could be ruled out. This information was submitted by SKB in 2019 [5-41] and reviewed by an expert committee under the auspice of SSM [5-40]. SSM, after a thorough technical review of the new material, reiterated its earlier statement that SKB's preferred site is suitable, the disposal concept is feasible and the safety case fulfils strict regulatory requirements [5-35]. The local council at Östhammar approved the repository. This, together with approval from SSM and MMB, allows the government to approve the licence so the construction of the repository and the encapsulation plants can be implemented.

Table 5.2.4-2. Chronology of DGR development and approval in Sweden

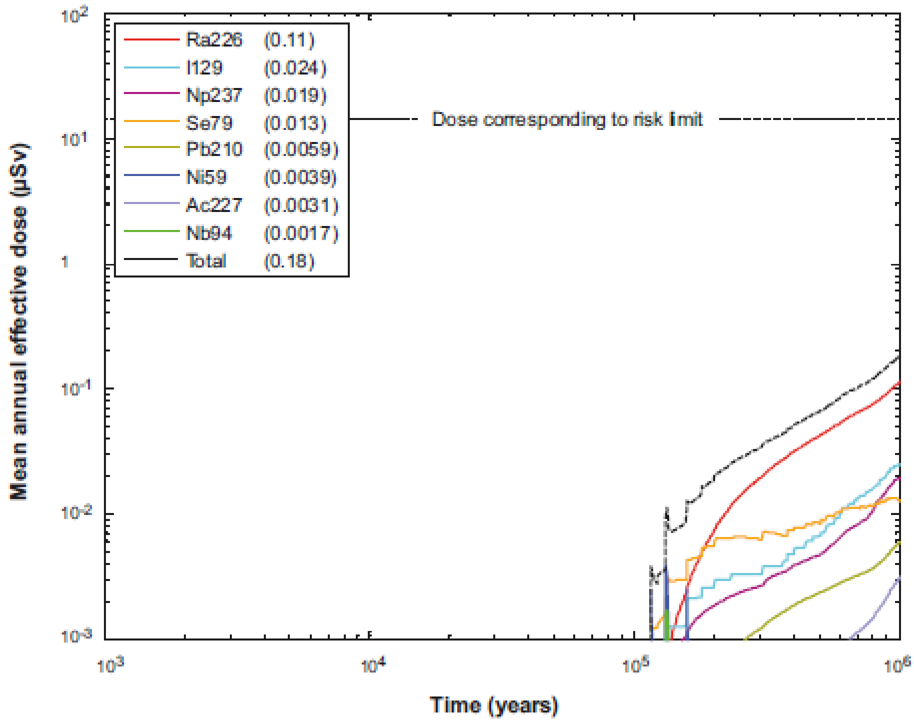
Year	Event
1983	Initial design KBS-3 repository
1990 -95	Construction of Äspö Underground Research Laboratory
2001	KBS-3 planning premise
2002	Oskarshamn and Östhammar selected as potential sites for repository
2006	First complete safety report SR-Can [5-36]
2009	Forsmark selected as site for final repository
2011	Complete safety report (SR-Site) [5-37] Licence application for encapsulation plant and disposal facility submitted by SKB to SSM and L&EC
2012-2015	Broad national consultation, NEA review [5-24], Review of licence application by SSM and supplementary information and documents including revision of preliminary safety analysis provided by SKB
2017-2018	Regulator SSM recommends approval of SKB's licence application under Nuclear Activities act, MMB positive but request supplemental documentation on copper corrosion of canister
2019	Supplement to licence application submitted by SKB to SSM & MMB [5-41], [5-40]
2020	Östhammar municipality council approves DGR.
2021	Approval expected for licence to construct DGR by the government; SSM & MMB to stipulate conditions for licence
2024	Expected start of construction

Figure 5.2.4-7. Flowchart safety case and safety assessment of the Swedish DGR

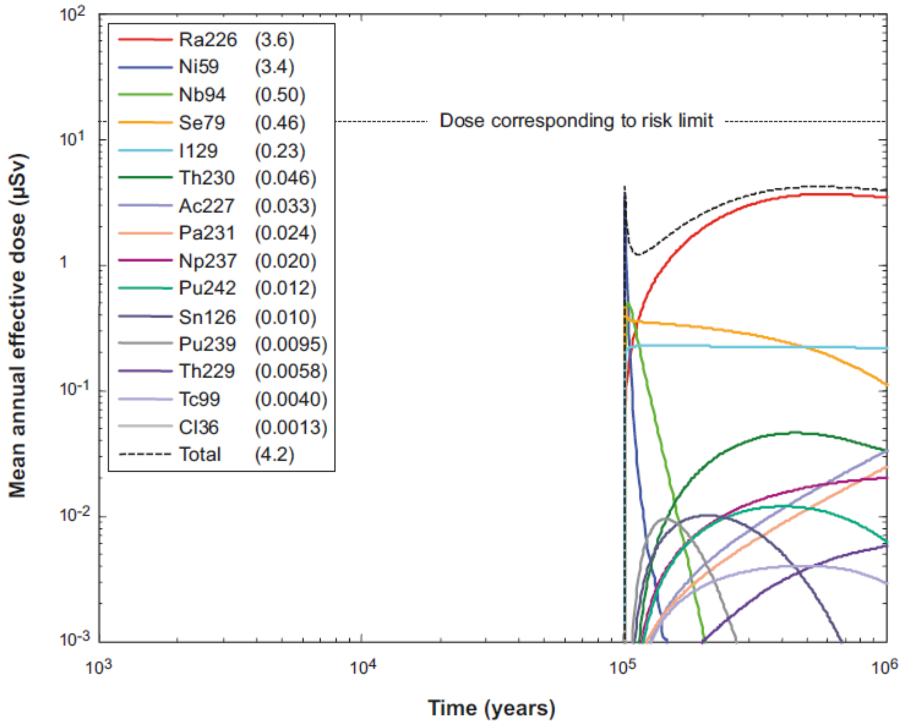


Source: [5-37]

Figure 5.2.4-8. Computed far-field mean annual effective dose caused by the DGR expressed in micro-Sv/year as a function of time for a probabilistic calculation (a) the reference scenario (central corrosion scenario with average 0.12 canister failed) after one million years; (b) probabilistic calculation one canister failing after 100 000 years. The legends report the dose contribution of each nuclide after 1 million year.



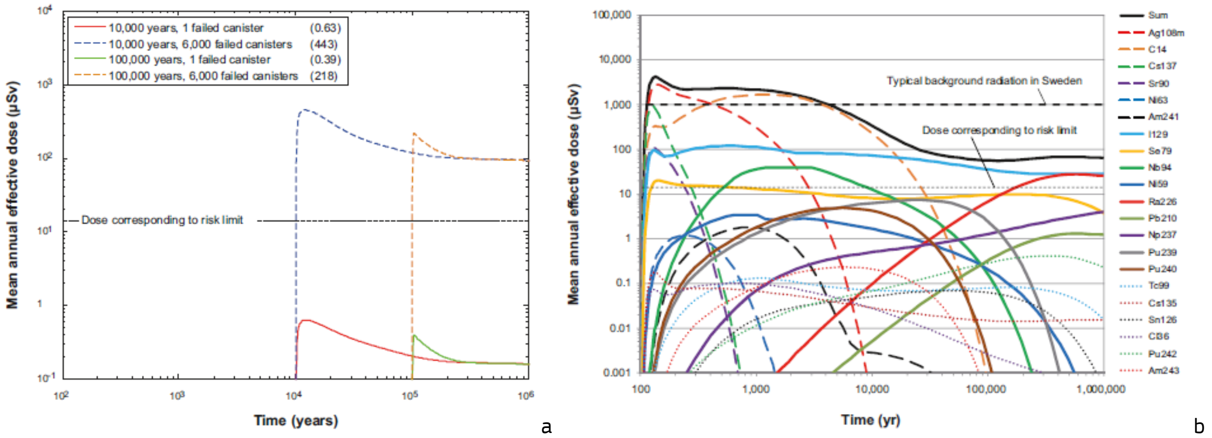
a



b

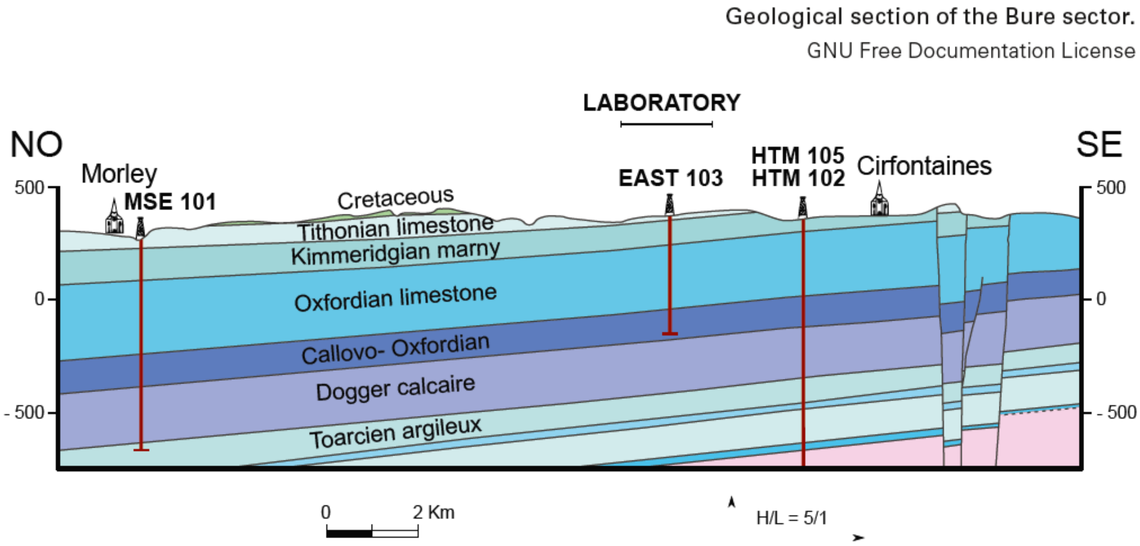
Source: [5-37]

Figure 5.2.4-9. Computed far-field mean annual effective dose caused by the DGR expressed in micro-Sv/year as a function of time for *what-if* scenarios: (a) comparison of the consequences of 1 canister vs. all 6 000 canisters failing after 10 000 or 100 000 years; the legend reports the dose contribution after 1 million year; (b) *what-if* scenario in which all canisters have an initial large defect and the buffer between the defect in the canister and the wall of the deposition hole is missing.



Source: [5-37]

Figure 5.2.4-10. Cross section of the geological formation at Bure where the French DGR Cigéo location is planned

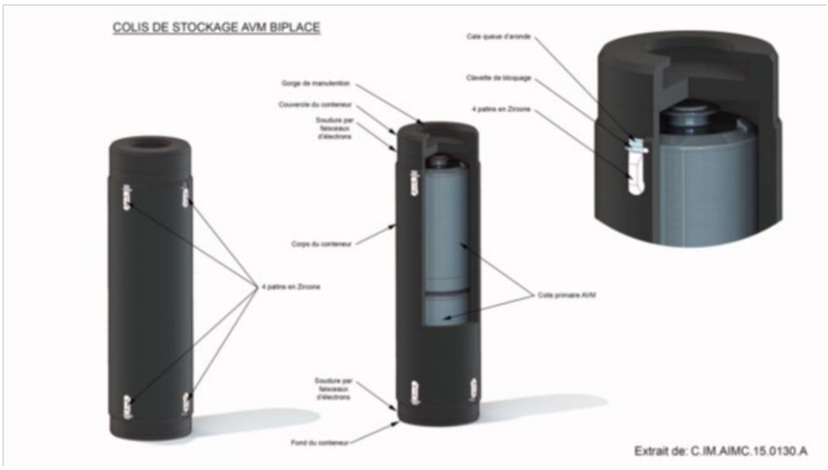


Source: [5-23]

Argillaceous formation: France [5-56] to [5-65]

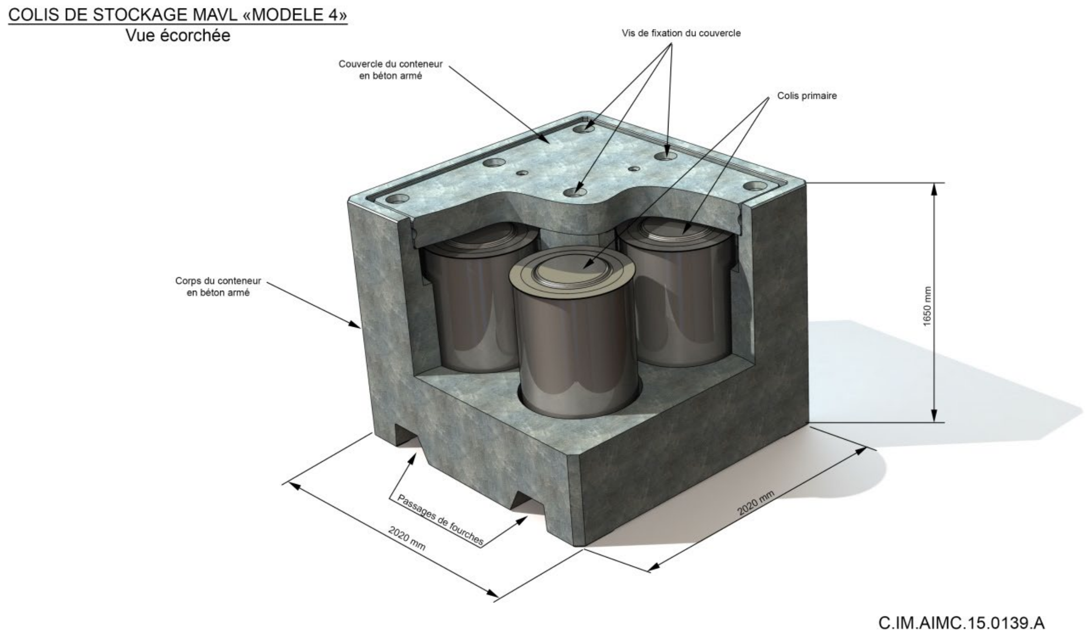
The French programme for geological disposal differs from those for Sweden and Finland on a number of points. The French nuclear programme is much larger than in the two Scandinavian countries; moreover, the spent fuel is reprocessed, generating vitrified high-level waste as the main HLW waste form that will be emplaced in the DGR. The Cigéo repository will host both high-level waste and long-lived intermediate level waste. By removing plutonium and uranium for recycling, reprocessing somewhat reduces the radioactivity and the long term heat generation of the HLW compared to spent nuclear fuel; however, the volumes of intermediate level waste generated by the reprocessing can be significant (see Chapters 3.3.5 of part A of this report and 2.3 of part B of this report). In terms of selected host rock, the French DGR will be based on Callovo-Oxfordian argillaceous clay with specifically good retention properties, located in the north east of the country, in the region of Bure. Another difference is that retrievability of the waste is stipulated by law before final closure of the repository. Figure 5.2.2-2 provides a schematic illustration of the repository layout, and Figure 5.2.4-10 shows the geological cross section of the Cigéo location. Callovo-Oxfordian clay is only available at a few locations and at a specific depth, which reduces significantly the number of available site locations compared to suitable crystalline rock in Sweden and Finland. On the other hand, Callovo-Oxfordian clay has an extremely low permeability and low hydraulic gradient, which limits groundwater flow and ensures excellent retardation properties that limit radionuclide mobilization and migration [5-61]. Thanks to the favourable properties of the host rock, in this repository concept there is less reliance on the waste packages than in the case of KBS-3. Figure 5.2.4-11 shows the disposal canister for vitrified HLW and Figure 5.2.4-12 a schematic concept of the configuration of a waste package for intermediate level long-lived waste [5-63].

Figure 5.2.4-11. Vitrified HLW canister for the French repository consisting of stainless steel cask with steel overpack (see Chapters 3.3.5, 3.3.8 of part A of this report, and 4.2 of part B of this report). Canister HA (HA)/HA1/HA2)



Source: [5-61]

Figure 5.2.4-12. Schematic concept of the configuration of a waste package for intermediate level long-lived waste for the Cigeo repository in France (ANDRA 2016, Dossier d'options techniques de récupérabilité



Source: [5-63]

Table 5.2.4-3 summarizes the chronology of the development and licensing for construction and operation of a DGR in France. The waste management programme is based on a series of laws and legal acts. The 1991 law stipulated the three options to be assessed for the back-end of the nuclear fuel cycle in France:

transmutation, long-term storage and disposal; the Waste Act 2006 endorsed reversible deep geological disposal; the 2016 law sets the requirements of the industrial deep geological disposal facility Cigéo.

Table 5.2.4-3. Chronology of the steps towards the construction and operation of a deep geological repository in France.

Year	Event
1991	Law "Bataille" No 91 – 1381: three strata for nuclear waste research (transmutation, geological disposal, long-term storage)
1993	Geological characterisation campaign began at four sites, call for candidate departments
1998	The government authorised the development of the Meuse/Haute-Marne Underground Research Laboratory in the Callovo-Oxfordian clay
2000	Start construction of the Underground Research Laboratory
2005	Dossier 2005, feasibility study for a reversible disposal facility [5-53]
2006	Public Debate & Planning Act Law 2006-739) Sustainable Management of Radioactive Materials and Waste (Waste Act), endorsing reversible geological disposal
2007	Implementation of the Permanent Environmental Observatory (OPE) to describe and monitor environmental effect of repository
2008	Publication Safety Guide for Geological Disposal of Radioactive Waste [5-57]
2009	Government approved Andra's proposal for work in the host rock zone planned for the disposal facility based on the advice of the ASN [5-59]
2013	Public debate on the Cigéo project was organised by the French National Public Debate Commission (CNDP).
2014	Andra submits a set of documents to the government consisting of a master plan for the operation of Cigéo, the Safety Options Report and the Retrievability Technical Options Report to prepare for the examination of the construction licence application for Cigéo". [5-58]
2016	Law 2016-1015 concerning Cigéo. Andra produced the safety option files on operational and post closure phases and submitted to ASN [5-56], [5-57], International peer review of Cigéo [5-64]
2018	ASN positive advice [5-65]
2019	Public Debate
2024	Expected licence application

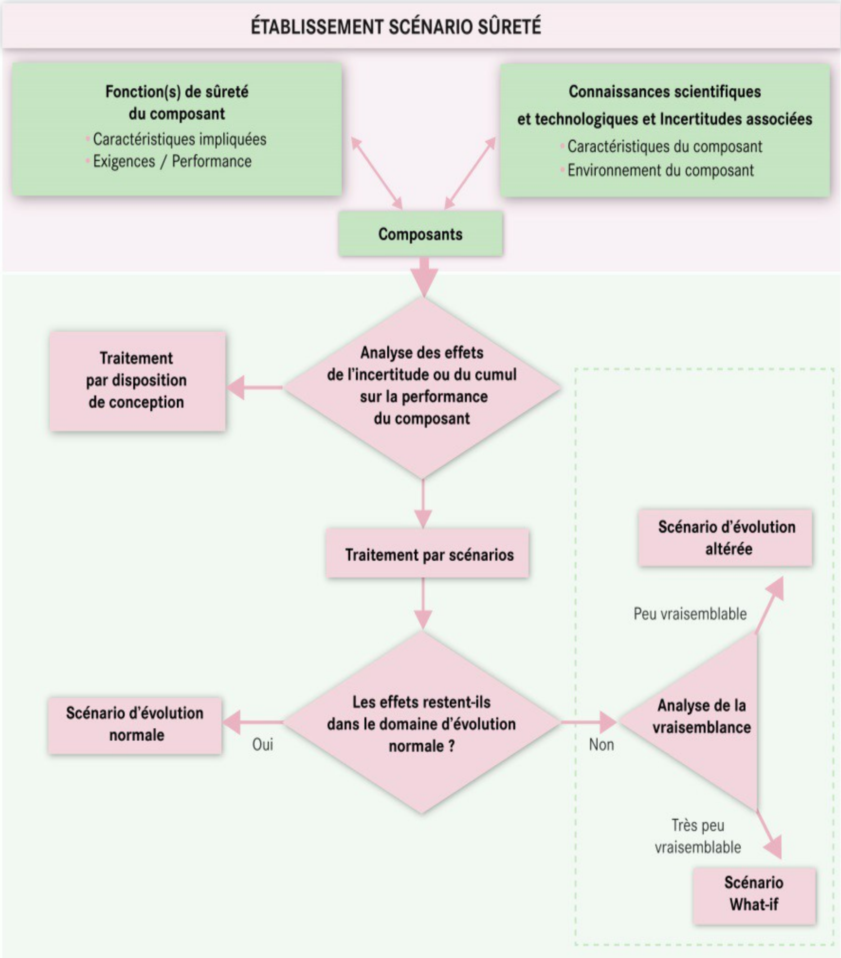
The waste management organization ANDRA is responsible for implementing a repository. ASN (Autorité de sûreté nucléaire) has the regulatory oversight. The Environment Code requires that the Government drafts a National Plan for Radioactive Materials and Waste Management (Plan National de Gestion des Matières et Déchets Radioactifs, or PNGMDR), every three years.

The site and geology were identified in the nineties and the construction of an underground research laboratory started in 2000. The feasibility of the site was confirmed in the 2005 Dossier [5-67]. By support of continued research and geological and technical investigation following the guidelines [5-57], the industrial disposal project Cigéo in the Meuse/Haute-Marne region was presented [5-69]. In 2016, ANDRA submitted to ASN a safety options report on the Cigéo project [5-61 to 5-63], subjected to an international peer review under IAEA supervision [5-64].

Figure 5.2.4-13 shows the flow chart for the safety assessments by ANDRA.

Figure 5.2.4-14 depicts the effective dose for a reference scenario with respect to two different climate evolution conditions and corresponding to wells 50 meters above or below the Callovo-Oxfordian layer [5-60]. In all cases the affective annual dose is well below the maximum allowed dose value set by ASN (0.25 mSv/year).

Figure 5.2.4-13. Flowchart scheme for safety analysis

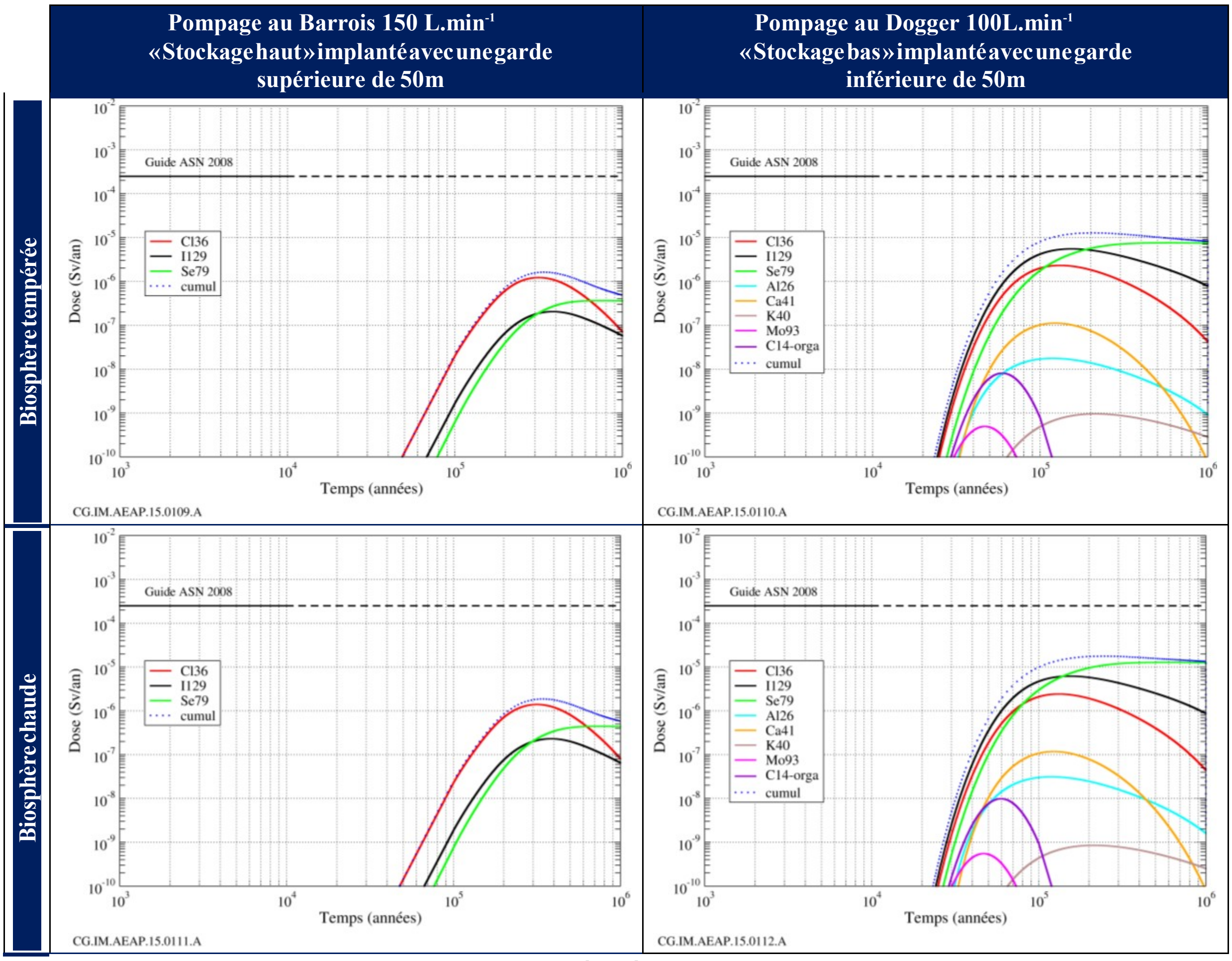


Source: [5-60]

Table 5.2.4-4 shows the computed maximum annual effective dose and the time when it will be reached for three radionuclides. A reference case and a what-if scenario in which all seals fail are illustrated. It is noteworthy that the difference in effective dose is very small, which confirms the efficiency of the clay formation to minimize radiological risks.

The ASN opinion in 2018 concerning the Cigéo DOS [5-65] is based on the recommendations of the Advisory committee for waste and on the report by a group of experts. While underlining the satisfactory technological maturity achieved at the DOS stage, the ASN opinion makes a number of specific recommendations, which concern the inventory of radioactive waste concerned, the disposal of bituminous waste packages and certain subjects which could lead to design changes (justification of the repository architecture, designing the installation against hazards, installation monitoring and post-accident situations) [5-56]. ANDRA has announced that the licence application will be finalised in 2021 with an expected licence for construction by 2024.

Figure 5.2.4-14. Computed annual dose due to the DGR (in Sv/year) for the reference case, corresponding to a well 50 meters above and below the clay formation and corresponding to climate evolution scenarios causing a tempered or a hot biosphere. The maximum allowed dose to the public set by ASN is 0.25 mSv/year



Source: [5-60]

Table 5.2.4-4. Computed annual effective dose in the reference case for the two climate change scenarios. The reference case with all sealing functional and a what-if scenario postulating that all sealing are dysfunctional are described.

	SEN(1) (tous scellements efficients)				**What-if «Dysfonctionnement de tous les scellements»**			
	Biosphère tempérée		**Biosphère chaude**		**Biosphère tempérée**		**Biosphère chaude**	
RN	**Dose maximale (mSv/an)**	**Date de dose maximale (milliers d'années)**	**Dose maximale (mSv/an)**	**Date de dose maximale (milliers d'années)**	**Dose maximale (mSv/an)**	**Date de dose maximale (milliers d'années)**	**Dose maximale (mSv/an)**	**Date de dose maximale (milliers d'années)**
^{36}Cl	$1.2 \cdot 10^{-03}$	320	$1.4 \cdot 10^{-03}$	320	$1.4 \cdot 10^{-03}$	310	$1.6 \cdot 10^{-03}$	310
^{129}I	$2.1 \cdot 10^{-04}$	380	$2.3 \cdot 10^{-04}$	380	$2.4 \cdot 10^{-04}$	350	$2.7 \cdot 10^{-04}$	350

^{79}Se	$3.7 \cdot 10^{-04}$	730	$4.4 \cdot 10^{-04}$	730	$7.3 \cdot 10^{-04}$	730	$7.3 \cdot 10^{-04}$	730
Cumul	$1.6 \cdot 10^{-03}$	340	$1.9 \cdot 10^{-03}$	340	$2.1 \cdot 10^{-03}$	350	$2.3 \cdot 10^{-03}$	340

(1) Scénario d'évolution normale

Source: [5-61].

5.2.5 Concluding remarks

There is broad consensus in the scientific community that deep geological disposal is the safest long-term solution for spent nuclear fuel and high level radioactive waste. The deep geological repositories (DGR) are based on a multi barrier combination including both engineered and natural barriers. The operational safety of geological disposal facilities is provided by means of engineered systems and active operational controls. Disposal facilities are designed to be passively safe after closure. The DGR are designed so that potential radioactive release from them occurring in the remote future are well below the maximum allowed dose limit set by the relevant regulation, which, in turn are orders of magnitude below natural background dose levels, and which ensure that no significant harm will be caused to humans by the repository. There are presently no deep geological repositories in operation, but after four decades of research and technology development the construction and operation of several repositories is expected in the present decade. The process for the design, licensing, construction, operation and final closure of deep geological repositories is regulated by national law, based on international conventions and European directives; this means that there is a common ground shared by all programmes based on the best available principles and concepts. The very long process to build a DGR is stepwise and reversible to various extents to ensure that the best available technology is used and that the radiological effects are and will be as low as reasonably achievable.

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6 Research and development for radioactive waste management

6.1 Introduction

R&D programmes and activities on the management of radioactive waste have accompanied and supported the development and implementation of nuclear technologies for peaceful applications throughout the history of this sector.

The Euratom Treaty of 1957 establishes a clear mandate to promote research and ensure dissemination of technical information on many aspects, including radioactive waste management and protection from the harmful effects of radiation.

Most R&D activities have been and are performed in support of national programmes. Additionally and as a complement to national efforts, many joint collaborative research programmes involving Member States organizations and supported by the EC have been implemented since the signing of the Treaty. More recently, the relevance and necessity of research on radioactive waste management has been affirmed in the Council Directive 2011/70/Euratom on the Responsible and Safe Management of Spent Fuel and Radioactive Waste[139] [6-4] Research activities are an important component of the national plans; as such they are included in the periodical reports from the Member States highlighting the relevant progress of the national plans.

Already in the 1950's, geological disposal was considered one of the most effective solutions (if not the most effective) for the disposal of high level radioactive waste [6-1]. In Europe, since the 1970s systematic studies on geological disposal as the reference option for the long-term management of high level and/or long-lived radioactive waste have been performed [6-2]. Early projects included a *European catalogue of geological formations having favourable characteristics for the disposal of solidified high level and/or long-lived radioactive wastes*, [6-3] and, in the 1980s and 1990s, investigations were carried out on radioactive waste behaviour in repository conditions, radionuclide migration through barriers and geological media, and overall behaviour of barriers.

6.2 Scope of R&D activities

Research activities on radioactive waste management support all technology areas and waste typologies described in the previous chapters. Throughout the history of nuclear energy, they have contributed to maintain the required safety standards and to implement innovation for all stages of waste management, from preparation and consolidation of waste packages to final disposal. The R&D support addresses a broad spectrum of scientific and technological objectives, from basic scientific studies, to improving the technology readiness levels (TRL) of new concepts, to demonstration and validation of optimization and innovation methods for well-established processes.

In particular, the scope of research programmes includes:

Basic knowledge
Acquiring basic knowledge on physical and chemical properties of radioactive species and compounds allows optimizing their immobilization in corrosion resistant wasteforms for final disposal, and allows understanding mechanisms and processes affecting the long term behaviour of the wasteforms after disposal. As such, research is a necessary component informing safe management of radioactive waste. In the case of HLW and spent fuel, the long term evolution of the wasteform during extended interim storage and after disposal in a geological repository is studied, with particular attention to solid state ageing effects during the pre-disposal stages and to mechanisms that may affect corrosion resistance, and release of radionuclides in groundwater in the final repository. Similar studies are performed on the waste package and containment barriers ensuring that the safety function is maintained. There is a large body of knowledge collected over the years through numerous scientific projects and collaborations, which provides a solid basis for implementation of final disposal options; this is reflected in the safety case demonstration and in the documentation supporting the disposal licence application submitted to the relevant national regulatory authorities.

The current focus of basic research is to extend the body of knowledge to cover special cases, e.g. new or unconventional wasteforms, and to reduce uncertainties associated with the very long timeframe of final disposal, e.g. the accurate determination of the inventory of radionuclides relevant to the waste repository evaluations and/or the properties of "hard to characterize" radionuclides [6-29]

139 Article 12.1(f) states the need to perform "the research, development and demonstration activities that are needed in order to implement solutions for the management of spent fuel and radioactive waste"; see also Art. 8.

Pre-disposal stages
In this domain R&D contributes by assessing the various stages of radioactive waste management preceding disposal in terms of individual components and overall system effectiveness, including e.g. the performance and synergies of multiple shielding and containment barriers protecting against exposure to and release of radionuclides from the waste. In the case of spent fuel before final disposal, the relevant stages include: cooling, packaging and transportation, storage, retrieval, and encapsulation for disposal, or reprocessing and vitrification of the non-recycled waste (see Chapters 3.3.8 of part A and 4.2 of part B). All these activities are industrially mature and implemented, so the research support for this domain is focused on optimization of the processes and possible innovation aspects.

Currently, a particular area of interest for research in the storage domain is the assessment of potential effects of extended spent fuel storage on the subsequent spent fuel management and disposal stages. Lists of R&D-relevant topics in various countries are available e.g. in the Strategic Research Agenda of the European Joint Programme on Radioactive Waste Management EURAD [6-5].

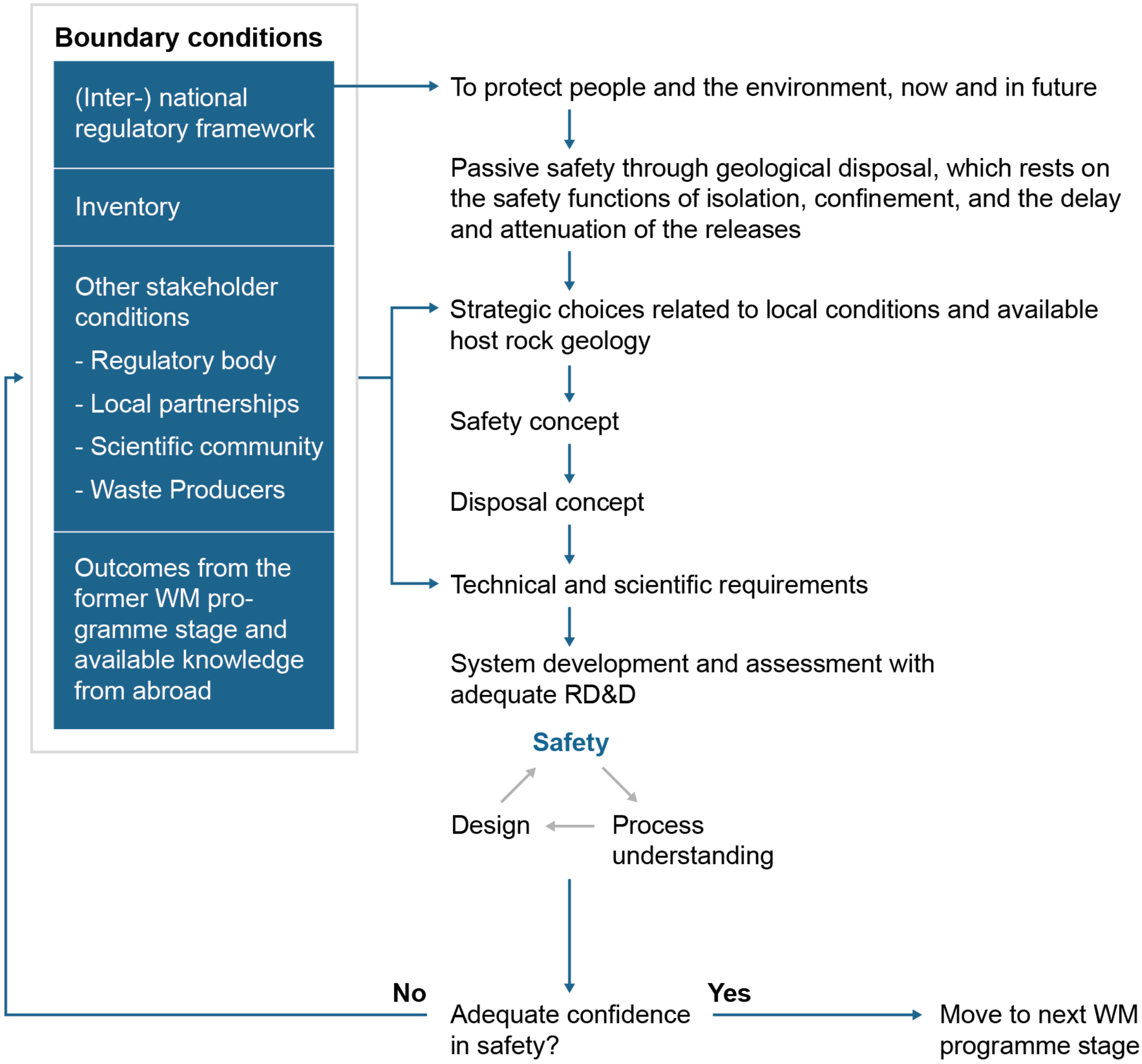
The definition of pre-disposal stages encompasses also conditioning and management procedures to treat radioactive waste other than nuclear fuel and high level radioactive waste, and bring it to a form compatible with waste acceptance criteria for storage and disposal. This is the scope addressed by the new project Pre-Disposal Management of Radioactive Waste PREDIS. The goal of this project is to increase the Technological Readiness Level and achieve highly optimized treatment and conditioning methodologies for specific low and intermediate waste (typically, relatively small batches of unconventional waste and/or legacy waste generated by research activities) for which no industrially mature solutions are currently available in the market. [6-7]

Disposal in geological repository
During the last five decades in Europe a very large amount of knowledge has been generated on geological disposal as the reference option for the long term management of HLW and spent fuel. The research was driven mostly by the end-user and by inputs from regulators, technical support organisations, and research organisations (see Chapter 6.3.1 below). The outcome of the research is peer reviewed and, especially the components directly used for safety and licensing applications, subjected to independent critical assessment and review by the regulators, including comparisons and cross-referencing among different programmes. R&D (or, in this case, RD&D: research, development and demonstration) provides the knowledge and the technical and scientific assessment basis for system design, siting and optimisation as well as contributions to fundamental understanding of the underlying processes affecting the behaviour of the repository (see Figure 6.2-1). Experimental and modelling activities provide an important input to the safety case and the performance assessment of the radioactive disposal, and consequently contribute to the licensing process. RD&D activities stretch from the initial decision to build a disposal for radioactive waste through all implementation and operation stages until disposal closure and possibly through post-closure monitoring (see Chapter 5.2 of part B of this report). Over the decades, in the case of final disposal in geological repository, the R&D contributions have been deployed along two main dimensions: space and time.

- *Space*. R&D supports siting campaigns to identify geologic formations suitable to host a deep geologic repository; each national programme includes siting campaigns (see also [6-3]). Additionally, research programmes contribute to the assessment and validation of the safety performance of the so-called "near field" and "far field" of the repository (see Chapter 5.2 of part B of this report). In the near field, the research efforts address properties, effectiveness and interplay of the multiple man-made and natural containment barriers acting against the release and mobilization of radionuclides from the wasteform. The local conditions and processes expected to govern the release of radionuclides from the waste form in case of direct reaction between waste and groundwater, and the subsequent transport through the barriers are investigated. In the far field, the research activities study the mechanisms that affect radionuclides migration through the geologic media. R&D contributes also to assess the potential impact of specific events (e.g. extreme climatic variations, such as a new ice age, etc.) on the overall assessment of the repository performance.

- *Time*. The deep geological repository for spent fuel and high level waste is designed to contain and isolate radioactive waste for a very long time. Engineered barriers and natural conditions will contribute to delay the occurrence of direct reaction between radioactive waste and groundwater, and radionuclide release from the wasteform for thousands of years or more. Moreover, the properties of the selected geologic media in the far field will ensure very slow migration of released radionuclides. R&D efforts include determining the timeframe of interest for eventual radionuclides mobilization, and extrapolating the safety functions to the set of conditions expected at that time, to ensure that the potential exposure of the public does not reach the limits established by the relevant regulations.

Figure 6.2-1. Schematic description of the iteration step implemented to move forward a waste management (disposal) process. The technical and scientific requirements associated with the proposed concept must be assessed and fulfilled by an adequate Research, Development and Demonstration (RD&D) set of activities



Source: IGD-TP, 2020 [6-2]

The deep geological repository for spent fuel and high level waste is designed to contain and isolate radioactive waste for a very long time. Engineered barriers and natural conditions will contribute to delay the occurrence of direct reaction between radioactive waste and groundwater, and radionuclide release from the wasteform for thousands of years or more. Moreover, the properties of the selected geologic media in the far field will ensure very slow migration of released radionuclides. R&D efforts include determining the timeframe of interest for eventual radionuclides mobilization, and extrapolating the safety functions to the set of conditions expected at that time, to ensure that the potential exposure of the public does not reach the limits established by the relevant regulations.

Decommissioning and remediation

Nuclear decommissioning is an industrially mature technology, which will experience a strong growth trend during the current and next decades. Some nuclear power plants and reactors have been successfully decommissioned in Europe.

Currently, R&D activities mostly support process optimization efforts aimed at improving process efficiency and standardization, minimizing dose to operators, and establishing shared grounds for exchanges of experience and best practices, and for training and education [6-8]; (see also ongoing Euratom research projects SHARE [6-9] and INSIDER [6-10] and [6-11]). Specific research activities aim at the development of advanced methods for fast, in-situ characterization of contaminated surfaces and components (e.g. the development of suitable analytical tools for "Hard To Characterize" nuclides). Non-standard cases (e.g. decommissioning of graphite reactors or special research facilities, legacy site remediation, etc.) are characterized by relatively wider knowledge gaps, requiring dedicated and also more basic R&D.

6.3 Innovative options for the back-end of the nuclear fuel cycle

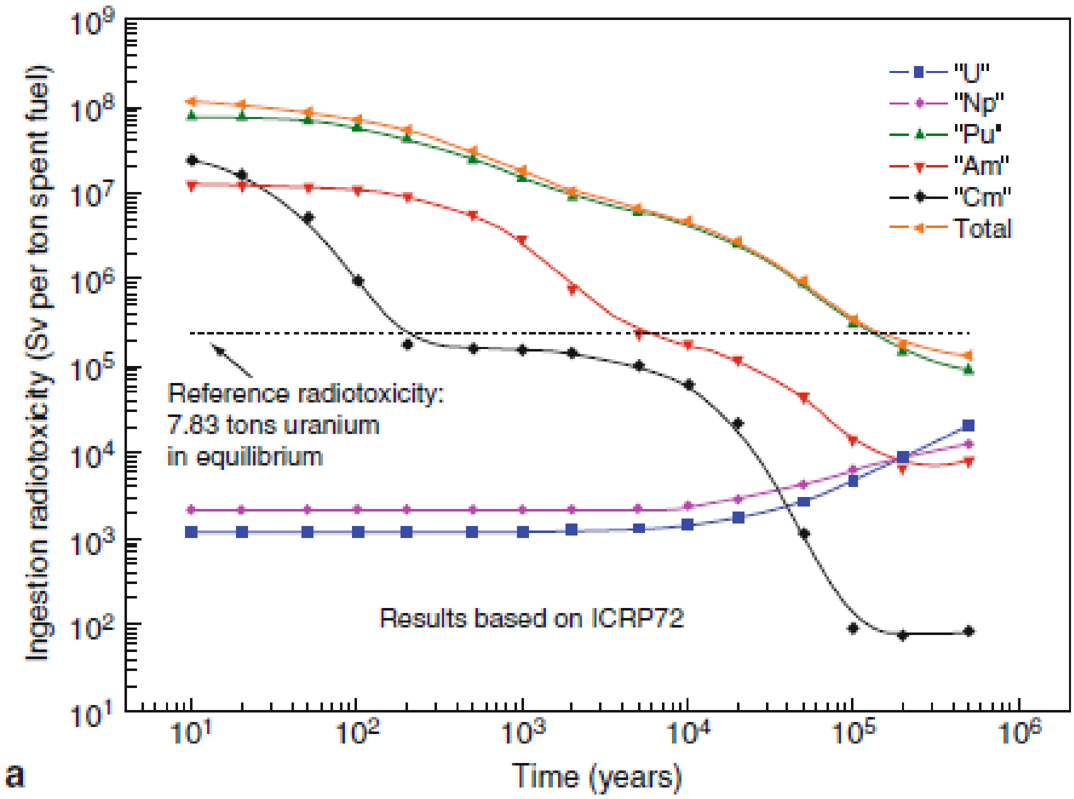
The main objectives of "closing" the nuclear fuel cycle are relevant to the environmental objective "Transition to a circular economy, waste prevention and recycling" [6-31]:

- maximize the fraction of spent nuclear fuel that can be recycled in nuclear reactors
- minimize/optimize the long term radiotoxicity of HLW to be disposed of in the geological repository.

These objectives are achieved by chemically dissolving spent fuel, and separating reusable and/or burnable species (partitioning). The partitioned reusable/burnable species are incorporated in special fuel elements and irradiated in nuclear fast reactors, which recycle the fuel component and destroy long-lived radiotoxic species (transmutation). Due to the fact that fast reactors allow multiple (re)cycling of the fractions of fuel/waste not consumed/burned, the final result of iterating this process would be an almost complete use of the fuel and an increasingly reduced fraction of long-lived species (mostly in terms of minor actinides fraction) in the irradiated fuel.

Figure 6.3-1 highlights the contribution of the minor actinides to the long term radiotoxicity of spent fuel. The radiotoxicity of different actinides is compared to the total radiotoxicity profile (dominated by recyclable plutonium) and to the reference radiotoxicity level of the natural mineral required to fabricate 1 metric ton of uranium fuel. Implementing partitioning and transmutation would reduce the time necessary for the HLW to decay down to the natural reference level to some centuries instead of some hundred thousand years. In particular, P&T of Am, in addition to Pu that can be re-used as fuel, could be an effective strategy to achieve such a goal. Removing Pu and Am would leave curium as the main radiotoxic minor actinide in the waste [6-12]. However, due to the relatively short half-life of Cm, the presence of this species in the waste would not affect significantly the time required to decay down to the natural minerals level, as also shown in Figure 2.4-1 in Chapter 2.4 of part B of this report. The non-recycled residual waste destined for vitrification (or immobilization in crystalline wasteforms) and disposal in a geological repository would include the fission products. Another potential benefit from the adoption of a closed nuclear fuel cycle would be the significant reduction of the footprint of the geologic repository for HLW (see Figure 3.3.8-11 of part A of this report).

Figure 6.3-1. Radiotoxicity as a function of time of spent nuclear fuel, of minor actinides Np, Am, Cm, and of U and Pu. The diagram shows that by recycling U and Pu, and by partitioning and transmuting Am, the resulting radiotoxicity is governed by the curium inventory, which decays below the reference radiotoxicity level in a few centuries. The reference radiotoxicity line represents the radiotoxicity of the natural minerals from which the uranium was extracted.



Source: [6-12]

Significant R&D effort at national, European and international level, has been dedicated to investigating options aimed at implementing a closed fuel cycle which includes P&T. Table 6.3-1 lists Euratom Research and Training programmes dedicated to P&T since the 5th Framework Programme (FP) of the European Commission ([6-14]; see also e.g. [6-13]. Although essentially all steps of P&T have been demonstrated at laboratory scale, the Technology Readiness Level is not yet corresponding to industrial maturity. Therefore, the input required from research activities includes a broad spectrum of applications, to fill some remaining knowledge gaps and to support implementing prototype level demonstrations to increase the TRL of these concepts. The progress in this area is associated also with the development of new irradiation facilities. R&D programmes involving Member States, the EC and international partners and organizations are continuing the effort.

Table 6.3-1. Euratom Research and Training programmes dedicated to P&T implemented since the 5th Framework Programme of the European Commission.

Topic	**FP5**	**FP6 EUROTRANS DM**	**FP7**	**H2020**
Coupling	–	DM2-ECATS	FREYA	(MYRTE)
Fuels	CONFIRM FUTURE	DM3-AFTRA	FAIRFUELS, F-BRIDGE, ASGARD, ALICE	
Thermal-Hydraulics	ASCHLIM		THINS, SEARCH, MAXSIMA	SESAME

Materials	MEGAPIE, SPIRE, TECLA	DM4-DEMETRA	MATTER, GETMAT, MATISSE	GEMMA, INSPYRE, M4F
Design	PDS-XADS	DM1-DESIGN	CDT, MAX, SARGEN-IV, SILER	ESNII+, MYRTE
LFR	_	ELSY	LEADER	(ESNII+, MYRTE)
Infra-structures	_	VELLA, MTR-I3	HELIMNET, ADRIANA, DELOITTE Study, MARISA,	(ESNII+, MYRTE)
Scenario studies	_	PATEROS	ARCAS	
Partitioning	PYROREP, PARTNEW	EUROPART	ACSEPT, SACSESS, TALISMAN	GENIORS
Total budget / EU contribution	**~€51M/€20M**	**~€71M/€36M**	**~€167M/€92M**	**~€59M/€38M**

Source: [6-14].

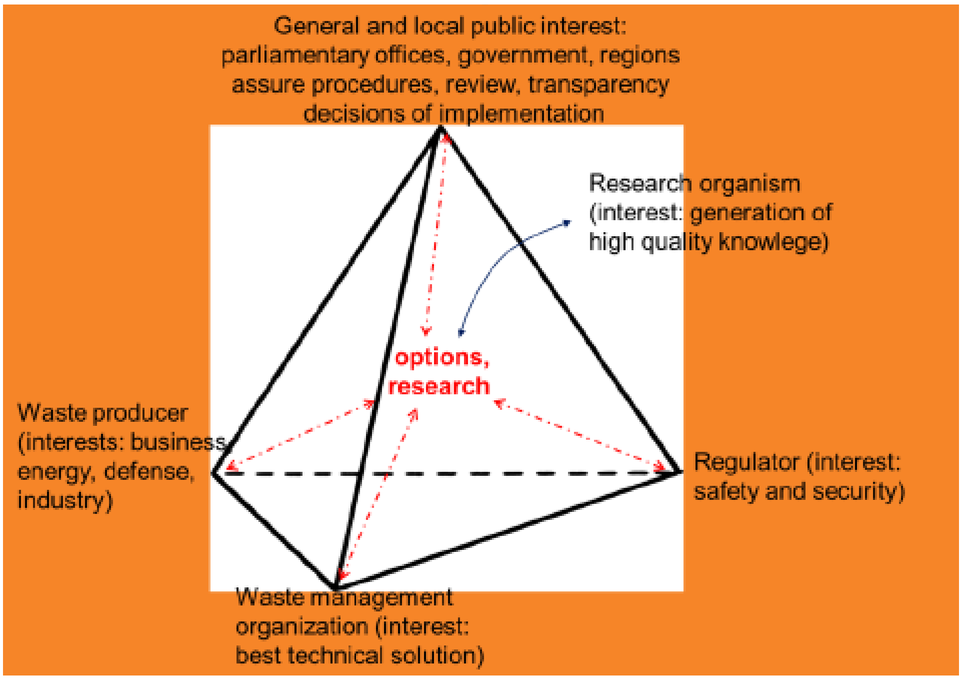
6.4 European research in radioactive waste management: who does it and how it is structured

6.4.1 Actors

The research, development and demonstration (RD&D) carried out in support of safe radioactive waste management, including disposal, is a key component of each National Programme. Given the long timescales and socio-political dimension, RD&D provides primarily the scientific basis for implementing safe radioactive waste management solutions, whilst also contributing to building stakeholder trust, public acceptance, and training for next generations of experts (see e.g. Chapter 5.2 in part B of this report).

Figure 6.4.1-1 schematically illustrates the separation and interconnection among the different actors involved in RD&D planning, implementation, review and use [6-15].

Figure 6.4.1-1. Schematic illustration of roles and interconnection among the main actors involved in radioactive waste management RD&D.



Source: [6-15]

The realization of a radioactive waste disposal programme including RD&D planning and implementation, licence application, and, finally, the actual disposal of radioactive waste, is performed by an implementer organisation. The implementer can be appointed/ licensed by the government, for instance as a private Waste Management Organisation (WMO) representing industry. Alternatively, it can be a public/governmental entity depending on a ministry, as determined by the Member State regulatory frame.

In order to achieve its objective, the implementer builds its own internal competences and/or acquires them in the open market. The necessary spectrum of competences can be found in the industry (technical), in national and international research organizations (scientific and technological), and in universities (scientific and academic). Significant competence is available also within TSO, in support of independent reviewing and assessment carried out by the national regulator. The interaction between implementer and TSO is affected by the necessity of ensuring adequate separation of roles between supervising and supervised entities.

As described in the preceding section, research organizations or entities (national or supra-nationals as in the case of the JRC) supply scientific data addressing basic and/or applied open issues, and perform validation modelling and experimental campaigns, often providing input to the performance assessment of a geological repository or waste disposal concept. Recipients of data and deliverables produced by the RD&D organizations are industry (WMO), Technical Support Organisations and/or regulators, policy makers and the scientific community. National research organizations primarily contribute their outcome to the corresponding national programme. However, scientific data and knowledge is generally disseminated among and reviewed by international partners, including the scientific community at large and the public. This ensures optimized (peer) reviewing of relevant findings and helps to build international consensus on the most effective solutions.

An essential role is played by international organizations such as IAEA and OECD/NEA to maintain and develop a global dimension of RD&D, thus extending cross referencing and review beyond Europe.

6.4.2 Structure

Historically, most R&D activities are performed in support of the national programmes on the safe management and disposal of radioactive waste. The relevant outcomes are described in the periodical progress report of the national plans, as prescribed in the Council Directive 2011/70/Euratom [6-4].

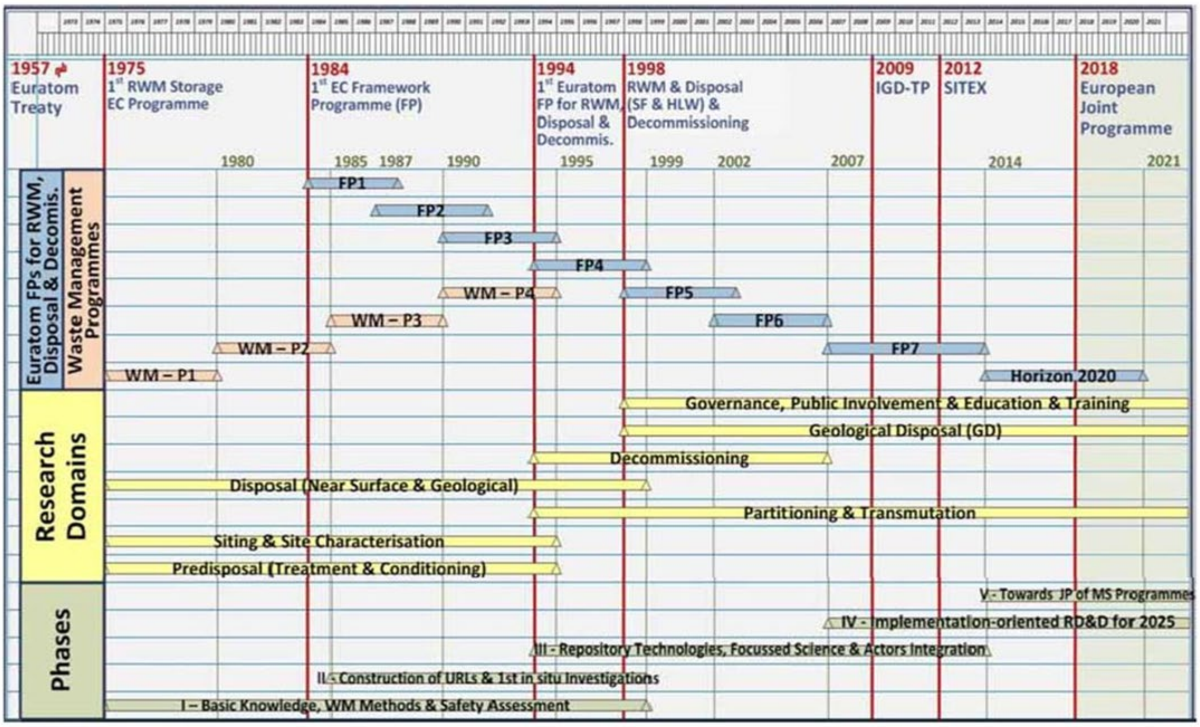
The state of advancement differs among the Member States, reflecting slightly different national strategies and their implementation timelines. Some Member States have already progressed significantly in the licensing process (Finland, Sweden, France) and are building geologic repositories for final disposal of radioactive waste and spent fuel (Finland). Other countries have adopted a definition and siting process that will be completed in a few decades (e.g. Germany). Other Member States have longer implementation timeframes, and will have the possibility of identifying best practices and options possibly leveraging knowledge established by the most advanced programmes, including, in some cases, possible synergies at regional level (as in the case e.g. of the European Repository Development Organisation Working Group (ERDO WG)). Such variety of strategies is in line with the frame of options contemplated by the waste Council Directive 2011/70/Euratom [6-4] (see also 6-30].

Figure 6.4.2-1 illustrates the evolution of Euratom Research and Training (R&T) programmes of the European Commission since their inception. Less than two decades after the ratification of the Euratom Treaty in 1957, the first EC R&T programme on radioactive waste management started in 1975 to support MS national RD&D programs. Since then, seven Framework Programmes (FP) have been successfully executed. The 8th Framework Programme, Horizon 2020, is followed by Horizon Europe, which started in 2021. The EC gives support through co-financed joint programming (Indirect Actions), or through R&D activities performed by the Joint Research Centre of the EC (Direct Actions). During 45 years, the multi-partner projects have evolved to become, especially after the turn of the century, large integrated projects, including demonstration projects (e.g. [6-16]), EC FP6 Integrated Project, 2004 – 2009). .

The introduction of a common European Radioactive Waste Directive on management of all types of radioactive waste (Council Directive 2011/70/Euratom) accelerated the collaborative trend among the Member States. The EC R&D actions support the implementation of the Radioactive Waste Directive in the Member States, taking into account the various stages of advancement of the national programmes.

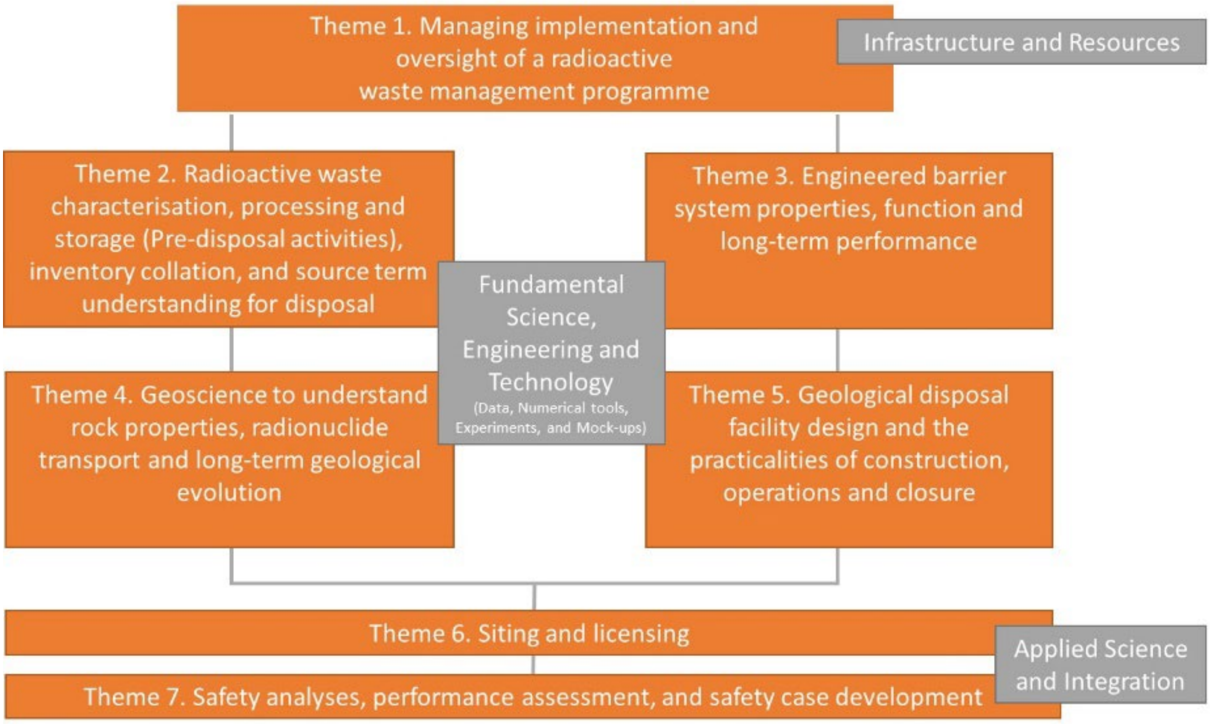
A significant part of spent nuclear fuel research is now performed by research organizations as a contribution to the EC research framework programmes. The MS national radioactive waste management R&D programmes have benefited from the joint research projects and from networking actions (e.g. Coordination and Support Action, CSA). The most recent evolution is represented by the current European Joint Programme configuration. Following decades of research, development and demonstration in support of the safe management and disposal of radioactive waste, the European Joint Programme on Radioactive Waste Management (EURAD) was started in 2019. EURAD builds upon existing networks of European actors such as IGD-TP [6-17], SITEX [6-18] and EURADScience [6-19], on past coordination and support actions (in particular, SecIGD2, SITEX-II project) and on the preparatory work by the EC JOPRAD project. It coordinates activities on agreed priorities of common interest among European Waste Management Organisations, Technical Support Organisations and Research Entities (with actors mandated by the MS ministries/governments). Figure 6.4.2-2 shows the prioritised activities in the EURAD Strategic Research Agenda [6-5].

Figure 6.4.2-1. Evolution of the Research and Training Euratom programmes of the European Commission in the radioactive waste management domain since their inception in 1975.



Source: [6-14]

Figure 6.4.2-2. Schematic illustration of the Strategic Research Agenda priorities of the European Joint Programme on Radioactive Waste Management (EURAD).



Source: (EURAD https://www.ejp-eurad.eu/strategic-research-agenda).

One of the primary objectives of EURAD is to consolidate existing knowledge for the implementation of the first of a kind deep geological disposal for spent nuclear fuel and high level waste. The research activities included in the current 5-year cycle of EURAD are focused on related R&D tasks. The success of the first 5-year EURAD could lead to its extension for a second joint programming period 2024-2029[140]. In order to cover the whole chain of waste management from cradle to grave, the European Commission launched two complementary projects: the Research and Innovation Action on waste pre-disposal activities PREDIS and the Coordination and Support Action related to decommissioning activities SHARE.

Joint programming complements MS national RD&D Programmes, by establishing and carrying out joint activities where there is an added value at the European level. It is a significant change, which deepens collaboration among European actors in the field of radioactive waste management. The European Joint Programming in combination with the Radioactive Waste Directive foster a fusion of national RD&D activities into actions at European level, thus optimizing the use of resources, achieving sharing of methodology and knowledge, development of common strategies, and overall promotion of a faster and cost-effective implementation of radioactive waste disposal in the Member States.

Additionally, global partnerships in the radioactive waste management area complement European RD&D initiatives. Such global partnerships with, e.g. with USA and Japan have been in existence for a long time. The many ongoing projects in collaboration with IAEA, OECD/NEA and other international agencies further extend the global dimension and vision of European research.

6.4.3 R&D infrastructure, methods and tools

As described in Chapter 6.2 above, the main objective of research studies in the radioactive waste management domain is to characterize and understand the mechanisms that may affect radionuclides release from the wasteform and transport/migration through containment barriers and/or through geologic media.

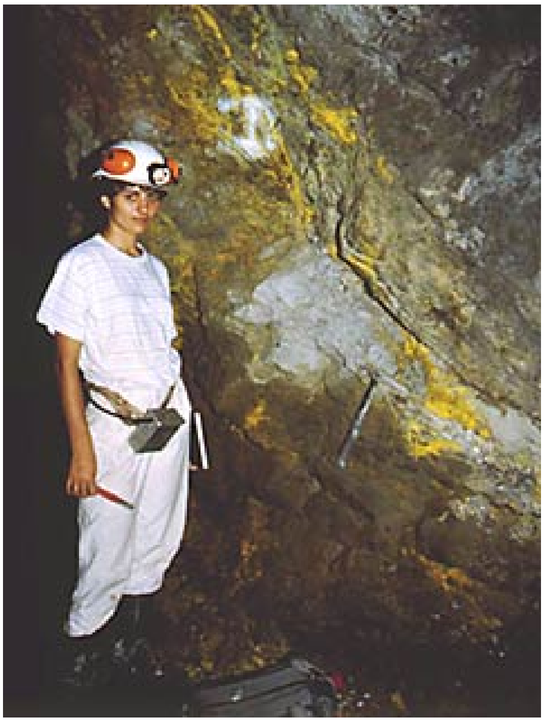
Direct studies involving radioactive waste require using shielded experimental facilities (glove boxes and/or hot cells) in laboratories equipped and licensed to handle radioactive substances. In particular, in the case of experiments on spent fuel/HLW, hot cell facilities are needed. In hot cells, highly radioactive specimens are manipulated remotely through a very thick biologic shielding. The availability of such installations is essential to ensure adequate R&D support in which the “real” materials are studied. A list of hot laboratories can be found e.g. at [6-20] or at [6-21]. Due to the relative scarcity of large infrastructures and to the necessity to optimize research resources, there is a growing trend associated with the evolution of joint R&D programming, as described in the preceding section, towards implementing “sharing” schemes for the use/access of large infrastructure among research organizations and relevant actors in the waste management R&D domain.

The research performed in hot laboratories is effectively complemented by studies on tailor-made analogue compounds, i.e. materials incorporating stable isotopes chemically similar to the radionuclides of interest. Investigation on non-radioactive analogues can be performed in “cold” laboratories, allowing organizations which do not possess hot lab capabilities (e.g. universities) to contribute useful data. Although this type of studies cannot replace hot cell or hot lab investigations, using non-radioactive specimens allows applying characterization methods, which may not be suitable for implementation in highly radioactive environments.

Particularly relevant for the assessment of long term behaviour of a deep geologic repository for spent fuel and HLW is the study of natural analogues. Natural analogues are very old minerals from geologic formations which billions of years ago hosted naturally occurring nuclear reactors. The Oklo uranium ore site in Gabon has been confirmed to have hosted multiple natural reactors two billion years ago (Figure 6.4.3-1). By looking at the composition of the rocks surrounding the natural reactor site, in particular by examining the distribution of stable isotopes which are the end “daughters” of the nuclear decay chain of relevant radionuclides, it is possible to obtain useful information about the migration of radionuclides through the geologic media over very long times [6-23]. In fact, geologic media which hosted relatively high concentrations of nuclear material and reveal limited radionuclide migration may be considered good candidates to host a geologic repository for spent fuel and HLW. Figure 6.4.3-2 shows a comparison between the Oklo natural reactor site and the layout of the planned Swiss geological repository for spent fuel and high level waste in opalinus clay, highlighting the differences.

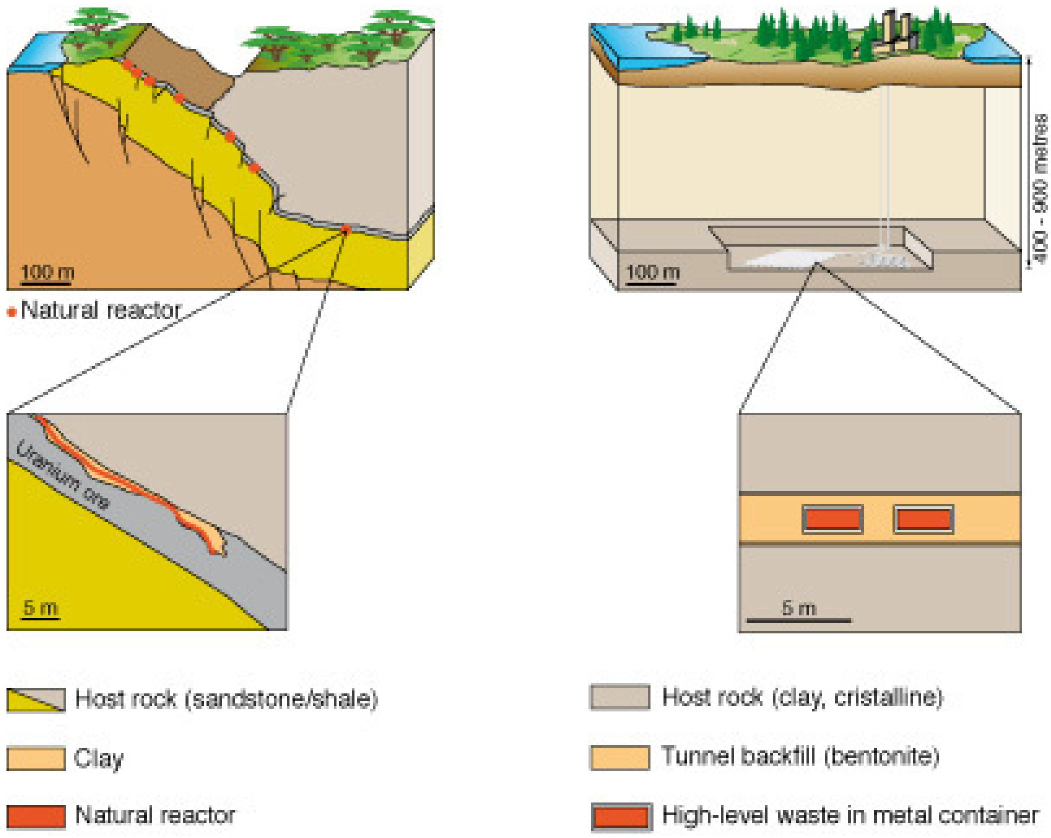
[140] If the programme implementation will confirm to have an EU-added value beyond the national programme, to carry a joint vision and forward looking plan of action, to be inclusive of MS actors, and to deploy a transparent and fair mode of operation, the European Joint Research Programme scheme could become the reference contractual instrument for the Commission's support in future Euratom Framework Programmes.

Figure 6.4.3-1. A natural reactor spot at Oklo. The yellow phases on the rock wall are uranium-rich uraninite minerals.



Source: [6-23]

Figure 6.4.3-2. Comparison of the uranium ore deposit at Oklo (left), where natural reactors were present 2 billion years ago, with the planned Swiss geological repository for high level waste (right). In contrast to a natural reactor, there can be no spontaneous chain reaction in a deep geological repository as the content of uranium-235 in the spent fuel assemblies is too low, and the design of the repository ensures that criticality conditions cannot occur



Source: [6-24]

The geologic repository is a complex system, and many physical and chemical mechanisms can play a role in determining the long term corrosion behaviour of wasteforms and the transport of radionuclides. It is thus necessary to consider multi-angle approaches to the characterization of such systems. On the one hand, simplified experiments in which a limited number of “ingredients” is present, allow determining single cause-effect relationships; however, they may miss the interplay among multiple agents. Integral tests, in which most or all of the components of the repository system are present, are more representative of the “real” configuration in the repository; however, the simultaneous effect of different factors makes it difficult to determine unequivocally cause–effect correspondences. Moreover, integral experiments are very difficult to set up in a hot cell or in a glove box. Relevant studies under realistic conditions can be performed *in-situ* in underground facilities. Several countries have established underground research laboratories (URL) in geological formations corresponding to the actual medium considered for the final repository [6-25]. Although highly radioactive specimens cannot be tested in such facilities, migration studies using radioactive tracers can be performed, to help determine relevant radionuclide transport kinetics affecting the far field of the repository. URL can be site-specific, i.e. dedicated to test properties relevant for a specific site evaluation, or “generic”, i.e. used to perform more general studies and provide information that may support disposal elsewhere. Table 6.4.3-1 describes the main features characterizing the two types of URL [6-26].

Table 6.4.3-1. Objectives of Generic and site-specific URLs.

Generic URL	**Site-specific URL**
Development and testing of technology and methodology – test methods for characterisation, construction techniques, monitoring. Development of understanding of processes and collection of generic data for safety assessment – sensitivity of rock mechanics, host rock-barrier properties and their interaction. Concept testing and demonstration – testing of disposal design concept and alternatives, operational options, demonstration of industrial-scale projects. Building confidence and fostering international co-operation – experts from different disciplines interact to build technical confidence, develop experience among international professional communities, interaction between various stakeholders and interested public.	Evaluation of site and confirmation – characterisation of geosphere immediately adjacent to repository and development of upscaling rules. Collection of site-specific data – data required for performance assessment and for future optimisation of repository design, reduction of inherent conservatism in conceptual and safety assessment models. Demonstration of technology and techniques – monitoring of near field responses of the repository for regulatory purposes, addressing environmental impact assessment issues. Testing of final repository design as well as other operational aspects – testing the robustness of the EBS or other testing linked specifically to safety assessment requirements for licensing. Building confidence – demonstration of specific system design/techniques to regulators and the public.

Source: [6-26]

An important component of the R&D studies is the modelling of the experimental data using state of the art codes based on physico-chemical and thermodynamic laws and correlations. Modelling is extensively used to understand behaviours and trends observed experimentally and to obtain prediction capabilities for complex systems. The most sophisticated codes applied to deep geological repository modelling allow covering both near field and far field, generating the outcome for the overall performance assessment of the repository (see e.g. chapter 5.2 of part B and [6-27]).

The best results leading to converging conclusions are obtained by combining all the above-mentioned approaches, and leveraging their complementary nature. Such synergistic approach is well suited within the frame of collaborative multi-partner projects and of joint programming as described in the preceding section.

6.5 Future role and perspectives for research on radioactive waste management

A recurrent question is if there is still need for R&D in domains such as HLW and spent fuel disposal (or on other processes industrially available or deemed mature for implementation), given the fact that there is generalized consensus, based on the outcome of decades of dedicated R&D, that the deep geological disposal concept is safe and technologically mature. The reply to this question is unequivocally positive. There is still a clear role for R&D, and a strong necessity to maintain adequate capabilities to investigate all waste management and disposal domains.

In general, there are two basic reasons that justify continuing research activities, which apply to mature cases such as HLW and spent fuel disposal: the first is to be able to improve accuracy and reduce uncertainties affecting the existing knowledge, taking advantage of innovative/improved experimental tools and improved understanding of basic phenomena as they become available; the second reason is to be able to address new questions and needs, which may arise from new knowledge on spent fuel and radioactive waste management, from other connected fields, and/or from the society (including new political priorities, public demands, and queries from societal actors (see e.g.: [6-28])).

Moreover, there are new or evolving specific domains that require maintaining and further developing the full spectrum of research capabilities, from basic to applied research, and the corresponding research infrastructure. Nuclear technology is not static: as mentioned in Chapter 6.2, the spent fuel and waste forms are changing in correspondence with the introduction of accident tolerant fuel and certain types of small modular reactors, the increasing burnup of spent fuel, the emerging perspective of direct disposal of spent mixed oxide fuel Additionally, research activities will continue to address specific or new domains that are not yet fully mature, such as closed cycle concepts for minimization of long term radiotoxicity of long lived waste in a deep geological repository. The variety of topics where the contribution by RD&D is currently required is illustrated e.g. in the Strategic Research Agendas of EURAD and of IGD-TP ([6-5] and [6-2]).

As stated in Art. 8 of the Radioactive Waste Directive 2011/70/Euratom, there is a need to provide adequate education and training and to maintain R&D activities in order to ensure the availability of the necessary expertise and skills to cover the needs of the national programmes on spent fuel and radioactive waste management. Compared to other domains, the case of nuclear energy and radioactive waste management is somewhat special. On the one hand, the timescale affected by radioactive long-lived waste management tasks will encompass many generations, requiring a strong, robust knowledge transmission system; on the other hand, nuclear technologies have been implemented for a relatively short time span since their inception, corresponding to only one or two generations until now. The lack of a consolidated system for transmitting knowledge across generations tested over multiple generational changes as in the case of "older" technologies highlights as a very relevant and high priority the necessity to consider preserving, managing, disseminating and forwarding nuclear science and technology knowledge across generations. Thus the education and training, and the knowledge management dimensions are considered key complementary components of R&D programmes at national, European and global level.

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List of abbreviations and definitions

Acronym	Definition
AC	Alternating current
ADP	Abiotic Depletion Potential
ADS	Accelerator Driven System
ADU	Ammonium Diuranate Process
AEL	BAT-AEL Best Available Techniques - Associated emission levels
AGR	Advanced Gas-cooled Reactors
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AMAD	Activity Median Aerodynamic Diameter
ANDRA	French national radioactive waste management agency (French: Agence nationale pour la gestion des déchets radioactifs)
AP	Acidification Potential
ARAO	Waste Management Organisation of Slovenia (Slovenian: Agenciji za radioaktivne odpadke)
ARTEMIS	Integrated Review Service for Radioactive Waste and Spent Fuel Management, Decommissioning and Remediation (IAEA review service)
ASN	French regulatory body (French: Autorité de sûreté nucléaire)
AUC	Ammonium Uranyl Carbonate Process
BAM	German Federal Scientific and Technical Institute (German: Bundesanstalt für Materialforschung und -prüfung)
BAT	Best Available Technique
BBC	British Broadcasting Corporation
BEFAST	Coordinated Research Project: Behaviour of Spent Fuel Assemblies in Storage
BGE	German Federal Company for radioactive waste disposal (German: Bundesgesellschaft für Endlagerung)
BGZ	German Federal Company for the interim storage of radioactive waste (German: BGZ Gesellschaft für Zwischenlagerung mbH)
BSA-RC	Biosphere Safety Assessment - Reference case
BSS	Basic Safety Standards
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium Reactor (Pressurised Heavy water Reactor)
CASTOR	Cask for Storage and Transport of Radioactive Material
CCS	Carbon Capture and Sequestration
CFC	Chlorofluorocarbon
CIGEO	French Industrial Centre for Geological Disposal (French: Centre industriel de stockage géologique)

CLAB	Swedish Central Interim Storage Facility for Spent Nuclear Fuel (Swedish: Centralt mellanlager för använt kärnbränsle)
CLB	Current Licensing Basis
CLIMA	Directorate-General for Climate Action
CML	Leiden University's Centre of Environmental Science (Centrum voor Milieuwetenschappen)
CNDP	French National Public Debate Commission (French: Commission nationale du débat public).
CNNC	China National Nuclear Corporation
CNS	Convention of Nuclear Safety
COMEAP	Committee on the Medical Effects of Air Pollutants (UK)
CONSTOR	Cask for long term storage of spent fuel
COVRA	Waste Management Organisation of the Netherlands (Dutch: Centrale Organisatie Voor Radioactief Afval)
CPPNM	Convention on the Physical Protection of Nuclear Material
CSA	Coordination and Support Action
CSP	Concentrating Solar Power
CT	Computerised Tomography
DALY	Disability Adjusted Life Years
DCB	Dichlorobenzene
DEVCO	Directorate-General for International Cooperation and Development
DG	Directorate-General
DGR	Deep Geological Repository
DNSH	Do No Significant Harm
DOE	Department of Energy (USA)
DOS	Safety Options File (French: dossier d'options de sûreté)
DU	Depleted Uranium
EBRD	European Bank for Reconstruction and Development
EBS	Engineered Barrier System
EC	European Commission
ECURIE	European Community Urgent Radiological Information Exchange
EERA	European Energy Research Alliance
EF	Ecological Footprint
EFR	European Fast Reactor
EHS	Environment, Health and Safety
EIA	Environmental Impact Assessment
EIO	Economic Input-Output
ENER	Directorate-General for Energy

ENRESA	Waste Management Organisation of Spain (Spanish: Empresa Nacional de Residuos Sociedad Anónima)
ENSI	Swiss Federal Nuclear Safety Inspectorate (German: Eidgenössische Nuklearsicherheitsinspektorat)
ENV	Directorate-General for the Environment
EP	Eutrophication Potential
EPA	Environmental Protection Agency (USA)
EPD	Environmental Product Declaration
EPR	European Pressurised-water Reactor
EPREV	Emergency Preparedness Review
EPS	Environmental Priority Strategies
ERDO	European Repository Development Organisation
ESCP	Extended Storage Collaboration Program
ESG	Environmental, Social and Governance
EU	European Union
EUCO	EUCO scenarios: A set of policy scenarios designed to achieve EU climate and energy targets.
EURAD	European Joint Programme on Radioactive Waste Management
EURDEP	European Radiological Data Exchange Platform
FA	Fuel Assembly
FAETP	Freshwater Aquatic Eco-toxicity Potential
FA-VL	French radioactive waste classification, equivalent to English LLW-LL (French: Déchets de Faible Activité à Vie Longue)
FBR	Fast Breeder Reactor
FEP	Features, Events and Processes
FISMA	Directorate-General for Financial Stability, Financial Services and Capital Markets Union
FMA-VC	French radioactive waste classification, equivalent to English LILW-SL (French: Déchets de Faible et Moyenne Activité à Vie Courte)
FP	Fission Product
GDR	German Democratic Republic
GES	Good Environmental Status
GHG	Greenhouse Gas
GIIP	Good International Industry Practice
GNS	Gesellschaft für Nuklear-Service mbH
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
GW	Gigawatt
GWP	Global Warming Potential
HA	French radioactive waste classification, equivalent to English HLW (French: Déchets de Haute Activité)

HABOG	Interim storage facility for spent fuel and high level waste of the Netherlands (Dutch: Hoogradioactief Afval Behandelings- en OpslagGebouw)
HCL	Hydrochloric Acid
HEPA	High Efficiency Particulate Air
HEU	Highly Enriched Uranium
HF	Hydrofluoric Acid
HLW	High-Level Waste
HM	Heavy Metal
HTP	Human Toxicity Potential
IAEA	International Atomic Energy Agency
ICMM	International Council on Mining and Metals
ICRP	International Commission on Radiological Protection
IDR	Integrated Dry Route process
IEA	International Energy Agency
IED	Industrial Emissions Directive
IFC	International Finance Corporation
IGD-TP	Implementing Geological Disposal Technology Platform
ILCD	International Life Cycle Data
ILW	Intermediate-Level Waste
ILW-LL	Intermediate-Level Waste - Long-Lived
ILW-SL	Intermediate-Level Waste - Short-Lived
IPA	Impact Pathway Approach
IPCC	Intergovernmental Panel on Climate Change
IPPAS	International Physical Protection Advisory Service
IRMIS	International Radiation Monitoring Information System
IRRS	Integrated Regulatory Review Service
ISL	In-Situ Leaching
ISO	International Organization for Standardization
ISR	In Situ Recovery
JPNM	Joint Programme on Nuclear Materials
JRC	Joint Research Centre
KBS	A concept for high-level radioactive waste disposal developed in Sweden (Swedish: kärnbränslesäkerhet)
KOH	Potassium Hydroxide
KPI	Key Performance Indicators
LC	Life Cycle
LCA	Life Cycle Analysis
LCI	Life Cycle Inventory

LCIA	Life Cycle Impact Assessment
LET	Linear Energy Transfer
LFR	Lead-cooled Fast Reactor
LILW	Low- and Intermediate-Level Waste
LILW-LL	Low- and Intermediate-Level Waste - Long-Lived
LILW-SL	Low- and Intermediate-Level Waste - Short-Lived
LL	Long-Lived (ILW-LL: Intermediate-Level Waste - Long-Lived)
LLW	Low-Level Waste
LLW-LL	Low-Level Waste - Long-Lived
LTE	Lifetime Extension
LTO	Long Term Operation
LUCAS	LCIA method Used for a CAnadian-Specific context
LWGR	Light Water Graphite Reactor
LWR	Light Water Reactor
MA	Minor Actinide
MA-VL	French radioactive waste classification, equivalent to English ILW-LL (French: Déchets de Moyenne Activité à Vie Longue)
MCPD	Medium Combustion Plant Directive
MIMAS	MIcronized MASter blend
MLA	Swedish near-surface repositories for low level waste (Swedish: Markdeponi Lågaktivt Avfall)
MMB	Sweden Environmental Board
MOX	Mixed Oxide
MS	Member State
MTR	Materials Testing Reactor
MW	Megawatt
NACE	Statistical classification of economic activities in the EC (Nomenclature statistique des Activités économiques dans la Communauté Européenne)
NAGRA	Waste Management Organisation of Switzerland (German: Nationale Genossenschaft für die Lagerung radioaktiver Abfälle)
NCDA	No Comparable Data Available
NDA	Nuclear Decommissioning Authority (UK)
NE	Nuclear Energy
NEA	Nuclear Energy Agency (of the OECD)
NEEDS	New Energy Externalities Developments for Sustainability (project of the EU's 6th Framework Programme for Research and Technological Development - FP6)
NEI	Nuclear Energy Institute
NIRAS	Belgian National Agency for Radioactive Waste and enriched Fissile Material (Flemish: Nationale Instelling voor Radioactief Afval en verrijkte Splijtstoffen). See also ONDRAF
NL	Netherlands

NMHC	Non-methane hydrocarbon
NMVOC	Non-methane volatile organic compounds
NNWS	Non-Nuclear Weapon States
NORM	Naturally Occurring Radioactive Materials
NPP	Nuclear Power Plant
NRC	The U.S. Nuclear Regulatory Commission
NSD	Nuclear Safety Directive
NSS	Nuclear Security Series
NWS	Nuclear Weapon States
ODP	Ozone Depletion Potential
OECD	Organisation for Economic Cooperation and Development
ONDRAF	Belgian National Agency for Radioactive Waste and enriched Fissile Material (French: Organisme National des Déchets Radioactifs et des matières Fissiles enrichies). See also NIRAS
OSART	Operational Safety Review Team
OSPAR	The Oslo/Paris (OSPAR) convention: Convention for the Protection of the Marine Environment of the North-East Atlantic
OTC	Once Through Cycle
PAH	Polyaromatic hydrocarbons
PCF	Product Carbon Footprint
PHWR	Pressurised Heavy Water Reactor
PM	Particulate Matter
PNGMDR	Plan national de gestion des matières et déchets radioactifs
POCP	Photochemical Oxidant Creation Potential
POSIVA	Finnish Company for the disposal of radioactive waste
PPE	Personal Protective Equipment
PRIMES	Price-Induced Market Equilibrium System
PSA	Probabilistic Safety Assessment
PSI	Paul Scherrer Institute
PSR	Periodic Safety Review
PURAM	Waste Management Organisation of Hungary (Public Limited Company for Radioactive Waste Management)
PUREX	Plutonium Uranium Extraction
PV	Photovoltaic
PWR	Pressurised Water Reactor
RA	Radioactive
RAMON	Reference and management of nomenclatures database
RBBA	Spanish radioactive waste classification, equivalent to English VLLW (Spanish: Residuos de muy baja actividad)

RBMK	High-Power Channel-type Reactor (Russian: реактор большой мощности канальный - Reaktor Bolshoy Moshchnosti Kanalnyy)
RD&D	Research, development and demonstration
RDD	Research, development and demonstration
REPA	Resource and Environmental Profile Analysis
REPU	Reprocessed Uranium
RHWG	Reactor Harmonisation Working Group
RK&M	Records, Knowledge and Memory
RN	Radionuclide
RTD	Directorate-General for Research and Innovation
RW	Radioactive Waste
RWM	Radioactive Waste Management
SALTO	Safety Aspects of Long Term Operation
SBO	Station Blackout
SCC	Siberian Chemical Combine
SDG	Sustainable Development Goal
SEA	Strategic Environmental Assessment
SETAC	Society of Environmental Toxicology and Chemistry
SF	Spent Fuel
SFR	Final Repository for Short-Lived Radioactive Waste (Swedish: Slutförvaret för kortlivat radioaktivt avfall)
SKB	Swedish Nuclear Fuel and Waste Management Co (Svensk Kärnbränslehantering AB)
SL	Short-Lived (ILW-SL: Intermediate-Level Waste - Short-Lived)
SOARCA	State-of-the-art reactor consequence analyses
SOM	Soil Organic Matter
SPAR	Coordinated Research Project: Spent Fuel Performance Assessment and Research
SR	Safety Report
SRA	Strategic Research Agenda
SSM	Swedish Radiation Safety Authority (Swedish: Strålsäkerhetsmyndigheten)
STUK	Finnish Radiation and Nuclear Safety Authority (Finnish: Säteilyturvakeskus)
SW	Special Waste
SWU	Separative Work Unit
TEG	Technical Expert Group
TENORM	Technologically Enhanced Naturally Occurring Radioactive Material
TETP	Terrestrial Eco-toxicity Potential
TFA	French radioactive waste classification, equivalent to English VLLW (French: Déchets de Très Faible Activité)
THORP	Thermal Oxide Reprocessing Plant (Sellafield, UK)
TP	Technology Platform

TR	Taxonomy Regulation
TRACI	Tool for Reduction and Assessment of Chemicals and Other Environmental Impacts
TRL	Technology Readiness Levels
TSC	Technical Screening Criteria
TSO	Technical Support Organisation
TTC	Twice Through Cycle
TW	Technological Waste
UAE	United Arab Emirates
UG	Underground
UK	United Kingdom
UN	United Nations
UNCLOS	United Nations Convention on the Law of the Sea
UNCPC	United Nations Central Product Classification
UNECE	United Nations Economic Commission for Europe
UNEP	United Nations Environment Programme
UNESCO	United Nations Educational, Scientific and Culture Organisation
UNL	Uranyl Nitrate Liquor
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
UOC	Uranium Ore Concentrates
UOX	Uranium Oxide
URL	Underground Research Laboratories
US	United States (of America)
USA	United States of America
USEC	United States Enrichment Corporation
USNRC	The U.S. Nuclear Regulatory Commission
VLLW	Very Low-Level Waste
VOC	Volatile Organic Compounds
VVER	Water-Water Power Reactor (Russian: водо-водяной энергетический реактор - Russian PWR)
WENRA	Western European Nuclear Regulators Association
WFD	Water Framework Directive
WG	Working Group
WMO	Waste Management Organisation
WNA	World Nuclear Association
WOMARS	Worldwide Marine Radioactivity Studies
WTP	Willingness To Pay
YVL	Finnish Regulatory Guides on nuclear safety (Finnish: Ydinturvallisuusohjeet)

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Annex 1. Legal and regulatory background of nuclear energy

1. Introduction

The international community has agreed and implemented several international treaties in order to ensure that the benefits of nuclear energy, like electricity production (with low-carbon footprint), medical diagnosis and industrial/agricultural uses, can be realised while the risks that it poses to human health and the environment are controlled and maintained within acceptable levels. Therefore countries developed an international legal framework for conducting activities related to nuclear energy and ionizing radiation in a way that adequately protects individuals, property and the environment [A1-1].

International treaties are international binding instruments, creating rights and obligations for States. The States are bound by them and must implement their provisions. Should a disagreement occur between States, they can resort to different dispute resolution mechanisms, varying from consultations and arbitrations, to the submitting of disputes to the International Court of Justice, or the International Tribunal for the Law of the Sea.

Such international treaties are reflected in national legislation. General international law also applies to the uses of nuclear energy and ionizing radiation. For instance, the law of the sea covers sea transports including ships carrying nuclear materials, and environmental law covers industrial activities including nuclear activities. However, because of the specificity of nuclear technology, the international community concluded that in some instances international law did not provide sufficient rules and agreed on nuclear law treaties like the Convention on Nuclear Safety.

Many countries also adopted international guidelines and standards like the recommendations of the International Commission on Radiological Protection, the Code of Conduct on the Safety of Radioactive Sources, the International Basic Safety Standards and other International Atomic Energy Agency (the IAEA) Safety Standards. These are not binding, but they are internationally recognized principles which can be implemented at the level of national law. Such guidelines, when taking the form of domestic laws become binding at the national level.

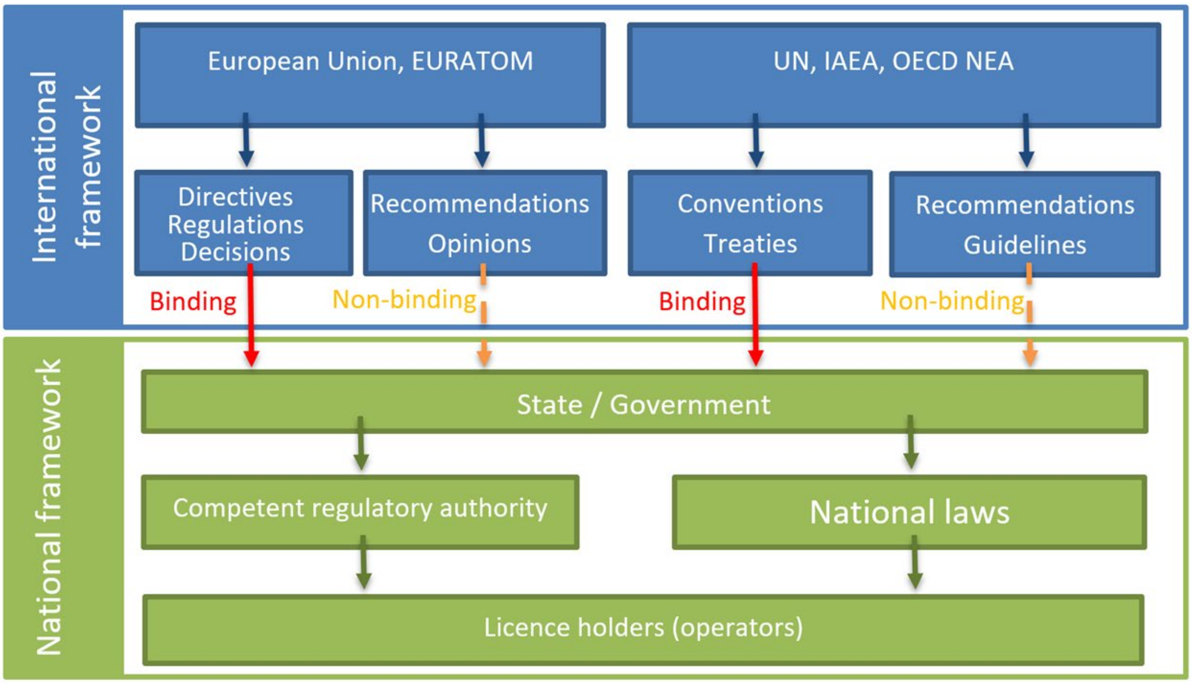
International nuclear law which includes both binding and non-binding instruments is comprehensive: it covers radiation protection, nuclear and radiation safety (radioactive sources and radioactive material, nuclear facilities, emergency preparedness and response, mining and milling, transport of radioactive material, radioactive waste and spent fuel), nuclear liability, and non-proliferation and physical protection (safeguards, export and import controls, physical protection). International nuclear law evolves as it is updated and revised based on lessons learned, best practice and technology development.

At the European Union (EU) level, Member States have agreed on a set of rules reflecting international treaties to which they are party. EU law can go beyond the requirements of the international treaties, such as the Euratom spent fuel and Radioactive Waste Directive providing more obligations (than the IAEA Joint Convention) for Member States, e.g. the establishment of the national programme for waste management. The EU nuclear legal framework is binding on all Member States; it is enforceable and Member States can be sanctioned for non-compliance. The Court of Justice of the EU has also issued judgements that provide further information on the interpretation and application of Euratom rules. In addition, certain EU law provisions and pieces of legislation that are not part of the EU nuclear legal framework, nevertheless also apply to the nuclear sector, e.g. environmental legislation.

At the national level, nuclear law can be a standalone law or included and covered by other laws primarily relating to subjects such as environmental protection, industrial safety, land use planning, administrative procedure, mining, transport, government ethics and electricity rate regulation. All EU Member States have established competent regulatory authorities to supervise nuclear activities. These authorities have a right to inspect and control the application of nuclear safety rules. The responsibility to ensure nuclear safety and security remains with each Member State and this responsibility is conferred on the licensees of nuclear installations.

This annex provides a non-exhaustive overview of the legal and regulatory background of nuclear energy. It describes the legal instruments of relevance to this report.

Figure A.1-1. International and national legal framework



Source: JRC

2. International agreements, standards and tools

International treaties are binding legal instruments, creating rights and obligations for States. This chapter provides a summary of various relevant multilateral treaties.

2.1. Multinational agreements – UN treaties

2.1.1. United Nations Convention on the Law of the Sea (UNCLOS)

The 1982 United Nations Convention on the Law of the Sea (UNCLOS) is an international agreement establishing rules governing all uses of the seas and oceans as well as their resources. It came into force in 1984 and is relevant to all States since all, even land-locked States, have the right of access to the sea and enjoy freedom of transit through the territory of transit States. Some of the features of the convention include prevention and control of marine pollution, liability for damage caused by violation of the international obligations to combat such pollution, and freedoms of navigation, overflight, scientific research and fishing on the high seas. All EU Member States and the EU are party to UNCLOS. The convention covers traffic management of nuclear-powered ships and ships carrying nuclear substances or materials in the territorial sea.[141]

2.1.2. Convention for the Protection of the Marine Environment of the North-East Atlantic (OSPAR Convention)

The Convention for the Protection of the Marine Environment of the North-East Atlantic (OSPAR Convention) was open for signature at the Ministerial Meeting of the Oslo and Paris Commissions in Paris on 22 September 1992. It entered into force on 25 March 1998.

The convention has been signed and ratified by all of the Contracting Parties to the original Oslo or Paris Conventions (Belgium, Denmark, the European Union, Finland, France, Germany, Iceland, Ireland, the Netherlands, Norway, Portugal, Spain, Sweden and the United Kingdom of Great Britain and Northern Ireland) along with Luxembourg and Switzerland. Its aim is to prevent and eliminate pollution of the maritime area.

The dumping of all wastes or other matter is prohibited, including the dumping of low and intermediate level radioactive substances, including wastes.

141 https://www.un.org/Depts/los/convention_agreements/convention_overview_convention.htm

2.1.3. Convention on the Protection of the Marine Environment of the Baltic Sea Area (Helsinki Convention)

The Helsinki Convention [A1-50] seeks to prevent and eliminate pollution of the marine environment of the Baltic Sea Area caused by harmful substances from all sources including radioactive substances and radioactive waste. It also commits the signatories to take measures on conserving habitats and biological diversity and for the sustainable use of marine resources. The original Convention was signed in Helsinki on 22 March 1974, and entered into force on 3 May 1980. It was then updated in 1992. There are currently ten Contracting Parties: Germany, Denmark, Estonia, the European Union, Finland, Lithuania, Latvia, Poland, Russia, and Sweden[142]

2.1.4. Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention and Protocol)

The Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter ("London Convention") was adopted in 1972 and entered into force in 1975. In 1996 it was updated by the London Protocol that came into force on 24 March 2006. Twenty one (21) Member States[143] are parties to the London Convention and thirteen (13)[144] are parties to the London Protocol.

The convention is aimed at controlling all sources of pollution of the marine environment, especially to prevent any deliberate disposal at sea of wastes that could create hazards to human health, to harm living resources and marine life, and to damage amenities or to interfere with other legitimate uses of the sea. Only deliberate disposal of wastes from vessels, aircraft, platforms or other man-made structures at sea is covered by the convention.

The London Convention includes Annex I listing all the wastes and other matter the dumping of which is prohibited. High level waste and other high level radioactive material are included in this annex. Annex II to the convention lists all the wastes and other matter that can be dumped but require a special permit issued by the Contracting Parties. This list includes all the radioactive waste and other radioactive matter not covered by Annex I. Although the London convention introduced some limitations to the dumping of radioactive waste, it was not completely banned and was still practiced by some States [A1-2]. Efforts to completely ban sea dumping of radioactive wastes continued and in November 1993 the London Convention was amended by listing all radioactive wastes and radioactive material in Annex I. By this amendment, the London convention effectively introduced prohibition of the dumping of any radioactive waste into the marine environment.

With the adoption of the London Protocol a different approach has been taken towards the listing of prohibited materials – the Contracting Parties were obliged to prohibit dumping of any waste that is not listed in the Annex I of the protocol (so called "reverse list") and Annex II was abandoned.

This annex covers some of the international agreements seeking to protect the marine environment. There are other agreements such as the Barcelona Convention for the Protection of the Marine Environment and the Coastal Region of the Mediterranean, and the Convention on the Protection of the Black Sea Against Pollution.

2.1.5. Aarhus Convention

The United Nations Economic Commission for Europe (UNECE) Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters was adopted on 25 June 1998 in the Danish city of Aarhus (Århus) at the Fourth Ministerial Conference as part of the "Environment for Europe" process. It entered into force on 30 October 2001. All EU Member States and European Union are Contracting Parties to the convention.

The Aarhus Convention establishes a number of rights of the public (individuals and their associations) with regard to the environment. The Parties to the convention are required to make the necessary provisions to ensure that public authorities at national, regional or local level adopt measures that allow the public to exercise these rights. The convention provides for:

[142] Ratification status as of February 2021 (https://helcom.fi/about-us/contracting-parties/)

[143] Belgium, Bulgaria, Croatia, Cyprus, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Ireland, Italy, Luxembourg, Malta, Netherlands, Poland, Portugal, Slovenia, Spain, Sweden (ratification status: October 2020)

[144] Belgium, Bulgaria, Denmark, Estonia, France, Germany, Ireland, Italy, Luxembourg, Netherlands Slovenia, Spain, Sweden (Ratification status date: October 2020)

- the right of everyone to receive environmental information that is held by public authorities ("access to environmental information"). This can include information on the state of the environment, but also on policies or measures taken, or on the state of human health and safety where this can be affected by the state of the environment. Applicants are entitled to obtain this information within one month of the request and without having to say why they require it. In addition, public authorities are obliged, under the convention, to actively disseminate environmental information in their possession;
- the right to participate in environmental decision-making. Arrangements are to be made by public authorities to enable the public affected and environmental non-governmental organisations to comment on, for example, proposals for projects affecting the environment, or plans and programmes relating to the environment. These comments are to be duly taken into account in decision-making, and information is to be provided on the final decisions and the reasons for them ("public participation in environmental decision-making");
- the right to review procedures to challenge public decisions that have been made without respecting the two aforementioned rights or environmental law in general ("access to justice").

2.1.6. Convention on Environmental Impact Assessment in a Transboundary Context (Espoo Convention)

The United Nations Economic Commission for Europe (UNECE) Convention on Environmental Impact Assessment in a Transboundary Context (Espoo Convention) was adopted in 1991 and entered into force on 10 September 1997. The convention was amended twice, in 2001 (entered into force on 26 August 2014) and in 2004 (entered into force on 23 October 2017). In 2003, the Protocol on Strategic Environmental Assessment to the convention on Environmental Impact Assessment in a Transboundary Context (Kyiv protocol to Espoo Convention) has been added to the Espoo convention. The Espoo Convention counts 45 Parties, including the European Union and its Member States [A1-3]

The Espoo convention aims at stepping up international cooperation in order to prevent, reduce and control the adverse transboundary impact of certain activities on the environment with a view to ensuring ecologically sound and sustainable development. Each Contracting Party must take the necessary legal, administrative or other measures, establish an environmental impact assessment (EIA) procedure and prepare the environmental impact assessment documentation. The convention obliges Parties to assess the environmental impact of certain activities at an early stage of planning as well as to notify and consult each other on all major projects under consideration that are likely to have a significant adverse environmental impact across boundaries. This is a key international instrument that aims to ensure that Contracting Parties shall take all appropriate and effective measures to prevent, reduce and control significant adverse transboundary environmental impact from proposed activities.

A Protocol on the assessment of the environmental impact of strategic decisions (Strategic environmental assessment - SEA) was signed by the EC on 21 May 2003 and approved on 12 November 2008. This Protocol requires its Parties to evaluate the environmental consequences of their official draft plans and programmes. SEA is undertaken much earlier in the decision-making process than EIA - it is therefore seen as a key tool for sustainable development.

During the 8th session of the Meeting of the Parties to the Espoo Convention and the 4th session of the Meeting of the Parties to the SEA Protocol (8-11 December 2020) the Guidance on the applicability of the Convention to the lifetime extension of nuclear power plants has been adopted. It intends to help the Contracting Parties planning to extend the lifetime of nuclear power plants to deal with complex issues associated with the related transboundary environmental impact assessment procedure [A1-4].

2.1.7. The Convention on Biological Diversity

Known informally as the Biodiversity Convention, the Convention on Biological Diversity is a multilateral treaty that has three main goals: the conservation of biological diversity; the sustainable use of its components; and the fair and equitable sharing of benefits arising from genetic resources.

Each Contracting Party to the convention is required to develop national strategies, plans or programmes for the conservation and sustainable use of biological diversity. As part of the monitoring and identification process, the Contracting Parties are required to identify processes and categories of activities which have or are likely to have significant adverse impacts on the conservation and sustainable use of biological diversity, and monitor their effects through sampling and other techniques. Appropriate procedures should be introduced requiring environmental impact assessment for the proposed projects that are likely to have significant adverse effects on biological diversity, and where appropriate allowing for public participation in such procedures.

More details on the link between the Biodiversity Convention and the Espoo Convention can be found in [A1-5].

2.2. Multilateral agreements – IAEA Conventions

The International Atomic Energy Agency (IAEA) is the central intergovernmental organisation and forum for scientific and technical cooperation in the nuclear field at global level. The Director General of the IAEA serves as depository for several important international conventions, forming part of the international nuclear legal framework. Euratom and EU Member States are Contracting Parties to many important conventions in the area of nuclear energy:

- in the area of nuclear installation safety, the Convention on Nuclear Safety;
- in the area of radioactive waste management, the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management;
- in the area of emergency preparedness, the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency;
- in the area of physical protection of nuclear material, the Convention on the Physical Protection of Nuclear Material and its amendment;
- in the area of nuclear liability, the Vienna Convention on Civil Liability for Nuclear Damage, its amending Protocol, the Convention on Supplementary Compensation, and the Joint Protocol establishing treaty relations between the Contracting Parties to the Vienna and the OECD Paris conventions.

In the area of nuclear non-proliferation, all EU Member States ratified the UN Treaty on the Non-proliferation of nuclear weapons, known as the Non-Proliferation Treaty, and have relevant IAEA Safeguards Agreements in force. The Safeguards agreements establish a system of accountancy and control ensuring that the use of nuclear materials is not diverted from its original peaceful purpose. As this report focuses on the effects originating from the authorised use of radioactive materials in the nuclear fuel cycle, the nuclear safeguards' legal framework is only briefly described here. The EU has its own supranational safeguards system based on Chapter 7 of the Euratom Treaty which entrusts the European Commission, the executive body of the EU, with the responsibility to verify the management of nuclear materials and special fissile materials, including fully independent inspection capabilities. The European Commission has sole competence for Euratom nuclear safeguards. Users and holders of nuclear material in the EU are obliged to keep records and to declare all flows of these materials to the Commission. Euratom Safeguards are fully implemented in both EU Non-Nuclear Weapon States (NNWS) and EU Nuclear Weapon States (NWS), which co-exist in the Union. Verification agreements have been established with the IAEA (INFCIRC/193 for all EU NNWS, INFCIRC/263 for the UK and INFCIRC/290 for France), allowing the IAEA to verify that all EU Member States comply with their commitments as Contracting Party to the Non-Proliferation Treaty.

2.2.1. Convention on Nuclear Safety

The Convention on Nuclear Safety (CNS) is an international treaty that was adopted on 17 June 1994 and entered into force on 24 October 1996. All EU Member States and Euratom are Contracting Parties to the Convention.

The Convention seeks to:

- achieve and maintain a high level of nuclear safety through the enhancement of national measures and technical co-operation;
- establish and maintain effective defences against radiological hazards in nuclear installations in order to protect people and the environment, etc.;
- prevent nuclear accidents and limit their consequences.

The Convention does not lay down detailed safety standards but represents a commitment to the application of fundamental safety principles for nuclear installations – that are defined as land-based civil nuclear power plants including facilities for storage, handling and treatment of radioactive materials that are on the same site and are directly related to the operation of the nuclear power plant[145].

The Parties to the Convention are committed to establishing a legislative, regulatory and administrative framework to ensure the safety of nuclear installations, which provides for:

- the establishment of sufficient national safety requirements and regulations;
- a system for licensing nuclear installations and the prohibition of operating without a licence;
- a system of inspection and assessment. Comprehensive and systematic assessments shall be carried out before the construction and commissioning of an installation and throughout its life;
- measures to enforce the regulations and the terms of licensing (suspension or revocation of licences, etc.).

The main elements envisaged in the Conventions to ensure the safety of installations are:

- The Parties must set up an independent regulatory body to grant licences and to ensure that the regulations are correctly implemented. This body must be effectively separated from those of any other organisation whose task is to promote or use nuclear energy.
- The regulatory body is in charge of granting licences to nuclear installations for each phase of the life of the installation. The convention specifies assessment criteria for each phase: siting, design and construction, and operation.
- In choosing the site, consideration must, among other things, be given to its effect on the safety of the installation and the effects of the installation on individuals and the environment. Appropriate measures should be taken in order for other Contracting Parties in the vicinity of the site to be consulted if the installation is likely to have consequences for them.
- Regarding design and construction, several reliable levels and methods of protection i.e. defence in depth must be put in place against the release of radioactive materials to prevent the occurrence of accidents and to mitigating their radiological consequences. The techniques and equipment used must be proven by experience or testing.
- Authorisation to operate a nuclear installation is based on safety analysis and a commissioning programme. The operation and management of the installation must conform to the regulations established by the national authorities. Programmes to collect and analyse data must also be introduced. The generation of radioactive waste resulting from the operation shall be kept to the minimum practicable for the process concerned.
- Licence holders must establish policies prioritising safety and must draw up a quality assurance programme to ensure that the requirements are met.
- Each installation must also have on-site and off-site emergency plans to protect workers, the general public, the environment, etc. in the case of a radiological emergency.

The Vienna Declaration on Nuclear Safety in 2015 at a Diplomatic Conference [A1-6] lays down principles for the implementation of the objective of the Convention on Nuclear Safety to prevent accidents and mitigate radiological consequences:

- New nuclear power plants are to be designed, sited and constructed with the objective of preventing accidents and, should an accident occur, mitigating the consequences in order to avoid releases of radionuclides causing long-term off site contamination, early radioactive releases or radioactive releases large enough to require long-term protective measures and actions
- Existing nuclear installations must undergo regular comprehensive safety assessments
- IAEA Safety Standards and good practices from the Convention Review Meetings must be taken into account in national requirements and regulations
- Technical criteria and standards used by State Parties must be peer reviewed.

At least once every 3 years each party to the convention must submit to the other Parties a report on the measures that they have taken to meet their obligations under the convention, and those reports are

[145] As defined in Article 2 of the Convention.

reviewed during the regular review meetings of the Contracting Parties. Contracting Parties are encouraged to make public their national reports, questions and comments (or corresponding summaries) [A1-7]. However, this is an 'incentive' instrument (i.e. it incites action, greater effort) that has no enforcement mechanism, but relies peer pressure.

2.2.2. The Joint Convention on Safety of Spent Fuel and Radioactive Waste

The Joint Convention on Safety of Spent Fuel and Radioactive Waste (Joint Convention) was adopted on 5 September 1997 and entered into force on 18 June 2001. This is the first legally binding instrument devoted to the safe management of radioactive waste and spent fuel on a global scale. All EU Member States and Euratom are Contracting Parties to this convention which is relevant to all countries, even those with no nuclear power.

The Joint Convention seeks to:

- achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management, through the enhancement of national measures and international co-operation, including where appropriate, safety-related technical co-operation;
- ensure that during all stages of spent fuel and radioactive waste management there are effective defences against potential hazards so that individuals, society and the environment are protected from harmful effects of ionizing radiation, now and in the future, in such a way that the needs and aspirations of the present generation are met without compromising the ability of future generations to meet their needs and aspirations;
- prevent accidents with radiological consequences and to mitigate their consequences should they occur during any stage of spent fuel or radioactive waste management.

The Joint Convention applies to the radioactive waste (including planned and controlled discharges) and spent fuel resulting from the operation of civilian nuclear reactors or to the radioactive waste resulting from other civilian applications. Some of the spent fuel or radioactive waste is excluded from the scope of the Joint Convention, unless it is declared as spent fuel or radioactive waste for the purposes of this Convention by the Contracting Party (i.e. spent fuel or radioactive waste resulting from the military or defence programmes, waste that contains only naturally occurring radioactive materials and that does not originate from the nuclear fuel cycle).

The Parties to the Joint Convention are committed to establishing a legislative, regulatory and administrative framework for the safety of spent fuel and radioactive waste management to ensure adequate protection of individuals, society and the environment against radiological and other hazards and which provides for:

- The establishment of applicable national safety requirements and regulations for radiation safety;
- A licensing system for spent fuel and radioactive waste management activities and facilities;
- A system of appropriate institutional control, regulatory inspection and documentation and reporting as well as enforcement of applicable regulations and of the terms of licences;
- A clear allocation of responsibilities of the bodies involved in the different steps of spent fuel and of radioactive waste management;
- Appropriate siting, design and construction of facilities;
- Provisions to ensure safety of facilities during their operation and after their closure;
- Transboundary movement – each Contracting Party involved in transboundary movement shall ensure that such movements are undertaken in line with the provisions of the Joint Convention and relevant binding international instruments;
- Management of disused sealed radioactive sources (possession, remanufacturing or disposal), which should take place in a safe manner.

The Joint Convention not only requires protecting individuals, society and the environment from radiological hazards, but it also requires avoiding actions that impose reasonably predictable impacts on future generations greater than those permitted for the current generation (Art. 4(vi) and 11(vi)) as well as to avoid imposing undue burdens on future generations (Art. 4(vii) and 11(vii)). Moreover, articles 13 to 15 set out requirements concerning the safety of waste management facilities during the siting, design, construction, and operation phases. Some of those requirements relate to the long term safety, e.g. the requirement to evaluate all relevant site-related factors on the safety of a disposal facility after closure, and the requirement to evaluate the safety impact of a facility on individuals, society and the environment taking into account possible evolution of the disposal facility's site conditions after closure.

Article 17 of the Joint Convention lists institutional measures that have to be ensured after closure of the disposal facility. They include measures such as preservation of records (location, design and inventory), active or passive institutional controls (monitoring or access restrictions, if required), etc.

Additionally, article 22(iii) requires financial provisions that enable the appropriate institutional controls and monitoring arrangements to be continued for the period deemed necessary following the closure of a disposal facility.

Contracting Parties meet at least once every 3 years to report on measures they have taken to fulfil the treaty obligations. Contracting Parties are encouraged to make public their national reports, questions and comments (or corresponding summaries) [A1-8]. However, this is an 'incentive' instrument that has no enforcement mechanism, but relies peer pressure.

2.2.3. Convention on Early Notification of a Nuclear Accident

The Convention on Early Notification of a Nuclear Accident was adopted on 26 September 1986 and entered into force on 27 October 1986. The EU Member States (excl. Malta) and Euratom are Contracting Parties to this convention[146].

It is aimed at establishing a notification system of any accident resulting in a release of radioactive material or is likely to result in a release and which has resulted or might result in an international transboundary release that could be of radiological safety significance for another State. The convention applies to any nuclear reactor/fuel cycle facility, radioactive waste management facility, transport and storage of nuclear fuels or radioactive wastes, as well as to manufacture, use, storage, disposal and transport of radioisotopes (States may notify other nuclear accidents as well to minimize radiological consequences).

The Contracting Parties are required to notify affected States, directly or through the IAEA, of the nuclear accident, its nature, the time of its occurrence and its exact location where appropriate. Additionally, the affected States should be provided with any other information relevant to minimizing the radiological consequences in those States.

2.2.4. Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency

The Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency was adopted on 26 September 1986 and entered into force on 26 February 1987. The EU Member States (excl. Malta) and Euratom are Contracting Parties to the Convention[147].

The convention sets out an international framework enabling the Contracting Parties to cooperate between themselves and with the International Atomic Energy Agency to facilitate prompt assistance in the event of a nuclear accident or radiological emergency to minimize its consequences and to protect life, property and the environment from the effects of radioactive releases.

Any Contracting Party may call, directly or through IAEA, for assistance from any other Contracting Parties in the event of a nuclear accident or radiological emergency. By doing so, the Contracting Party has to specify the scope and type of assistance required and, where practicable, provide the assisting party with such information as may be necessary for that party to determine the extent to which it is able to meet the request. Each State Party to which a request for such assistance is directed shall promptly notify the requesting State Party, directly or through the Agency, whether it is in a position to render the assistance requested, and the scope and terms of the assistance that might be rendered.

2.2.5. Convention on the Physical Protection of Nuclear Material (CPPNM)

The Convention on the Physical Protection of Nuclear Material was adopted on 26 October 1979 and entered into force on 8 February 1987. It obliges Contracting Parties to ensure, during international transport, the protection of nuclear materials within their territory or on board their ships or aircrafts. It also obliges Contracting Parties to include a list of crimes in national legislation.[148] The convention was amended on 8 July 2005 with a view to strengthen its provisions and the amendment entered into the force on 8 May 2016. With

[146] Ratification status date: October 2020

[147] Status of ratification: October 2020

[148] 1: unlawful receipt, possession, use, transfer, dispersal of nuclear material causing or likely to cause death or damage to property; 2: theft; 3: illicit trade; 4: threat; 5: attempt; 6: participation in any of the above

the amendment, the scope of the convention is broadened. It includes physical protection requirements for nuclear facilities, and nuclear materials in domestic use, storage and transport; it expands existing offences and introduces the smuggling of nuclear materials and the threat of sabotage of a nuclear facility; it strengthens international cooperation; and it adds damage to the environment to the list of damages covered by the CPPNM. All EU Member States and Euratom are Contracting Parties to the CPPNM and to the CPPNM amendment.

Each State that is party to the convention and its amendment must establish and implement measures to guarantee this effective protection to prevent, in particular, the theft or disappearance of nuclear material for which it is responsible, as well as sabotage of nuclear facilities on its territory. The Euratom Treaty is broader in that it states that EU Member States must prevent any diversion of nuclear material to purposes other than those for which it is intended.

In implementing the convention and its amendment, the States that are party to it must respect a certain number of basic principles, in particular the principles of responsibility of the State and licence-holders, of a culture of security, insurance and confidentiality.

The contracting States must ensure that the nuclear material they import, export or accept in transit on their territory is protected in accordance with the applicable safety level.

The contracting States must designate a competent authority responsible for the application of the convention, as well as a point of contact, and give this information to the other signatory countries directly or through the intermediary of the International Atomic Energy Agency. Furthermore, they must cooperate in the event of theft, sabotage or risk of theft or sabotage. This cooperation in particular takes the form of an exchange of information, while respecting the confidentiality of this information vis-à-vis third parties.

The contracting States must apply appropriate penalties to certain infringements, in line with their severity. In particular, it is punishable to act without authorisation in a way that causes or is likely to cause death or serious injury, theft of nuclear material, sabotage of a nuclear installation, the threat of using nuclear material to cause death or serious injury of a third party or cause significant damage to property; attempts to commit one of these acts, involvement in such acts and organisation thereof are also punishable.

Any contracting State has jurisdiction for infringements committed on its territory or on board a vessel or aircraft registered in the said State and when the person presumed to have committed the infringement is a native of the said State. These infringements are grounds for extradition between the contracting States, who must also provide each other with the most extensive judicial assistance in the event of these infringements. Political motives for the infringement are not a reason for refusing extradition or mutual judicial assistance.

Since 2006, the IAEA has issued Nuclear Security Series publications to help States establish effective national nuclear security regimes. The IAEA Nuclear Security Guidance Committee, established in March 2012 and made up of Member States' representatives, reviews and approves draft publications in the Nuclear Security Series (NSS) as they are developed. The Nuclear Security Series are non-legally binding instruments. They provide international consensus guidance on all aspects of nuclear security to support States as they work to fulfil their responsibility for nuclear security.

These publications complement international legal instruments on nuclear security, such as the CPPNM and its Amendment, the International Convention for the Suppression of Acts of Nuclear Terrorism, the United Nations Security Council resolutions 1373 and 1540, and the IAEA Code of Conduct on the Safety and Security of Radioactive Sources.

Member States rely on the NSS for the effective implementation of their nuclear security regime. For the CPPNM's effective implementation, the main Nuclear Security Series (NSS) publications are the Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) (NSS 13) [A1-51], the Nuclear Security Recommendations on Radioactive Material and Associated Facilities (NSS 14) [A1-52], and the Nuclear Security Recommendations on Nuclear and Other Radioactive Material out of Regulatory Control (NSS 15) [A1-53].

In addition, the International Physical Protection Advisory Service, or IPPAS, assists Member States - on request - and helps strengthen nuclear security globally. The IAEA has worked together with Member States to improve the programme by creating modular services, by establishing a database of good practices identified during the advisory missions, and by taking into account changes in the nuclear security framework.

2.2.6. Conventions on the civil liability for nuclear damage

In the area of nuclear liability, several EU member States are Contracting Parties to the Vienna Convention on Civil Liability for Nuclear Damage, its amending Protocol, the Convention on Supplementary Compensation, and the Joint Protocol to the Vienna and the OECD Paris conventions. Further information on EU countries' participation to nuclear liability conventions is available in Chapter 2.3 below.

2.3. Multilateral agreements – OECD/NEA and IAEA nuclear liability conventions

In recognition of the potential magnitude and possible cross-border character of the damage to health, property, and the economy that a nuclear incident can cause, many States consented to develop treaty relations among themselves that would ensure adequate compensation for damage suffered by victims. Nuclear liability conventions not only protect victims by providing clarity in where to bring a claim and by providing efficient indemnification but also protect nuclear investors and suppliers from ruinous liability claims. This international approach is even more relevant when it comes to an incident that can have transboundary consequences. It allows victims to quickly make a claim and get harmonised compensation for the damage suffered whereas basic international law would be more time and money consuming and therefore would not benefit the victims. The existing conventions have undergone several additions, revisions and amendments in order to increase the compensation amounts available, the types of damage, the time to make a claim, and the types of victims who can make a claim.

There is currently no EU nuclear liability regime. EU Member States rely on the international nuclear liability conventions that they are party to. Under the OECD auspices, there are the 1960 Paris Convention, the 1963 Brussels Convention, and the 2004 Protocols amending the Paris and Brussels Convention that are not in force yet. Under the auspices of the IAEA, there are the 1963 Vienna Convention, the 1997 Protocol amending the Vienna Convention[149], and the Convention on Supplementary Compensation that came into force in 2015. There is also the 1988 Joint Protocol to the Vienna and the Paris conventions that eliminates conflicts that could otherwise arise from the simultaneous application of both Conventions to the same nuclear incident.

The liability regimes developed in parallel. The OECD Paris Convention covers most Western European countries and is open to all OECD Member States and to any non-member with the consent of the other members. Belgium, Denmark, Finland, France, Germany, Greece, Italy, the Netherlands, Portugal, Slovenia, Spain, and Sweden (12) are party to the Paris Convention[150]; the same countries except for Greece and Portugal are party to the Brussels Convention.[151] As for Austria and Luxembourg, both counties signed the Paris and Brussels Conventions but they have not ratified them. For instance, Luxembourg adopted a new domestic law in the field of nuclear liability because it is more protective than the international regime.[152]

The rest of the EU Member States are party to the IAEA conventions, which are open to any state. Bulgaria, Croatia, Czech Republic, Estonia, Hungary, Latvia, Lithuania, Poland, Romania and Slovakia (10) are members of the Vienna Convention. Most of them are also members of the revised Vienna Convention.[153] Romania is also party to the latest Convention on Supplementary Compensation.[154]

Whatever convention a country is party to, the nuclear liability principles included in the conventions and implemented at the national level are the same. Liability and compensation for damage occur in the case of a nuclear incident at a nuclear installation or during the transport of nuclear substances, involving nuclear fuels, radioactive products or waste. The liability of the operator is strict (i.e. victims do not need to prove fault or negligence), exclusive, which means that the liability is channelled to the operator (e.g. suppliers of goods are protected), limited to a specific amount, and limited in time (10-30 years after the accident). The operator must be financially sound in order to provide liability amount when needed (e.g. via an insurance pool). In addition, the international conventions bring uniformity in the sense that victims are treated equally and that the national court who is competent is the one where the incident occurred, or if unclear, the one from the country where the liable operator is located.

The majority of EU Member States is bound by one of the existing nuclear liability conventions or have a domestic law in place.[155] Even though the international regime provides for harmonisation, at the national level, State Parties can decide to apply stricter rules than those in the conventions. For instance, Germany

[149] also called revised Vienna Convention
[150] https://www.oecd-nea.org/jcms/pl_20196/paris-convention-on-third-party-liability-in-the-field-of-nuclear-energy-paris-convention-or-pc (ratification status date: October 2020)
[151] https://www.oecd-nea.org/jcms/pl_20318/brussels-supplementary-convention-to-the-paris-convention-brussels-supplementary-convention-or-bsc (status date: October 2020)
[152] https://gouvernement.lu/fr/actualites/toutes_actualites/articles/2018/01-janvier/12-responsabilite-nucleaire.html
[153] Bulgaria, Croatia, Estonia, and Slovenia are not members of the revised convention yet (status date: October 2020)
[154] https://www.iaea.org/topics/nuclear-liability-conventions
[155] Malta, Ireland, Cyprus, Austria, and Luxembourg are not part of any yet (status date: October 2020).

opted for the unlimited liability of the operator. Discussions at the EU level are still ongoing regarding the development of an EU nuclear liability regime.

2.4. International non-legally binding instruments, standards, guidance documents and tools

Complementary to the treaties and conventions that are legally binding on their State Parties, there are international recommendations that are not legally binding but that are internationally accepted. In general, treaties and conventions establish main principles whereas recommendations like the IAEA Safety Standards describe detailed safety measures that can help State Parties implement the conventions they ratified.

2.4.1. IAEA Codes of Conduct

2.4.1.1. Code of Conduct on the Safety and Security of Radioactive Sources

There is no convention on the safety and security of radioactive sources. Therefore the Code of Conduct is the only international instrument covering this topic [A1-9]. The Code is a non-legally binding international instrument and helps to ensure that radioactive sources are used within an appropriate framework of radiation safety and security. All EU Member States made a political commitment to implement this Code.

Two documents supplement the Code. The Guidance on the Import and Export of Radioactive Sources aims to provide for an adequate transfer of responsibility when a source is being transferred from one State to another [A1-10]. The Guidance on the Management of Disused Radioactive Sources provides further guidance regarding the establishment of a national policy and strategy for the management of disused sources, and on the implementation of management options such as recycling and reuse, long term storage pending disposal and return to a supplier [A1-11]. International meetings for the exchange of experience on the implementation of the Code and the Guidance documents occur every three years. The first meeting on the Management of Disused Radioactive Sources is planned for 2020-2021.

2.4.1.2. Code of Conduct on the Safety of Research Reactors

Research reactors are not covered by the Convention on Nuclear Safety. Therefore this code is the only international instrument covering the safety of research reactors [A1-12]. This Code strengthens the international nuclear safety arrangements for civil research reactors and sets out parameters for the management of research reactor safety and provides guidance to governments, regulatory bodies and operating organizations for the development and harmonization of the relevant policies, laws and regulations. International meetings on the application of the Code of Conduct on the Safety of Research Reactors occur every three years.

2.4.2. IAEA Safety Standards

The IAEA Safety Standards provide the fundamental principles, requirements and recommendations to ensure nuclear safety. They are in principle not binding but serve as a global reference for protecting people and the environment and contribute to a high level of safety.

Activities such as the operation of nuclear installations, the production, transport and use of radioactive material, the management of radioactive waste and the medical uses of radiation are subject to standards of safety.

The Safety Standards consists of three sets of publications categorized into:

- Fundamental safety principles, stating the basic objective, concepts and principles of safety;
- Safety Requirements, establishing the requirements that must be fulfilled to ensure safety; and
- Safety Guides, recommending measures for complying with these requirements for safety.

The IAEA Safety Standards Series are subdivided into General Safety Requirements and General Safety Guides (GSR and GSG), which are applicable to all types of facilities and activities, and Specific Safety Requirements and Specific Safety Guides (SSR and SSG), which are for application in particular thematic areas.

IAEA Fundamental Safety Principles

The fundamental safety objective — to protect people and the environment from harmful effects of ionizing radiation — applies to all circumstances that give rise to radiation risks. The safety principles are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes, and to protective actions to reduce existing radiation risks. They provide the basis for requirements and measures for the protection of people and the environment against radiation risks and for the safety of facilities and activities that give rise to radiation risks, including, in particular, nuclear installations and uses of radiation and radioactive sources, the transport of radioactive material, and the management of radioactive waste [A1-13].

As regards the long-term management of spent fuel and radioactive waste, the IAEA Safety Fundamentals provide that the responsibility for safety of licensees and regulators must be fulfilled not only in relation to present operations but also to future operations (para 3.7). Moreover, protection of future generations is among the main safety principles and it is stated that subsequent generations have to be adequately protected without any need for them to take significant protective actions (para 3.27). The IAEA Safety Fundamentals also stresses the importance of avoiding imposing an undue burden on future generations. The waste producers should not only keep generation of the radioactive waste to the minimum but also seek and apply safe, practicable and environmentally acceptable solutions for its long term management (para 3.29).

IAEA General Safety Requirements

The fundamental safety objective and principles are provided in the Safety Fundamentals IAEA SF-1 and they establish the basis for the safety requirements. The IAEA has published several general safety requirements documents as part of the IAEA Safety Standard Series that provide international consensus on requirements and cover various topics, such as:

- Governmental, Legal and Regulatory Framework for Safety (No. GSR Part 1 (Rev.1)). This document covers the essential aspects of the framework for establishing a regulatory body and taking other actions necessary to ensure the effective regulatory control of facilities and activities utilized for peaceful purposes [A1-14].
- Leadership and Management for Safety (No. GSR Part 2). This document defines requirements for establishing, assessing, sustaining and continuously improving effective leadership and management for safety in organisations concerned with, and facilities and activities that give rise to, radiation risks [A1-15].
- Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards (No. GSR Part 3). This document establishes requirements for the protection of people and the environment from harmful effects of ionizing radiation and for the safety of radiation sources [A1-16].
- Safety Assessment for Facilities and Activities (No. GSR Part 4 (Rev.1)). This document describes the generally applicable requirements to be fulfilled in safety assessments for facilities and activities, with special attention paid to defence in depth, quantitative analyses and the application of a graded approach to the range of facilities and activities that are addressed [A1-17].
- Predisposal Management of Radioactive Waste (No. GSR Part 5). This document presents requirements for the management of radioactive waste prior to its disposal and it provides the safety imperatives on the basis of which facilities can be designed, operated and regulated [A1-18].
- Decommissioning of Facilities (No. GSR Part 6). This document establishes requirements for the safe decommissioning of a broad range of facilities and it addresses all the aspects of decommissioning that are required to ensure safety [A1-19].
- Preparedness and Response for a Nuclear or Radiological Emergency (No. GSR Part 7). This document establishes the requirements for ensuring an adequate level of preparedness and response for a nuclear or radiological emergency, irrespective of its cause [A1-20].

The following subsections will primarily focus on the Safety Standards applicable to various installations of the nuclear fuel cycle and related topics, such as nuclear power plants or management of radioactive waste.

2.4.2.1 IAEA Safety Standards for the safety of nuclear power plants

<u>Design of Nuclear Power Plants (IAEA Safety Standards Series No. SSR-2/1 (Rev. 1))</u>

This publication establishes requirements applicable to the design of nuclear power plants and elaborates on the safety objective, safety principles and concepts that provide the basis for deriving the safety requirements that must be met for the design of a nuclear power plant. It is useful for organisations involved in design, manufacture, construction, modification, maintenance, operation and decommissioning of nuclear power plants, as well as for regulatory bodies [A1-21].

<u>Commissioning and Operation of Nuclear Power Plants (IAEA Safety Standards Series No. SSR-2/2 (Rev. 1))</u>

This standard describes the requirements to be met to ensure the safe commissioning, operation, and transition from operation to decommissioning of nuclear power plants including long term operation of nuclear power plants, plant ageing, periodic safety review, probabilistic safety analysis review and risk informed decision making processes [A1-22]

Site Evaluation for Nuclear Installations (IAEA Safety Standards Series No. SSR-1)

This publication establishes requirements and provides criteria for ensuring safety in the site evaluation for nuclear installations [A1-23].

2.4.2.2. IAEA Safety Standards for the management of radioactive waste

The IAEA has established Safety Standards, Fundamentals, Requirements and Guides applicable to the management of radioactive waste.

Disposal of Radioactive Waste (IAEA Safety Standards Series No. SSR-5)

This standard establishes requirements applicable to all types of radioactive waste disposal facility. It is linked to the fundamental safety principles for each disposal option and establishes a set of strategic requirements that must be in place before facilities are developed. Consideration is also given to the safety of existing facilities developed prior to the establishment of present day standards [A1-24]

Classification of Radioactive Waste (IAEA Safety Standards Series No. GSG-1)

This publication sets out a classification system for the management of waste prior to disposal and for disposal, driven by long term safety considerations. It includes a number of schemes for classifying radioactive waste that can be used to assist with planning overall national approaches to radioactive waste management and to assist with operational management at facilities [A1-25].

In addition, the IAEA has issued several specific safety guides concerning different types of disposal:

- SSG-23 Safety case and safety assessment for disposal of radioactive waste [A1-54];
- SSG-14 Geological disposal facilities for radioactive waste [A1-55];
- SSG-1 Borehole disposal facilities for radioactive waste [A1-56];
- SSG-29 Near surface disposal facilities for radioactive waste [A1-57];
- SSG-31 Monitoring and surveillance of radioactive waste disposal facilities [A1-58].

2.4.2.3 IAEA Safety Standards for Preparedness and Response for a Nuclear or Radiological Emergency (Series No. GSR Part 7)

This General Safety Requirements publication establishes the requirements for preparedness and response for a nuclear or radiological emergency and ensures an adequate level of preparedness and response for a nuclear or radiological emergency, irrespective of its cause. These Safety Requirements are intended to be used by governments, emergency response organizations, other authorities at the local, regional and national levels, operating organizations and the regulatory body as well as by relevant international organizations at the international level [A1-20]

2.4.2.4 IAEA Safety Standards for the Safety of Research Reactors (Series No. SSR-3)

This Specific Safety Requirements document establishes requirements for all main areas of safety for research reactors, with particular emphasis on requirements for design and operation. It explains the safety objectives and concepts that form the basis for safety and safety assessment for all stages in the lifetime of a research reactor. The safety requirements apply to site evaluation, design, manufacturing, construction, commissioning, operation, and planning for decommissioning of research reactors [A1-26].

2.4.3. International peer-review services

The IAEA offers its Member States a wide array of peer review and advisory services. These services play key roles for global nuclear safety and security, enabling countries to benefit from the independent insights of leading international experts, based on the common reference frame of the IAEA safety standards and security guidance in which an IAEA-led team of experts compares actual practices with IAEA standards.

Each of these services is undertaken by a team of international experts whose conclusions and recommendations are compiled in a report which advises the Member State on ways of improving its nuclear safety and security. A follow-up mission assesses progress made in implementing the recommendations.

The following IAEA peer review services are relevant for the present report:

— Integrated Regulatory Review Service (IRRS)

  The Integrated Regulatory Review Service helps strengthen and enhance the effectiveness of a Member State's regulatory infrastructure for nuclear, radiation, radioactive waste and transport safety.

  The service offers an integrated approach to the review of common aspects of any State's national, legal and governmental framework and regulatory infrastructure for safety. The IRRS regulatory review process provides a peer review of both regulatory technical and policy issues and is suitable for any State, regardless of the level of development of its activities and practices that involve ionizing radiation or a nuclear programme.

  IRRS teams evaluate a State's regulatory infrastructure for safety against IAEA safety standards. The teams compile their findings in reports that provide recommendations and suggestions for improvement, and note good practices that can be adapted for use elsewhere to strengthen safety. Mission reports describe the effectiveness of the regulatory oversight of nuclear, radiation, radioactive waste and transport safety and highlight how it can be further strengthened.

  Art. 8e(1) of the Council Directive 2014/87/Euratom requires EU Member States at least every 10 years to invite international peer review of their national framework and competent regulatory authority with the aim of continuously improving nuclear safety. EU Member States are using IAEA IRRS peer review services to meet this obligation, especially under Article 4 (Legal, regulatory and organisational framework) and Article 5 (Competent regulatory authority) under this Directive.

  Additionally, the Directive requires outcomes of such peer review to be reported to the Commission and the other Member States, when available. Outcomes of IRRS peer review missions are published on the IAEA website (see https://www.iaea.org/services/review-missions/calendar).

— Integrated Review Service for Radioactive Waste and Spent Fuel Management, Decommissioning and Remediation (ARTEMIS),

  ARTEMIS review is intended for facilities and activities involving radioactive waste or spent fuel management, radiological impact assessments for human health and the environment, the management of residues arising from uranium production as well as the decommissioning and remediation of sites contaminated by radioactive materials. Both government and private sector entities can call upon this service, which is also available to international organizations.

  Reviews may involve detailed assessments and technical advice on the implementation of specific programmes and project activities, with an emphasis on technology, on safety, or both.

  ARTEMIS review missions comprise meetings, interviews, site visits and document reviews. Observations, preliminary findings and recommendations are provided to the Member State in a draft review report for clarifications and fact-checking before a final approved report is delivered. The recipient entity remains fully responsible for all ensuing decisions and actions. The final review report, unless otherwise requested by the Member State, is made public three months after delivery.

  Art. 14(3) of the Council Directive 2011/70/Euratom requires EU Member States at least every 10 years to invite international peer review of their national framework, competent regulatory authority, national programme and its implementation with the aim of ensuring that high safety standards are achieved in the safe management of spent fuel and radioactive waste. EU Member States are using IAEA ARTEMIS and IRRS peer review services to meet this obligation. Additionally, the Directive requires outcomes of any peer review to be reported to the Commission and the other Member States, and to be made available to the public where there is no conflict with security and proprietary information. Outcomes of ARTEMIS peer review missions are published on the IAEA website[156] [A1-27].

— Operational Safety Review Team (OSART),

  Conservative design, careful manufacture and sound construction are prerequisites for the safe operation of nuclear power plants. The IAEA's OSART programme assists Member States in strengthening the safety

156 https://www.iaea.org/services/review-missions/calendar

of their nuclear power plants during commissioning and operation, comparing actual practices with IAEA safety standards.

The safety of nuclear installations depends on several factors, for example: capable management, sound policies, procedures, processes and practices; the competence of commissioning and operating personnel; sound accident management and emergency preparedness; and adequate resources. The OSART programme considers these and other aspects in assessing a facility's operational safety performance.

While OSART reviews have a strong technical focus, the expert reviewers also identify safety culture and organizational issues.

— Emergency Preparedness Review (EPREV) Service,

  EPREV services appraise level of preparedness for nuclear or radiological emergencies and facilitate the development of national emergency response capabilities, consistent with the IAEA safety standard.

— Safety Aspects of Long Term Operation (SALTO).

  The SALTO peer review is a comprehensive safety review directly addressing strategy and key elements for the safe long-term operation of nuclear power plants. Long-term operation (or lifetime extension) in this context means the continuation of the nuclear power plant operation beyond its originally anticipated lifetime and which has been justified by a safety assessment. The evaluation of programmes and performance is made on the basis of the IAEA's Safety Standards and other guidance documents.

A full list of available peer review services can be found on the IAEA webpage[157] [A1-28]

3. European Union legal framework

3.1. Euratom Treaty

Each Member State of the European Union has the right to decide whether to allow the use of nuclear energy for electricity generation on its own territory. Those that choose to generate electricity using nuclear energy must do so in accordance with Euratom law.

The Treaty establishing the European Atomic Energy Community (Euratom Treaty) constitutes Euratom's primary law. It is the original supreme source of law, by which the Community is established and secondary legislation can be adopted.

The Treaty was signed in Rome on 25 March 1957, for an indefinite duration, under the general objective of tackling the shortage of conventional energy in the 1950s. The Euratom Treaty also mandated the adoption by the Community of basic safety standards for the protection of workers and the general public. In addition, it provided for a safeguards system to prevent nuclear materials from being diverted from their intended uses. The powers of Euratom are limited to civil applications of nuclear energy.

Following the entry into force of the Treaty of Lisbon in 2009, the European Community was dissolved into the European Union ('EU') and the Treaty establishing the European (Economic) Community of 1957 was thus renamed ''Treaty on the Functioning of the European Union'' ('TFEU'). However, Euratom was not dissolved into the EU, and although Euratom has the same members as the EU and is governed by the EU institutions, it is a separate legal entity.

The institutional structure of Euratom is identical to that of the EU. Thus, the fulfilment of the tasks entrusted to Euratom is ensured by the Council of the European Union ('Council'), the European Parliament, the European Commission, the Court of Justice and the Court of Auditors, which are all institutional organs of the EU. Each of these institutions acts within the limits of the powers conferred to it by the common institutional framework provided in the Treaties.

As mentioned above, the Euratom Treaty provides that Euratom is to establish and enforce appropriate basic safety standards. Article 2(b) of Title I of the Euratom Treaty states that in order to perform its task, Euratom shall "*establish uniform safety standards to protect the health of workers and of the general public and ensure that they are applied*". This is detailed further in Article 30 of the Treaty, which stipulates: "*Basic standards shall be laid down within the Community for the protection of the health of workers and the general public against the dangers arising from ionising radiations…*", whereas Article 32 thereof lays down the procedure for the establishment of such standards. Article 33 requires that Member States' lay down the appropriate provisions to ensure compliance with the basic standards. Articles 34 et seq. ensure, in various

[157] https://www.iaea.org/services/review-missions

ways, the monitoring by the Commission of national health and safety measures, including any plans for the disposal of radioactive waste which may have a cross border impact.

The binding measures adopted by the Institutions are referred to as Euratom secondary law. Those measures can be regulations, directives and decisions. Non-binding measures provided for by the Euratom Treaty are recommendations and opinions.

A directive needs to be transposed into national legislation; regulations and decisions are directly applicable in the Member States.

Since the establishment of the Euratom Treaty, a substantial corpus of Euratom binding secondary legislation has been adopted and then updated. The most important legal acts in the context of the present report, i.e. in the fields of nuclear safety and radiation protection, and environment, are presented in the next chapters of this report.

3.2. Directives – Nuclear Safety and Radiation Protection

The Euratom legislative framework includes three key legal acts, namely Directive 2009/71/Euratom, as amended by Directive 2014/87/Euratom (Nuclear Safety Directive), Directive 2011/70/Euratom on the responsible and safe management of spent fuel and radioactive waste, and Directive 2013/59/Euratom (revised Basic Safety Standards Directive). All three directives have now entered into force and are applicable in EU Member States. Further details on these and other relevant Euratom directives are provided in the following sections.

3.2.1. The Basic Safety Standards Directive (2013/59/Euratom)

One of the main pillars of the Euratom secondary legislation over the years since the entry into force of the Treaty has been the Basic Safety Standards Directive ('BSS Directive'), which was first adopted in 1959 and subsequently updated in 1962, 1966, 1976, 1980, 1984, 1996 and 2013 to take into consideration new international standards.

Its latest version, Council Directive 2013/59/Euratom [A1-29] lays down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation.

This Directive was adopted in order:

- To take account of the scientific and technological progress since 1996, in particular the new recommendations in Publication 103(2007) of the International Commission on Radiological Protection (ICRP) but also of the operational experience with the then existing requirements, and
- To consolidate the existing set of Euratom radiation protection legislation into one single piece of legislation, by repealing five older Directives – the Medical Exposure Directive, the High Activity Sealed Sources Directive, the Outside Workers Directive, the Public Information Directive and the previous BSS Directive – and by "upgrading" a Commission Recommendation related to the protection of the public against indoor exposure to radon, to become legally binding.

The BSS Directive establishes uniform standards for the protection of the health of individuals subject to occupational, medical and public exposures against the dangers arising from ionising radiation. The directive applies to any planned, existing or emergency exposure situation involving a risk from exposure to ionising radiation that cannot be disregarded from a radiation protection point of view or with regard to the environment in view of long-term human health protection.

The directive introduces a graded approach to regulatory control of practices by way of notification, authorisation and appropriate inspections commensurate with the magnitude and likelihood of exposures resulting from the practice, and commensurate with the impact that regulatory control may have in reducing such exposures or improving radiological safety. Authorisation can take the form of a registration or a licence. Justified practices, such as the disposal or storage of radioactive waste, need to be notified prior to the practice commencement. Member States shall require licensing, inter alia, for the operation, decommissioning and closure of any facility for the long-term storage or disposal of radioactive waste, including facilities managing radioactive waste for this purpose.

The directive clearly defines the responsibilities of an undertaking or an employer for the radiation protection of their workers, including emergency workers, and provides for detailed requirements on the radiation protection programme for workers. The operational protection of exposed workers is based on:

- prior evaluation to identify the nature and magnitude of the radiological risk to exposed workers;

— optimisation of radiation protection in all working conditions;

— classification of exposed workers into different categories;

— control measures and monitoring relating to the different areas and working conditions, including individual monitoring;

— medical surveillance of workers;

— education and training of workers.

It provides also for the protection of members of the public in normal circumstances, as well as in emergency exposure situations. The operational protection of members of the public from practices subject to licensing, in normal circumstances, shall include:

— examination and approval of the proposed siting of the facility from a radiation protection point of view;

— acceptance into service of the facility subject to adequate protection being provided against any exposure or radioactive contamination liable to extend beyond the perimeter of the facility or radioactive contamination liable to extend to the ground beneath the facility;

— examination and approval of plans for the discharge of radioactive effluents;

— measures to control the access of members of the public to the facility.

For practices where a discharge authorisation is granted, the radioactive discharges into the environment need to be monitored and reported. Further to this, the BSS Directive requires the estimation of doses to members of the public from authorised practices, and the set-up of an environmental monitoring programme.

The Directive establishes limits for the effective radiation dose[158] for both workers (occupational exposures) and members of the public. The limit for members of the public shall be set at 1 mSv/year. Importantly, these dose limits for public exposure shall "… *apply to the sum of annual exposures of a member of the public resulting from all authorised practices*". This means that Member States need to evaluate all authorised practices which may contribute to the exposure of an individual member of the public and ensure that the sum of exposures remain below the dose limit. Member States have to consider this when establishing regulatory limits to radiological emissions from nuclear installations.

The directive requires that Member States shall establish an adequate legislative and administrative framework ensuring the provision of appropriate radiation protection education, training and information to all individuals whose tasks require specific competences in radiation protection. In addition, it contains detailed requirements for radiation protection education, training and information of workers, including emergency workers, and members of the public.

The BSS Directive stipulates that Member States shall ensure that account is taken of the fact that emergencies may occur on their territory and that they may be affected by emergencies occurring outside their territory. Member States shall establish an emergency management system and adequate administrative provisions to maintain such a system. It requires that emergency response plans are established in advance for the various types of emergencies identified by an assessment of potential emergency exposure situations and that these emergency response plans are tested, reviewed and, as appropriate, revised at regular intervals, taking into account lessons learned from past emergency exposure situations and the results of the participation in emergency exercises at national and international level. Undertakings are requested to notify the competent authority immediately of any emergency in relation to the practices for which it is responsible and to take all appropriate action to mitigate the consequences. Further requirements concern the protective measures to be taken. In addition, the directive requires prior information provision to the members of the public likely to be affected by an emergency, as well as an information provision to the affected members of the public in the event of an emergency.

Article 24 of the BSS Directive establishes the concept of a graded approach to regulatory control. It states that the regulatory controls should be commensurate with the magnitude and likelihood of exposures resulting from the practice, and commensurate with the impact that regulatory control may have in reducing such exposures or improving radiological safety. Exemption and release of practices/materials complying with the exemption and clearance criteria (established in annex VII of the BSS Directive) from regulatory control are options implementing the principle of the graded approach.

[158] Specific limits are also established for the equivalent doses to the lens of the eye and to the skin.

3.2.2. The Nuclear Safety Directive (2009/71/Euratom)

The Nuclear Safety Directive, Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations [A1-30], as amended by the Council Directive 2014/87/Euratom of 8 July 2014 [A1-31], supplements the basic standards referred to in Article 30 of the Treaty, as regards the safety of nuclear installations. It applies to nuclear power plants, fuel enrichment plants, nuclear fuel fabrication plants, reprocessing plants, research reactor facilities and spent fuel storage facilities. It also applies to storage facilities for radioactive waste that are on the same site and are directly related to the aforementioned installations. It reflects the principles of the Convention on Nuclear Safety.

The goal of the Nuclear Safety Directive is to promote the continuous improvement of nuclear safety and to ensure that a high level of nuclear safety is provided by the Member States to protect workers and the public against dangers arising from ionizing radiations from nuclear installations.

The Directive requires Member States to establish and maintain a national legislative, regulatory and organisational framework for nuclear safety. As part of this framework, Member States shall establish:

- national nuclear safety requirements covering all stages of the lifecycle of nuclear installations,
- a system of licensing and prohibition of operation of nuclear installations without a licence,
- a system of regulatory control of nuclear safety performed by the competent regulatory authority, which shall include verification of compliance with the national nuclear safety requirements, inspections, effective and proportionate enforcement actions, including, where appropriate, corrective action or suspension of operation and modification or revocation of a licence,
- education and training arrangements for staff of all parties having responsibilities related to the nuclear safety of nuclear installations.

In line with the above, Member States must establish and maintain a competent regulatory authority in the field of nuclear safety of nuclear installations. The effective independence from undue influence of the competent regulatory authority in its regulatory decision-making must also be ensured. It shall be functionally separate from any other body or organisation concerned with the promotion or utilisation of nuclear energy and shall be provided with sufficient budget and human resources for the effective discharge of its responsibilities.

The Nuclear Safety Directive recognizes the principle of national responsibility and the principle of prime responsibility of the licence holder for the nuclear safety of a nuclear installation under the supervision of its national competent regulatory authority. Licence holders are required to undertake systematic and verifiable safety assessments, including the verification of "defence-in-depth" measures.

The national framework should be improved when appropriate, taking into account: operating experience, insights gained from safety analyses for operating nuclear installations, development of technology, and results of safety research. In addition, periodic safety assessments of their national framework and competent regulatory authorities shall be organised by the Member States, supplemented with international peer reviews.

The Directive sets out an EU-wide safety objective for nuclear installations. Member States shall ensure that the national nuclear safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding early and large radioactive releases. It applies to new nuclear installations and shall be used as a reference for the timely implementation of reasonably practicable safety improvements to existing nuclear installations.

The Directive provides for regular safety reassessments of nuclear installations, to be carried out by the licence holder under the supervision of the competent regulatory authority, to identify further safety improvements, taking into account, inter alia, ageing issues, using as a reference the aforementioned nuclear safety objective.

With reference to transparency issues, Member States are to ensure that necessary information in relation to the nuclear safety of nuclear installations and its regulation is made available to workers and the general public. It also requires Member States to ensure that the general public is given the appropriate opportunities to participate effectively in the decision-making process relating to the licensing of nuclear installations.

The Directive recognises the fundamental safety principles set by the International Atomic Energy Agency which should constitute a framework of practices that Member States should have regard to when implementing the Directive. It also recognises the body of work of the Western European Nuclear Regulators Association (WENRA).

Following the amendment of the Nuclear Safety Directive in 2014 (2014/87/Euratom), the EU significantly enhanced its leadership in nuclear safety worldwide. The amendment is based on nuclear risk and safety assessments (stress tests) carried out in 2011 and 2012, the lessons learned from the Fukushima nuclear accident, and the safety requirements of the Western European Nuclear Regulators Association and the International Atomic Energy Agency.

The amended directive requires EU countries to give the highest priority to nuclear safety at all stages of the lifecycle of a nuclear power plant. This includes carrying out safety assessments before the construction of new nuclear power plants and ensuring significant safety enhancements for old reactors. Specifically, the directive:

- strengthens the role of national regulatory authorities by ensuring their independence from national governments. EU countries must provide the regulators with sufficient legal powers, staff, and financial resources;
- creates a system of peer reviews. EU countries choose a common nuclear safety topic every six years and organise a national safety assessment on it. They then submit their assessment to other countries for review. The findings of these peer reviews are made public;
- requires a safety re-evaluation for all nuclear power plants to be conducted at least once every 10 years;
- increases transparency by requiring operators of nuclear power plants to release information to the public, both in times of normal operation and in the event of incidents.

3.2.3. The Spent Fuel and Radioactive Waste Management Directive (2011/70/Euratom)

The Spent Fuel and Radioactive Waste Management Directive applies to all stages of management of spent fuel and radioactive waste from civilian activities (except waste from extractive industries which may be radioactive and which falls within the scope of Directive 2006/21/EC, c.f. 3.3.8) [A1-32]. It supplements the basic standards referred to in Article 30 of the Euratom Treaty, as regards the safety of spent fuel and radioactive waste, and is without prejudice to the basic safety standards directive referred to above. It ensures that Member States make provisions for a high level of safety in spent fuel and radioactive waste management to protect workers and the general public against the dangers arising from ionising radiation and to avoid imposing undue burdens on future generations.

The Directive reflects and enhances some of the main principles and requirements of the IAEA Joint Convention[159] and the IAEA Safety Standards. It imposes legal obligations on the Member States to establish and maintain a national policy, a national programme to implement this national policy, as well as a national legislative, regulatory and organisational framework for the management of spent fuel and radioactive waste that allocates responsibilities and provides for coordination between relevant competent bodies. The national framework must provide for the following:

- a national programme for the management of spent fuel and radioactive waste setting out how Member States intend to implement their national policies for the responsible and safe management of spent fuel and radioactive waste in a way that meets the aims of the directive;
- national arrangements for the safety of spent fuel and radioactive waste management;
- a system of licensing of spent fuel and radioactive waste management activities and/or facilities, including the prohibition of such activities and/or of the operation of such facilities without a licence, and, if appropriate, the setting of licence conditions;
- a system of appropriate control, a management system, regulatory inspections, documentation and reporting obligations for radioactive waste and spent fuel management activities and/or facilities, including appropriate measures for the post-closure periods of disposal facilities;
- enforcement actions, including the suspension of activities and the modification, expiration or revocation of a licence together with requirements, if appropriate, for alternative solutions that lead to improved safety;

[159] As pointed out by recital 25 of the Directive, the principle of national responsibility, as well as the principle of prime responsibility of the licence holder for the safety of spent fuel and radioactive waste management are enhanced and the role and independence of the competent regulatory authority is reinforced.

- the allocation of responsibility to the bodies involved in the different steps of spent fuel and radioactive waste management;
- national requirements for public information and participation;
- the financing scheme(s) for spent fuel and radioactive waste management. In particular, the Directive requires that the costs for the management of spent fuel and radioactive waste shall be borne by those who generated those materials, in accordance with the "polluter-pays" principle.

The effective independence of national regulatory bodies in the field of safety of spent fuel and radioactive waste management is also addressed by Article 6 of the Directive:

> ''1. Each Member State shall establish and maintain a competent regulatory authority in the field of safety of spent fuel and radioactive waste management.
>
> 2. Member States shall ensure that the competent regulatory authority is functionally separate from any other body or organisation concerned with the promotion or utilisation of nuclear energy or radioactive material, including electricity production and radioisotope applications, or with the management of spent fuel and radioactive waste, in order to ensure effective independence from undue influence on its regulatory function.
>
> 3. Member States shall ensure that the competent regulatory authority is given the legal powers and human and financial resources necessary to fulfil its obligations in connection with the national framework (...)''

With reference to transparency issues, Member States are to ensure that necessary information on the management of spent fuel and radioactive waste is made available to workers and the public. This obligation includes ensuring that the competent regulatory authority informs the public in the fields of its competence. Member States are also required to ensure that the public is given the necessary opportunities to participate effectively in the decision-making process regarding spent fuel and radioactive waste management in accordance with national legislation and international obligations.

Moreover, Member States shall ensure that the national framework is improved where appropriate, taking into account operating experience, insights gained from the decision-making process and the development of relevant technology and research.

As regards the long-term management of spent fuel and radioactive waste, the Directive affirms the ethical obligation of each Member State to avoid any undue burden on future generations (recital 24, article 1(1)). Moreover, it stresses that the storage of radioactive waste, including long term storage, is only an interim solution, but not an alternative to disposal (recital 21).

Article 4(3) defines the principles for the national policy. One of them specifically addresses long-term management by stating that spent fuel and radioactive waste shall be safely managed, including in the long term with passive safety features. As, due to the long timeframes, it is impossible to rely on institutional controls and active safety systems, passive safety features are crucial in ensuring long term safety. Article 5(1)(e) indicates that the national framework, among other elements, shall provide for appropriate measures for the post-closure periods of disposal facilities. In their national programmes Member States have to indicate concepts or plans for the post-closure period of a disposal facility's lifetime, including the period during which appropriate controls are retained and the means to be employed to preserve knowledge of that facility in the longer term (Article 12(1)(e)). As part of the licensing of the facility, the safety demonstration shall cover the operation or closure of a disposal facility as well as the post- closure phase of a disposal facility (Article 7(3)).

At least every ten years Members States have to carry out self-assessments and invite international peer reviews of their national framework, competent authority and/or national programme. The Member States shall also inform the Commission and the other Member States about the outcome of such reviews, which may be made available to the public where there is no conflict with security and proprietary information. In practice the Member States fulfil these obligation by relying on the IAEA ARTEMIS and IRRS missions as described in detail in Chapter 2.4.3.

Table A1-1 provides an overview of the main provisions of the Radioactive Waste Directive.

3.2.4. The Radioactive Waste and Spent Fuel Shipment Directive (2006/117/Euratom)

The Radioactive Waste and Spent Fuel Shipment Directive lays down a Community system of supervision and control of transboundary shipments of radioactive waste and spent fuel in, through and outside the

Community [A1-33]. In particular, it provides for a compulsory and common scheme of notification and a standard control document, for shipments of radioactive waste or spent fuel which have a point of departure, transit or destination in an EU Member State, provided that the quantities in question exceed certain limits.

Such a directive is relevant in view of the Spent Fuel and Radioactive Waste Management Directive 2011/70/Euratom requirement that radioactive waste must be disposed of in the Member State in which it was generated (Article 4(4)). It introduces an exception if, at the time of the shipment, an agreement taking into account the criteria in accordance with Article 16(2) of Directive 2006/117/Euratom, has entered into force between the Member State concerned and another Member State or a third country to use a disposal facility in one of them. Commission Recommendation 2008/956/Euratom of 4 December 2008 lists the criteria for the export of radioactive waste and spent fuel to third countries (as stated in Article 16(2) of Directive 2006/117/Euratom).

3.2.5. The Euratom Drinking Water Directive (2013/51/Euratom)

In addition to the Basic Safety Standards Directive, a directive laying down requirements for the protection of the health of the general public with regard to radioactive substances in drinking water was adopted in 2013 [A1-34]. In view of the importance for human health of the quality of water intended for human consumption, the EU laid down quality standards at Community level and provided for the monitoring of compliance with those standards, with the aim of enhancing radiation protection legislation.

In particular, the Directive sets out parametric values triggering remedial action for radon and tritium concentrations in drinking water, as well as indicative dose[160]. It also sets out frequencies and methods for monitoring radioactive substances in drinking water.

3.3. Other provisions of Euratom law

With regard to Euratom health and safety measures, there are some additional provisions in the Euratom Treaty that are not covered in the foregoing:

- "Each Member State shall establish the facilities necessary to carry out continuous monitoring of the level of radioactivity in the air, water and soil and to ensure compliance with the basic standards. The Commission shall have the right of access to such facilities; it may verify their operation and efficiency" (Article 35).
- "The appropriate authorities shall periodically communicate information on the checks referred to in Article 35 to the Commission so that it is kept informed of the level of radioactivity to which the public is exposed" (Article 36).
- "Each Member State shall provide the Commission with such general data relating to any plan for the disposal of radioactive waste in whatever form will make it possible to determine whether the implementation of such plan is liable to result in the radioactive contamination of the water, soil or airspace of another Member State. The Commission shall deliver its opinion within six months, after consulting the group of experts referred to in Article 31" (Article 37).
- "The Commission shall make recommendations to the Member States with regard to the level of radioactivity in the air, water and soil. In cases of urgency, the Commission shall issue a directive requiring the Member State concerned to take, within a period laid down by the Commission, all necessary measures to prevent infringement of the basic standards and to ensure compliance with regulations. Should the State in question fail to comply with the Commission directive within the period laid down, the Commission or any Member State concerned may forthwith, by way of derogation from Articles 258 and 259 of the Treaty on the Functioning of the European Union, bring the matter before the Court of Justice" (Article 38).

In relation to the above articles of the Treaty, the European Commission has developed the European Radiological Data Exchange Platform (EURDEP), a web-based platform for the exchange of radiological monitoring data between participating countries almost in real-time. Monitoring information is collected from automatic surveillance systems in 39 countries. The arrangements for the data exchange for the EU Member States are laid down in the Commission Recommendation 2000/473/Euratom, while non-EU countries participate on a voluntary basis. Those countries that send their national radiological monitoring data have access to the data of all the other participating countries. The system is continuously operating with a daily

[160] The committed dose for one year of ingestion resulting from all the radionuclides whose presence has been detected in a supply of water intended for human consumption, of natural and artificial origin, but excluding tritium, potassium-40, radon and short-lived radon decay products.

data exchange routine and it is expected that the data transmissions will continue during a radiological or nuclear emergency. A freely accessible version of the website allows the public to view graphical information on radioactivity levels over the EURDEP area.

In 2010, the Commission concluded a Memorandum of Understanding with the IAEA concerning the EURDEP system. This Memorandum makes EURDEP technology available to the IAEA, for creating a global on-line environmental radiation data exchange application. The Commission supported the establishment of the IAEA's International Radiation Monitoring Information System (IRMIS) and established data transmission from EURDEP.

With regard to emergencies, Council Decision 87/600/Euratom outlines the requirements for the early exchange of information in the event of a radiological emergency. The resulting arrangements cover Euratom Member States, Switzerland, Norway, North Macedonia and Montenegro, and "apply to the notification and provision of information whenever a Member State decides to take measures of a wide-spread nature in order to protect the general public in case of a radiological emergency". A radiological emergency may be declared either due to an accident at a facility where a significant release of radioactive material occurs or is likely to occur, or due to detection of abnormal levels of radioactivity which are likely to be detrimental to public health. The Decision sets out the actions to be taken by the Member State that initially decides to take measures, as follows:

- forthwith notify the Commission and those Member States which are - or are likely to be - affected of such measures and the reasons for taking them;
- promptly provide the Commission and those Member States which are - or are likely to be - affected with available information relevant to minimising the foreseen radiological consequences, if any, in those States.

The decision also specifies the nature of the information to be provided and requires that the initial information be supplemented at appropriate intervals. The Commission forwards the information it receives from a Member State to all Member States.

In support to the above, the European Commission has developed ECURIE, a 24h emergency notification and information exchange system. The system notifies the competent authorities of the participating States and the Commission when a Member State notifies a major nuclear accident or radiological emergency. During an emergency, the system provides an information exchange platform for the participating States, in order to inform about the current and foreseeable status of the accident, meteorological conditions, national protective actions, etc. The legal basis for the participation of Euratom Member States in ECURIE is the aforementioned Council Decision 87/600/Euratom and the Agreement between Euratom and non-Member States on the participation of the latter in the Community arrangements for the early exchange of information in the event of a radiological emergency. The Commission is responsible for ECURIE management and development and the practical arrangements for the exchange of information under ECURIE are reviewed and agreed with the Competent Authorities at their biennial meetings.

With regard to the contamination of foodstuffs, following the nuclear accidents of Chernobyl in 1986 and of Fukushima in 2011, specific EU Regulations on import conditions into the EU of agricultural products, food and feed were put in place ([A1-35], [A1-36]).

Council Regulation (Euratom) 2016/52 of 15 January 2016 on radioactive contamination of food and feed following a nuclear accident or any other case of radiological emergency lays down maximum permitted levels of radioactive contamination of food and feed [A1-37].

3.4. EU Directives – Environmental

Nuclear energy is at the crossroad of several sectors such as economy, industry, and environment. This intricacy is reflected in the various laws that one should look at before developing a nuclear project or carrying out a nuclear activity since any action can have an impact on other sectors. For instance, nuclear facilities just as other industrial activities have to be compliant with applicable rules and regulations, including environmental requirements, during their entire lifecycle i.e. design, construction, commissioning, operation and decommissioning. In addition, they are subject to specific nuclear legislation and a licensing regime that focus on radiation protection, nuclear safety and security (see Chapters 3.1 – 3.3).

This section describes the EU directives that are applicable to the nuclear energy sector. There is a wide range of EU legislation in force concerning the environment with the aims of preserving the quality of the environment, protecting human health, and ensuring rational use of natural resources. The main areas covered are nature and biodiversity, integrated pollution control, waste management, air pollution, water pollution, noise pollution, environmental impact assessment, and genetically modified organisms. Such legislation reflects international obligations covered in international conventions ratified by EU Member States like the Biological Diversity Convention, the London Convention and its Protocol, and the Espoo Convention (cf. 2.1. United Nations Treaties).

3.4.1. EU Directive on Environmental Impact Assessment (2011/92/EU[161])

An environmental impact assessment is a procedure ensuring that the environmental implications of decisions on projects are taken into account before decisions are made. An environmental assessment can be undertaken for individual projects, based on the Environmental Impact Assessment Directive [A1-38] or for public plans or programmes based on Strategic Environmental Assessment Directive (cf. 3.4.2). The main objective of the Directive is to ensure that projects likely to have significant effects on the environment be subject to an environmental impact assessment, prior to their authorisation. Consultation with the public is a key feature of environmental assessment impact procedures[162]. Requirements for transparency, public hearings and consultation have been part of the nuclear legislation for a long time. Already in the late fifties, the Euratom treaty included in Articles 41 to 44 that investments in nuclear industrial activities relating to new installations and replacements or conversions be communicated to the Commission and to the Member States concerned.

The initial EIA Directive of 1985 and its three amendments were consolidated in Directive 2011/92/EU, which was amended by Directive 2014/52/EU [A1-39] in 2014. The EIA shall identify, describe and assess the direct and indirect significant effects of a project on: the population and human health, biodiversity with particular attention to species and habitats protected under Directive 92/43/EEC [A1-40] and Directive 2009/147/EC [A1-41] (see description below), land, soil, water, air and climate, material assets, cultural heritage, the landscape and the interaction between all these factors.

The EIA procedure is as follows:

- the developer may request the competent authority to define what should be covered by the EIA (scoping stage);
- the developer must provide information on the environmental impact (EIA report – Annex IV);
- the environmental authorities, local or regional the public and affected Member States must be informed and consulted; and
- the competent authority decides taking into consideration the results of consultations. The public is informed of the decision afterwards and can challenge the decision before the courts.

In the nuclear energy field, projects for the construction of new nuclear facilities as well as dismantling or decommissioning of such facilities, or changes/extensions representing environmental risks similar to the initial project for construction of such facilities, have to undergo an assessment of their impacts on the environment during the construction, operation and where relevant demolition stages. The environmental impact assessment shall identify, describe and assess the direct and indirect significant effects of a project on the following factors:

- population and human health;
- biodiversity;
- land, soil, water, air and climate;
- material assets, cultural heritage and the landscape;
- interaction between all these factors.

[161] Directive 2011/92/EU of the European Parliament and of the Council of 13 December 2011 on the assessment of the effects of certain public and private projects on the environment (OJ L 026, 28.1.2012, p.1). It has been amended by Directive 2014/52/EU of the European Parliament and of the Council of 16 April 2014 amending Directive 2011/92/EU on the assessment of the effects of certain public and private projects on the environment (OJ L 124, 25.4.2014, p. 1).

[162] art. 6 in order to ensure the effective participation of the public concerned in the decision-making procedures, the public shall be informed electronically and by public notices of the following matters early in the environmental decision-making procedures
Art. 7 The Member States concerned shall enter into consultations regarding, inter alia, the potential transboundary effects of the project and the measures envisaged to reduce or eliminate such effects and shall agree on a reasonable time- frame for the duration of the consultation period.

Projects related to the supply chain of nuclear fuel, storage and disposal of radioactive waste and spent nuclear fuel are also subject to an environmental impact assessment which must demonstrate compliance with the applicable nuclear regulations.

3.4.2. EU Directive on Strategic Environmental Assessment (2001/42/EC)

The SEA Directive [A1-42] applies to a wide range of public plans and programmes (e.g. on land use, transport, energy, waste, agriculture, etc.) which must be prepared or adopted by a national, regional or local authority and be required by legislative, regulatory or administrative provisions.

A SEA is mandatory for plans/programmes prepared for agriculture, forestry, fisheries, energy, industry, transport, waste/water management, telecommunications, tourism, town & country planning or land use, and which set the framework for future development consent of projects listed in the EIA Directive or have been determined to require an assessment under the Habitats Directive.

The SEA procedure can be summarized as follows: an environmental report is prepared in which the likely significant effects on the environment and the reasonable alternatives of the proposed plan or programme are identified. The public and the environmental authorities are informed and consulted on the draft plan or programme and the environmental report prepared. As regards plans and programmes which are likely to have significant effects on the environment in another Member State, the Member State in whose territory the plan or programme is being prepared must consult the other Member State(s). On this issue the SEA Directive follows the general approach taken by the SEA Protocol to the UNECE Convention on Environmental Impact Assessment in a Transboundary Context (Espoo Convention).

The environmental report and the results of the consultations are taken into account before adoption. Once the plan or programme is adopted, the environmental authorities and the public are informed and relevant information is made available to them. In order to identify unforeseen adverse effects at an early stage, significant environmental effects of the plan or programme are to be monitored.

The SEA and EIA procedures are very similar, but there are some differences:

— the SEA requires the environmental authorities to be consulted at the screening stage;

— scoping (i.e. the stage of the SEA process that determines the content and extent of the matters to be covered in the SEA report to be submitted to a competent authority) is obligatory under the SEA.

Other environmental directives aiming at protecting biodiversity, water, and air also apply to nuclear facilities as presented below. Nuclear facilities must comply with such directives as they are applicable to all industrial activities, which clearly includes nuclear installations. For instance, nuclear power plants have conventional parts, such as diesel generation stations for which emissions have to be within a required range (as its fuel is diesel) and chemical plants to treat water. So nuclear plants have to carry out assessments, monitoring, and comply with relevant environmental requirements.

3.4.3. Habitants Directive (92/43/EEC) and Birds Directive (2009/147/EC)

The EIA directive requires assessing the direct and indirect significant effects of a project on the biodiversity with particular attention to habitats and species protected under the Habitats [A1-40] and Birds [A1-41] directives.

The Habitats Directive helps maintain biodiversity. It protects over 1000 animals and plant species as well as over 200 types of habitat. It also established the EU-wide Natura 2000 network of protected areas. For sites/operations located in or near to biodiversity-sensitive areas (including the Natura 2000 network of protected areas, UNESCO World Heritage sites and Key Biodiversity Areas), it requires that an appropriate assessment be conducted in compliance with the provisions of the Natura 2000.

The Birds Directive provides comprehensive protection to all wild bird species naturally occurring in the EU and requires carrying out assessments of the effects on wild birds.

3.4.4. Water Directives

EU Water Framework Directive (Directive 2000/60/EC)

The purpose of the EU Water Framework Directive is to establish a framework for the protection of inland surface waters, transitional waters, coastal waters and groundwater in Europe. The directive lists priority hazardous substances (toxic, persistent and liable to bio-accumulate - radioactive substances are excluded) whose pollution, discharges and losses need to be progressively reduced or phased out. The ultimate environmental objective is to achieve “good” ecological and chemical status for surface waters[163] and good chemical and quantitative status for groundwater with no deterioration of existing water. The directive´s obligations are in line with international agreements, including those aiming at preventing and eliminating pollution of the marine environment.

Directives on environmental quality standards (2008/105/EC) in the field of water policy

Directive 2008/105/EC (amended by Directive 2013/39/EU) is a ‘daughter’ directive of the Water Framework Directive setting out environmental quality standards (EQSs) concerning the presence in surface water of certain substances or groups of substances identified as priority pollutants because of the significant risk they pose to or via the aquatic environment. The priority substances include the metals cadmium, lead, mercury and nickel, and their compounds; benzene; polyaromatic hydrocarbons (PAH); and several pesticides. Several of these priority substances are classed as hazardous. The directive also requires EU countries to designate mixing zones near the points of discharge, where the EQSs may be exceeded. These areas must be clearly identified in the river basin management plans established in accordance with the Water Framework Directive. These standards are in line with the strategy and objectives of the EU’s Water Framework Directive (Directive 2000/60/EC).

Directive 2013/39/EU updated the EQSs for 7 of the 33 original priority substances in line with the latest scientific and technical knowledge concerning the properties of those substances. It also requires the Commission to establish a watch list of substances in surface water for which EU-wide monitoring data are to be gathered to support future prioritisation exercises. Implementing Decision (EU) 2020/1161 establishes the latest watch list.

Groundwater directive (2006/118/EC)

The Groundwater Directive is also a ‘daughter’ directive of the Water Framework Directive, and further clarifies the obligation under the Water Framework Directive for groundwater bodies to achieve “good” chemical status and reverse upward trends of pollutants. For instance, article 17 prohibits direct discharges into groundwater apart from those listed, including artificial recharge.

There are other water directives which may be relevant in this context, in particular Directive 2007/60/EC on the assessment and management of flood risks.

Drinking Water Directive (98/83/EC)

The Drinking Water Directive [A1-44] concerns the quality of water intended for human consumption. Its objective is to protect human health from adverse effects of any contamination of water intended for human consumption by ensuring that it is wholesome and clean. The Directive lays down the essential quality standards at EU level. A total of 48 microbiological, chemical and indicator parameters, including parameters related to radioactivity[164], must be monitored and tested regularly.

Urban Waste Water Directive (91/271/EEC)

The objective of the Urban Waste Water Treatment Directive [A1-59] is to protect the environment from the adverse effects of urban waste water discharges and discharges from certain industrial sectors. It requires Member States to ensure that their towns, cities and settlements properly collect and treat waste water. Untreated waste water can be contaminated with harmful chemicals, bacteria and viruses and thus can also present a risk to human health.

EU Marine Strategy Framework Directive (2008/56/EC)

The Marine Strategy Framework Directive [A1-45] was adopted on 17 June 2008. Its aim is to protect more effectively the marine environment across Europe. The Directive enshrines in a legislative framework the ecosystem approach to the management of human activities having an impact on the marine environment, integrating the concepts of environmental protection and sustainable use.

The Marine Strategy Framework Directive aims at achieving Good Environmental Status (GES) of the EU's marine waters by 2020 and to protect the resource base upon which marine-related economic and social activities depend. It is the first EU legislative instrument related to the protection of marine biodiversity, as it

[163] Surface waters are rivers, lakes, transitional waters and coastal waters. They also include territorial waters as far as their chemical status is concerned

[164] Annex I part C of the Council Directive 98/83/EC of 3 November 1998 on the quality of water intended for human consumption

contains the explicit regulatory objective "biodiversity is maintained by 2020", as the cornerstone for achieving GES.

Maritime Spatial Planning Directive (2014/89/EU)

The Directive [A1-60] establishes a framework for maritime spatial planning aimed at promoting the sustainable growth of maritime economies, the sustainable development of marine areas and the sustainable use of marine resources. Member States shall establish maritime spatial plans which include a planning of the possible activities in their marine waters. If there is a nuclear plant in the vicinity of the sea, some activities could not be possible or should be better framed.

3.4.5. Ambient Air Directives

EU Directive on ambient air quality (2008/50/EC) and (2004/107/EC)

The ambient air quality directives [A1-46] and [A1-61] include the following elements:

- defining and establishing objectives for ambient air quality designed to avoid, prevent or reduce harmful effects on human health and the environment as a whole;
- assessing the ambient air quality in Member States on the basis of common methods and criteria;
- obtaining information on ambient air quality in order to help combat air pollution and nuisance and to monitor long-term trends and improvements resulting from national and Community measures;
- ensuring that such information on ambient air quality is made available to the public;
- maintaining air quality where it is good and improving it in other cases;
- promoting increased cooperation between the Member States in reducing air pollution.

Member States shall assess ambient air quality with respect to the following pollutants: sulphur dioxide, nitrogen dioxide and oxides of nitrogen, particulate matter (PM10 and PM2.5), lead, benzene and carbon monoxide (as per Directive 2008/50/EC), as well as arsenic, cadmium, mercury, nickel and polycyclic aromatic hydrocarbons (as per Directive 2004/107/EC).

One can note that EU Directive 2015/1480/EC of 28 August 2015 amends several annexes to the ambient air quality directives by laying down the rules concerning reference methods, data validation and location of sampling points for the assessment of ambient air quality.

National Emission Reduction Commitments Directive (2016/2284)

This Directive [A1-47] sets national reduction commitments for the Member States' anthropogenic atmospheric emissions of five air pollutants (sulphur dioxide, nitrogen oxides, volatile organic compounds, ammonia and fine particulate matter) responsible for acidification, eutrophication and ground-level ozone pollution, which lead to significant negative impacts on human health and the environment.

3.4.6. EU Waste Framework Directive (2008/98/EC)

The directive [A1-48] provides for a general framework of waste management requirements and sets the basic concepts and definitions related to waste management, such as definitions of waste, recycling, recovery. It explains when waste ceases to be waste and becomes a secondary raw material (so called end-of-waste criteria), and how to distinguish between waste and by-products. The Directive lays down some basic waste management principles: it requires that waste be managed without endangering human health and harming the environment, and in particular without risk to water, air, soil, plants or animals, without causing a nuisance through noise or odours, and without adversely affecting the countryside or places of special interest. The European Union's approach to waste management is based on the "waste hierarchy" which sets the following priority order when shaping waste policy and managing waste at the operational level: prevention, (preparing for) reuse, recycling, recovery and, as the least preferred option, disposal (which includes landfilling and incineration without energy recovery).

The management of radioactive waste is out of the scope of the waste framework directive as requirements for the management of radioactive waste are set out in Directive 2011/70/Euratom (cf. 3.2.3). For further information, a comparison of the legal requirements for both the radioactive waste and waste including hazardous waste is available in Chapter 5 of this Annex.

3.4.7. Management of Waste from Extractive Industries Directive (2006/21/EC)

Directive 2006/21/EC on the management of waste from extractive industries [A1-62] aims at preventing or reduce as far as possible any adverse effects on the environment, in particular on water, air, soil, fauna and flora and the landscape, and any resultant risks to human health, brought about as a result of the management of waste from the extractive industries.

3.4.8. Industrial Emissions Directive (2010/75/EU)

The Industrial Emissions Directive (IED) [A1-63] regulates the emissions from around 52 000 of the largest (agro)industrial installations. IED installations must operate according to a permit issued by the competent authority of the Member State covering all environmental aspects of the installation's activities. Permit conditions must be based on the use of Best Available Techniques (BAT).

Nuclear activities are not included in the scope of the IED, nor are the use and release of radioactive substances. However, activities carried out at nuclear installations but not related to radioactive substances may fall under the scope of the IED.

3.4.9. Medium Combustion Plants Directive (EU) 2015/2193

Directive (EU) 2015/2193 on the limitation of emissions of certain pollutants into the air from MCPs known as the Medium Combustion Plant Directive (MCPD) [A1-64] regulates pollutant emissions from the combustion of fuels in plants with a rated thermal input equal to or greater than 1 MWth and less than 50 MWth. This includes such combustion plants which may be located at nuclear installations.

3.4.10. Control of major-accident hazards involving dangerous substances (2012/18/EU)

Directive 2012/18/EU on the control of major-accident hazards involving dangerous substances [A1-43] aims at controlling major accident hazards involving dangerous substances, especially chemicals.

Considering the very high rate of industrialisation in the EU, this directive, also known as the Seveso Directive, has contributed to achieving a low frequency of major accidents. The Directive is widely considered as a benchmark for industrial accident policy and has been a role model for legislation in many countries worldwide. It has been amended twice since its introduction in 1982. The current amendment is referred to as Seveso-III.

Directive 2012/18/EU covers some 12 000 establishments where dangerous substances may be present (e.g. during processing or storage) in quantities exceeding certain threshold. Depending on the amount of dangerous substances present, establishments are categorised in lower or upper tier, the latter being subject to more stringent requirements.

Certain industrial activities or hazards subject to other legislation providing a similar level of protection are excluded from the Directive. In particular, Article 2.2(b) specifies that the directive shall not apply to 'hazards created by ionising radiation originating from substances'. However, dangerous substances which do not pose a hazard created by ionizing radiation are covered by the Seveso-III Directive, even if they are within a nuclear establishment.

4. National nuclear legislative and regulatory frameworks

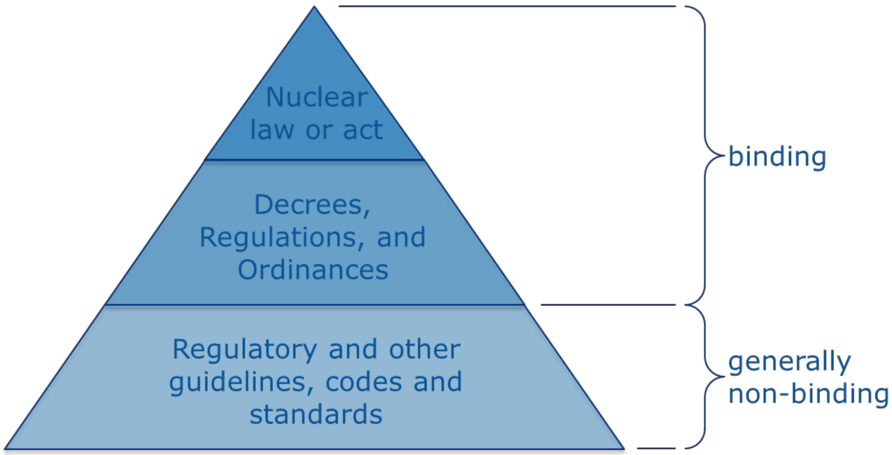
Radiation protection, safe operation of nuclear facilities as well as safe and responsible management of spent fuel and radioactive waste is achieved not only by means of appropriate technical solutions, but also by means of an administrative, legal and regulatory framework that establishes clearly the duties, responsibilities and obligations for each of the stakeholders (operators, radioactive waste management organisations, authorities), the requirements for the activity (e.g. the need for a licence, safety assessments, sufficient skills and resources, availability of sufficient funds, etc.) as well as the different mechanisms for control, verification and enforcement (e.g. regulatory approval of documents, inspections, fines, etc.). Ensuring nuclear safety is a national responsibility of each country that operates nuclear installations and this is the fundamental principle on which nuclear safety legislation has been developed at the international level.

The main actors in ensuring the safe operation of nuclear installations during their complete lifecycle (from design/construction to decommissioning, including management and disposal of residual wastes) are: (i) national governments responsible for establishment of a national legislative and regulatory framework for nuclear safety; (ii) operators of nuclear installations (licence holders) primarily responsible for the safety of their installations; and (iii) national regulatory authorities responsible for supervision and control of nuclear activities to ensure nuclear safety.

European Union Member States operating nuclear installations have mature legislative and regulatory frameworks that have been developed over a number of years and amended regularly to take into account feedback of experience from home and abroad, as well as new developments in related knowledge. The overall objective of these systems is to ensure that priority is given to safety and to protect workers, the public and the environment from the dangers of ionising radiation. All involve the establishment of a nuclear regulatory authority and include a system of licensing of nuclear activities, so that such activities may not be performed without a licence. Before a licence may be granted, the licensee must provide a comprehensive demonstration, to the satisfaction of the nuclear regulatory authority, that the activities will be carried out in full respect of the national nuclear legislation and that risk is reduced as low as reasonably achievable[165]. The regulatory authority may impose conditions in the licence and will perform inspections and supervision during its lifetime to ensure that the licence conditions are respected and that the safety demonstration that formed the basis of the licence application remains valid. Furthermore, recognising that the lifetime of a nuclear facility can be long and that developments in science and technology will advance during that time, licensees are required periodically to review the safety of the installations and to implement reasonably practicable safety improvements taking into account the feedback of operating experience and developments in knowledge.

Nuclear legislative frameworks generally comprise a hierarchy of documents that can be represented by the classical pyramid shown in the figure below.

Figure A.1-2. Nuclear legislative and regulatory pyramid



At the top of the pyramid is the nuclear law or act that sets out the basic provisions dealing with fundamental safety principles and protection goals. Lower tiers become successively more detailed and voluminous.

The upper two tiers, comprising the nuclear act and the decrees, ordinances and regulations are binding. Obligations resulting from the ratification of any international conventions and treaties will also be reflected in these two levels. This applies also to the requirements and objectives laid down in the Euratom legislation that have to be integrated into national legislation.

The lower tier is generally non-binding but may be made binding by, for example, regulations or licence conditions issued by the nuclear regulatory authority. For example, in Finland, the regulatory guides are mandatory. National measures reflecting detailed technical international standards and guides can also be included at this level.

Regulatory frameworks may be highly prescriptive or may take a more 'goal-setting' approach, leaving the details of how to meet the overall objectives to the licensee. Both require a robust demonstration of how the law and regulations are to be met for a particular project, which has to be endorsed by the regulatory authority before any licence to construct or operate can be granted.

The IAEA safety standards establish fundamental principles, requirements and recommendations to ensure nuclear safety, serving as a global reference (see Chapter 2.4). They are in principle not binding on IAEA

165 Nuclear installations must also respect the applicable non-nuclear legislation, like all other industrial facilities, including for example environmental protection legislation. This may be subject to additional permitting.

Member States but may be adopted by them. In many EU countries with nuclear power plants, the requirements specified in the IAEA safety standards are taken into account when developing the upper two tiers of the above pyramid.

WENRA[166] has developed sets of so-called Safety Reference Levels covering existing nuclear reactors, radioactive waste disposal facilities, radioactive waste and spent fuel storage, radioactive waste treatment and conditioning, and decommissioning. These represent proposed harmonised approaches and WENRA members are committed to incorporating them into their national frameworks. WENRA has also developed a set of safety objectives for new NPPs, which define the expectations for a high level of safety for future NPPs in WENRA countries.

5. Analysis of the legal frameworks for carbon capture and sequestration and radioactive waste/spent fuel disposal

This chapter provides an analysis and comparison of radioactive waste (RW) management with carbon capture and sequestration (CCS). The CCS technology is based on the long-term disposal of waste in geological facilities and it has been included in the taxonomy and received a positive assessment. The Taxonomy Expert Group therefore considers that the challenges of safe long-term disposal of CO_2 in geological facilities, which are similar to the challenges facing disposal of high level radioactive waste, can be adequately managed.

Comparison is also made with the legal framework for the management, including final disposal, of waste and hazardous waste.

5.1. Comparison of radioactive waste management with carbon capture and sequestration

5.1.1. Similar legal frameworks in the Community

At the EU level, Member States adopted the Radioactive Waste Directive [A1-32] and the CCS storage directive [A1-49] for the safe long term disposal of spent fuel/radioactive waste and carbon dioxide respectively. Both directives are currently applicable in all Member States and are binding on them.

From a legal perspective, the set of rules governing the safe management of carbon dioxide disposal and radioactive waste disposal are similar. They regulate an activity in order to protect health and the environment, i.e such activity would be prohibited without a permit. A system of control with roles for Member States, operators, competent authorities and the public ensures that the system works. To illustrate this, the common legal provisions are listed below.

Common legal provisions ensuring safe long-term management of radioactive waste, and carbon dioxide storage
National legal framework, policies and plans
Competent regulatory authority
Clear allocation of responsibilities
Licensing process, control regime, and records
Appropriate finances
Long term responsibility after storage/disposal is closed or sealed
Public participation, stakeholder's involvement and transparency

5.1.2. Similar containment types and long-term management provisions

Both the CCS storage and the Radioactive Waste Directives envision permanent containment/ emplacement in geological formations. In the first case, it is the permanent containment of carbon dioxide in an

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environmentally safe geological storage. In the latter case, it is the safe emplacement of spent fuel and high level radioactive waste in a geological disposal without intention of retrieval.

Both legal frameworks include provisions to deal with the challenges of long-term management. In the case of CCS storage, monitoring is reduced after 20 years, which is the period after which the operator can transfer its responsibility to the competent authority. The operator covers 30 years of monitoring costs after that. In the case of the Radioactive Waste Directive, the safety of the post-closure phase is demonstrated during the licensing process, and long term passive safety measures (e.g. built-in system that can reduce the consequences of a potential accident) are included in national policies. The national programme, which is the practical implementation of the Radioactive Waste Directive, specifies the plans for post closure, the period of controls and the means to preserve knowledge in the longer term. A financing plan covers the finances for radioactive/spent fuel management including the post-closure period. Finally, both legal frameworks provide for transparency and public participation, which, in the particular case of the Radioactive Waste Directive, includes opportunities for the public to participate effectively in the decision-making process.

However, a number of important differences also exist between the two legal instruments. First, the Radioactive Waste Directive proclaims the principle of ultimate responsibility of the Member States, which means that radioactive waste producing Member States must ensure the safe disposal of their radioactive waste and spent fuel, while the CCS Directive does not make disposal compulsory for CO_2 emitters. Second, the Radioactive Waste Directive obliges the Member States to create the three-layered, interrelated system of national policy, national framework and national programme and defines their compulsory elements to ensure efficient implementation, while the CCS Directive relies on a system of storage permits.

5.1.3. Similarity in status of implementation

There is an advanced regulatory framework in place in the Community for both carbon dioxide storage and radioactive waste management. In terms of practical implementation, there is currently no operational geological disposal for carbon dioxide or for radioactive waste.

5.2. Comparison with the legal framework for management of waste and hazardous waste

The directive for the management of waste [A1-48], which includes hazardous waste, sets out an EU legislative framework for the handling of waste in order to prevent a negative impact on the environment or human health. It defines key concepts such as waste, recovery and disposal and puts in place the obligation for an entity carrying out waste management operations to have a permit or to be registered. It also requires Member States to draw up waste management plans.

In order to provide a larger comparison spectrum, the provisions for the safe management of waste that are common to those for the safe management of carbon dioxide storage and radioactive waste are included in the comparison table below. Both the waste directive and the Radioactive Waste Directive have a comprehensive approach towards waste management as they provide for legislation, policies, programmes and plans. The CCS directive, on the other hand, focuses on the licensing system regulating the development and use of a geological storage for dioxide carbon.

All three directives fall within the scope of the Environmental Impact Assessment (EIA) directive. The EIA directive [A1-38] applies to the assessment of the environmental effects of projects which are likely to have significant effects on the environment. Such projects include the final disposal of radioactive waste/spent fuel, storage sites for carbon dioxide, waste disposal installations for incineration, chemical treatment, and landfill of hazardous waste. Those projects require the preparation of an EIA with public participation and examination by the competent authority.

Table A1-1 focuses on the common provisions of the EU legal frameworks for the safe management of radioactive waste/spent fuel, of carbon dioxide storage, and of waste and hazardous waste[167].

[167] MS stands for Member States, EC for European Commission, art. for article

Table A.1-1. Common provisions of the EU legal frameworks for the safe management of radioactive waste/spent fuel, of carbon dioxide storage, and of waste and hazardous waste

	Safe management of radioactive waste (RW) and spent fuel (SF)	Safe management of carbon dioxide (CO_2) storage	Safe management of waste including hazardous waste
Specific objective	MS develop legislation and policies for the safe management of SF and RW to protect people and the environment against ionizing radiation dangers	MS develop legislation to permanently contain carbon dioxide in an environmentally safe geological storage which protects health and environment and combat climate change (art. 1)	MS develop waste legislation and policy to protect health and environment against the adverse impacts of waste and waste management (art. 1)
National legal framework, policies and plans	The **national framework** includes a national programme (practical implementation), legislation, a licensing system with appropriate control and enforcement actions, allocation of responsibilities of the bodies involved, information and participation of the public, and a financing plan. MS improve the framework based on operational experience and technological improvement (art. 5) **National policies** include principles such as keeping the waste generated to a reasonable minimum, the use of proportionate measures, a decision-making process based on evidence, passive safety measures in the long term, and placing the cost burden on the waste generators (art. 4.3) The **national programme** includes the goals of the national policy, their timeframes, the KPI to monitor progress and the responsible body/ies. It also includes an inventory of SF and RW and future quantities, the technical solutions and RDD needed to implement solutions, plans and controls for the post-closure period, a cost assessment and current financing plan, and a transparency policy (art.12)	The directive regulates the development, use and post-closure of a geological containment for CO_2. It includes provisions on a licensing system with appropriate control and enforcement actions, finances, obligations for the bodies involved, and information of the public. MS transposed those provisions in their national laws.	MS develop waste legislation and policy that includes recycling, recovery or disposal (art. 4) The **waste management plans** analyse the national waste management situation, the measures needed, and how to implement the plan. They include the type, quantity and source of waste generated, how it is collected, where it is disposed of, if more waste installations are needed, and criteria to identify a site for future disposal. They also include (art. 28) waste management **policies**, technologies and methods. The **waste prevention programmes** set out the waste prevention objectives, and assess with indicators whether the measures are useful to meet the objectives (art. 29)
Competent regulatory authority	MS establish/designate a competent regulatory authority in the field of SF and RW management that is independent, impartial and has appropriate human	MS establish/designate a competent authority (art.23)	There is a competent authority

	Safe management of radioactive waste (RW) and spent fuel (SF)	Safe management of carbon dioxide (CO_2) storage	Safe management of waste including hazardous waste
	and financial resources (art. 6)		
Clear allocation of responsibilities	**Member States** • establish and maintain national policies on SF and RW management for a high level of safety (art. 1,4) • have ultimate responsibility for the management SF/RW generated on their territory (art. 4) • have an export responsibility: they shall apply strict rules if they want to export SF/RW (art. 4.4) • have an ethical obligation to avoid any undue burden on future generations (recital 24, art. 1) • report on how they implement the directive and submit their SF/RW inventory every 3 years to the EC • carry out self-assessments and invite international peer reviews of their national framework, competent authorities and/or national programme at least every 10 years The **licence holder** is responsible for a SF/RW management activity or facility (art. 3.6). It is primarily responsible for SF/RW management and cannot delegate such responsibility (art. 5f, art. 7.1). It: • regularly assesses safety with evidentiary support (art. 7.2) • demonstrates safety for all phases (art. 7.3) • has measures in place to prevent accidents and mitigate consequences (art. 7.3)	**Member States** • establish a licensing system with control • develop penalties (art. 28) • report on how they implement the directive every 4 years (art. 27) • establish a dispute settlement mechanism (art. 22) **The operator** who operates or controls the storage site and who can delegate such responsibility (art.3.10): • establishes a Monitoring Plan according to criteria in Annex II, and updates it every 5 years to integrate changes in risk assessment, new scientific knowledge, and technology developments (art. 13) • monitors behaviour of CO_2 & formation water, significant irregularities, migration of CO_2, leakage of CO_2, and drinking water, and updates safety assessment of storage in short and long term (art. 13) • reports every year results of monitoring and CO_2 quantities injected (art. 14) • notifies the authority in case of leakages and takes corrective measures (art. 16) The operator can transfer his responsibility for monitoring, reporting and corrective measures to the competent authority after the storage is closed (art. 17-18) Liabilities for the after transfer period should be dealt	**Member States** • develop waste legislation and policy that includes recycling, recovery or disposal (art. 4) • make abandonment, dumping or uncontrolled management of waste illegal and set up penalties (art. 36) • report to the EC every 3 years on how they implement the directive and meet targets for re-use and recycling (art. 37 and 11.5) • review national waste management plans and prevention programmes every 6 years (art. 30) The **waste producer or holder** is responsible for the waste management/treatment. It can delegate such responsibility. It is also financially responsible for the waste management costs (art. 14-15). The producer's responsibility can extend to accept remaining waste after a product is used. The waste treatment operator, collector, transporter and broker: • monitors and controls operations • keeps records of waste type, destination, transport mode and planned treatment (art. 35) • labels hazardous waste during collection, transport and temporary storage according to international standards, and traces it from

	Safe management of radioactive waste (RW) and spent fuel (SF)	Safe management of carbon dioxide (CO_2) storage	Safe management of waste including hazardous waste
	• has adequate financial and human resources (art. 7.5) The **competent regulatory authority** grants permits, inspects regulated activities, and has power to enforce actions (e.g. suspend activity)	with at national level (rec. 34) The **competent authority** grants permits, inspects regulated activities (art. 15), and has power to enforce actions (art.11.3)	production to final destination (art. 17, 19) The **competent authority** establish waste management plans (art. 28.1), grants permits or registers activities (art. 23, 26), and inspects activities (art. 34)
Licensing process, control regime, and records	A licence is needed for the: • siting • design • construction • commissioning • operation • decommissioning, and • closure of a SF/RW facility (art. 3.5) The safety assessment for a disposal facility includes areas of uncertainty, the understanding of natural (geological) and engineered barriers, and how the disposal system develops over time (Recital 34)	A permit is needed for: • exploration, and • storage The storage permit requirements are: • Site characterisation and assessment of expected security • CO_2 quantity and location of facilities • Measures to prevent significant irregularities and corrective measures • Monitoring plan • Provisional post-closure plan • Proof of financial security, and • Proof of financially sound operator which is technically competent with trained staff (art. 7-9) The competent authority establishes a register for storage permits granted and for all closed storage sites (art. 25)	A permit is needed for: • carrying out waste treatment (art. 23) The permit requirements are • identification of waste types and quantities • technical requirements of the site • safety and precautionary measures • monitoring and control operations • closure and after-care provisions (art. 23) The competent authority keeps a register of waste collectors and transporters who do not need a permit or who are exempted from it, e.g. for waste disposal at the production place (art. 26)
Appropriate	Financing plans are required in the national	The operator's financial security is part of the permit	The waste producer or holder is financially responsible for the waste management costs -

	Safe management of radioactive waste (RW) and spent fuel (SF)	Safe management of carbon dioxide (CO_2) storage	Safe management of waste including hazardous waste
finances	programme (art. 11) Adequate financial resources are available to implement national programmes for SF/RW within the national frameworks. They are based on the responsibility of the waste generator (Recital 27, art. 9). Finances should be adequate, secure and transparent as described in a recommendation for the management of financial resources for SF/RW management (1)	requirements (art. 19) The operator makes finances available to authorities to cover post-closure monitoring costs for at least 30 years (art. 20)	polluter-pays principle (art. 14) The investments for future waste installations are included in the waste management plans (art. 28.3c)
Long term responsibility after storage/disposal is closed or sealed	National policies include passive safety measures in the long term (Art. 4.3) The national programme specifies plans for post closure, period of controls and means to preserve knowledge in the longer term (art. 12e) Measures for post-closure periods of disposal facilities are controlled (Art. 5d) Licensing requires safety demonstration of the post-closure phase (Art. 7.3)	**After closure** Inspections are reduced (art. 15.3) Operator is responsible for monitoring, reporting and corrective measures until the responsibility is transferred to the competent authority. The transfer occurs if: • evidence shows that CO_2 is permanently contained • after min 20 years • financial obligations have been fulfilled • site is sealed and injection facilities removed (art. 18) Operator prepares report that shows conformity of actual CO_2 behaviour with the modelled one, the absence of leakage, and that the site evolves towards long-term stability **After transfer of responsibility** routine inspections cease and monitoring is reduced (art.1 8.6) Operator makes finances available to authorities to	Closure and after-care provisions are included in the permit requirements (art. 23) The need for closure of existing waste installations is assessed in the waste management plans (art. 28.3c)

	Safe management of radioactive waste (RW) and spent fuel (SF)	Safe management of carbon dioxide (CO_2) storage	Safe management of waste including hazardous waste
		cover post-closure monitoring costs for at least 30 years (art. 20)	
Public participation, stakeholder's involvement and transparency	MS make available to the public: • international peer reviews of the national framework, regulator and national programme (art. 14.3) • necessary information on SF/RW management (art. 10) MS give the public opportunities to participate effectively in the decision-making process (art. 10, Recital 31) A transparency policy is included in the national programme (art. 12j)	MS make available to the public environmental information on geological storage of CO_2 (art. 26)	MS develop waste legislation and policy in a transparent manner, i.e consulting and involving citizens (art. 4.2) The public can participate in elaboration of waste management plans and programmes which are publicly available (art. 31)

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Source: JRC, 2020

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Annex 2. Summary of LCA results for all lifecycle phases of nuclear energy

In the preparatory phase, a large number of nuclear energy LCA studies from the last two decades were carefully analysed. The main conclusions of this analysis can be summarized as follows:

- The relevance of the studies and the traceability of the conclusions varies to a considerable extent. This is often caused by the quality and actuality of the applied input data, as well as the clarity of the assumptions, estimations and boundary conditions used. Some studies use obsolete and/or inadequate inventory data, simply because they were the only available.
- Some studies mix technologies in a manner that is incompatible with the state-of-the-art of the contemporary nuclear industry and arrive at conclusions, which do not relate to any existing nuclear-based electricity production chain. For example, the combination of mining of very low-grade uranium ores with the simultaneous application of a gas-diffusion enrichment plant using a CO_2-intense electricity mix will result in unrealistically high CO_2 emissions. However, no civilian nuclear company uses such production chain any more around the world.
- The analytical comparison of results from individual LCIA (lifecycle impact assessment) studies is hampered by the fact that a wide variety of environmental impact indicators can be calculated, depending on the impact categories selected by the authors. The GHG emission analysis is an exception, because it can be considered as fully standardized.

In Chapter 3.2, the objective was to compare the lifecycle impact of nuclear energy with other electricity generating technologies. Therefore, studies were selected in which the lifecycle impact of nuclear electricity generation is assessed and compared with other electricity generation technologies in the same study. This was important to ensure that technologies are compared using the same assessment methodologies and consistent assumptions. Furthermore, studies covering a broad range of environmental impact indicators and providing the actual data to allow comparison between them, were also favoured.

After careful consideration of the objectives of the present report, for the purposes of providing detailed and comprehensive lifecycle assessment data to illustrate the environmental impacts of contemporary nuclear electricity generation in Europe, three studies were identified. According to our professional opinion, these studies satisfy the following criteria:

- application of a state-of-the-art LCA and LCIA methodology by recognized authors;
- determination of impact indicators which are well suited to assessing the fulfilment of the environmental objectives of the EU taxonomy;
- utilization of traceable and reliable inventory and other input data in the study;
- proper description of assumptions and estimations made in the study;
- modelling of a “real” nuclear electricity generation chain consisting of production units that adequately represent the current status of nuclear energy in Europe.

The selected three studies were as follows:

- **[A.2-1]** Ch. Poinssot, et al.: Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211
- **[A.2-2]** A. Simons, C. Bauer: Life cycle assessment of the European pressurized reactor and the influence of different fuel cycle strategies, Proc. of the Institution of Mechanical Engineers, Part A: Journal of Power and Energy, April 2012
- **[A.2-3]** X. Zhang, C. Bauer: Life cycle assessment (LCA) of nuclear power in Switzerland, Final Report, Paul Scherrer Institute (PSI), 16 July 2018

Obviously there are many studies of similarly good quality in the literature, but the above three reports have certain merits which are important for the current report:

- They all provide detailed information on the contribution of each stage of the fuel cycle to the environmental impact, thereby allowing identification of the life cycle stages having the major environmental impacts.

— **Ref. [A.2-1]** analyses the entire French nuclear fleet, which in 2019 consisted of 56 production units and delivered 382 TWh (a value corresponding to 52% of the total nuclear electricity generated in EU-28 countries, see https://pris.iaea.org/PRIS/). A special value of this reference is that it analyses a closed fuel cycle and reprocessing of spent nuclear fuel by using reliable data from a large number of NPPs.

— **Ref. [A.2-2]** was mainly selected because it provides a thorough analysis of the EPR. The EPR is a Gen III PWR design and it is expected that in the near future mostly Gen III PWRs will be constructed and operated in Europe.

— **Ref. [A.2-3]** deals with the analysis of nuclear power in Switzerland by performing the LCA for a large PWR (Gösgen NPP, 970 MW_e) and a large BWR (Leibstadt NPP, 1220 MW_e). The special value of this reference is that it is rather recent (2018); it analyses two different reactor types (PWR and BWR) and it deals with the open fuel cycle.

An additional report certainly worth mentioning, because it is a very detailed and accurate description of the potential environmental impacts of the NPP units operated at the Ringhals and Forsmark sites in Sweden. The Vattenfall report is a Type III environmental declaration which was prepared according to the ISO 14025 standard (International Standard ISO 14025 – Environmental labels and declarations – Type III environmental declarations – Principles and procedures):

— [**A.2-4**] Certified Environmental Product Declaration EPD® of Electricity from Vattenfall Nordic Nuclear Power Plants, UNCPC Code 17, Group 171 – Electrical energy, S-P 00923, 2019-12-31, Vattenfall AB, Sweden

Note that the above selected studies are by no means considered as "best" or the "LCA of the European nuclear energy". They are merely used in our study to provide realistic quantitative illustrations of the environmental impacts potentially represented by the various lifecycle phases of nuclear-based electricity production activities.

The following tables provide an overview of results from the [A.2-1] study, which provides the LCA for the French nuclear energy production and focuses on the closed (TTC) fuel cycle, but publishes data for an assumed open fuel cycle (OTC), as well.

Table A.2-1. Summary of LCA results for the closed fuel cycle showing all lifecycle phases of nuclear energy

Non-radioactive impact indicators I.

Lifecycle phase	GHG emission [gCO_2eq/kWh_e]	Atmospheric pollution SO_x [mg/kWh_e]	Atmospheric pollution NO_x [mg/kWh_e]	Water pollution [mg/kWh_e]
Mining	1.704	14.242	19.73	263.07
Conversion	0.278	0.058	1.04	0.087
Enrichment	0.626	0.547	1.06	2.548
Fuel fabr.	0.035	0.013	0.05	0.021
Operation	2.140	0.938	2.84	16.366
Reprocessing	0.376	0.484	0.50	5.433
MOX fabr.	0.027	0.004	0.035	-
Disposal	0.104	0.024	0.097	-
Total	**5.29**	**16.252**	**25.35**	**287.53**

Note: the cyan background means larger than 15% contribution to the total emission, while the yellow background indicates the dominant (largest) contributor.

Non-radioactive impact indicators II.

Lifecycle phase	Land use [m^2/GWh_e]	Water consumption [L/MWh_e]	Water withdrawal [L/MWh_e]	Technological waste [g/MWh_e]
Mining	144.1	17.0	17.0	1.5
Conversion	1.82	4.6	4.6	2.0
Enrichment	1.88	23.0	23.0	0.65
Fuel fabr.	0.93	0.2	0.2	0.23
Operation	45.1	1460.0	72318.0	20.15
Reprocessing	4.98	1.7	1.7	0.63
MOX fabr.	0.13	0.1	0.1	0.18
Disposal	12.01	0.1	0.1	1.11
Total	**211.0**	**1507.0**	**72365.0**	**26.45**

Potential impact indicators

Lifecycle phase	Acidification potential [gSO_2eq/MWh_e]	POCP [gC_2H_4eq/MWh_e]	Eutrophication [gPO_4eq/MWh_e]	Eco-toxicity [g1,4-DCBeq/MWh_e]	Human toxicity [g1,4-DCBeq/MWh_e]
Mining	28.06	2.436	2.774	637.597	1225.207
Conversion	0.90	0.149	0.148	0.205	1.348
Enrichment	1.25	0.055	0.918	0.229	1.428
Fuel fabr.	0.05	0.002	0.015	-	0.064
Operation	2.89	0.151	0.760	0.005	4.331
Reprocessing	0.84	0.039	0.583	0.185	0.779
MOX fabr.	0.03	0.001	0.005	-	0.043
Disposal	0.09	0.007	0.013	-	0.124
Total	**34.11**	**2.840**	**5.216**	**638.221**	**1233.32**

Radioactive impact indicators

Lifecycle phase	Gaseous radioactive emissions [Bq/kWh_e]	Liquid radioactive emissions [Bq/kWh_e]	Solid radioactive waste production [m^3/TWh_e]			
			VLLW	LILW-SL	ILW-LL	HLW
Mining	666 744.0	-	3 190.0	-	-	-
Conversion	-	53.8	1.97	1.19	-	-
Enrichment	-	-	-	-	-	-
Fuel fabr.	-	-	-	-	-	-
Operation	162.0	2717.0	22.94	24.61	0.32	-
Reprocessing	554 628.0	24 444.0	2.63	4.31	0.80	0.36
MOX fabr.	-	-	0.019	0.10	0.05	-
Total	**1 221 534.0**	**27 215.0**	**3 217.56**	**30.21**	**1.17**	**0.36**

Data from Ref. [A.2-1]

With regard to the production of solid radioactive waste in the closed cycle:

Solid VLLW

The dominant contributor to solid VLLW (3218 m^3/TWh_e) is uranium mining (>**99%**), while reactor operations and other fuel cycle operations do not produce significant amount of VLLW.

Solid LILW-SL

The total volume of solid LILW-SL is around 30 m^3/TWh_e: reactor operations are responsible for **81%** of this volume, while reprocessing contributes **14%**, the rest mainly comes from conversion.

Solid ILW-LL

Wastes belonging to the ILW-LL category (1.17 m^3/TWh_e) mainly come from reprocessing (**68%**) and reactor operations (**27%**), the rest can be attributed to MOX fuel fabrication.

Solid HLW

HLW (0.36 m^3/TWh_e) are only produced during the spent fuel reprocessing operations, where separated fission products and minor actinides are vitrified and stored in the form of nuclear glass waste in steel canisters.

Note that for the open fuel cycle the HLW production rate is 1.17 m^3/TWh_e, which is more than **three times higher** than for the closed cycle (in the open cycle all spent fuel is treated as high level **waste**).

Table A.2-2. Summary of LCA results for the open fuel cycle showing all lifecycle phases of nuclear energy

Non-radioactive impact indicators I.

Lifecycle phase	**GHG emission [gCO_2eq/kWh_e]**	**Atmospheric pollution SO_x [mg/kWh_e]**	**Atmospheric pollution NO_x [mg/kWh_e]**	**Water pollution [mg/kWh_e]**
Mining	2.037	17.03	23.60	314.60
Conversion	0.308	0.06	1.16	0.1
Enrichment	0.696	0.61	1.18	2.83
Fuel fabr.	0.039	0.01	0.06	0.02
Operation	2.141	0.94	2.84	16.37
Disposal	0.227	0.09	0.239	-
Total	**5.45**	**18.74**	**29.08**	**333.92**

Non-radioactive impact indicators II.

Lifecycle phase	**Land use [m^2/GWh_e]**	**Water consumption [L/MWh_e]**	**Water withdrawal [L/MWh_e]**	**Technological waste [g/MWh_e]**
Mining	172.4	20.0	20.0	1.5
Conversion	2.0	5.1	5.1	2.2
Enrichment	2.1	25.0	25.0	0.7
Fuel fabr.	1.0	0.2	0.2	0.3
Operation	45.1	1 460.0	72 318.0	20.1
Disposal	12.0	0.1	0.1	4.1
Total	**234.6**	**1 510.4**	**72 368.0**	**28.9**

Potential impact indicators

Lifecycle phase	Acidification potential [gSO_2eq/MWh_e]	POCP [gC_2H_4eq/MWh_e]	Eutrophication [gPO_4eq/MWh_e]	Eco-toxicity [g1,4-DCBeq/MWh$_e$]	Human toxicity [g1,4-DCBeq/MWh$_e$]
Mining	33.551	2.914	3.317	761.117	1463.489
Conversion	1.002	0.165	0.164	0.226	1.493
Enrichment	1.395	0.061	1.018	0.252	1.587
Fuel fabr.	0.053	0.002	0.017	-	0.071
Operation	2.887	0.151	0.760	0.005	4.331
Disposal	0.253	0.023	0.031	-	0.313
Total	**39.14**	**3.32**	**5.31**	**761.60**	**1471.3**

Radioactive impact indicators

Lifecycle phase	Gaseous radioactive emissions [Bq/kWh$_e$]	Liquid radioactive emissions [Bq/kWh$_e$]	Solid radioactive waste production [m^3/TWh_e]			
			VLLW	LILW-SL	ILW-LL	HLW
Mining	797 352.0	-	3 815.0	-	-	-
Conversion	-	60.0	2.15	1.29	-	-
Enrichment	-	-	-	-	-	-
Fuel fabr.	-	-	-	-	-	-
Operation	162.0	2 717.0	22.95	24.61	0.32	1.17
Total	**797 514.0**	**2 777.0**	**3 840.0**	**25.90**	**0.32**	**1.17**

Data from Ref. [A.2-1]

Table A.2-3. Comparison of LCA results for the closed and open fuel cycles in all lifecycle phases of nuclear energy

Non-radioactive impact indicators I.

Fuel cycle type	GHG emission [gCO_2eq/kWh_e]	Atmospheric pollution SO_x [mg/kWh$_e$]	Atmospheric pollution NO_x [mg/kWh$_e$]	Water pollution [mg/kWh$_e$]
Open	5.45	18.74	29.08	333.92
Closed	5.29	16.25	25.35	287.53

Non-radioactive impact indicators II.

Fuel cycle type	Land use [m^2/GWh_e]	Water consumption [L/MWh$_e$]	Water withdrawal [L/MWh$_e$]	Technological waste [g/MWh$_e$]
Open	234.6	1 510.4	72 368.0	28.9
Closed	211.0	1 507.0	72 365.0	26.5

Potential impact indicators

Fuel cycle type	Acidification potential [gSO_2eq/MWh_e]	POCP [gC_2H_4eq/MWh_e]	Eutrophication [gPO_4eq/MWh_e]	Eco-toxicity [g1,4-DCBeq/MWh_e]	Human toxicity [g1,4-DCBeq/MWh_e]
Open	39.14	3.32	5.31	761.60	1 471.3
Closed	34.11	2.84	5.22	638.22	1 233.3

Radioactive impact indicators

Fuel cycle type	Gaseous radioactive emissions [Bq/kWh_e]	Liquid radioactive emissions [Bq/kWh_e]	Solid radioactive waste production [m^3/TWh_e]			
			VLLW	LILW-SL	ILW-LL	HLW
Open	797 514.0	2 777.0	3 840.0	25.90	0.32	1.17
Closed	1 221 534.0	2 7215.0	3 217.6	30.21	1.17	0.36

Data from Ref. [A.2-1]

Table A.2-4. Comparison of GHG emissions published in Refs. [A.2-1], [A2.-2] and [A.2-3] in all lifecycle phases of nuclear energy

GHG emissions [gCO2eq/kWhe]

Lifecycle phase	PWR-CH (mixed)	EPR-CH (open)	EPR-FR (closed)	French-TTC (closed)	French-OTC (open)	PWR-CH2 (open)	BWR-CH (open)
Mining	1.96	2.36	2.00	1.704	2.037	2.52	4.70
Conversion	1.21	1.18	0.96	0.278	0.308	0.28	0.38
Enrichment	0.42	0.33	0.15	0.626	0.696	0.56	1.69
Fuel fabr.	0.04	0.03	0.02	0.035	0.039	0.05	0.02
Operation	1.05	0.75	0.80	2.140	2.141	1.18	1.32
Reprocessing	0.21	-	0.41	0.376	-	-	-
MOX fabr.	0.05	-	0.03	0.027	-	-	-
Disposal	0.36	0.35	0.23	0.104	0.227	1.01	1.32
Total	**5.30**	**5.00**	**4.60**	**5.29**	**5.45**	**5.60**	**9.40**

PWR-CH = 1000 MW_e Gen II PWR; "mixed" fuel cycle (8% use of MOX fuel, see [A.2-2])

Enrichment = 60% diffusion / 40% centrifuge

EPR-CH = 1600 MW_e Generation III PWR (EPR); "open" fuel cycle (see [A.2-2])

Enrichment = 100% centrifuge

EPR-FR = 1600 MW_e Generation III PWR (EPR); "closed" fuel cycle (see [A.2-2])

Enrichment = 100% centrifuge

French-TTC = analysis of the French nuclear fleet; "closed" fuel cycle (see [A.2-1])

Enrichment = 100% centrifuge (gaseous diffusion but with nuclear energy)

French-OTC = analysis of the French nuclear fleet; "open" fuel cycle (see [A.2-1])

Enrichment = 100% centrifuge (gaseous diffusion but with nuclear energy)

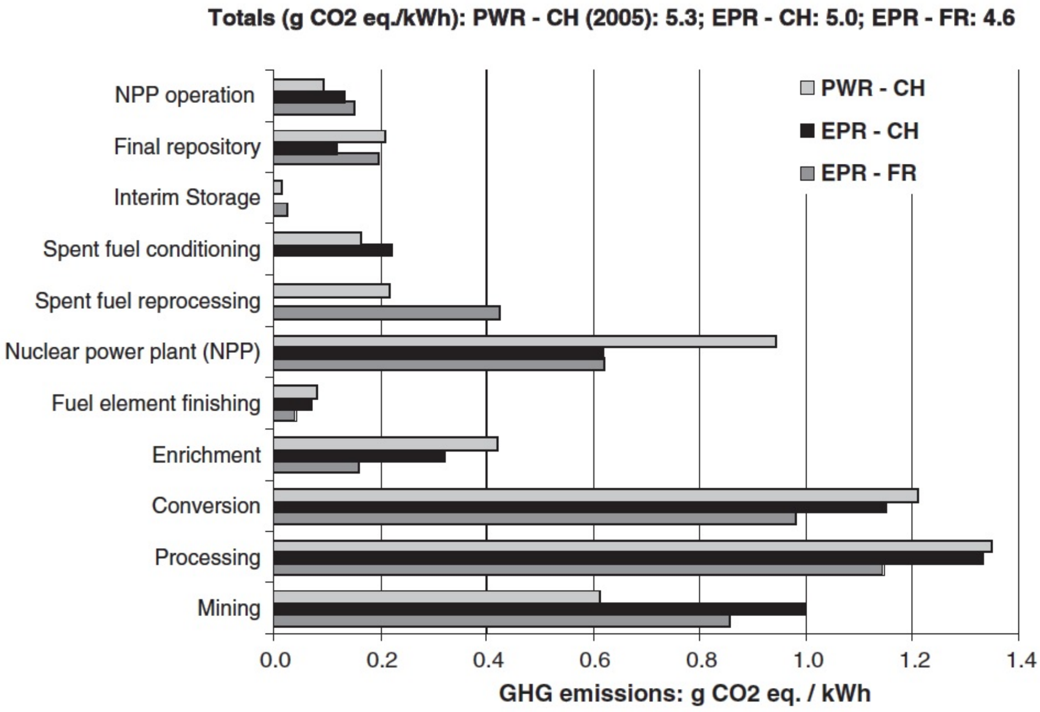
PWR-CH2 = analysis of a large Swiss PWR (Gösgen NPP, 970 MWe); "open" fuel cycle (see [A.2-3])

"Baseline scenario", Enrichment = 100% centrifuge

BWR-CH = analysis of a large Swiss BWR (Leibstadt NPP, 1220 MWe); "open" fuel cycle (see [A.2-3])

"Baseline scenario", Enrichment = 100% centrifuge

Figure A.2-1. GHG emissions for the three reactor types investigated in Ref. [A.2-2]



Source: [A.2-2]

Table A.2-5 Comparison of LCA results for the open fuel cycle (French-OTC) and the Vattenfall EPD in all lifecycle phases of nuclear energy

Non-radioactive impact indicators I.

Fuel cycle type	GHG emission [gCO_2eq/kWh_e]	Atmospheric pollution SO_x [mg/kWh_e]	Atmospheric pollution NO_x [mg/kWh_e]	Water pollution [mg/kWh_e]
French-OTC	5.45	18.74	29.08	333.92
Vattenfall	4.13 (2.48)	11.80 (4.92)	10.3 (6.70)	NCDA

Non-radioactive impact indicators II.

Fuel cycle type	Land use [m^2/GWh_e]	Water consumption [L/MWh_e]	Water withdrawal [L/MWh_e]	Technological waste [g/MWh_e]
French-OTC	234.6	1 510.4	72 368.0	28.9
Vattenfall	NCDA	NCDA	NCDA	NCDA

Potential impact indicators

Fuel cycle type	Acidification potential [gSO_2eq/MWh_e]	POCP [gC_2H_4eq/MWh_e]	Eutrophication [gPO_4eq/MWh_e]	Eco-toxicity [g1,4-DCBeq/MWh_e]	Human toxicity [g1,4-DCBeq/MWh_e]
French-OTC	39.14	3.32	5.31	761.60	1 471.3
Vattenfall	18.50 (8.61)	2.14 (0.76)	8.56 (3.55)	NCDA	NCDA

Radioactive impact indicators

Fuel cycle type	Gaseous radioactive emissions [Bq/kWh$_e$]	Liquid radioactive emissions [Bq/kWh$_e$]	Solid radioactive waste production [m^3/TWh$_e$]			
			VLLW	LILW-SL	ILW-LL	HLW
French-OTC	797 514.0	2 777.0	3 840.0	25.90	0.32	1.17
Vattenfall	165.9(1)	NCDA	NCDA	NCDA	NCDA	2.26(2)

(1) Includes 14C, 85Kr and 222Rn, but does not include radon emissions due to mining activities

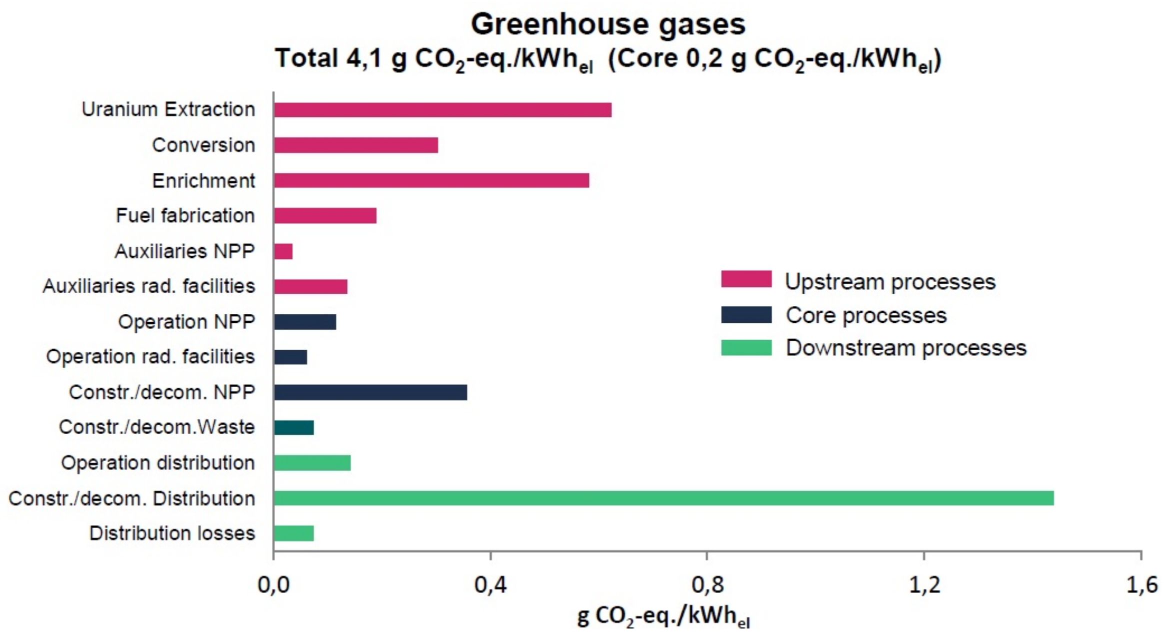
(2) Includes fuel assembly structural components such as steel, Zircaloy and Inconel

NCDA = No Comparable Data Available (e.g. only the total LLW + ILW amounts were reported)

In the "Vattenfall" row usually two values are given, the first number (without brackets) shows the "total distributed" value which includes also contribution from the grid. The number in the bracket shows the "total generated" value which does not include the grid contribution.

Data from Ref. [A.2-1] and Ref. [A.2-4]

Figure A.2-2. Contribution to the GHG emission from the various lifecycle phases for the Vattenfall NPP units



Note the significant (dominant) contribution from the transmission grid ("downstream" processes)

Source: Ref. [A.2-4]

References for Annex 2

[A.2-1] Ch. Poinssot, et al.: Assessment of the environmental footprint of nuclear energy systems. Comparison between closed and open fuel cycles, Energy 69 (2014) 199-211

[A.2-2] A. Simons, C. Bauer: Life cycle assessment of the European pressurized reactor and the influence of different fuel cycle strategies, Proc. of the Institution of Mechanical Engineers, Part A: Journal of Power and Energy, April 2012

[A.2-3] X. Zhang, C. Bauer: Life cycle assessment (LCA) of nuclear power in Switzerland, Final Report, Paul Scherrer Institute (PSI), 16 July 2018

[A.2-4] Certified Environmental Product Declaration EPD® of Electricity from Vattenfall Nordic Nuclear Power Plants, UNCPC Code 17, Group 171 – Electrical energy, S-P 00923, 2019-12-31, Vattenfall AB, Sweden.

Annex 3. NACE codes corresponding to main lifecycle phases of nuclear energy

The TEG reports refer to the various investigated activities by using their NACE codes. The NACE provides the statistical classification of economic activities in the EU and the abbreviation comes from the French name "Nomenclature statistique des Activités économiques dans la Communauté Européenne"). The current version is defined in Regulation (EC) Nº 1893/2006.

Table A.3-1 shows the NACE codes belonging to those activities that can potentially be present in the lifecycle of nuclear energy, depending on whether an "open" or a "closed" fuel cycle is applied. This table is not exhaustive and may be further specified, if required.

Table A.3-1. – List of nuclear energy related NACE codes (the list is not exhaustive)

NACE macro-sector code(1)	Activity as defined in NACE	NE lifecycle phases covered
B – Mining and quarrying	B.07 – Mining of non-ferrous metal ores *B.07.21 – Mining of uranium and thorium ores* Note that this item also includes the concentration of such ores and the manufacture of "yellowcake"	- Uranium mining & milling - Preparation of yellow cake
C – Manufacturing	C.20 – Manufacture of chemicals and chemical products *C.20.13 - Manufacture of other inorganic basic chemicals* Note that this item also includes the enrichment of uranium and thorium ores	- Enrichment of uranium - Nuclear fuel reprocessing
C – Manufacturing	C.24 – Manufacture of basic metals *C.24.46 – Processing of nuclear fuel* Note that this item also includes production of uranium metal from ores and smelting / refining of uranium	- Manufacture of nuclear fuel elements and fuel assemblies
C – Manufacturing	C.25 – Manufacture of fabricated metal products, except machinery and equipment *C.25.30 – Manufacture of steam generators, except central heating hot water boilers*	- Manufacture of nuclear reactors
C – Manufacturing	C.33 – Repair and installation of machinery & equipment *C.33.11 – Repair of fabricated metal products*	- Repair and maintenance of nuclear reactors
D – Electricity, gas, steam and air conditioning supply	D.35 – Electricity, gas, steam and air conditioning supply *D.35.11 – Production of electricity*	- NPP operations
D – Electricity, gas, steam and air conditioning supply	D.35 – Electricity, gas, steam and air conditioning supply *D.35.30 – Steam and air conditioning supply*	- NPP operations (cogenerating plants)
E – Water supply; sewerage, waste management and remediation activities	E.38 – Waste collection, treatment and disposal activities; materials recovery E.38.12 – Collection of hazardous waste E.38.22 – Treatment and disposal of hazardous waste E.39 – Remediation activities & other waste management services E.39.00 – Remediation activities & other waste management services	- RW management - RW management - Decommissioning and remediation of NPP sites
F – Construction	F.42 – Civil engineering F.42.22 – Construction of utility projects for electricity and telecommunications	- Construction of NPPs

(1) Based on EUROSTAT – RAMON – Reference and management of nomenclatures (database), see also https://nacev2.com/ for on-line searches. Note that RW stands for "radioactive waste".

In Table A.3-1 the coloured background indicates that the NACE code shown in the specific row is elaborated in the TEG reports, i.e. TSC were developed in [A.3-2] for the (non-nuclear) activity.

References for Annex 3

[A.3-1] Financing a Sustainable European Economy, Technical Report, Taxonomy: Final report of the Technical Expert Group on Sustainable Finance, March 2020

[A.3-2] Taxonomy Report, Technical Annex, Updated methodology & Updated Technical Screening Criteria, March 2020

[A.3-3] Taxonomy Regulation: Regulation (EU) 2020/852 of the European Parliament and of the Council of 18 June 2020 on the establishment of a framework to facilitate sustainable investment, and amending Regulation (EU) 2019/2088

Annex 4. Illustrative TSC tables

1. Electricity generation from nuclear energy

Description of the activity

Construction or operation of electricity generation facilities that produce electricity from nuclear energy.

The activity is classified under NACE codes D35.11[168] and F42.22[169] in accordance with the statistical classification of economic activities established by Regulation (EC) No 1893/2006.

Technical screening criteria

Substantial contribution to climate change mitigation
Life cycle GHG emissions from the generation of electricity from nuclear energy are lower than 100 gCO_{2e}/kWh. Life-cycle GHG emission savings are calculated using Commission Recommendation 2013/179/EU or, alternatively, using ISO 14067:2018 or ISO 14064-1:2018. Quantified life-cycle GHG emissions are verified by an independent third party. ***1. Extension of the service time of existing nuclear power plants*** Nuclear safety characteristics of the existing facility – including its accident prevention and mitigation features – shall at least comply with the WENRA Safety Reference Levels for Existing Reactors[170] and with the provisions of the Euratom Nuclear Safety Directive (NSD)[171], taking into account the guidance outlined in Art 8a Par 2(b) of the NSD[172]. Radioprotection provisions of the facility shall at least comply with the requirements laid down in the Euratom Basic Safety Standards (BSS)[173]. ***2. Construction and operation of new nuclear power plants*** Nuclear safety characteristics of the new facility – including its accident prevention & mitigation features – shall at least fulfil the WENRA Safety Objectives for New Nuclear Power Plants[174] and comply with the provisions of the Euratom Nuclear Safety Directive. The same conditions apply when the new facility is constructed and operated outside the EU. Radioprotection provisions of the facility shall at least comply with the requirements laid down in the Euratom Basic Safety Standards. Outside the EU, the radiological protection shall at least comply with the latest recommendations of the International Commission on Radiological Protection (ICRP)[175].

Do no significant harm ('DNSH')	
(2) Climate change adaptation	The activity complies with the criteria set out in Appendix E to this Annex. ***1. Extension of the service time of existing nuclear power plants*** Compliance with the WENRA Safety Reference Levels for Existing Reactors and the Euratom NSD ensures that the existing facility is able to cope with extreme

168 NACE D35.11 = Production of electricity (D.35 = Electricity, gas, steam and air conditioning supply)

169 NACE F42.22 = Construction of utility projects for electricity and telecommunications (F.42 = Civil engineering)

170 http://www.wenra.org/media/filer_public/2014/09/19/wenra_safety_reference_level_for_existing_reactors_september_2014.pdf

171 Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014

172 Art 8a Par 2(b) of the NSD states that "...the objective set out in paragraph 1 is used as a reference for the timely implementation of reasonably practicable safety improvements to existing nuclear installations, including in the framework of the periodic safety reviews as defined in Article 8c(b)"

173 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

174 http://www.wenra.org/media/filer_public/2013/08/23/rhwg_safety_of_new_npp_designs.pdf

175 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

	natural hazards (such as floods and extreme weather conditions) potentially resulting from future climate change. The resilience of the EU nuclear power plants against extreme natural hazards (including earthquakes) was demonstrated in the EU stress-tests exercise[176]. ***2. Construction and operation of new nuclear power plants*** Fulfilling the WENRA Safety Objectives for New Nuclear Power Plants and compliance with the Euratom NSD guarantees that the new facility will be able to cope with extreme natural hazards (such as floods and extreme weather conditions) potentially resulting from future climate change.
(3) Sustainable use and protection of water and marine resources	Environmental degradation risks related to preserving water quality and avoiding water stress are identified and addressed, in accordance with a water use and protection management plan, developed in consultation with relevant stakeholders[177]. In order to limit thermal anomalies associated with the discharge of waste heat, inland nuclear power plants utilizing once-through wet cooling by taking water from a river or a lake shall control the: — maximum temperature of recipient freshwater body after mixing; — maximum temperature difference between the discharged cooling water and the recipient freshwater body The temperature control shall be implemented according to the individual licence conditions for the specific operations, where applicable, and/or national threshold values in line with the EU regulatory framework[178]. For activities performed outside the EU, the relevant IFC standards[179] are applicable.
(4) Transition to a circular economy	A plan for the management of conventional and radioactive waste is in place and ensures maximal reuse or recycling at end of life in accordance with the waste hierarchy, including through contractual agreements with waste management partners, reflection in financial projections or official project documentation. During operation and decommissioning, the amount of radioactive waste is minimized and the amount of free-release waste is maximized. Plans for the long-term safe disposal of high-level radioactive waste resulting from the activity during its whole life cycle are in place and their adequacy is demonstrated by science-based evidence or empirical data.
(5) Pollution	Non-radioactive emissions are within or lower than the emission levels

[176] Communication from the Commission to the Council and the European Parliament on the comprehensive risk and safety assessments ("stress tests") of nuclear power plants in the European Union and related activities, COM(2012) 571 final, Brussels, 4.10.2012

[177] As required by Directive 2000/60/EC (Water Framework Directive) for activities subject to Union law, or as required by equivalent national provisions or international standards addressing environmental degradation risks related to preserving water quality and avoiding water stress for activities in third countries.
Where an Environmental Impact Assessment is carried out in accordance with Directive 2011/92/EU (Consolidated EIA Directive) and includes an assessment of the impact on water in accordance with Directive 2000/60/EC, no additional assessment of impact on water is required, provided the risks identified have been addressed.

[178] As required by Directive 78/659/EEC (repealed by the WFD in 2013) on the quality of fresh waters needing protection or improvement in order to support fish life and the BREF ICS (Integrated Pollution Prevention and Control (IPPC) Reference Document on the application of Best Available Techniques to Industrial Cooling Systems, European Commission, December 2001)

[179] a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007 and b) IFC Environmental, Health and Safety Guidelines for Thermal Power Plants, IFC, December 2008

prevention and control	associated with the best available techniques (BAT-AEL) ranges set out in the best available techniques (BAT) conclusions for large combustion plants[180]. No significant cross-media effects occur[181]. For nuclear power plants greater than 1 MW thermal input but below the thresholds for the BAT conclusions for large combustion plants to apply, emissions are below the emission limit values set out in Annex II, part 2, to Directive (EU) 2015/2193 of the European Parliament and of the Council[182]. Radioactive discharges to air, water bodies and ground (soil) shall comply with individual licence conditions for the specific operations, where applicable, and/or national threshold values in line with the EU regulatory framework (i.e. BSS[183] and Drinking Water Directive[184]). For activities performed outside the EU, the ICRP recommendations[185] are applicable. The ALARA (as low as reasonably achievable) principle should be applied during the control of radioactive discharges consistently.
(6) Protection and restoration of biodiversity and ecosystems	An Environmental Impact Assessment (EIA) or screening[186] has been completed, for activities within the Union, in accordance with Directive 2011/92/EU. For activities in third countries, an EIA has been completed in accordance with equivalent national provisions or international standards[187]. Where an EIA has been carried out, the required mitigation and compensation measures for protecting the environment are implemented. For sites/operations located in or near biodiversity-sensitive areas (including the Natura 2000 network of protected areas, UNESCO World Heritage sites and Key Biodiversity Areas, as well as other protected areas), an appropriate assessment[188], where applicable, has been conducted and based on its conclusions the necessary mitigation measures[189] are implemented.

[180] Commission Implementing Decision (EU) 2017/1442 of 31 July 2017 establishing best available techniques (BAT) conclusions, under Directive 2010/75/EU of the European Parliament and of the Council, for large combustion plants (OJ L 212, 17.8.2017, p. 1).

[181] Cross-media effects are characterized by global warming, human- and aquatic toxicity, acidification, eutrophication, ozone depletion and POCP (photochemical ozone creation potential) impact categories, see "Integrated Pollution Prevention & Control", Ref. Document on Economics and Cross-Media Effects, EC, July 2006

[182] Directive (EU) 2015/2193 of the European Parliament and of the Council of 25 November 2015 on the limitation of emissions of certain pollutants into the air from medium combustion plants (OJ L 313, 28.11.2015, p. 1).

[183] Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

[184] Council Directive 98/83/EC of 3 November 1998 on the quality of water intended for human consumption

[185] The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007.

[186] The procedure through which the competent authority determines whether projects listed in Annex II to Directive 2011/92/EU is to be made subject to an environmental impact assessment (as referred to in Article 4(2) of that Directive).

[187] For example, IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks.

[188] In accordance with Directives 2009/147/EC and 92/43/EEC, or, for activities located in third countries, in accordance with equivalent national provisions or international standards, for example IFC Performance Standard 6: Biodiversity Conservation and Sustainable Management of Living Natural Resources.

[189] Those measures have been identified to ensure that the project, plan or activity will not have any significant effects on the conservation objectives of the protected area.

2. Mining and processing of uranium ore

Description of the activity

Construction and operation of uranium mines and associated uranium ore processing facilities, including "yellowcake" manufacturing factories.

The activity is classified under NACE codes B.07.21[190] in accordance with the statistical classification of economic activities established by Regulation (EC) No 1893/2006.

The activity is an enabling activity in accordance with Article 10(1), point (i) of Regulation (EU) 2020/852[191] where it complies with the technical screening criteria set out in this section.

Technical screening criteria

Substantial contribution to climate change mitigation
The economic activity explores, retrieves and processes uranium ore to produce "yellowcake"[192], which is the most important base material for manufacturing nuclear fuel used for low-carbon electricity generation in nuclear power plants. As of today, three basic mining technologies are used: open-pit or underground mines and in-situ leaching. During all phases of these uranium extraction technologies, but especially for those taking place in underground mines, efficient and comprehensive worker's health protection shall be implemented, with particular attention to the prevention of radon inhalation and incorporation of other radioactive substances. Radioprotection provisions of the associated facilities shall at least comply with the requirements laid down in the Euratom Basic Safety Standards (BSS)[193]. For activities performed outside the EU, the ICRP recommendations[194] are applicable.

Do no significant harm ('DNSH')	
(2) Climate change adaptation	The activity complies with the criteria set out in Appendix E to this Annex.
(3) Sustainable use and protection of water and marine resources	Environmental degradation risks related to preserving water quality and avoiding water stress are identified and addressed, in accordance with a water use and protection management plan, developed in consultation with relevant stakeholders[195]. In uranium mines, where large waste rock dumps are created and/or large tailings are piled up, adequate management of these dumps and tailings shall be ensured, in order to prevent: — contamination of surface- and groundwater by heavy metals and radioactive

190 NACE B.07.21 = Mining of non-ferrous metal ores / Mining of uranium and thorium ores, including the concentration of such ores and the manufacture of "yellowcake"

191 Article 10(1), point (i) of Regulation (EU) 2020/852 refers to activities enabling any of the activities listed in points (a) to (h) of paragraph 10(1) in accordance with Article 16. Article 10(1) point (a) refers to "generating, transmitting, storing, distributing or using renewable energy in line with Directive (EU) 2018/2001" and in this sense concerns only activities related to renewable energy. However, if nuclear energy is to be included into the Taxonomy as an activity significantly contributing to the climate change mitigation objective, then all related raw material mining, processing and fuel manufacturing activities should be included as activities enabling nuclear energy based electricity production.

192 The "yellowcake" is a yellow-coloured powder containing 70 to 90 weight% uranium oxide (U_3O_8).

193 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

194 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

195 As required by Directive 2000/60/EC (Water Framework Directive, WFD) for activities subject to Union law, or as required by equivalent national provisions or international standards addressing environmental degradation risks related to preserving water quality and avoiding water stress for activities in third countries.
Where an Environmental Impact Assessment is carried out in accordance with Directive 2011/92/EU (Consolidated EIA Directive) and includes an assessment of the impact on water in accordance with Directive 2000/60/EC (WFD), no additional assessment of impact on water is required, provided the risks identified have been addressed.

	substances; — acidification of groundwater; — seepage or large – accidental – discharge of contaminated water; The prevention measures shall ensure adequate isolation (containment) of the dumps and tailings from their environment. If in-situ leaching is used to extract uranium then groundwater quality shall be maintained or restored after completing the extraction activities. Prevention and restoration measures shall be implemented according to individual licence conditions for the specific operations, where applicable, and/or national regulations in line with the EU regulatory framework[196]. For activities performed outside the EU, the relevant IFC standards and guidelines[197] are applicable.
(4) Transition to a circular economy	A plan for the management of conventional and radioactive waste is in place and ensures maximal reuse or recycling at end of life in accordance with the waste hierarchy. During operation and facility closure (including site remediation), the amount of radioactive waste is minimized and the amount of free-release waste is maximized.
(5) Pollution prevention and control	Non-radioactive emissions are within or lower than the emission levels associated with the best available techniques (BAT) ranges set out in the best available techniques (BAT) conclusions for extractive industries[198]. For activities performed outside the EU, the relevant IFC standards and guidelines[199] are applicable. Radioactive discharges to air, water bodies and ground (soil) shall comply with the individual licence conditions for the specific operations, where applicable, and/or national threshold values in line with the EU regulatory framework (i.e. BSS[200], Drinking Water Directive[201] and Water Framework Directive[202]). Special protection measures shall be implemented wherever necessary, to prevent radon emanation from dumps/tailings and dispersion of radioactive dust. For activities performed outside the EU, the ICRP recommendations[203] are applicable. The ALARA (as low as reasonably achievable) principle should be applied during the control of radioactive discharges consistently.
(6) Protection and restoration of	An Environmental Impact Assessment (EIA) or screening[204] has been completed, for activities within the Union, in accordance with Directive 2011/92/EU. For

196 Directive 2006/21/EC of the European Parliament and of the Council of 15 March 2006 on the management of waste from extractive industries and amending Directive 2004/35/EC

197 a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007; b) IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007 and c) IFC EHS Guidelines – Mining, IFC, December 2007

198 Best Available Techniques (BAT) Reference Document for the Management of Waste from Extractive Industries in accordance with Directive 2006/21/EC, JRC Science for Policy Report, JRC109657 (EUR 28963 EN), 2018

199 a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007; b) IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007 and c) IFC EHS Guidelines – Mining, IFC, December 2007

200 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

201 Council Directive 98/83/EC of 3 November 1998 on the quality of water intended for human consumption

202 Directive 2000/60/EC of the European Parliament and of the Council of 23 October 2000 establishing a framework for Community action in the field of water policy

203 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

204 The procedure through which the competent authority determines whether projects listed in Annex II to Directive 2011/92/EU is to be made subject to an environmental impact assessment (as referred to in Article 4(2) of that Directive).

biodiversity and ecosystems	activities in third countries, an EIA has been completed in accordance with equivalent national provisions or international standards[205]. Where an EIA has been carried out, the required mitigation and compensation measures for protecting the environment are implemented. For sites/operations located in or near biodiversity-sensitive areas (including the Natura 2000 network of protected areas, UNESCO World Heritage sites and Key Biodiversity Areas, as well as other protected areas), an appropriate assessment[206], where applicable, has been conducted and based on its conclusions the necessary mitigation measures[207] are implemented.

205 For example, IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks.

206 In accordance with Directives 2009/147/EC and 92/43/EEC, or, for activities located in third countries, in accordance with equivalent national provisions or international standards, for example IFC Performance Standard 6: Biodiversity Conservation and Sustainable Management of Living Natural Resources.

207 Those measures have been identified to ensure that the project, plan or activity will not have any significant effects on the conservation objectives of the protected area.

3. Reprocessing of spent nuclear fuel

Description of the activity

Construction or operation of facilities for the reprocessing of spent nuclear fuel.

The activity is classified under NACE codes C.20.13[208] in accordance with the statistical classification of economic activities established by Regulation (EC) No 1893/2006.

Technical screening criteria

Substantial contribution to climate change mitigation

The economic activity chemically separates fission products, plutonium and unused uranium from spent nuclear fuel. The basic aim of the reprocessing activity is to recover plutonium and uranium for reuse, as these materials still have significant potential as nuclear fuel for electricity production. The reprocessed plutonium can be utilized to manufacture MOX[209] fuel for nuclear power reactors. The reprocessed uranium can be re-enriched and re-used as uranium oxide fuel.

Nuclear safety characteristics of the facility shall comply with the provisions of the Euratom Nuclear Safety Directive (NSD)[210]. Facilities constructed and operated outside the EU shall be designed and operated in accordance with applicable IAEA[211] safety standards or national legislation having equivalent requirements.

Radioprotection provisions of the facility shall at least comply with the requirements laid down in the Euratom Basic Safety Standards (BSS)[212].

Outside the EU, the radiological protection shall at least comply with the latest recommendations of the International Commission on Radiological Protection (ICRP)[213].

Do no significant harm ('DNSH')

(2) Climate change adaptation	The activity complies with the criteria set out in Appendix E to this Annex. Compliance with the Euratom NSD ensures that the facility is able to cope with extreme natural hazards (such as floods and extreme weather conditions) potentially resulting from future climate change. The resilience of the EU nuclear installations against extreme natural hazards (including earthquakes) was demonstrated in the EU stress-tests exercise[214,215].
(3) Sustainable use and protection of water and marine resources	Environmental degradation risks related to preserving water quality and avoiding water stress are identified and addressed, in accordance with a water use and protection management plan, developed in consultation with relevant stakeholders[216].

208 NACE C.20.13 = Manufacture of other inorganic basic chemicals (C.20 = Manufacture of chemicals and chemical products). Note that this item also includes the enrichment of uranium and thorium ores and nuclear fuel reprocessing.

209 MOX = Mixed (plutonium and uranium) Oxide fuel.

210 Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Council Directive 2014/87/Euratom of 8 July 2014

211 International Atomic Energy Agency.

212 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

213 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

214 Communication from the Commission to the Council and the European Parliament on the comprehensive risk and safety assessments ("stress tests") of nuclear power plants in the European Union and related activities, COM(2012) 571 final, Brussels, 4.10.2012

215 See e.g. France's 7th National Report on compliance with the Joint Convention on the safety of the management of spent fuel and on the safety of the management of radioactive waste, October 2020 on the La Hague facility

216 As required by Directive 2000/60/EC (Water Framework Directive) for activities subject to Union law, or as required by equivalent national provisions or international standards addressing environmental degradation risks related to preserving water quality and avoiding water stress for activities in third countries.
Where an Environmental Impact Assessment is carried out in accordance with Directive 2011/92/EU (Consolidated EIA Directive) and includes an assessment of the impact on water in accordance with Directive 2000/60/EC, no additional assessment of impact on water is required, provided the risks identified have been addressed.

	Radioactive discharges shall be in accordance with regulatory limits established to ensure fulfilment of the requirements of the Euratom Basic Safety Standards Directive[217] and the Euratom Drinking Water Directive[218], or, outside the EU, to be in accordance with ICRP recommendations[219]. The ALARA (as low as reasonably achievable) principle should be applied during the control of radioactive discharges consistently. Control of chemical and thermal pollution shall be implemented in line with the EU legislative framework and relevant BAT (Best Available Techniques). For activities performed outside the EU, the relevant IFC standards[220] are applicable.
(4) Transition to a circular economy	A plan for the management of conventional and radioactive waste is in place and ensures maximal reuse or recycling at end of life in accordance with the waste hierarchy. During operation and decommissioning, the amount of radioactive waste is minimized. Plans for the long-term safe disposal of high-level radioactive waste resulting from the activity during its whole life cycle are in place and their adequacy is demonstrated by science-based evidence or empirical data.
(5) Pollution prevention and control	Non-radioactive pollutant emissions shall be in accordance with limits established by national authorities in order to meet the requirements of the National Emission Reduction Commitments Directive[221]. For activities performed outside the EU, the relevant IFC standards and guidelines[222] are applicable. Radioactive discharges shall be in accordance with regulatory limits established to ensure fulfilment of the requirements of the Euratom Basic Safety Standards Directive[223] and the Euratom Drinking Water Directive[224], or, outside the EU, to be in accordance with ICRP recommendations[225]. The ALARA (as low as reasonably achievable) principle should be applied during the control of radioactive discharges consistently. For activities performed outside the EU, the relevant IFC standards[226] are applicable.
(6) Protection and restoration of	An Environmental Impact Assessment (EIA) or screening[227] has been completed, for activities within the Union, in accordance with Directive 2011/92/EU. For

217 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

218 COUNCIL DIRECTIVE 2013/51/Euratom of 22 October 2013 laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption.

219 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

220 a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007 and b) IFC Environmental, Health and Safety Guidelines for Thermal Power Plants, IFC, December 2008

221 Directive (EU) 2016/2284 of the European Parliament and of the Council of 14 December 2016 on the reduction of national emissions of certain atmospheric pollutants.

222 a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007; b) IFC – Environmental, Health, and Safety (EHS) General Guidelines, IFC, April 2007

223 Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

224 COUNCIL DIRECTIVE 2013/51/Euratom of 22 October 2013 laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption.

225 The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

226 a) IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks, IFC, April 2007 and b) IFC Environmental, Health and Safety Guidelines for Thermal Power Plants, IFC, December 2008

227 The procedure through which the competent authority determines whether projects listed in Annex II to Directive 2011/92/EU is to be made subject to an environmental impact assessment (as referred to in Article 4(2) of that Directive).

biodiversity and ecosystems	activities in third countries, an EIA has been completed in accordance with equivalent national provisions or international standards[228]. Where an EIA has been carried out, the required mitigation and compensation measures for protecting the environment are implemented. For sites/operations located in or near biodiversity-sensitive areas (including the Natura 2000 network of protected areas, G World Heritage sites and Key Biodiversity Areas, as well as other protected areas), an appropriate assessment[229], where applicable, has been conducted and based on its conclusions the necessary mitigation measures[230] are implemented.

228 For example, IFC Performance Standard 1: Assessment and Management of Environmental and Social Risks.

229 In accordance with Directives 2009/147/EC and 92/43/EEC, or, for activities located in third countries, in accordance with equivalent national provisions or international standards, for example IFC Performance Standard 6: Biodiversity Conservation and Sustainable Management of Living Natural Resources.

230 Those measures have been identified to ensure that the project, plan or activity will not have any significant effects on the conservation objectives of the protected area.

4. Interim storage and final disposal of high-level radioactive waste (including high-level vitrified waste)

Description of the activity

The activity includes the interim storage of spent fuel and high level (vitrified) waste and the final disposal of this waste in a deep geological disposal facility. The activity is classified under NACE codes E.38.12[231] and E.38.22[232] in accordance with the statistical classification of economic activities established by Regulation (EC) No 1893/2006.

This activity can be considered as an enabling activity[233].

Technical screening criteria

Substantial contribution to climate change mitigation
The activity aims at containing and isolating the radioactive waste from the accessible biosphere. Radiation protection and nuclear safety requirements shall be fulfilled during all operational stages of waste management (collecting, handling, treatment, conditioning, interim storage, transport, and disposal). These are laid down in the national regulatory framework, and for EU Member States, shall comply with the provisions of the Euratom Directive for the safe and responsible management of radioactive waste and spent fuel[234], as well as with those of the Euratom Basic Safety Standards (BSS)[235]. Outside the EU, the nuclear safety and radiation protection requirements are laid down in the national regulatory framework, and shall comply with the provisions of International Conventions[236], IAEA relevant standards and the latest recommendations of the International Commission on Radiological Protection (ICRP)[237]. Notwithstanding, Council Directive on the safe and responsible management of spent fuel and radioactive waste establishes that radioactive waste shall be disposed of in the Member State in which it was generated unless there is an agreement between the Member State concerned and the destination Member State or Third Country, the country of destination has radioactive waste management and disposal programmes and a suitable disposal facility in operation compliant with the requirements of the Directive.

Do no significant harm ('DNSH')	
(2) Climate change adaptation	The activity complies with the criteria set out in Appendix E to this Annex. The design and construction of the facilities applied for the interim storage and disposal of high level radioactive waste shall ensure the containment of the waste and its isolation from the accessible biosphere also during the occurrence of extreme natural hazards, such as earthquakes, tornados, flooding, etc. Deep geological disposal facilities shall be located in stable geological formations, and the calculated impact of different climate evolutions over very long periods, including severe climate alterations such as glaciation, shall be considered in the safety case.

[231] NACE E.38.12 Collection of hazardous waste. E.38 Waste collection, treatment and disposal activities; materials recovery.

[232] NACE E.38.22 Treatment and disposal of hazardous waste. E.38 Waste collection, treatment and disposal activities; materials recovery.

[233] Article 10(1), point (i) of Regulation (EU) 2020/852 refers to activities enabling any of the activities listed in points (a) to (h) of paragraph 10(1) in accordance with Article 16. Article 10(1) point (a) refers to "generating, transmitting, storing, distributing or using renewable energy in line with Directive (EU) 2018/2001" and in this sense concerns only activities related to renewable energy. However, if nuclear energy is to be included into the Taxonomy as an activity significantly contributing to the climate change mitigation objective, then all related raw material mining, processing, fuel manufacturing and waste storage and disposal activities should be included as activities enabling nuclear energy based electricity production.

[234] Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste. OJ L 199, 2.8.2011, p. 48–5

[235] Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom. OJ L 13, 17.1.2014, p. 1–73

[236] INFCIRC/546. 24 December 1997. International Atomic Energy Agency Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

[237] The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007

(3) Sustainable use and protection of water and marine resources	The Environmental Impact Assessment addresses the potential impacts to water and marine resources[238] associated with the construction and operation of radioactive waste management facilities. Environmental degradation risks related to preserving water quality and avoiding "water stress" are identified and adequately addressed, in accordance with a water use and protection management plan, developed in consultation with relevant stakeholders. During operation of storage or disposal facilities, liquid radioactive discharges shall be insignificant, or not present. The design and construction of the facilities shall ensure that the radioactive waste remains contained and shielded and cannot contaminate water streams or marine resources. During the operation phase, adequate surveillance shall ensure that, if necessary, timely remedial actions can be initiated before water streams or marine resources are impacted. Any release shall be below regulatory limits established to ensure fulfilment of the requirements of the Euratom Basic Safety Standards Directive and the Euratom Drinking Water Directive[239], or, outside the EU, to be in accordance with ICRP recommendations. Concerning spent fuel/HLW disposal, in the deep geological repository post-closure phase the compliance with the basic safety objective (namely, to maintain the dose contribution to humans and the environment below the regulatory limit) shall be retained thanks to the multi-barrier design of the containment and isolation barriers and by the overall engineered safety properties of the repository. In particular, the mobility and migration of the radionuclides from the emplacement location shall be hindered for the required timespans. The safety case demonstration shall address the long term evolution of the reference case and shall include also consideration of extreme scenarios (e.g. loss of functionality by the engineered barriers, external events).
(4) Transition to a circular economy	Reuse and recycle of technological waste materials is limited to materials from the decommissioning and dismantling of the auxiliary facilities (including storage facilities, encapsulation plant, etc.) after the end of the operational phase. A plan for the management of waste shall be in place which ensures maximal reuse or recycling at end of life in accordance with the waste hierarchy. The largest need for materials that are neither recyclable nor reusable results from the encapsulation and backfilling, but their amount is very small. During operation and decommissioning, the amount of radioactive waste shall be minimized. Recycling of unused fuel fractions is enabled by reprocessing. Innovative closed fuel cycle concepts, currently under development, aim at optimizing recycling of fuel fractions and minimizing long term radiotoxicity of resulting HLW.
(5) Pollution prevention and control	Non-radioactive pollutant emissions shall be in accordance with limits established by national authorities in order to meet the requirements of the National Emission Reduction Commitments Directive[240]. The Environmental Impact Assessment addresses the impacts related to pollution during construction and operation of radioactive waste storage or disposal facilities. No pollution shall

[238] As required by Directive 2000/60/EC (Water Framework Directive) for activities subject to Union law, or as required by equivalent national provisions or international standards addressing environmental degradation risks related to preserving water quality and avoiding water stress for activities in third countries.
Where an Environmental Impact Assessment is carried out in accordance with Directive 2011/92/EU (Consolidated EIA Directive) and includes an assessment of the impact on water in accordance with Directive 2000/60/EC, no additional assessment of impact on water is required, provided the risks identified have been addressed.

[239] Council Directive 2013/51/Euratom of 22 October 2013 laying down requirements for the protection of the health of the general public with regard to radioactive substances in water intended for human consumption. OJ L 296, 7.11.2013, p. 12–2

[240] Directive (EU) 2016/2284 of the European Parliament and of the Council of 14 December 2016 on the reduction of national emissions of certain atmospheric pollutants, amending Directive 2003/35/EC and repealing Directive 2001/81/EC (Text with EEA relevance). OJ L 344, 17.12.2016, p. 1–3

	occur during the post-closure phase of the disposal facility. For activities performed outside the EU, the relevant national standards and guidelines are applicable. During operation of storage and disposal facilities, small gaseous radioactive discharges may occur during waste handling in controlled environment. After loading the waste in the storage or disposal container, no release shall occur; the design and construction of the facilities and the containers shall ensure that the radioactive waste remains contained and shielded and cannot contaminate the atmosphere. Adequate surveillance shall ensure that, if necessary, timely remedial actions can be initiated. Any gaseous radioactive discharges shall be below regulatory limits established to ensure fulfilment of the requirements of the Euratom Basic Safety Standards Directive, or, outside the EU, to be in accordance with ICRP recommendations. Concerning spent fuel/HLW disposal, in the deep geological repository post-closure phase the compliance with the basic safety objective (namely, to maintain the dose contribution to humans and the environment below the regulatory limit) shall be achieved thanks to the multi-barrier design of the containment and isolation barriers and by the overall engineered safety properties of the repository. In particular, the mobility and migration of the radionuclides from the emplacement location shall be hindered for the required timespans. The safety case demonstration shall address the long term evolution of the reference case, and shall also include consideration of extreme scenarios (e.g. loss of functionality by the engineered barriers, external events).
(6) Protection and restoration of biodiversity and ecosystems	Protection of biodiversity and ecosystems during operation of storage and disposal facilities is addressed in the Environmental Impact Assessment (EIA), for activities within the Union, in accordance with Directive 2011/92/EU. For activities in third countries, an EIA shall be completed in accordance with equivalent national provisions or international standards. The EIA shall incorporate the required measures for mitigation and for protecting the environment. Storage facilities are decommissioned and dismantled at the end of their operational phase, and the site where they were located is remediated and released from regulatory control. Disposal facilities are closed at the end of their operational phase, and the auxiliary facilities decommissioned and dismantled. Once closed, the site is environmentally remediated and the residual resulting waste shall be contained and isolated from the environment according to the relevant procedures. Concerning spent fuel/HLW disposal, in the deep geological repository post-closure phase the compliance with the basic safety objective (namely, to maintain the dose contribution to humans and the environment below the regulatory limit) shall be achieved thanks to the multi-barrier design of the containment This ensures that no significant harm is caused to the biodiversity and ecosystems. The safety case demonstration shall address the long term evolution of the reference case, and shall include also include consideration of extreme scenarios (e.g. loss of functionality by the engineered barriers, external events).

5. Appendix E (of Annex I of the Commission Delegated Regulation): Generic criteria for DNSH to climate change adaptation

Copy of Appendix E of Annex I of Commission Delegated Regulation[241]

I. Criteria

New activity

The physical climate risks that are material to the activity have been identified from those listed in the table in Section II of this Appendix by performing a robust climate risk and vulnerability assessment. The assessment is proportionate to the scale of the activity and its expected lifespan, such that:

1. for investments into activities with an expected lifespan of less than 10 years, the assessment is performed, at least by using downscaling of climate projections;
2. for all other activities, the assessment is performed using high resolution, state-of-the-art climate projections across a range of future scenarios consistent with the expected lifetime of the activity, including, at least, 10 to 30 years climate projections scenarios for major investments.

The economic operator has developed a plan to implement adaptation solutions to reduce material physical climate risks to the activity. Those adaptation solutions do not adversely affect the adaptation efforts or the level of resilience to physical climate risks of other people, of nature, of assets and of other economic activities and are consistent with local, sectoral, regional or national adaptation efforts.

Activity upgrading or altering existing assets or processes

The physical climate risks that are material to the activity have been identified from those listed in the table in Section II of this Appendix by performing a robust climate risk and vulnerability assessment. The assessment is proportionate to the scale of the activity and its expected lifespan, such that:

1. for investments into activities with an expected lifespan of less than 10 years, the assessment is performed, at least by using downscaling of climate projections;
2. for all other activities, the assessment is performed using high resolution, state-of-the-art climate projections across a range of future scenarios consistent with the expected lifetime of the activity, including, at least, 10 to 30 years climate projections scenarios for major investments.

The economic operator has developed a plan to implement adaptation solutions to reduce material physical climate risks to the activity. The adaptation solutions identified need to be implemented within five years from the start of the activity. These adaptation solutions do not adversely affect the adaptation efforts or the level of resilience to physical climate risks of other people, of nature, of assets and of other economic activities and are consistent with local, sectoral, regional or national adaptation efforts.

II. Classification of climate related risks

(from Annex I, same as Appendix A in Annex II)

	Temperature-related	**Wind-related**	**Water-related**	**Solid mass-related**
Chronic	Changing temperature (air, freshwater, marine water)	Changing wind patterns	Changing precipitation patterns and types (rain, hail, snow/ice)	Coastal erosion
	Heat stress	-	Precipitation or	Soil degradation

[241] Annex to the Commission Delegated Regulation (EU) xxx/xxx supplementing Regulation (EU) 2020/852 of the European Parliament and of the Council by establishing the technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change mitigation or climate change adaptation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives; ARES (2020)6979284 - 20/11/2020, European Commission, 20 November 2020
ANNEX I – Technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change mitigation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives
ANNEX II – Technical screening criteria for determining the conditions under which an economic activity qualifies as contributing substantially to climate change adaptation and for determining whether that economic activity causes no significant harm to any of the other environmental objectives

			hydrological variability	
	Temperature variability	–	Ocean acidification	Soil erosion
	Permafrost thawing	–	Saline intrusion	Solifluction[1]
	–	–	Sea level rise	–
	–	–	Water stress	–
Acute	Heat wave	Cyclone, hurricane, typhoon	Drought	Avalanche
	Cold wave/frost	Storm (including blizzards, dust- and sandstorms)	Heavy precipitation (rain, hail, snow/ice)	Landslide
	Wildfire	Tornado	Flood (coastal, fluvial, pluvial, ground water)	Subsidence
	–	–	Glacial lake outburst	–

(1) Slow creeping of soil down a slope that usually occurs in perennial frost regions due to the freeze-thaw activity

Annex 5. Ionising radiation: definitions, units, biological effects and radiation protection

Source: [A.5-1].

1. Biological effects of ionising radiation

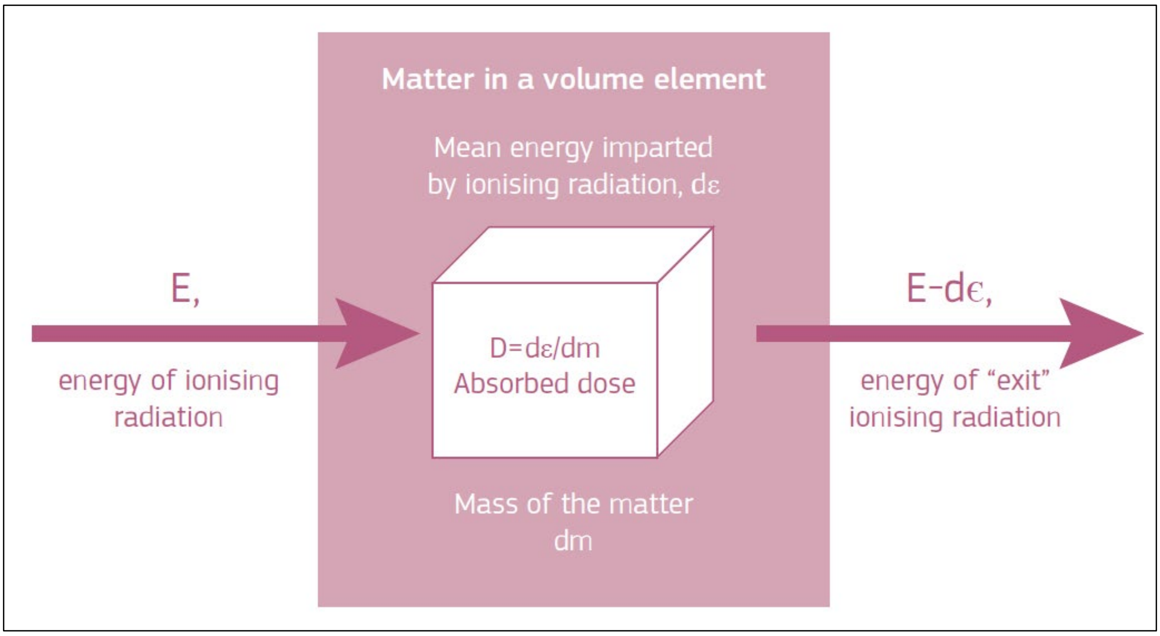
In daily life, we are exposed to various sources of radiation, for example natural radiation sources, medical applications, industrial practices, effluents from nuclear installations (which are generally controlled and negligible), fallouts from nuclear weapons testing and the impact of nuclear accidents (historical events). Exposure to increased levels of ionising radiation can be harmful to human health. Indeed, radiation can ionise or excite atoms while passing through tissue.

There are various quantities to specify the dose received and the biological effectiveness of that dose:

Absorbed dose (D): the energy absorbed per unit mass $D = d\varepsilon/dm$ where $d\varepsilon$ is the mean energy imparted by ionising radiation to the matter in a volume element and dm is the mass of the matter in this volume element. It is expressed in gray (Gy=J/kg).

The **absorbed dose rate** is the rate at which an absorbed dose is received (Gy/s).

Figure A.5-1. Simplified schema for defining the absorbed dose



Source: European Atlas of Natural Radiation, Ref. [A.5-1]

The biological effect of radiation depends not only on the energy deposited by radiation in an organism, but in addition on the type of radiation and the way in which the energy is deposited along the path of the radiation. So therefore the linear energy transfer (LET) is defined. It describes the mean energy deposited per unit path length in the absorbing material. The unit of the LET is keV/µm. So for the same absorbed dose, the biological effect of alpha particles or neutrons (high LET) is much greater than of beta or gamma rays (low LET). To characterise this difference in biological effects of various types of radiation, the radiation weighting factor w_R was established (Table A.5-1) and has been published in ICRP Recommendation 103 [A.5-2].

Table A.5-1. Radiation weighting factors

Radiation type	Radiation weighting factor, w_R
Photons	1
Electrons, and muons	1
Protons and changed pions	2
α particles, fission fragments, heavy ions	20
Neutrons	A continuous function depending on neutron energy (see Equation A.5-1)

Source: ICRP 2007 [A.5-2], adopted by the Basic Safety Standards Directive [A.5-3]

To calculate radiation weighting factors for neutrons, a continuous function in neutron energy, E_n (MeV), is used (Equation A.5-1).

$$w_R = \begin{cases} 2.5 + 18.2e^{-[ln\,(E_n)]^2/6}\,, & E_n < 1\,MeV \\ 5.0 + 17.0e^{-[ln\,(2E_n)]^2/6}\,, & 1\,MeV \leq E_n \leq 50\,MeV \\ 2.5 + 3.25e^{-[ln\,(0.04E_n)]^2/6}\,, & E_n > 50\,MeV \end{cases} \tag{A.5-1}$$

The **equivalent dose (H_T)** represents the radiation dose to tissue and thus makes the link between absorbed dose and its biological effect. H_T is calculated as absorbed dose multiplied by the weighting factor (w_R) of the radiation. If there are several types of radiation (R) present, the equivalent dose in the tissue (T) is the weighted sum over all contributions. Equivalent dose is also expressed in joule per kilogram, because of the dimensionless weighting factor. For differentiation the unit of the equivalent dose is named sievert (Sv) after the Swedish doctor and physicist Rolf M. Sievert (1896 – 1966). The relation with the former unit, roentgen equivalents man (rem), is 1 Sv = 100 rem.

$$H_T = \sum_R w_R\, D_{T,R} \tag{A.5-2}$$

The equivalent dose rate is the rate at which an equivalent dose is received, expressed for example in Sv/s or Sv/h.

The equivalent dose is always related to a defined tissue or organ. Different tissues and organs show different sensitivities to radiation, depending on their cell cleavage frequency and their cell renewal frequency. To take these effects into account, the equivalent doses in different tissues must be weighted (Table A.5-2; ICRP 2007 [A.5-2]).

Table A.5-2. Radiation weighting factors

Tissue or organ	Tissue-weighting factor
Bone marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Breast	0.12
Gonads	0.08
Bladder	0.04
Liver	0.04
Oesophagus	0.04
Thyroid	0.04
Skin	0.01
Bone surface	0.01
Salivary gland	0.01
Brain	0.01
Sum of remainder tissues or organs	0.12

Source: ICRP 2007 [A.5-2], adopted by the Basic Safety Standards Directive [A.5-3].

The equivalent dose (H_T) in tissue or organ T multiplied by this tissue weighting factor (w_T) reported in Table A.5-2, is called the **effective dose (E)**.

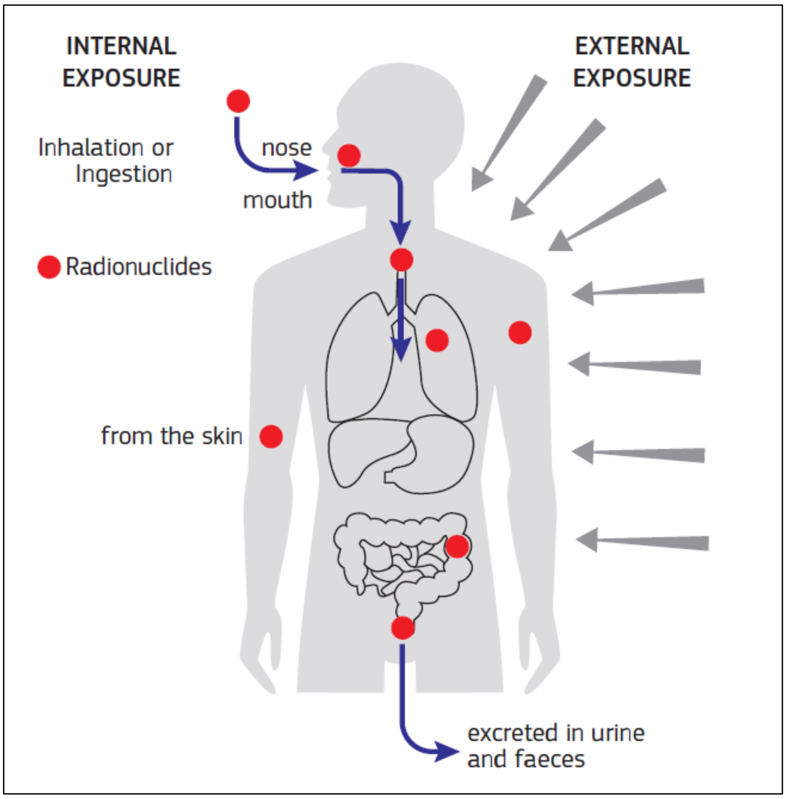
$$E = \sum_T w_T H_T = \sum_T w_T \sum_R w_R D_{T,R} \qquad \text{(A.5-3)}$$

The sum of the relative weighting factors is one; this means that the sum of the weighting risks for the organs is numerically equal to the risk for the whole body.

2. Calculating doses from intakes of radionuclides

Irradiation by ionising radiation outside the body causes only a dose during the period of irradiation. But by an intake through ingestion or inhalation some radionuclides can remain inside the body and irradiate the tissues for years. In these cases, the total radiation dose depends on the half-life of the radionuclide, its distribution in the body, and the rate at which it is excreted from the body. On the basis of mathematical models, doses can be calculated with consideration of the radionuclides intake each year. The resulting total effective dose delivered over a lifetime is called the committed effective dose.

Figure A.5-2. Schematic representation of the modes of exposure to ionizing radiation



Source: European Atlas of Natural Radiation [A.5-1]

ICRP develops effective dose coefficients to simplify the calculation of equivalent dose and effective dose for inhaled or ingested radionuclides: values for committed doses following the intake of 1 Bq of a radionuclide via ingestion and inhalation.

These coefficients have been calculated for members of the public at six standard ages and for intake by adult workers. The unit of the effective dose coefficient is Sv/Bq. The received dose via ingestion or inhalation of a radionuclide can be calculated as a product of the incorporated activity and the effective dose coefficient. Choosing the right dose coefficient depends on:

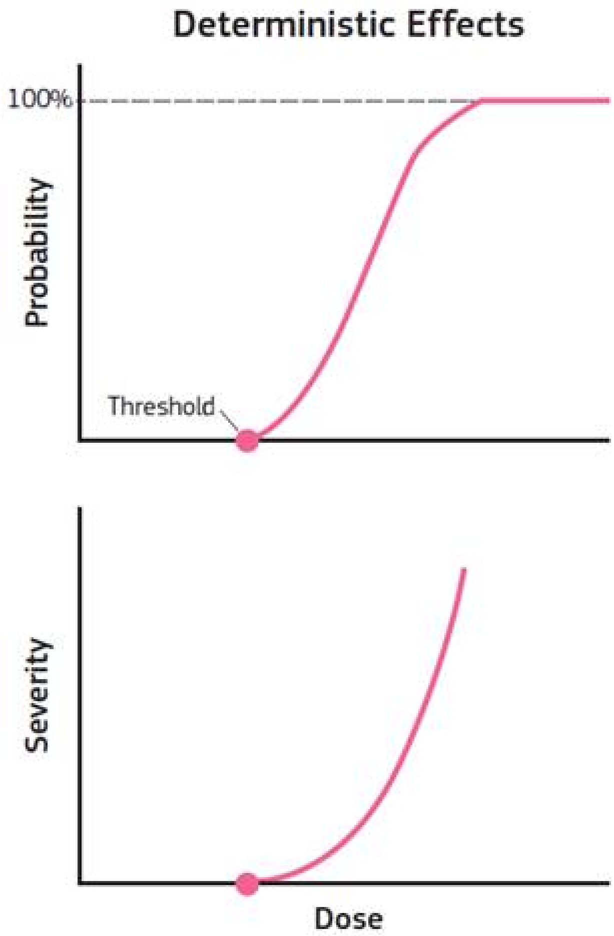
- The radionuclide
- Whether it is inhaled or ingested
- The particle size (for inhalation)
- The chemical form
- Population group
- The time since intake (if using bioassay data)
- Activity Median Aerodynamic Diameter (AMAD).

3. Deterministic and stochastic effects

Radiation can affect people's health in two different ways, called deterministic effects and stochastic effects.

Deterministic effects are characterised by a threshold (Figure A.5-3); below it, no damage is recognised; and above it, the damage increases with dose. Deterministic effects are the acute radiation syndrome, which occurs immediately after an irradiation with high doses and damages, which occur at a later time, but induce no cancer (opacity of lens, vitiation of fertility). Immediate symptoms after a whole body irradiation can be recognised above a dose between 0.5 and 1 Gy. For doses in the range 2 – 6 Gy mortality is between 5 and 95 % without treatment and between 5 and 50 % with treatment. These are estimates and recovery potential depends on treatment. If the whole body dose goes up to 10 Gy, the mortality would reach 100 % (Eisenbud & Gesell [A.5-4]).

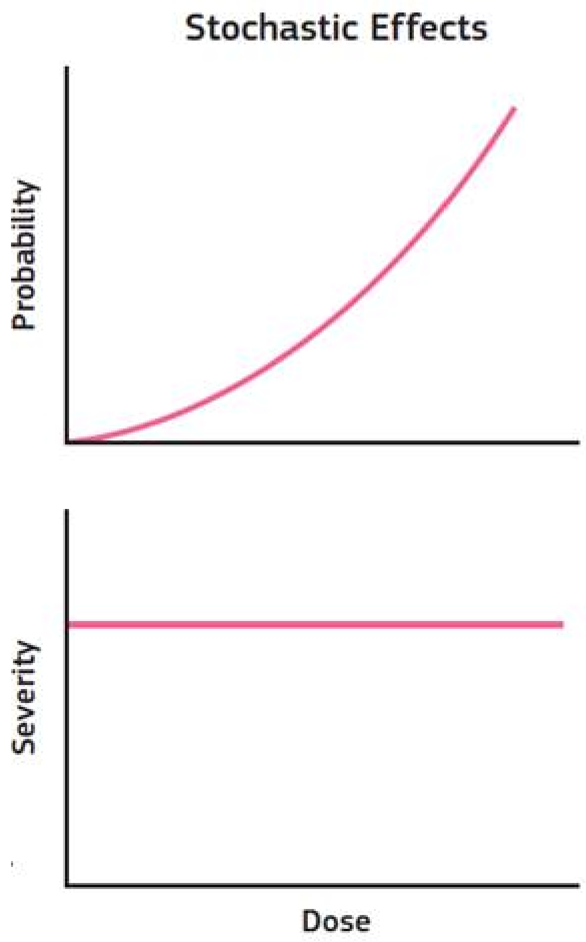
Figure A.5-3. Stylised probability-dose and severity-dose relationships for deterministic effects



Source: modified from Zanzonico et al. [A.5-5]

Stochastic effects of ionising radiation are chance events, with the probability of the effect increasing with dose, but the severity of the effect is independent of the dose received. Primarily cancer risk, but also hereditary disorders are stochastic effects [A.5-2]. Stochastic effects are assumed to have no threshold (Figure A.5-4). However it is not yet known what the curve looks like for small doses (i.e. < 0.1 Sv), and several hypotheses have been considered, including homeostatic (positive effect for very small doses) and the existence of a threshold (i.e. limit below which there is no effect). However, for regulatory purposes, simplicity and conservatism, the most prevalent assumption is linear-no-threshold. An approach is called 'conservative' if, according to the state of knowledge, it likely represents an unfavourable situation, i.e. it is pessimistic, or in other words, most likely the expected true effect is less severe. Its purpose is to be on the safe side.

Figure A.5-4. Stylised probability-dose and severity-dose relationships for stochastic effects



Source: modified from Zanzonico et al. [A.5-5]

4. General principles of radiation protection

The system of radiation protection is based on the following principles of justification, optimisation and dose limitation:

(a) The Principle of Justification: Any decision that alters the radiation exposure situation should do more good than harm. Decisions introducing or altering a radiation source, an exposure pathway or actual exposures shall be justified in the sense that such decisions shall be taken with the intent to ensure that the individual or societal benefit resulting from them offsets the detriment that they may cause;

(b) The Principle of Optimisation: In all exposure situations, radiation protection shall be optimised with the aim of keeping the magnitude and likelihood of exposure and the number of individuals exposed as low as reasonably achievable, taking into account economic and societal factors, whereby optimisation of the protection of individuals undergoing medical exposure shall be commensurate with the medical purpose of the exposure as described in Article 56 of the Basic Safety Standards Directive [A.5-3]. This principle shall be applied in terms of Effective Dose as well as organ doses, as a precautionary measure to allow for uncertainties as to health detriment below the threshold, for deterministic effects;

(c) The Principle of Dose Limitation: In planned exposure situations, the sum of doses to an individual from all regulated radiation sources may not exceed the dose limits laid down for occupational exposure or public exposure. Dose limits shall not apply to medical exposures [A.5-3].

Radiation protection (also called radiological protection) is defined as the protection of people from the harmful effects of exposure to ionising radiation, and the means for achieving this [A.5-6].

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Annex 6. Characteristics of radioactive waste

1. Radioactivity and radioactive substances

Radioactivity or radioactive decay is the phenomenon in which a 'parent' nuclide (also called 'radionuclide') is energetically unstable (its atomic nucleus possesses too much energy) and disintegrates/decays into a 'daughter' nuclide by emitting energy in the form of ionising radiation. If the daughter nuclide is also unstable, the radioactive decay process continues further in a decay chain until a stable nuclide (with no excess energy in the nucleus) is reached [A6-1]. The scientific unit for radioactive decay is named Becquerel (Bq), corresponding to one disintegration per second. Highly radioactive substances, e.g. high level waste (irradiated nuclear fuel or vitrified waste from reprocessing) can have a radioactivity level in excess of 1 billion Bq/g (depending on composition and age).

Ionising radiation is radiation that carries enough energy to remove electrons from atoms or molecules, thereby ionising them [A6-2]. There are different kinds of ionising radiation: alpha-, beta-, and gamma-radiation.

Alpha-radiation is the emission of an alpha particle from the nucleus of the radionuclide. An alpha particle consists of two neutrons and two protons (positively charged), which corresponds to the nucleus of a helium atom. As a heavy charged particle, it will not travel far and is easily stopped *e.g.* already by a sheet of paper or by the outer skin. Alpha radiation is most dangerous when emitted inside a human body after ingestion or inhalation: due to its high ionising power, it will damage the surrounding cells. Alpha radiation is predominantly emitted by heavy radionuclides.

Beta-radiation is the emission of a high-energy electron (negatively charged) or positron (positively charged) from the nucleus of the radionuclide. A beta particle is much smaller and lighter than an alpha particle. It can travel and penetrate into matter somewhat further than an alpha particle before being stopped. However, beta radiation has less ionising power compared to alpha radiation and thus causes less damage.

Gamma-radiation is electro-magnetic radiation and not a particle. In the electro-magnetic spectrum, gamma-radiation is more energetic than X-rays. To protect for gamma-radiation requires heavy shielding by *e.g.* lead, steel or heavy concrete. Daughter nuclides, especially from alpha- or beta-decay, are often still in an excited energy state. They can decay and emit their surplus energy by gamma-radiation.

The 'radiotoxicity' of a radionuclide indicates how harmful it is to the human body because of the radiological effects, after ingestion or inhalation [A6-3] (see Annex 5 for a comprehensive description of the effects of radiations on the human body). The following definition was proposed in [A6-6]: "The toxicity of a radionuclide is the ability of the nuclide to produce injury, by virtue of its emitted radiations, when incorporated in a body."[242]

From the standpoint of long term radioactive waste management, ingestion radiotoxicity and, especially, ingestion radiotoxicity as a function of time is often used as a source term reference to define the barriers and the measures to be implemented to ensure that no significant harm is caused by the waste [A6-8]

Radiation protection is defined as the protection of people from the effects of exposure to ionising radiation and the means for achieving this [A6-3]. Protection from external exposure is provided by distancing and by surrounding radioactive substances with adequate shielding. Protection from ingestion and/or inhalation is provided by effective insulation of radioactive substances (sealing, encapsulation, immobilization), and by avoiding or strictly limiting the presence of radiotoxic substances in food, water, and in the human environment. Radiation protection has a long tradition in Europe. The latest update of the Basic Safety Standards Directive (Directive 2013/59/Euratom) lays down basic safety standards for protection against the dangers arising from exposure to ionising radiation, fully in line and periodically updated with the latest scientific findings and taking account of technological progress and operational experience [A6-3].

2. Natural and artificial radionuclides

Due to the decay process, the radioactivity of a substance decreases with time. The half-life of a radionuclide tells how much time it takes for the radioactivity of a given mass of the radioactive substance to decay to half of its initial value. After one half-life, half of the mass of the initial 'parent' radionuclide has decayed into

[242] A definition of chemical toxicity was provided in [A6-8]: "*Toxicity is the ability of a chemical molecule or compound to produce injury once it reaches a susceptible site in or on the body. Toxicity hazard is the probability that injury may be caused by the manner in which the substance is used.*"

its daughter nuclide. Each radionuclide has its characteristic half-life: half-lives vary from fractions of a second to millions of years. Some radionuclides are of natural origin (they were formed during the early stage of the Big Bang and during final stages of the life of stars) and contribute to the natural radioactivity levels [A6-1] which characterize our planet. Thus radioactivity exists in nature. The origin of radionuclides can be sub-divided in three categories: primordial, cosmogenic, and anthropogenic.

Primordial radionuclides have existed since before the solar system was formed [A6-2]. They are the remains of nuclear reactions occurring in stars, the origin of all elements. Such high-energy reactions produce stable nuclides, but also energetically unstable (radioactive) radionuclides [A6-3]. Since the formation of our planet, most radionuclides have decayed. Only the primordial radionuclides with very long half-life (> 100 million years) are still to be found on earth today. The most important ones are potassium-40 (^{40}K, half-life: 1.251 billion years), uranium-238 (^{238}U, half-life: 4.5 billion years) and thorium-232 (^{232}Th, half-life: 14 billion years). Potassium is an important element for the human body: 0.0117% of the total potassium in our body is the radioactive ^{40}K. The average ^{40}K content in a human body weighing 70 kg is 4 400 Bq [A6-10]. The decay chains of ^{238}U and ^{232}Th include radon (Rn) isotopes. For the population in Europe, ^{222}Rn (half-life: 3.82 days) and its progeny represent on average ~60% of the total dose from natural sources of radiation (total average annual effective dose from natural sources being 3.20 mSv) [A6-2].

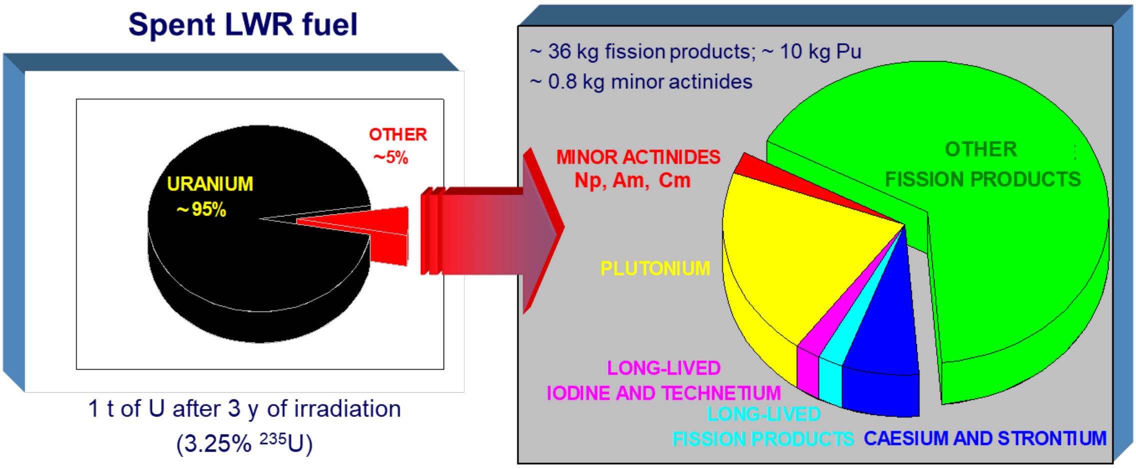
Cosmogenic radionuclides are continuously produced in the atmosphere by the interaction of cosmic rays with stable elements [1-4]. The most important cosmogenic radionuclides are tritium (^{3}H, half-life: 12.32 years), carbon-14 (^{14}C, half-life: 5730 years) and beryllium-7 (^{7}Be, half-life: 53.22 days). ^{14}C is the most important one for human radiation exposure. The dose received from cosmogenic radionuclides doubles every 2 000 m of increasing altitude [A6-5]; as such, significantly higher doses are received from prolonged stays in mountainous areas and in traveling aircrafts.

Anthropogenic radionuclides are produced by human technologies, in particular since the discovery of nuclear fission and the further development of nuclear science and applications. In Europe, traces of anthropogenic radionuclides originating from atmospheric nuclear weapon tests during the 50s and 60s and from the Chernobyl nuclear accident in 1986 can still be found in the environment.

Another source of anthropogenic radionuclides stems from peaceful nuclear applications. In a nuclear reactor, many of the fission and activation products resulting from the splitting (fission) of nuclear fuel atoms and from neutron irradiation are radioactive. Moreover, specific radionuclides are being produced in dedicated power plants and/or accelerators also for medical and industrial applications. The radioactive waste resulting from these anthropogenic nuclear activities has to be managed in a safe and responsible way, in order to avoid radionuclides dispersion and ensure that no harm is caused for the humans and the environment (Council Directive 2011/70/Euratom).

From the radioactive waste management standpoint, within the nuclear energy lifecycle the biggest and most relevant source of anthropogenic radionuclides in terms of specific radioactivity level and variety of radionuclides is fuel irradiation in the nuclear reactor. After irradiation, typically 4-5% of the fuel nuclei has experienced fission (Figure A6-1). Microstructural radiation damage processes and the formation of fission products modify chemical composition, morphology and microstructure, and overall properties of the fuel. Spent fuel is a heterogeneous and complex material [A6-6].

Figure A.6-1. Schematic illustration of the change in composition of spent nuclear fuel due to irradiation. Only a small fraction of the original fuel (here considered as 1 t of uranium) has been transformed, mostly by fission of the fissile U-235 nuclei. However, due to fission and other nuclear reactions, many elements have formed during irradiation and are present in the spent fuel.



Source: Handbook of Nuclear Energy, Ed. D.G. Cacuci; Chapter 26 "Transuranium Elements in the Nuclear Fuel Cycle", pp. 2935-2998, Springer 2010.

3. Relevant radionuclides in spent fuel and high level waste from light water reactors.

Tables A6-1 and A6-2 list the amount of relevant actinides and fission products in spent fuel irradiated in light water reactor and in the representative French vitrified waste formulation, respectively (www.nucleonica.net; see also [A6-11]). Table A6-3 lists some relevant properties of radionuclides of interest for high level waste management.

Table A.6-1. Relevant elemental/radionuclide inventory of transuranics elements and relevant fission products) in LWR (PWR) spent fuel with different burnup(1) ten years after discharge from the reactor; values in kg/tHM obtained from the Nucleonica database (www.nucleonica.net).

Fuel, burnup	**UO_2 35 GWd/tHM**	**UO_2 55 GWd/tHM**	**MOX 50 GWd/tHM**
initial fissile content	3.25% ^{235}U	4.7% ^{235}U	4.2% Pu
Pu	9.49	11.04	40.95
Np	0.48	0.83	0.20
Am	0.70	0.94	6.06
Cm	0.02	0.07	1.08
^{79}Se	0.002	0.004	0.003
^{90}Sr	0.447	0.693	0.281
^{93}Zr	0.756	1.187	0.737
^{99}Tc	0.857	1.284	1.144
^{107}Pd	0.240	0.364	0.807

^{126}Sn	0.025	0.039	0.070
^{129}I	0.171	0.265	0.320
^{135}Cs	0.388	0.739	1.125
^{137}Cs	1.038	1.595	1.437

(1) The burnup is the energy produced by the fuel during irradiation in the reactor. Here it is expressed in GWday per ton of heavy metal: 10 GWd/tHM correspond approximately to the fission of 1% of the nuclei in the fuel.

Source: JRC from www.nucleonica.net data (2020).

Table A6-2. Composition of French NC R7/T7 reference vitrified HLW

Component	Fraction wt%
SiO_2	45.1
B_2O_3	13.9
Al_2O_3	4.9
Na_2O	10
Fe_2O_3	2.9
NiO	0.4
Cr_2O_3	0.5
fission products	12.4
actinides (oxide)	0.37
metal particles	1.6

Source: JRC with data from Gras et al. 2007 [A6-12] and IAEA [A6-13]

Table A.6-3. Selected properties of relevant radionuclides present in HLW

Nuclide	$T_{1/2}$ y	Main decay mode	Spontaneous fission rate $s^{-1}g^{-1}$	Specific activity Bqg^{-1}	Specific power Wg^{-1}
^{234}U	$2.46 \cdot 10^{6}$	α	$3.91 \cdot 10^{-3}$	$2.3 \cdot 10^{8}$	$1.79 \cdot 10^{-4}$
^{235}U	$7.04 \cdot 10^{8}$	α	$5.76 \cdot 10^{6}$	$8.00 \cdot 10^{4}$	$6.0 \cdot 10^{-8}$
^{237}Np	$2.1 \cdot 10^{6}$	α	$5.1 \cdot 10^{-5}$	$2.61 \cdot 10^{7}$	$2.06 \cdot 10^{-5}$
^{238}Pu	87.7	α	$1.18 \cdot 10^{3}$	$6.33 \cdot 10^{11}$	0.567
^{239}Pu	$2.41 \cdot 10^{4}$	α	$7.11 \cdot 10^{-3}$	$2.3 \cdot 10^{9}$	$1.93 \cdot 10^{-3}$

^{240}Pu	6564	α	479	$8.4 \cdot 10^{9}$	$7.06 \cdot 10^{-3}$
^{241}Pu	14.3	β^-	$9.19 \cdot 10^{-4}$	$3.82 \cdot 10^{12}$	$3.28 \cdot 10^{-3}$
^{242}Pu	$3.75 \cdot 10^{5}$	α	805	$1.46 \cdot 10^{8}$	$1.17 \cdot 10^{-4}$
^{241}Am	432.7	α	0.545	$1.27 \cdot 10^{11}$	0.114
^{242m}Am	140	α, β^-	62	$3.88 \cdot 10^{11}$	$4.65 \cdot 10^{-3}$
^{243}Am	7370	α	0.27	$7.33 \cdot 10^{9}$	$6.43 \cdot 10^{-3}$
^{242}Cm	0.45	α	$7.47 \cdot 10^{6}$	$1.23 \cdot 10^{14}$	122
^{244}Cm	18.1	α	$4.0 \cdot 10^{6}$	$3 \cdot 10^{12}$	2.83
^{14}C	$5.73 \cdot 10^{3}$	β^-	-	$1.66 \cdot 10^{11}$	$1.31 \cdot 10^{-3}$
^{36}Cl	$3.01 \cdot 10^{5}$	β^-	-	$1.22 \cdot 10^{9}$	$5.34 \cdot 10^{-5}$
^{59}Ni	$7.6 \cdot 10^{4}$	β^-	-	$2.95 \cdot 10^{9}$	$3.39 \cdot 10^{-6}$
^{79}Se	$1.1 \cdot 10^{6}$	β^-	-	$1.52 \cdot 10^{8}$	$1.36 \cdot 10^{-6}$
^{90}Sr	28.81	β^-	-	$5.11 \cdot 10^{12}$	0.142
^{90}Y	$7.3 \cdot 10^{-3}$	β^-	-	$2.01 \cdot 10^{16}$	3010
^{93}Zr	$1.5 \cdot 10^{6}$	β^-	-	$9.31 \cdot 10^{7}$	$2.86 \cdot 10^{-7}$
^{93m}Nb	16.14	β^-	-	$8.83 \cdot 10^{12}$	$4.37 \cdot 10^{-2}$
^{94}Nb	$2 \cdot 10^{4}$	γ	-	$7.05 \cdot 10^{9}$	$1.96 \cdot 10^{-3}$
^{99}Tc	$2.1 \cdot 10^{5}$	β^-	-	$6.25 \cdot 10^{8}$	$8.54 \cdot 10^{-6}$
^{107}Pd	$6.5 \cdot 10^{6}$	β^-	-	$1.9 \cdot 10^{7}$	$2.8 \cdot 10^{-8}$
^{126}Sn	$2.3 \cdot 10^{5}$	γ	-	$4.57 \cdot 10^{8}$	$1.31 \cdot 10^{-5}$
^{126m}Sb	$3.6 \cdot 10^{-5}$	γ	-	$2.89 \cdot 10^{18}$	$1.02 \cdot 10^{6}$
^{129}I	$1.6 \cdot 10^{7}$	β^-, γ	-	$6.37 \cdot 10^{6}$	$8.9 \cdot 10^{-8}$

^{135}Cs	$2.3 \cdot 10^{6}$	β^-	-	$4.26 \cdot 10^{7}$	$6.1 \cdot 10^{-7}$
^{137}Cs	30.06	β^-	-	$3.22 \cdot 10^{12}$	0.0967
^{137m}Ba	$4.9 \cdot 10^{-6}$	γ	-	$1.99 \cdot 10^{19}$	$2.1 \cdot 10^{6}$

Source: JRC with data from www.nucleonica.net

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