Fuel cycle, spent fuel management and transport of radioactive materials
Background

In 1991, the General Conference (GC) in its resolution RES/552 requested the Director General to prepare 'a comprehensive proposal for education and training in both radiation protection and in nuclear safety' for consideration by the following GC in 1992. In 1992, the proposal was made by the Secretariat and after considering this proposal the General Conference requested the Director General to prepare a report on a possible programme of activities on education and training in radiological protection and nuclear safety in its resolution RES1584.

In response to this request and as a first step, the Secretariat prepared a Standard Syllabus for the Post-graduate Educational Course in Radiation Protection. Subsequently, planning of specialised training courses and workshops in different areas of Standard Syllabus were also made. A similar approach was taken to develop basic professional training in nuclear safety. In January 1997, Programme Performance Assessment System (PPAS) recommended the preparation of a standard syllabus for nuclear safety based on Agency Safely Standard Series Documents and any other internationally accepted practices. A draft Standard Syllabus for Basic Professional Training Course in Nuclear Safety (BPTC) was prepared by a group of consultants in November 1997 and the syllabus was finalised in July 1998 in the second consultants meeting.

The Basic Professional Training Course on Nuclear Safety was offered for the first time at the end of 1999, in English, in Saclay, France, in cooperation with Institut National des Sciences et Techniques Nucleaires/Commissariat a l’Energie Atomique (INSTN/CEA). In 2000, the course was offered in Spanish, in Brazil to Latin American countries and, in English, as a national training course in Romania, with six and four weeks duration, respectively. In 2001, the course was offered at Argonne National Laboratory in the USA for participants from Asian countries. In 2001 and 2002, the course was offered in Saclay, France for participants from Europe. Since then the BPTC has been used all over the world and part of it has been translated into various languages. In particular, it is held on a regular basis in Korea for the Asian region and in Argentina for the Latin American region.

In 2015 the Basic Professional Training Course was updated to the current IAEA nuclear safety standards. The update includes a BPTC text book, BPTC e-book and 2 “train the trainers” packages, one package for a three month course and one package is for a one month course. The” train the trainers” packages include transparencies, questions and case studies to complement the BPTC.

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Editorial Note

The update and the review of the BPTC was completed with the collaboration of the ICJT Nuclear Training Centre, Jožef Stefan Institute, Slovenia and IAEA technical experts.
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1 NUCLEAR FUEL CYCLE

Learning objectives
After completing this chapter, the trainee will be able to:
1. Broadly describe the nuclear fuel cycle;
2. List the main phases of the nuclear fuel cycle;
3. Recognize the difference between the open and closed fuel cycle;
4. Describe basic principles of each phase of the fuel cycle;
5. Describe basic features of the PWR and BWR fuel elements.

1.1 Introduction

The Nuclear Fuel Cycle (NFC) includes the set of processes and operations needed to: mine and extract uranium from ore; enrich the fissile content of the fuel if necessary; manufacture nuclear fuel; irradiate the fuel in nuclear power reactors; store the irradiated fuel; and either reprocess the fuel for recycling of uranium and plutonium or dispose of the fuel and in either case dispose of waste products.

Although several nuclear fuel cycles may be considered depending on the type of reactor and the type of fuel used and whether or not the irradiated fuel is reprocessed and recycled, they all include common or similar steps. They start with mining of uranium and end with disposal of spent fuel and/or other radioactive waste.

The raw material for the NFC is uranium, which is a relatively common metal found throughout the world. The first step of the NFC is uranium production, when uranium ore is extracted from the ground and processed to final product, “yellowcake”, a powder form of uranium oxide (U\textsubscript{3}O\textsubscript{8}). In the second step, conversion, this “yellowcake” is converted to uranium hexafluoride (UF\textsubscript{6}), which can be vaporized at a relatively low temperature. The hexafluoride (“hex”) can be converted to uranium metal for certain types of reactor, but is usually sent for enrichment. In this step the concentration of the fissile isotope \textsuperscript{235}U is increased in comparison with non-fissile \textsuperscript{238}U. In the next step, fuel fabrication, UF\textsubscript{6} is converted to UO\textsubscript{2} powder, which is then converted to ceramic pellets and loaded into long metal tubes forming fuel rods. More fuel rods are put into fuel assemblies for loading into nuclear reactors. This step, irradiation/nuclear reactor operation, is the purpose of the whole NFC since the energy hidden in nuclei is released and transformed to heat which can be used to generate electricity.

After typically 3-6 years in a nuclear reactor, spent fuel is transferred to spent fuel storage, where spent fuel assemblies are stored under water, which provides both cooling and radiation shielding. After a few years, the spent fuel can be transferred to interim storage, where it is kept in water pools (wet storage), or in casks (dry storage).
Depending on the chosen type of NFC and chosen spent fuel management option, spent fuel might be conditioned for longer term interim storage or for disposal (this is spent fuel conditioning), or transferred to facilities where uranium and plutonium are recovered from spent fuel for recycling. Recovered uranium can be converted to UF$_6$ and re-enriched. Plutonium (with uranium) can be used for production of mixed oxide fuel (MOX fuel) for certain types of reactors. All these activities are called spent fuel reprocessing and recycling.

Conditioned spent fuel, and vitrified unusable high level waste products from spent fuel reprocessing can be safely disposed of deep underground, in stable rock formations such as granite.

The steps of the NFC are presented in Figure 1.1. The NFC is “closed” if spent fuel is reprocessed, or partly reused. If this is not a case, the NFC is referred to as an “open” or “once-through” cycle.

![Figure 1.1: Nuclear Fuel Cycle](© IAEA [3], AREVA, Fortum, Posiva, TVO, WNA).

The NFC requires some activities that are omitted from Figure 1.1. These include uranium ore exploration, i.e. activities related to the finding and development of the uranium ores; heavy water production, which is necessary to run certain type of reactors; production of zirconium and nuclear grade stainless steel metal and tubing; management of high level and other wastes; and finally, transportation activities associated with moving materials.
between operations.

### 1.2 Non-Proliferation

Preventing the proliferation of nuclear weapons, their components and the technology to produce nuclear materials is a global imperative that requires the participation and cooperation of industry and governments.

The Nuclear Non-Proliferation Treaty (NPT or NNPT) is an international agreement aimed at preventing the spread of nuclear weapons and promoting cooperation in the commercial uses of nuclear energy and disarmament.

The NPT gave the International Atomic Energy Agency (IAEA) a duty to establish a system of international safeguards.

Created in 1968 and signed by 190 States, the NPT permits ownership of nuclear weapons only by the five countries that possessed them at the treaty’s inception: China, France, Russia, the United Kingdom and the United States. These five countries pledged not to transfer nuclear weapons technology to other states and to reduce their own weapons stockpiles.

IAEA inspectors work to ensure that commercial nuclear materials and technologies are not used for military purposes. Acting under the treaty, the IAEA regularly inspects civilian nuclear facilities. Under the Additional Protocol, adopted by the IAEA in 1997, the IAEA was granted expanded rights of access to information and sites.

To combat the threat of proliferation, the international nuclear energy community has adopted robust controls to ensure that it can secure and fully account for nuclear materials and their by-products. The industry does so throughout the NFC. Controls include global monitoring by international inspectors and stringent national inspection programs.

The principal materials of concern in terms of nuclear weapons production include highly-enriched uranium (HEU) and the plutonium created during reactor operation. It is impossible to create a nuclear weapon from natural uranium or low-enriched uranium (LEU).

With respect to non-proliferation of nuclear materials, the most sensitive phases of nuclear fuel cycle are enrichment and spent fuel reprocessing. Consequently also the respective technologies and equipment are subject to strict international control.
1.3 Uranium production

Uranium occurs with an abundance of 2.7 parts per million by weight in the Earth's crust. It is 600 times more abundant than gold and about as abundant as tin which makes it a rather common metal. Traces of it occur almost everywhere: in most rocks and solids, in many rivers and sea water. Uranium ores usually contain 0.1% to 0.5% uranium although higher grades (up to several per cent) have been found.

Historically, almost forty countries have produced uranium. Production has increased in recent years (prior to Fukushima), to 58 816 tU in 2012, with Kazakhstan producing more than a third of world production (see Figures 1.2 and 1.3).

![Figure 1.2: Country shares in 2012 uranium production [1].]
Thorium
Thorium is a possible alternative source of nuclear fuel, but the technology for using this is not yet fully established. The thorium NFC requires conversion of thorium to $^{233}\text{U}$ in a nuclear reactor, for use as fissile material.

Thorium is probably three to four times more abundant in nature than uranium and the element currently has little commercial value.

Uranium prospecting, mining and milling/recovery
Prospecting for uranium is in some ways easier than for other mineral resources because the radiation signature of uranium's decay products allows deposits to be identified and mapped from the air.

The prevailing method of uranium production in the last decade has been in situ leaching (ISL) with a 45% share in 2012. Conventional methods, like underground mining and open-pit mining provided 26% and 20% of production respectively. Other methods, such as recovery from copper, gold and phosphate operations provided 7%. “Heap-leaching” accounted for 1.7% of uranium production.

ISL is based on injecting chemical solutions into underground deposits to dissolve (leach) uranium “in situ”, i.e. without removing ore from the ground. In this method, mining and the next step, milling (also known as “recovery”), are combined in a single operation.

Conventional uranium mining does not differ from other kinds of mining unless the ore is very high grade. Techniques such as dust suppression, and in extreme cases remote handling, are employed to limit worker radiation exposure and to ensure the safety of the environment and the general public.
The conventional mining and milling activities associated with uranium recovery involve two distinct stages:

- Uranium ore is mined from the Earth, typically from deep underground shafts or shallow open pits (Figure 1.4).
- In milling, the mined ore is crushed, chemically processed to leach the uranium from the ore. After concentration, a material known as "yellowcake" is precipitated. It is a mixture of uranium oxides, ~85% U₃O₈. These oxides vary in proportion and colour from yellow to orange to dark green (blackish) depending on the temperature at which the material is dried, which affects the level of hydration and impurities. The yellowcake produced by most modern mills is actually brown or black, rather than yellow, but the name comes from the colour and texture of the concentrates produced by early milling operations.
Figure 1.5: Overview of uranium extraction methods.

The three milling processes are described below.

**Conventional milling** - A conventional uranium mill is a chemical plant that extracts uranium using a two-step process. In the first step, uranium ore is crushed into smaller particles (Figure 1.6). This material is subsequently extracted or leached by using sulphuric acid or alkaline solutions.

![Uranium ore crusher](image1)

Figure 1.6: Uranium ore crusher.

The second step is the extraction of uranium to precipitate yellowcake (Figure 1.7).

![Conventional uranium mill in the United States](image2)

Figure 1.7: Conventional uranium mill in the United States.
Conventional mills are typically located in areas of low population density, and they process ore from mines within a geographic radius of a few tens of kilometres.

Mill tailings are the fine-grained, sandy waste material that remains after the milling process has extracted and concentrated the uranium from the ore. Mill tailings are typically created in slurry form during processing, and are then deposited in an impoundment or "mill tailings pile," which must be carefully regulated, monitored, and controlled to contain the heavy metal constituents and radium.

Heap leach operations (Figures 1.8 and 1.9) involve the following processes:
1. Small pieces of uncrushed ore are placed in a "heap" on an impervious pad of plastic, clay, or asphalt, with perforated pipes under the heap.
2. An acidic solution (sometime alkali – depends on rock chemistry) is then sprayed over the ore to dissolve the uranium.
3. The uranium-rich solution is collected from the perforated pipes.
4. A solvent extraction or ion-exchange system extracts and concentrates the uranium to produce yellowcake.

![Figure 1.8: The Heap leach recovery process.](image-url)
In Situ Leaching (ISL)/Recovery
The steps in the ISL process are:

1. A solution called a lixiviant (typically containing water mixed with oxygen and/or hydrogen peroxide, as well as sodium carbonate or carbon dioxide) is injected through a series of injection wells into the ore body to dissolve the uranium (Figure 1.10).
2. The lixiviant solution is then collected in a series of recovery wells, from which it is pumped to a processing plant (Figure 1.11), where the uranium is extracted from the solution, usually by ion-exchange.
3. The uranium extract is then further purified, concentrated, and dried to produce yellowcake.

Monitoring wells are checked regularly to ensure that uranium and chemicals are not escaping from the drilling area.
Yellowcake produced by any method (conventional milling, Heap Leach or ISL) is transported to a uranium conversion facility for further processing.

Table 1.1 compares the main environmental and safety regulatory issues for the three main types of milling (also known as “recovery”) facilities.
**Table 1.1:** Characteristics of uranium milling facilities

<table>
<thead>
<tr>
<th>Feature</th>
<th>Conventional Uranium Mill</th>
<th>Heap Leach Facility</th>
<th>ISL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Recovery Method</td>
<td>Physical and chemical process to extract uranium from mined ore.</td>
<td>Physical and chemical process to extract uranium from mined ore that has been piled in a heap.</td>
<td>Chemical process to extract uranium from underground deposits.</td>
</tr>
<tr>
<td>Surface Features</td>
<td>Mill building(s), process tanks, tailings impoundment, and evaporation ponds.</td>
<td>Process buildings, heap pile consisting of crushed ore.</td>
<td>Well fields, header houses, pipes, processing facility, storage or evaporation pond.</td>
</tr>
<tr>
<td>Waste Generated</td>
<td>Mill tailings, pipes, pumps, and other process equipment that cannot be decontaminated.</td>
<td>Heap pile remains in place after processing; pipes, pumps, and other process equipment that cannot be decontaminated.</td>
<td>Liquid waste (disposed of in injection wells or through an evaporation system), pipes, pumps, and other process equipment that cannot be decontaminated.</td>
</tr>
<tr>
<td>Decommissioning</td>
<td>Demolition of mill and site buildings, final cover system installed over tailings pile, groundwater monitoring.</td>
<td>Demolition of site buildings, final cover system installed over heap pile, groundwater monitoring.</td>
<td>Restoration of groundwater, decommissioning of injection wells, removal of pipes and processing building.</td>
</tr>
<tr>
<td>Status at End of Use</td>
<td>Site permanently transferred to relevant authority for long-term care; annual inspections.</td>
<td>Site permanently transferred to relevant authority for long-term care; annual inspections.</td>
<td>Site released for unrestricted use when clean-up criteria are met.</td>
</tr>
</tbody>
</table>

**1.4 Conversion**

“Conversion” is the name given to the process to purify and convert yellowcake to uranium hexafluoride (UF₆). Fluorine has only one natural isotope and does not interfere with enrichment processes based on differences in molecular weight.

There are two principal methods of converting uranium oxide to UF₆:
- wet chemical process,
- dry fluoride volatility process.

In the **wet process**, the uranium concentrate is dissolved in nitric acid. The uranyl nitrate solution is purified and then calcined (heated strongly) to produce UO$_3$ powder. This is hydrofluorinated with anhydrous hydrogen fluoride, which converts it into UF$_4$, a green salt. In the second stage, the UF$_4$ is converted into uranium hexafluoride (UF$_6$) through fluorination.

In the **dry process** the uranium concentrate is pelletized and directly reduced with hydrogen to UO$_2$ in a fluidized bed reactor. The UO$_2$ product is then reacted with anhydrous hydrogen fluoride to form uranium tetrafluoride (UF$_4$). The tetrafluoride is then fed, with gaseous fluorine, into a production unit consisting of a flame-reactor and a fluidized bed reactor, to produce uranium hexafluoride, UF$_6$, gas. This hexafluoride is purified in a distillation process. This is necessary, because, in contrast with the wet process, no purification is carried out in earlier stages.

![Figure 1.12: Cylinder of UF$_6$.](image)

UF$_6$ is highly corrosive if moist. When warm (above ~60°C) it is a gas, suitable for use in the enrichment process. At lower temperatures and under moderate pressure, the UF$_6$ can be condensed. Then it flows as liquid into specially designed, thick walled, mild steel shipping cylinders weighing over 15 tonnes when full (Fig. 1.12). As it cools, the liquid UF$_6$ within the cylinder becomes a white crystalline solid and is shipped in this form.

Conversion plants exist in Argentina, Brazil, Canada and Iran as well as in states that have produced nuclear weapons. The industrial risks
involved are more related to the use of chemotoxic fluorine and its compounds than to nuclear or radiological risks. Furthermore, sensitive and secure measurements under international supervision (IAEA Safeguards) are used to verify the mass and enrichment of UF₆ transferred in and out of the facilities, with the aim of preventing the diversion of nuclear materials and limiting the potential for nuclear weapons proliferation.

### 1.5 Enrichment

Natural uranium is comprised of three isotopes: ²³⁸U (99.28% by mass), ²³⁵U (0.71% by mass) and ²³⁴U (0.005% by mass). ²³⁵U is a fissile nuclide and is the only naturally occurring nuclide which can be used as nuclear fuel in thermal reactors.

Increasing the ²³⁵U isotope above its natural concentration (0.71%) is termed uranium enrichment.

Heavy water reactors and early gas-cooled reactors can run on natural uranium. Light water reactors require enrichments from 2% to 5% ²³⁵U. Research reactors use fuel ranging from natural uranium to enrichment greater than 90% ²³⁵U, but most of them have enrichments just below 20%.

Uranium with enrichment less than 20% ²³⁵U is called low-enriched uranium (LEU), and uranium enriched to 20% ²³⁵U or greater is called highly enriched uranium (HEU). Uranium with a ²³⁵U content less than natural uranium is called depleted uranium (DU).

Because the chemical and physical properties of isotopes differ only very slightly, the separation of isotopes requires special techniques, based on the small difference in isotopic mass.

The basic component of an enrichment plant is the separation element. A separation element (SE) is a device that separates the incoming feed stream into two outgoing streams: an enriched gaseous UF₆ stream, in which the process material is enriched to some degree in the desired isotope, and a depleted gaseous UF₆ stream that is correspondingly depleted in this isotope (Fig. 1.13).

Enrichment processes are made up of many SEs; so it is usual to speak of separation factors per stage of the process. Because each process stage has only a small separation factor, many stages in series are needed to get the desired enrichment level. Also, when each stage has only a limited throughput, many stages are needed in parallel to get the required production rate.
The two of the most important features of a separation element (SE) are the separation factor and the throughput.

The separation factor ($\alpha$) is the degree of separation achieved in a given separation element or stage. It is the relative isotopic abundance of the $^{235}\text{U}$ in the enriched stream, relative to the depleted stream. This is approximately equal to the ratio of the concentration of $^{235}\text{U}$ in the enriched stream to the concentration of $^{235}\text{U}$ in the depleted stream. The magnitude of $\alpha$ is determined by process physics and engineering and varies widely among separation methods.

Throughput is measured by the mass that can be processed in unit time. Some elements can process kilograms of material per minute, while others might process only a few grams per minute. Multiple stages of elements are used to achieve the required enrichment. Elements in different stages may need to differ in physical characteristics due to the smaller amounts of product at later stages, criticality issues etc. Parallel identical elements are used to achieve the necessary throughput (rate of production).
Separative Work Unit (SWU)

Separative Work Unit (SWU) has a precise mathematical definition, but is the best thought of as related to the amount of energy required to take 1 kg of material from one enrichment level to another. A graph showing the energy effort in SWU to process 1 ton of natural uranium to different enrichment levels, and the respective amount of enriched uranium obtained, is shown on Fig. 1.14.

![Figure 1.14: The separative work required for different enrichments, and the respective amount of enriched uranium obtained from one ton of natural uranium (assuming depleted uranium with 0.25% $^{235}\text{U}$).](image)

To produce one kilogram of uranium enriched to 5% $^{235}\text{U}$ requires 7.9 SWU if the plant is operated at a tails assay 0.25%, or 8.9 SWU if the tails assay is 0.20% (thereby requiring only 9.4 kg instead of 10.4 kg of natural U feed). Thus there is a trade-off between the cost of enrichment SWU and the cost of uranium, depending on the tails assay to be achieved.

Commercial enrichment plants operate in France, Germany, Netherlands, UK, USA, and Russia, with smaller plants in about seven other countries. Their cumulative capacity is around 5 million SWU/year.

Many techniques for enriching uranium have been investigated. The gaseous diffusion, gas centrifugation and laser isotope separation techniques are briefly described in the following paragraphs.

Gaseous Diffusion

When a mixture of gas molecules (e.g., $^{235}\text{UF}_6$ and $^{238}\text{UF}_6$) is confined in a vessel and is in thermal equilibrium with its surroundings, the average thermal velocity of the lighter $^{235}\text{UF}_6$ molecules is slightly greater than that of the heavier $^{238}\text{UF}_6$ molecules. Therefore, the molecules of the lighter gas strike the vessel walls more frequently.
(relative to its concentration) than the molecules of the heavier gas. If the walls of the container are porous with holes large enough to permit the escape of individual molecules, but sufficiently small so that bulk flow of the gas is prevented, then the lighter $^{235}\text{UF}_6$ molecules escape more readily than the heavier $^{238}\text{UF}_6$ ones. The escaped gas is then enriched with respect to the lighter component of the mixture, and the remaining gas is depleted.

![Figure 1.15: Gas Diffusion stage.](image)

The basic unit of the gaseous diffusion process is the gaseous diffusion diffuser (Figure 1.15). Compressed UF$_6$ feed gas is made to flow inside a porous membrane or barrier tube. Approximately one-half of the gas passes through the barrier into a region of lower pressure. This gas is enriched in the component of lower molecular weight ($^{235}\text{U}$) and is sent to the next higher stage of the cascade. The gas that does not pass through the barrier is depleted with respect to $^{235}\text{U}$ and is sent back to the previous stage. Upon leaving the diffusion chamber, the enriched and depleted streams have to be recompressed to the barrier high-side pressure to make up for the pressure losses.

By a use of a large cascade of many stages, high overall separation factors can be achieved. It was the first process to be developed that was capable of producing enriched uranium in useful quantities. The gaseous diffusion process is very energy intensive and is no longer economically viable (although a small plant in Argentina was recently re-activated). A large gaseous diffusion plant is shown in Figure 1.16. Newer enrichment facilities are based on a more efficient gas centrifuge technology.
Gas Centrifugation

The gas centrifuge separation process uses the principle of centrifugal force to create a density gradient in a gas containing components of different molecular weights. The gas centrifuge is essentially a hollow, vertical cylinder (i.e., rotor) that is spun about its axis at a high angular velocity inside an evacuated casing (Figures 1.17 and 1.18).

Gaseous $\text{UF}_6$ is fed into the rotor and accelerated to the angular speed of the rotor. Higher centrifugal force on heavier $^{238}\text{UF}_6$ molecules increases their concentration near the outer wall of the cylinder more than the concentration of lighter $^{235}\text{UF}_6$ molecules. Consequently the gas near the axis is enriched with lighter molecules containing $^{235}\text{U}$. This separative effect is assisted by an axial counter current flow of gas within the centrifuge that moves the enriched and depleted streams to opposite ends of the rotor.
Laser Isotope Separation

Laser isotope separation (LIS) is based on the fact that electron energy states of atom are very precisely defined and depend on the mass of the nucleus. Hence different isotopes of the same element, while chemically identical, have different electronic energies and absorb different colours of laser light. If an atom absorbs light with energy exactly corresponding to one of its electronic states, it can become ionized, i.e. obtains a positive electric charge. Such charged ions can then be easily separated from other, neutral atoms, in an electric field.

In LIS enrichment, uranium metal is vaporized in a vacuum chamber. The vapour stream is then illuminated with laser light tuned precisely to a wavelength at which $^{235}\text{U}$ absorbs energy. Ionized $^{235}\text{U}$ atoms are collected on negatively charged surfaces inside the separator unit. The product material is condensed as liquid on these surfaces and then flows to a caster where it solidifies as metal nuggets. The atoms of $^{238}\text{U}$, which were unaffected by the laser beam, pass through the product collector, condense on the tailings collector, and are removed.
Figure 1.19: Schematics of Laser isotope Separation

Laser enrichment is more technically complicated than diffusion but consumes less power and is more efficient. There are no commercial laser enrichment facilities yet.

Management of depleted uranium tails
The depleted uranium that is a by-product of the enrichment process is in the form of UF$_6$. Its quantities are significantly larger than the amount of enriched UF$_6$ which is subsequently used to produce the fuel. A process called “de-conversion” can be used to chemically convert the depleted UF$_6$ to less reactive uranium oxide, by a process similar to that used in the production of fuel.

The inventory of depleted uranium should be managed safely and efficiently in a way that protects the health and safety of workers and the public, and protects the environment until the depleted UF$_6$ is either used or disposed of.

A depleted UF$_6$ Management Program involves three primary activities:
- Cylinder surveillance and maintenance,
- Conversion of depleted UF$_6$ to a more stable chemical form for use or disposal, and
- Development of beneficial uses for depleted uranium.

1.6 Fuel fabrication

Nuclear fuel can be made of uranium oxide, uranium carbide, metallic uranium and various other chemical compounds or alloys. Mixed uranium/plutonium oxide fuels have also been developed.

The vast majority of power reactors utilize fuel made of uranium dioxide (UO$_2$). This dioxide is in the form of ceramic pellets. These pellets are milled to a very precise size and shape, and loaded into
long metal tubes (cladding tubes) to form fuel rods. Many such fuel rods make up a fuel assembly.

Different types of reactor require different types of fuel. Over 80% of the world’s power reactors are Light Water Reactors (LWR), either Pressurized Water Reactors (PWR) or Boiling Water Reactors (BWR). Other types of reactors operating are Pressurized Heavy Water Reactors (PHWR, also known as CANDU), Advanced Gas cooled Reactors (AGR), Light Water Graphite Reactors (LWGR, also known as RBMK), and Fast Breeder Reactors (FBR).

A typical fuel fabrication process may be divided into three stages which are:
- Conversion of UF₆ back to UO₂ and pelletizing;
- Fuel rod manufacturing process; and
- Fuel assembly manufacturing process.

**Conversion and Pelletizing Process**

There are several dry or wet chemical processes for conversion of UF₆ to UO₂ powder. A dry process involving reaction with steam and hydrogen is called IDR (Integrated Dry Route). Two wet processes, are via ADU (ammonium diuranate) and AUC (ammonium uranyl carbonate) [3].

In order to attain the UO₂ powder quality, the physical and chemical properties of the powder have to be within a defined range. In the process of powder preparation; enrichment, uranium content, impurities content and specific surface are measured and confirmed. The powder slug is ball milled into a fine powder and a small quantity of lubricant may be added. In the pelletizing process, green pellets of about 60% theoretical density are produced by compaction and then sintered in the hydrogen sintering furnace at a temperature of greater than 1700 °C. The sintered pellets are ground by a centreless grinder to make the pellet diameter within the specification range. Before these pellets are loaded into the cladding tube, pellets are inspected for diameter, length, perpendicularity, cracks and chips, uranium content, O/U ratio, enrichment, moisture and impurities contents. As far as possible pellet grindings and reject pellets are recycled in the process.

**Fuel Rod Manufacturing Process**

For the purposes of this section of the course, we will focus on LWR fuel, although most of the process is common to most other types of reactor.

The fuel rod consists of a metal tube (cladding), top and bottom end plugs, fuel pellets and plenum spring (Figure 1.20).
The cladding is made of zirconium alloy (Zircaloy, more recently Zirlo). The pellets are loaded into the cladding tube which is bottom-end plug welded. A plenum spring is inserted into the cladding tube in order to compensate thermal expansion and structural changes of fuel pellets during power operation. The fuel rod is pressurized with helium and top end plug welding is performed. The initial pressure of helium in fresh fuel is approximately equal to the primary coolant pressure at operating temperatures. During operation, fission gases (mainly isotopes of krypton and xenon) are released from fuel pellets and the role of plenum (upper empty space of the fuel rod) is to accommodate these gases in order that the pressure in the fuel rod does not increase over its design limit.

Tests are performed, including: radiography; visual inspection; helium leak testing; discoloration at the welds; and checks on plenum length, overall length and straightness of the fuel rod.

**PWR Fuel Assembly Manufacturing**

Pressurised water reactors (PWRs) are the most common type of nuclear reactor accounting for two-thirds of current installed nuclear generating capacity worldwide. Fuel for western PWRs is built with a square lattice arrangement and assemblies are characterized by the number of rods they contain, typically, 17×17 in current designs. A PWR fuel assembly stands between four and five metres high, is about...
20 cm across and weighs about half a tonne. The assembly has vacant rod positions – space left for the vertical insertion of a control rod. Not every assembly position requires a fuel rod or a control rod, and a space may be designated as a "guide thimble" into which a neutron source rod, specific instrumentation, or a test fuel segment can be placed.

A PWR fuel assembly (see Figure 1.21) comprises a ‘skeleton’ that keeps the rods fixed within the fuel assembly. To create the assembly, rods are loaded through a lattice that includes spacer grids, instrumentation tubes and guide thimbles, crowned by a top nozzle. The bottom and top nozzles are heavily constructed as they provide much of the mechanical support for the fuel assembly structure. In the finished assembly most rod components will be fuel rods, but some will be guide thimbles, and one or more may be dedicated to instrumentation.

Various inspections are carried out to confirm that the distance between the fuel rods, fuel assembly torsion, length and other dimensions are correct.

![Figure 1.21: PWR fuel assembly (Mitsubishi).](image)

Russian PWR reactors are usually known by the Russian acronym VVER. These fuel assemblies are characterized by their hexagonal arrangement, but are otherwise of similar length and structure to
western PWR fuel assemblies (See Figure 1.22).

![Image of VVER-1000 fuel assembly](image)

**Figure 1.22:** VVER-1000 fuel assembly.

**BWR fuel**

Boiling water reactors (BWRs) are the second most common nuclear reactor type accounting for almost one-quarter of installed nuclear generating capacity.

BWRs also use zirconium-clad fuel rods containing uranium oxide ceramic pellets. Their arrangement into assemblies is again based on a square lattice, with pin geometries ranging from 6x6 to 10x10. Fuel life and management strategy is similar to that for a PWR.

But BWR fuel (See Figure 1.23) is fundamentally different from PWR fuel in certain ways:

- Four fuel assemblies and a cruciform shaped control blade form a 'fuel module'.
- Each assembly is isolated from its neighbours by a water-filled zone in which the cruciform control rod blades travel (they are inserted from the bottom of the reactor).
- Each BWR fuel assembly is enclosed in a Zircaloy sheath which directs the flow of coolant water through the assembly and during this passage it reaches boiling point, and
- BWR assemblies contain larger diameter water channels – flexibly designed to provide appropriate neutron moderation in the assembly.

BWR fuel fabrication takes place in much the same way as PWR fuel.

BWR fuel assemblies operate more as individual units, and different designs may be mixed in any core load, giving more flexibility to the utility in fuel purchases.
1.7 Fuel during power generation

The fuel assemblies are loaded into the reactor core (coloured orange in the cut-away of the PWR mock-up, Figure 1.24). Here, controlled nuclear fission take places, releasing energy to produce electricity.

The reactor core is housed in the reactor pressure vessel (RPV). It consists of a heavy-walled reactor vessel with all its necessary support and coolant flow guiding structures.
The arrangement of fuel assemblies in the core is dictated by three goals:

- Ensuring uniform power level over the core,
- Maintaining the integrity of the fuel elements,
- Minimising the cost of the fuel cycle.

Nuclear fuel operates in a harsh environment in which high temperature, chemical corrosion, radiation damage and physical stresses may attack the integrity of a fuel assembly. Fuel assemblies are designed so that at their projected maximum burn-up level their risk of failure is still low. Fuel ‘failure’ refers to a situation when the cladding has been breached and radioactive material leaks from the...
gap between ceramic pellet and cladding into the reactor coolant water. The elements with most tendency to be released from the fuel pellet into the gap and then to leak through a cladding breach into the reactor coolant are noble gases and volatile elements, such as krypton, xenon, iodine and caesium. Their radioactive isotopes contribute to radioactive contamination of the primary coolant.

Fuel leaks do not present a major risk to plant safety, though they have a big impact on reactor operations and potentially on plant economics. For this reason, primary coolant water is monitored continuously for these species so that any leak is quickly detected. The permissible level of released radioactivity is strictly regulated against specifications which take into account the continuing safe operation of the fuel. Depending on its severity a leak will require different levels of operator intervention:

- **Very minor leak**: no change to operations – the faulty fuel assembly with leaking rod(s) is removed at next refuelling, inspected, and possibly re-loaded.
- **Small leak**: allowable thermal transients for the reactor are restricted. This might prevent the reactors from being able to operate in a “load-follow” mode and require careful monitoring of reactor physics. The faulty fuel assembly with leaking rod is generally removed and evaluated at the next scheduled refuelling.
- **Significant leak**: the reactor is shut down and the faulty assembly located and removed.

Replacement fuel is one cost component associated with failed fuel. There is also the cost penalty and/or replacement power from having to operate at reduced power or having an unscheduled shutdown. There may also be higher operation and maintenance costs associated with mitigating increased radiation levels in coolant decontamination.

Fuel management is a balance between the economic imperative to burn fuel for longer and the need to keep within failure-risk limits. Improving fuel reliability extends these limits, and therefore is a critical factor in providing margin to improve fuel burn-up.

The nuclear industry has made significant performance improvements reducing fuel failure rates by about 60% in the last 20 years. At the same time there has been a gradual global trend toward higher fuel burnup. There is, however, a limit on how far fuel burnup can be stretched given the strict criticality safety limitation imposed on fuel fabrication facilities such that a maximum uranium enrichment level of 5% can be handled.

Higher burnup does not necessarily mean better energy economics. Utilities must carefully balance the benefits of greater cycle length against higher front-end fuel costs (uranium, enrichment). Refuelling outage costs may also be higher, depending on length, frequency and
the core re-load fraction.

An equally important trend in nuclear fuel engineering is to be able to increase the power rating for fuels, i.e., how much energy can be extracted per length of fuel rod. Currently this is limited by the material properties of the cladding.

### 1.8 Spent fuel storage

Spent nuclear fuel is generated from the operation of nuclear reactors of all types and needs to be safely managed following its removal from the reactor core. Spent fuel is considered waste in some circumstances or a future energy resource in others and, as such, management options may involve direct disposal (as part of what is generally known as the “once through fuel cycle” or “open cycle”) or reprocessing (as part of what is known as the “closed fuel cycle”).

Either management option will involve a number of steps, which will necessarily include storage of the spent fuel for some period of time. This time period for storage can differ, depending on the management strategy adopted, from a few months to several decades. The time period for storage will be a significant factor in determining the storage arrangements adopted. The final management option may not have been determined at the time of design of the storage facility, leading to some uncertainty in the storage period that will be necessary, a factor that needs to be considered in the adoption of a storage option and the design of the facility.

Storage options include wet storage in a storage pool or dry storage in a vault or storage casks built for this purpose. Storage casks can be located in a designated area on a site or in a designated storage building. A number of different designs for both wet and dry storage have been developed and used in different states.

Irrespective of the consideration of spent fuel (either waste or an energy resource), the safety aspects for storage remain the same as those for radioactive waste, which are established in the IAEA GSR Part 5 [4].

**Pool Storage**

After the reactor is shut down, the radionuclides in the fuel (fission products and minor actinides) still generate a significant amount of heat and they are also an extremely high source of radiation. Therefore robust radiation shielding and cooling is necessary. This is accomplished by a deep pool of water adjacent to the reactor to which spent fuel is transferred after discharge from the reactor (Figure 1.25). Damaged fuel elements are inserted in special racks to prevent radioactive contamination of the cooling water.
Water cooling and shielding is necessary for the first few years after discharge. During this period, loss of coolant could result in an overheating accident.

Example:
One week after discharge a ton of spent fuel generates about 100 kilowatts of heat. A full core (72 tons for 1000 MW_e reactor) of spent fuel loaded into a spent fuel pool with dimensions of 21 m × 9 m. If there is no cooling of spent fuel pool, how much time it would take for the water above the fuel to boil off? In the beginning, there is about 6 m of water, above the fuel.

Answer:
The mass of water above the fuel is:

\[ m_w = 21 \text{ m} \times 9 \text{ m} \times 6 \text{ m} \times 1 \text{ t/m}^3 = 1134 \text{ t} \]

The heat generated by the spent fuel is:

\[ P = 100 \text{ kW/t} \times 72 \text{ t} = 7.2 \text{ MW} \]

The time required to heat the water to 100°C:

\[ t_1 = (mc \Delta T)/P = (1134 \times 10^3 \text{ kg} \times 0.0042 \text{ MJ/kg K} \times 75 \text{ K})/(7.2 \text{ MJ/s}) = 49600 \text{ s} = 13.8 \text{ h} \]

The time required to evaporate all the water above the fuel:

\[ t_2 = (m q)/P = (1134 \times 10^3 \text{ kg} \times 2.26 \text{ MJ/kg})/(7.2 \text{ MJ/s}) = 356000 \text{ s} = 99 \text{ h} \]

Total time to evaporate the water is therefore

\[ t = t_1 + t_2 \approx 113 \text{ h} \approx 4.7 \text{ d} \]

It takes less than 5 days for the water to evaporate.

If the fuel in the spent fuel pool is exposed (i.e., the water has evaporated), the temperatures reached could be high enough so that the cladding of the fuel could oxidise in air and lose its integrity resulting in a release of volatile fission products – most importantly \(^{137}\text{Cs}\) which has a half-life of 30 years. The Fukushima accident exposed the vulnerability of spent fuel pools.

![Figure 1.25: Spent Fuel Pool.](image-url)
when most of today’s reactors were designed, the expectation was that, within a few years, the spent fuel would be shipped to a reprocessing plant. For many reactors, this expectation was not realized. Their operators responded first by increasing the storage density of the spent fuel in the pools by a factor of five — to almost the density in the core. In such dense-packed pools, each fuel assembly may be enclosed in a rack lined with neutron-absorbing plates to assure that the arrangement is sub-critical.

**Dry storage**

The generation of heat and radiation reduces with time as shorter half-life radionuclides decay away. After several years, air cooling is sufficient, but significant shielding is still required for radiation protection. Spent fuel can therefore be stored in a dry storage. The intense gamma radiation emitted by spent fuel also requires that fuel is filled in the casks for dry storage under water or remotely behind shielding.

Compared to spent fuel pools, casks for dry storage are passive, and resistant to aircraft crash and earthquakes.

The designs for the dry storage casks evolved initially from transport casks, designed to take spent fuel from the reactor sites to reprocessing plants. The first dry-storage casks were thick-walled cast iron and could be used for either storage or rail transport. Later, less costly dry storage was built by using a relatively thin steel canister to hold the spent fuel, and surrounding it at the storage site with a heavy shell of reinforced concrete for protection and radiation shielding. Cooling is provided by natural convection of air. A more compact design has the canisters inserted horizontally or vertically into a concrete monolith (vault) sized to hold six or more canisters with channels for convective air cooling.

The area density of dry storage is about 0.1 ton per square meter. The lifetime output of a 1 GW\textsubscript{e} LWR, about 1200 tons of spent fuel discharged during a 60-year lifetime, could therefore be stored on an area about 12000 m\textsuperscript{2}. Such an area is easily available within the exclusion zone associated with most nuclear power plants.
In some places (e.g. United States), dry storage is in the open (example on Fig. 1.26). In some other countries, a thick-walled vault provides an extra layer of protection against attack and also additional radiation shielding if the storage area is near a perimeter fence.

### 1.9 Geological Disposal

Spent nuclear fuel or the high-level wastes generated by reprocessing will have to be disposed and isolated from the biosphere for hundreds of thousand years. Most technical experts agree that this could be accomplished by burying the spent fuel or HLW in a mined repository some hundreds of meters underground (see example, Fig. 1.27).

The radionuclides in spent nuclear spent fuel include a wide array of fission products and activation products. Much of the radioactivity decays within the first 100 years, but other radionuclides, such as some transuranic isotopes, are very long-lived. The remaining radioactivity is dominated by plutonium and americium isotopes. For this reason, the geochemistry of these long-lived actinides in a geological medium is important to the science of geological disposal.
The geological conditions of the repository should minimize release from the waste form. More details on final disposal are provided in the Module 19.

1.10 Spent fuel reprocessing

Reprocessing or recycling is a mechanical and chemical process in which spent fuel is separated into different materials (uranium, plutonium, minor actinides, fission products, structural materials). The main purpose is to extract the remaining fissile material, in particular plutonium, from the spent fuel.

Reasons for reprocessing

Spent fuel reprocessing is complicated by a wide array of radionuclides with varying oxidation states in the fuel after dissolution in acid. In spite of this, reprocessing was developed before power reactors, to support weapon programmes.

Several countries, including Russia and Japan have a policy to reprocess used nuclear fuel from power reactors to:

1. Extract fissile materials for recycling - some 25% to 30% extra energy may be extracted from the uranium thus contributing to energy security.
2. Reduce the volume of high-level wastes thereby closing the fuel cycle. Also, due to separating out the uranium and plutonium, the level of radioactivity in the waste from reprocessing falls more rapidly than in spent fuel itself.

Today there is a significant amount of separated uranium which may
be recycled, including from ex-military sources. It is equivalent to about three years' supply of natural uranium from world mines. In addition, a significant amount of plutonium is recycled into MOX fuel and there is about 1.6 million tonnes of enrichment tails, with recoverable fissile uranium.

At present the output of reprocessing plants exceeds the rate of plutonium usage in MOX and the rate of uranium recycling, resulting in inventories of plutonium and reprocessed uranium in several countries.

New reprocessing technologies are being developed to be deployed in conjunction with fast neutron reactors to convert long-lived isotopes, including actinides such as plutonium, to short-lived fission products. This policy is driven by two factors: reducing the long-term radioactivity in high-level wastes, and increasing proliferation resistance of the fuel cycle. Note, however, that other long-lived radiotoxic isotopes may be created. Also, fabrication of the highly radioactive irradiation targets may involve extremely challenging remote handling technology.

Reprocessed uranium
Most of the uranium separated by reprocessing remains in storage, though its conversion and re-enrichment has been demonstrated, along with its re-use in fresh fuel.

The composition of reprocessed uranium is mostly $^{238}\text{U}$ with about 0.5% $^{235}\text{U}$ depending on the initial enrichment and the time the fuel has been in the reactor. The small amounts of $^{236}\text{U}$ created in the reactor act as a neutron absorber requiring more enrichment than fresh uranium if reprocessed uranium is used as nuclear fuel. Traces of $^{232}\text{U}$, make reprocessed uranium difficult to handle since it has strong gamma-emitting daughter nuclides.

Mixed Oxide (MOX) Fuel
Mixed oxide (MOX) fuel is produced by mixing uranium dioxide ($\text{UO}_2$) and plutonium dioxide ($\text{PuO}_2$). It provides about 2% of the new nuclear fuel used today. It is manufactured from plutonium recovered from used reactor fuel. MOX fuel also provides a means of burning plutonium from military sources to produce electricity.

Since 1963, about 2000 tonnes of MOX fuel have been fabricated and loaded into power reactors. Currently over 30 thermal neutron reactors (LWRs and CANDU) in Europe, Japan and Canada have used MOX. LWRs generally use MOX fuel for about one third of their core, but some newer reactors are able to accept complete fuel loadings of MOX.

MOX was first developed for prototype fast neutron reactors in
several countries, but only Russia plans to build more fast reactors.

### 1.11 Questions

1. List the main stages of the fuel cycle.
2. What is yellowcake?
3. What is the principle of gas centrifuge enrichment technology?
4. What kind of fuel is used in reactors in the open fuel cycle?
5. What kind of fuel is used in reactors in the closed fuel cycle?
6. What is the typical enrichment of fuel in current PWR reactors?
7. What is MOX fuel?
8. Describe a typical PWR fuel assembly.
9. Why is spent fuel stored under water for several years?
10. What are the reasons for reprocessing spent fuel?
2 TRANSPORT OF NUCLEAR MATERIALS

Learning objectives
After completing this chapter, the trainee will be able to:
1. To state major Safety standards related to safe transport of radioactive materials;
2. Summarise objectives of regulations for safe transport;
3. Describe the graded approach and safety principles in regulations;
4. Describe different forms of radioactive materials;
5. Explain the importance of $A_1$ and $A_2$ values;
6. Describe different packages;
7. Describe the usage of Transport index and Criticality safety index;
8. Describe the importance of labelling;
9. State what type of packaging is used for transport of LILW waste;
10. Discuss requirements for packages for transportation of spent fuel.

2.1 Regulatory framework

Since 1961 the IAEA has published advisory regulations for the safe transport of radioactive material. These regulations have come to be recognised throughout the world as the uniform basis for both national and international transport safety requirements in this area. Requirements based on the IAEA Regulations have been adopted worldwide by IAEA Member States and international and regional organisations, as the basis for relevant national and international regulations facilitating safe and effective national and international transport of radioactive material.

Some of the regional agreements are The Regulations Concerning the International Carriage of Dangerous Goods by Rail (RID), The European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR), The European Agreement concerning the International Carriage of Dangerous Goods on Inland Waterways (ADN), and The Regulations for the Transport of Dangerous Goods on the Rhine (ADNR).

The IAEA has regularly issued revisions to the transport regulations in order to keep them up to date. The latest set of regulations is published as Safety Requirements - SSR-6, Regulations for the Safe Transport of Radioactive Material [6]. The Regulations are based on the Fundamental Safety Principles, Safety Fundamentals No. SF-1 [7] and on the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115 [8]. Compliance with the Regulations is deemed to satisfy the principles of the Basic Safety Standards in respect of transport.
These Regulations apply to the transport of radioactive material by all modes on land, water, or by air.

In accordance with Fundamental Safety Principles, the prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

The Regulations for the Safe Transport of Radioactive Material are supplemented by the hierarchy of Safety Guides listed as references [9], [10], [11], [12], and [13].

The objective of the regulations is to establish requirements that must be satisfied to ensure safety and to protect people, property and the environment from the effects of radiation during the transport of radioactive material. Protection is achieved by:

- containment of radioactive contents;
- control of external radiation levels;
- prevention of criticality; and
- prevention of damage caused by heat.

The requirements are satisfied by a graded approach to contents limits for packages and conveyances and a graded approach to performance standards applied to package designs depending on the hazard of the radioactive contents, by imposing conditions on the design and operation of packages and on the maintenance of packagings, including consideration of the nature of the radioactive contents, and by requiring administrative controls, including, where appropriate, approval by competent authorities. Confidence in compliance with regulations is achieved through management systems and compliance assurance programmes.
Application of the requirements relies on the principles of inherent safety, passive safety and active safety controls. These principles are incorporated in regulations through:

- limiting the nature and activity of the radioactive material which may be transported in a package of a given design;
- specifying design criteria for each type of package;
- providing information on hazards by labels, marking, placards, and shipping papers;
- applying simple rules of handling and stowage of the packages during transport and in-transit storage.

<table>
<thead>
<tr>
<th>TERMINOLOGY</th>
<th>Definition</th>
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<tr>
<td><strong>Package:</strong></td>
<td>radioactive content + packaging.</td>
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<tr>
<td><strong>Overpack:</strong></td>
<td>an enclosure such as a box, used by a single consignor to consolidate one or more packages so they may be treated as one.</td>
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<tr>
<td><strong>Freight container:</strong></td>
<td>an article of transport equipment that enables goods to be easily transferred between conveyances.</td>
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<tr>
<td><strong>Consignment:</strong></td>
<td>package(s) or load of radioactive material that is presented for transport.</td>
</tr>
<tr>
<td><strong>Consignor:</strong></td>
<td>the individual or organization that prepares a consignment for transport.</td>
</tr>
<tr>
<td><strong>Consignee:</strong></td>
<td>the corresponding agent that receives the consignment.</td>
</tr>
<tr>
<td><strong>Carrier:</strong></td>
<td>an individual or organization that undertakes the carriage of radioactive material.</td>
</tr>
<tr>
<td><strong>Conveyance:</strong></td>
<td>any means by which the package is transported.</td>
</tr>
<tr>
<td><strong>Exclusive use:</strong></td>
<td>when a single consignor has sole use of the conveyance, such that all loading and unloading is carried out in accordance with the directions of the consignor or consignee.</td>
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The philosophy of the regulations:
- Packages of radioactive material should be dealt with in the same way as other hazardous goods;
- Safety depends primarily upon the package rather than on operational controls;
- The consignor should be responsible for ensuring safety during transport through proper characterization of the contents, proper packaging of those contents, and proper operational actions including adequate communications (i.e. shipping papers, marking, placarding and labelling, Transport indexes, Criticality safety indexes, approval certificates, proper shipping names and UN numbers).

2.2 Forms of radioactive materials

In general, radioactive materials being transported have just one common characteristic, i.e. elevated (or high) content of radionuclides. Other characteristics (state, composition, shape, size, chemical properties, etc.) could vary extensively. Some materials could also possess other characteristics that present threats to people and the environment. However, not all characteristics are equally important when it comes to radiation safety during transport. The most important characteristics are those that determine exposure of people during normal transport activities and potential exposures in accident conditions. For that purpose, radioactive materials are categorised in different classes regarding radioactivity, radiation levels and possibility of dispersion of radioactive material.

Materials with very low radioactivity; very low specific activity; or approved consumer products with incorporated radioactive materials; present an insignificant hazard in any situation and are exempted from regulations (i.e. they are not considered radioactive materials). The basis and numerical values for these exemptions (nuclide specific limits) are listed in Regulations and are the same as those in International Basic Safety Standards.

Materials with low activities, or low specific activities above exemption levels and below certain prescribed levels (which are specified in the Regulations and will be also explained later in the text) are classified as excepted materials. These materials, if released from a package due to an accident, present insignificant hazard in foreseeable situations. Examples of excepted materials could be empty packaging, articles manufactured from natural or depleted uranium or thorium, limited quantities of material, instruments or articles, and also small quantities of uranium hexafluoride.

Low specific activity (LSA) material is radioactive material that by its nature has low activity per unit mass. There are three subcategories
of LSA material, namely LSA-I being intrinsically radiologically safe due to the low specific activity of material (it is impossible to inhale or ingest enough material to commit significant dose), LSA-II are materials with radioactivity distributed throughout the material (it could be solid, gas or liquid) and specific activity below a certain level. This level is higher for LSA-III materials, but it must be solid material which is relatively insoluble (with a low leach rate).

Examples of LSA-I material are: unirradiated natural or depleted uranium and thorium compounds and ores; and certain materials with low radiotoxicity and low specific activity. LSA-II could be solids, liquids or gases with limited specific activity, while LSA-III materials could be concrete or bitumen with uniformly distributed activity.

Surface contaminated object (SCO) is another form of radioactive material. It is a solid object which is not radioactive by itself, but has a contaminated surface. Depending on the level of fixed and non-fixed contamination, there are two subcategories SCO-I and SCO-II, the second having higher allowed levels of contamination. SCO material could be for example: parts of the primary circuit in an NPP or other equipment that has come into contact with primary coolant or process waste.

Special form material denotes either indispersible solid radioactive material or a sealed capsule containing radioactive material. This expression refers to material which we usually describe as “sealed source”. It has a very high degree of physical integrity so while it can give rise to high radiation levels, it is unlikely that it will cause any contamination hazard. For special form material this property must be valid also in a transport accident scenario, therefore these sources must be subjected to very stringent qualification. For materials not qualified in this way contamination must be always considered as a probable consequence of an accident. Examples of special form materials are radiography sources, sources for industrial gauges and sources for brachytherapy.

Fissile material is material containing $^{233}\text{U}$, $^{235}\text{U}$, $^{239}\text{Pu}$, $^{241}\text{Pu}$, or any combination of these radionuclides that has the capability of undergoing nuclear fission, and thus requires additional package design considerations and controls to assure nuclear criticality safety during transport. Excluded from this category are natural uranium and depleted uranium that is unirradiated, or has been irradiated in thermal reactors only, and material with fissile nuclides less than a total of 0.25 g.

### 2.3 $A_1$ and $A_2$ values

In the description of forms of radioactive materials we related “low activity” to insignificant hazard if released from a package due to an
accident in foreseeable situations. In transport regulations the approach of hazard assessment is used to determine radioactive material activities which could be transported in packages with certain properties and an ability to withstand particular conditions of transport. We will use $A_1$ and $A_2$ values, which are the result of dose assessment (which includes external effective and committed effective dose) to a person involved in transport accident, where radioactive material was transported in a package designed to withstand normal conditions of transport (but not accidents). Starting from the assessed total effective dose, which is equal to an annual effective dose limit for exposed workers of 50 mSv, values $A_1$ and $A_2$ are calculated for special form material and other than special form radioactive material for most common radionuclides.

Values $A_1$ and $A_2$ are listed in regulations and serve to determine the type of packaging necessary for a particular radioactive material, to limit annual worker doses to 50mSv. Activity limits for excepted packages for most of the radionuclides in solid form are $10^{-3}A_1$ for special form material and $10^{-3}A_2$ for other forms. For liquids, this activity limit is $10^{-4}A_2$, and for instruments with radioactive materials $10^{-2}A_1$ and $10^{-2}A_2$. Packages are designated as “Type A” in cases where $A_1$ and $A_2$ are limiting values.

### 2.4 Classification of packages

A graded approach to packaging requirements implies that the package integrity is a function of the hazard associated with the radioactive material. Therefore more hazardous material (higher activity of the same radionuclide) requires more “resistive” packaging to different conditions of transport. In the regulations, three categories of conditions are used:

- **Routine conditions** (incident free, only conventional stresses and strains resulting from transport and handling);
- **Normal conditions** (with minor mishaps, such as being rained upon, being dropped, having other packages stacked on top);
- **Accident condition**.

Packages for radioactive materials with activities below $A_1$ for special form (or $A_2$ for other forms) are designed to withstand normal conditions (or even only routine conditions), while packages for materials with higher activities must survive accident conditions. Standard types of package for transport of radioactive materials are as follows:

**Excepted packages** are intended for transport of small quantities of radioactive materials (excepted materials or a limited number of instruments containing radioactive material). They should withstand
routine conditions of transport, but there are no special test requirements. It must be assumed that the package will fail in any form of accident. An example of an excepted package is in Figure 2.1.

**Industrial packages** are used for transport of LSA and SCO material. There are three types of industrial packages: Type IP-1 (designed for routine conditions of transport); Type IP-2 and Type IP-3 (which must withstand normal conditions of transport, verified through testing). Although special packaging could be acquired, many packages from industry, such as steel drums and bins could pass the testing. An example of an industrial package is in Figure 2.1.

**Type A packages** are intended for safe and economical transport of radioactive materials with activities up to $A_1$ (special form) or $A_2$ (other forms). They must maintain integrity under normal conditions of transport, which also includes falling from the vehicle, being exposed to water for a limited time, being stuck by a sharp object, or having other objects stacked on top. Specific tests simulate these conditions. An example of a Type A package is in Figure 2.1.

**Type B packages** should withstand most accident conditions without failure of containment or decrease of shielding ability. They must pass a series of mechanical and thermal tests where the effects are

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**Figure 2.1:** Examples of excepted package (upper left), Industrial package (upper right), Type A package (lower right) and Type B package (lower left)
cumulative. The limit to the content is not imposed by regulations, but by an approval certificate. Type B(U) packages are unilaterally approved (in the country of origin of design), while Type B(M) packages are multilaterally approved (country of origin, and countries through and into which the package is transported). But there are limits for transport of Type B packages by air, where upper limits for special form and other forms are 3000 A\textsubscript{1} and 3000 A\textsubscript{2}.

Type B packages are used for a variety of highly radioactive materials, from sources for industrial radiography, gauges, to spent fuel (“casks”) and vitrified waste. Examples of Type B packages are in Figures 2.1, 2.4, 2.5 and 2.6.

**Type C packages** are designed to survive the most demanding severe accident. They are intended for air transport and are tested to conditions that may be encountered in a severe air accident.

**Fissile packages** are designed to fulfil requirements for the radioactive material and also to ensure criticality safety under a variety of postulated conditions.

### 2.5 Transport index (TI) and Criticality safety index (CSI)

**Transport index (TI)** is a number that is assigned to a package (or overpack, freight container, or conveyance), which is used to provide control over groups of packages for the purposes of minimizing radiation exposure risks. It is measured (or calculated for uranium and thorium ores or concentrates) as maximum dose rate at 1 m from the external surface of package in mSv/h multiplied by 100. For example, if the highest measured dose rate at 1 m from the surface of package is 0.1 mSv/h, then the TI is 10. This is also the **limiting value for TI** permitted for a package (or an overpack), which is not a part of consignment under exclusive use (i.e. a part of sole consignment on conveyance). The highest value of dose rate on external surface of the package is limited to 2 mSv/h.

TI for conveyance is sum of TIs of packages or overpacks. TIs of conveyances, which are not under exclusive use, are **limited to 50** (or 200 for cargo planes, or even without limit for large freight containers on seagoing vessels).

**Criticality safety index (CSI)** is a number used to provide control over the accumulation of packages, overpacks or freight containers containing fissile material. CSI is result of calculation which takes into account the maximum number of identical packages under consideration which is still subcritical under conditions that provide the maximum neutron multiplication, considering proper safety margin. Similarly, as in the case of TI, a lower CSI means that more packages can be combined in a single consignment.
The CSI for a package or overpack in a consignment, which is not under exclusive use, is limited to 50. The sum of the CSIs for a conveyance, which is not under exclusive use, must be below 50. In case of exclusive use, the sum of the CSIs must be below 100, or even without limit for large freight containers on seagoing vessels.

2.6 Marking, labelling and placarding and shipping papers

While limiting the nature and activity of radioactive materials and specifying design criteria for packages are considered constituents of inherent and passive safety in the transport of radioactive materials, marking, labelling and placarding are part of active safety controls. They provide permanent information on hazards to people involved in normal transport operations, and also to first responders involved in emergency response in case of an accident.

All packages with radioactive material must have on the outer surface marking “UN” followed by the UN number. UN number uniquely identifies the package content as a specific category of dangerous goods.

All packages but excepted must have marked also proper shipping name and type of package approval (e.g. TYPE IP-1, TYPE A, or TYPE B(M)). Packages TYPE B(U), TYPE B(M) and TYPE C also must have identification mark allocated to that design by the competent authority.

Each package, overpack and freight container shall bear one of the labels with trefoil symbol for radioactivity presented on Figure 2.1. The name of the radionuclide, activity of the radioactive material, and Transport index should be written on the label. The category of label depends on the Transport index: for $TI = 0$ (in fact for $TI < 0.05$) category I-WHITE is chosen, for $0 < TI < 1$ category II-YELLOW is chosen, and for $1 < TI < 10$ category III-YELLOW is chosen. For $TI > 10$ category III-YELLOW should be used for consignment under exclusive use.

Packages with fissile material should also have additional label with inscription “FISSILE” and Criticality safety index (see Figure 2.2)

Large freight containers, rail and road vehicles carrying packages, overpacks or freight containers labelled with any of the labels on the Figure 2.2 should bear placards with trefoil symbol presented on Figure 2.3. The placard is similar to labels for packages, except for the size (it is larger) and there is no data printed on the placard. The orange placard shown in Figure 2.3 must contain the UN number of the radioactive material.
Module XVII: Fuel cycle, spent fuel management and safety transport of radioactive materials

Figure 2.2: Labels for packages with radioactive materials and additional label for packages with fissile material. Below labels corresponding dose rates are written. Number 7 on labels denotes Class 7 according to UN classification of dangerous goods.

Figure 2.3: Placards for large freight containers and rail and road vehicles. Placard on the right must contain corresponding UN number.

For each consignment, the consignor shall deliver transport documents with basic data regarding radioactive material and packages (UN number, proper shipping name, primary hazard Class number 7, subsidiary hazard class division number, name of radionuclide, maximum activity, category of package, TI, etc.); a statement regarding actions, if any, that are required to be taken by the carrier; supplementary requirements for loading, stowage, carriage, handling and unloading of the package, overpack or freight container; restrictions on the mode of transport or conveyance and any necessary...
routeing instructions; and emergency arrangements appropriate to the consignment.

### 2.7 Transport of LLW and ILW

Low-level and intermediate-level wastes (LLW and ILW) are generated throughout the nuclear fuel cycle and from the production of radioisotopes used in medicine, industry and other areas. The transport of these wastes is commonplace and they are safely transported to waste treatment facilities and storage sites.

Low-level radioactive wastes are a variety of materials that emit low levels of radiation, slightly above normal background levels. They often consist of solid materials, such as clothing, tools, or contaminated soil. Low-level waste is transported from its origin to waste treatment sites, or to an intermediate or final storage facility.

A variety of radionuclides give low-level waste its radioactive character. However, the radiation levels from these materials are very low and the packaging used for the transport of low-level waste does not require special shielding.

Low-level wastes are transported in drums (TYPE IP packages), often after being compacted in order to reduce the total volume of waste. The drums commonly used contain up to 200 litres of material. Typically, 36 standard, 200 litre drums go into a 6-metre transport container (freight container). Low-level wastes are moved by road, rail, and by sea. Most low-level waste is only transported within the country where it is produced.

The composition of intermediate-level wastes is broad, but they require shielding. Much ILW comes from nuclear power plants and reprocessing facilities.

Intermediate-level wastes are taken from their source to an interim storage site, a final storage site (as in Sweden), or a waste treatment facility. They are transported by road, rail and sea.

The radioactivity level of intermediate-level waste is higher than low-level wastes. The classification of radioactive wastes is decided for disposal purposes, not on transport grounds. The transport of intermediate-level wastes takes into account any specific properties of the material.

### 2.8 Transport of spent fuel

In some cases (eg for reprocessing) spent fuel has to be transported off site. Some countries like France, Russia and the United Kingdom have
a considerable amount of experience because they have been shipping large quantities of spent-fuel to their reprocessing plants for decades. Sea transport is used from Japan to Europe and from the continent to the UK. Most of the transport within continental Europe and Russia is by train. Smaller casks, containing 0.5 – 2 tons of spent fuel, are transported by truck. Details on commercial casks for spent fuel transport are provided in the IAEA-TECDOC-1532 [14].

Transport casks are TYPE B(U) packages with thick metal walls, usually incorporating an inner layer of lead for gamma-ray absorption and an outer layer that includes both hydrogen in plastic to slow neutrons and boron to absorb the slowed neutrons.

The basis for safety requirements is the IAEA Safety Requirement, SSR-6 [6]. The application for a transport license is made by Safety Analysis Report (SAR), which is similar in format to a storage SAR. However, this specifically provides assessments to demonstrate that the cask can satisfy the requirements for routine, normal and accidental conditions identified within the IAEA regulations. The contents of a SAR may include:

- General – package identification, use, purpose, description, contents, principal design criteria, compliance regulations, requirements and acceptance criteria:
  - Structural
  - Thermal
  - Containment
  - Shielding
  - Criticality
- Operating procedures
- Maintenance programme

Some countries have adopted the IAEA Regulations by reference while others have incorporated them into their national regulations with possibly some minor variations. Modal regulations for road, rail inland waterways, sea and air are listed in references [15] to [21].

**Acceptance tests, maintenance programme and monitoring**

Prior to first use, it is necessary to demonstrate that the cask conforms to the safety requirements outlined in the SAR. These tests are performed during manufacturing, commissioning and before the cask’s first shipment to confirm serviceability, structural integrity and leak tightness.

**Maintenance programme**

The purpose of the maintenance programme is to maintain the integrity of the cask so that it remains compliant with the SAR and the licence conditions. Therefore, the SAR is required to outline the maintenance programme for the cask once the cask is in operation. For transport casks, this is based on either the number of transport cycles
completed or by periodic maintenance based on the cask’s time in operation.

**Monitoring**
Monitoring is completed to ensure that the operation of the cask is in accordance with the license conditions prior to cask loading, during transport and following cask unloading.

**Physical protection**
Physical protection measures include designed features, security measures, and various administrative controls. For the case of spent fuel transport, these may include the attachment of IAEA type seals to the cask prior to transport and the confirmation that the seals are intact at receipt of the cask following transport.

**Integrated Management System (IMS)**
All operation and maintenance steps must be subject to IMS procedures, including unambiguous step-by-step instructions that are easy for the personnel to follow. IMS programmes are required to cover the design, manufacture, testing, operation and maintenance of the cask.

![Figure 2.4: Spent fuel transport.](image)

### 2.9 Transport of plutonium

Plutonium is usually transported either as plutonium dioxide or as MOX fuel. Special attention is necessary since plutonium has fissile isotopes and because it presents a serious health risk if dispersed leading to inhalation. Security during transport also requires special measures. Transportation has to be designed so that criticality is avoided.
Plutonium is transported in several different types of sealed packages and each can contain several kilograms of material. Criticality is prevented by the design of the package, limitations on the amount of material contained within the package, and on the number of packages carried on a transport vessel.

A typical transport consists of a truck or a ship carrying a protected shipping container. The container holds a number of packages with a total weight ranging up to 200 kg of plutonium dioxide.

![Figure 2.5: Cask for mixed oxide fuel (MOX) transport.](image)

### 2.10 Transport of vitrified waste

Highly radioactive wastes (especially fission products) created in the nuclear reactor are segregated may be recovered during the reprocessing operation. These wastes may be incorporated in a glass matrix by a process known as 'vitrification', which stabilises the radioactive material. The molten glass is held in a stainless steel canister where it cools and solidifies. A lid is welded into place to seal the canister. The canisters are then placed inside a Type B cask, similar to those used for the transport of spent fuel. Typically a vitrified waste transport cask contains up to 28 canisters of glass.

Of the order of 1000 tons of vitrified waste has been returned from France and the UK to Japan and more transfers are planned.
2.11 Questions

1. Explain the safety principles applied in regulations for safe transport of radioactive materials.
2. Name and describe forms of radioactive material in terms of transport regulations.
3. Describe the importance of $A_1$ and $A_2$ values.
4. List basic types of packagings.
5. What are the differences between “routine” and “normal” conditions of transport?
6. What types of packages are designed to survive a serious road traffic accident?
7. What is the difference between TI and CSI?
8. How is spent fuel (or high level waste) transported?
3 SAFETY ASPECTS OF THE NUCLEAR FUEL CYCLE

Learning objectives
After completing this chapter, the trainee will be able to:
1. Describe main criticality safety aspects of the NFC;
2. Describe main radiation safety aspects of the NFC;
3. Describe main chemical safety aspects of the NFC;
4. Describe main fire and explosion hazards of the NFC;
5. Describe main effluent concerns in the NFC;
6. Summarise other safety aspects of the NFC.

3.1 Safety aspects in different phases of the fuel cycle

The main hazards associated with different phases of the fuel cycle can be summarized below. Prevention and detection of, and response to, theft, sabotage, unauthorised access and illegal transfer or other malicious acts involving nuclear material and other radioactive substances must be ensured in all stages of nuclear fuel cycle.

Mining and milling
The primary risk are the occupational hazards found in any ore mining and milling operation. In addition, there is likely to be exposure to naturally-occurring radioactive materials (NORM). The drums for transportation of $\text{U}_3\text{O}_8$ are barely radioactive, but it has a chemical toxicity similar to lead, so occupational hygiene precautions similar to those in a lead smelter need to be followed. Most of the radioactivity present in the ore ends up in the tailings. Because the uranium produced at these facilities is not enriched, there is no criticality hazard and little fire or explosive hazard.

Conversion
Chemical hazards, associated with highly reactive HF and UF$_6$ dominate safety assessments. There is a risk of fire and explosion due to hydrogen. The radiological hazard is fairly low although the process to convert yellowcake (uranium oxide) powder to soluble forms could present risks of inhalation and ingestion of uranium.

Enrichment
This is the first phase in the nuclear fuel cycle where criticality safety becomes important. There are also chemotoxic and radiological hazards from the potential release of UF$_6$. Radiological hazards dominate at higher enrichments.

Fuel fabrication
Chemical, radiological, and criticality hazards at fuel fabrication facilities are similar to hazards at enrichment plants. Criticality and
radiotoxicity risks are significantly higher in the case of MOX fuel fabrication, where large quantities of highly radioactive PuO$_2$ have to be handled.

**Power operation**
Radioactive material in nuclear fuel can present a significant hazard for people and the environment during its use in a reactor, and should be treated in accordance with relevant safety standards. Fuel integrity should be assured in normal operational conditions and in most accident conditions in the reactor. Coolant chemistry and operational practice should be designed to avoid adverse effects on fuel materials and to reduce hazards arising from the transport of radioactive contamination in the coolant.

**Spent fuel storage**
Safe, secure and robust spent fuel storage, in wet and dry conditions, should be maintained to minimize fuel degradation, the potential for criticality and radiological hazards to people and the environment.

**Reprocessing**
It is a complex technology involving a wide range of significant radiological, criticality and chemical risks (particularly fire and explosion).

**Transport of spent fuel**
The primary nuclear risk is from radiation following an accident.

### 3.2 Criticality

Criticality safety is concerned with preventing inadvertent nuclear chain reactions. Such chain reactions would create potentially lethal gamma and neutron radiation doses to workers and possible release of fission products. Criticality becomes important when handling enriched uranium or plutonium and is one of the dominant safety issues for fuel cycle facilities.

All areas of fuel cycle facilities that process or contain plutonium or enriched uranium need to be evaluated for criticality hazards. The evaluation must show whether the presence of fissile materials presents a plausible scenario for an inadvertent criticality. As regards nuclear criticality, fuel cycle facilities may be split into two groups:

- facilities where a criticality hazard is not plausible, such as mining, milling, and conversion of natural uranium;
- facilities where the criticality hazards may be plausible, such as enrichment, reprocessing, (enriched) uranium fuel fabrication, MOX fuel fabrication, fresh fuel storage (and transportation), spent fuel storage (and transportation), waste treatment, and waste disposal facilities.
Facilities in the latter group need to be designed to ensure subcriticality in all areas, first by engineering design. This includes assurance that excessive amounts of fissile material do not accumulate above the specified limits in vessels, transfer pipes, sumps, drains, ventilation systems and other parts of the facility.

To prevent inadvertent nuclear criticality, process safety limits must not be violated, and safety limits must have enough margins to preclude criticality during postulated abnormal operating conditions. Particular attention should be paid to fissile material in the waste stream, process changes or modifications that may be inadequate from the point of view of criticality. Also, nuclear material accounting and control procedures may lack the appropriate accuracy to ensure subcriticality. Controls are needed to prevent the accumulation of nuclear materials beyond the intended locations.

Modes of control of criticality safety in any process include, but are not limited to, any one or a combination of the following:

- control of the mass of fissile material present in an equipment;
- control of the geometry (limitation of the dimensions and/or shape) of the process equipment;
- control of the concentration of fissile material in solutions;
- presence of appropriate neutron absorbers;
- limitation of moderation, when it can be guaranteed that (additional) hydrogenous substances cannot be present.

Double contingencies should be provided by appropriate limitation of at least two of the above factors.

Fuel cycle facilities with credible criticality hazards should have in place a programme to ensure subcriticality. Provision should be made to cope with an accident and to notify the facility personnel should an inadvertent criticality occur. Adequate emergency arrangements should be in place, where appropriate.

**Criticality accidents**

Most of criticality accidents have happened in the early years of using nuclear power. Twenty-two criticality accidents have been reported in process facilities up to the year 2000 [23].

A criticality accident occurred at the uranium processing facility in Tokaimura, Japan, in 1999. Three operators were engaged in processes combining uranium oxide with nitric acid to produce a uranium-containing solution for shipment. The material involved was 18.8% enriched uranium. The procedure used deviated from that licensed for the facility. In particular the uranium solution was being placed in a geometrically unfavourable precipitation tank for dispensing into shipment containers, not the more narrow vessel (geometrically favourable to avoid a criticality risks) prescribed by the
license. When two workers were adding a seventh batch of uranium solution to the tank, a criticality excursion occurred. The two workers, along with a third worker nearby, observed a blue flash and fled the location; simultaneously, gamma-radiation detectors went off in the building and two adjacent buildings, prompting all workers to evacuate to a muster area (there was no criticality evacuation alarm system). Two workers subsequently died from radiation exposure during accident.

3.3 Radiation safety

Radiation safety is an important consideration at most nuclear fuel cycle facilities. The theoretical background and the basics of operational radiation protection are described in Module 2.

Radiation protection programme

A facility should have in place a radiation protection programme that is adequate to protect the radiological health and safety of workers and the public and ensure that protection is optimised. To accomplish this, facilities evaluate and characterise the radiological risk and provide sufficient robust controls to minimise this risk. Potential accident sequences are considered in assessing the adequacy of the controls, which aim to minimise radiological risk. Fuel cycle facility radiation protection practices include:

- an effective documented programme to ensure that occupational radiological protection is optimised;
- adequate qualification requirements for the radiation protection personnel;
- approved written procedures for conducting activities involving radioactive materials;
- radiation protection training for all personnel who have access to radiologically restricted areas;
- a programme to measure and control airborne concentrations of radioactive material by engineering protection, administrative controls (designation of areas by contamination risk etc.) and, where necessary, respiratory protection;
- a radiation survey and monitoring programme that includes requirements for control of radioactive contamination within the facility and monitoring of external and internal radiation exposures and taking appropriate action;
- monitoring of both internal (where appropriate) and external radiation exposure to individual personnel, including visitors;
- programmes to maintain records, to report radiation exposures to the regulating authority, and to reinstate an acceptable in-plant radiological environment in the event of an incident.
Design features of nuclear installations to control radiation exposure

The levels of radioactivity in fuel cycle installations vary with the type and capacity (throughput) of the facility, the stage of the process and the radionuclides concerned. Preventive and protective measures, however, are always taken to reduce the external exposure, and especially to reduce the hazard to workers associated with ingestion or inhalation of radioactive substances. There are several engineered measures to accomplish this, most important of them are:

- Containment (e.g. hot cells, glove boxes),
- Ventilation
- Exhaust-gas (from ventilation) cleaning equipment (e.g. scrubbers, chemical traps, HEPA filters and electrostatic precipitators)

3.4 Chemical hazards

Fuel cycle facilities also pose hazards to workers and members of the public from releases of chemotoxic and corrosive materials. They may be considered as chemical plants, which handle and store large volumes of toxic products and corrosive materials. Major steps in the nuclear fuel cycle consist of chemical processing of fissile materials. This processing involves the use of strong reagents to dissolve the materials so that the subsequent chemical reactions may take place.

The use of uranium UF$_6$ in conversion facilities involves the handling of significant quantities of hydrogen fluoride. This poses a significant hazard to workers since hydrogen fluoride and uranium hexafluoride are both highly reactive, chemotoxic chemicals.

Other chemicals encountered at fuel cycle facilities in industrial quantities include ammonia, nitric acid, sulphuric acid, phosphoric acid, hydrogen and hydrazine. Strong acids are used to dissolve uranium and other materials. They are also used to dissolve the spent fuel during fuel element reprocessing, enabling the separation of the plutonium and uranium from the fission products. In addition, the separated fission products, which comprise approximately 99% of the total radioactivity in the spent fuel, pose a significant radiological hazard in what is typically a complex chemical slurry (after evaporation and concentration for storage).

In addition, unplanned release of the chemicals may adversely affect safety controls. For example, a release of hydrogen fluoride could disable an operator whose normal activities may be relied upon to ensure safe processing. In order to reduce risks, the chemically reactive and toxic substances are classified (by hazard or risk as appropriate) and controlled. A robust chemical risk control process
will include process descriptions with sufficient detail to support an understanding of the chemical process risks (including radiological risks caused by or involving chemical accidents) and would allow understanding of potential accident sequences involving chemical processes.

Appropriate methods should be used to estimate the concentration or to predict the toxic footprint of a release to the environment of hazardous chemicals. The tolerability of the toxicological consequences should be assessed against appropriate internationally accepted chemical exposure and appropriate national standards.

Chemical exposure standards are available from a variety of national and international organisations, e.g. relevant ISO standards. Fuel cycle facilities should be designed and operated in a manner that ensures that the risks of hazardous chemical exposure and contamination are controlled and protection optimised.

### 3.5 Fire hazards and explosions

Fire and explosion safety is also an important issue for fuel cycle facilities. Many of the facilities use combustible, and explosive materials in their process operations, such as a TBP-kerosene mixture for solvent extraction, bitumen for conditioning radioactive waste, hydrogen in sintering furnaces, and chemical reagents for oxide reduction. A significant source of hazard in higher radiation facilities is the radiolytic generation of molecular hydrogen (radiolysis is the process in which radiation breaks molecules of water into hydrogen and oxygen).

The design of the facilities should minimise the inventories of combustible materials and ensure adequate control of thermal processes and ignition sources to reduce the potential for fire and explosions. For example, extreme care is taken to prevent accumulation of radiolytic hydrogen in reprocessing plants.

In addition, fire can become the driving force for significant releases of radioactive and toxic material from the facilities as well as a threat to containment barriers themselves. Consequently, extensive fire detection, suppression, and mitigation controls are usually necessary.

A fuel cycle facility safety assessment typically considers the radiological and other consequences from fires and explosion. Suitable safety controls are instituted to protect the workers, the public and the environment from the potential consequences of fires and explosions. These safety controls are designed to provide the requisite protection during normal operations, anticipated operational occurrences, and credible accidents at a facility.
In comparison with nuclear power plants, where a major effort has been made to standardise protection procedures, the hazards encountered in fuel cycle facilities vary considerably and a special fire hazard analysis should be carried out for each individual installation. With older installations, analyses of this kind allow safety authorities to determine what improvements are needed to meet current safety standards. In order to carry out these analyses, some member countries have developed special design codes and expert systems that they use in conjunction with existing technical rules and regulatory requirements or guides. Analysis of fire hazards involves a sequential review of the provisions made for preventing, detecting and fighting fires.

**Explosion hazards**

Unlike fire, explosion hazards are limited to specific areas and specific combinations of circumstances that require specific prevention measures.

Potential generators of explosions are (mostly in reprocessing or waste conditioning processes):
- the use of hydrogen in the sintering furnaces of fuel fabrication;
- the explosive combustion of zirconium powder;
- the decomposition of hydrazoic acid;
- the reaction of solvent (“red oil”) with nitric acid in evaporators;
- the production of hydrogen by radiolysis;
- the oxidation of U;
- the use of reducing agents (hydrazine, etc.);
- the use of solvents and diluents;
- the use of formaldehyde in evaporators;
- the presence of nitrites in resins and bitumen;
- processing involving molten metal.

### 3.6 Effluents

Some fuel cycle facilities may pose special safety hazards to the environment because they produce large quantities of effluents or they have the potential to produce highly hazardous effluents.

Effluents in the forms of liquid or gas must be treated in order to optimise the environmental and health impacts from release. Appropriate filtration systems are used to prevent unacceptable dispersion of aerosol substances within the plant or to control the external release of hazardous substance. A liquid recovery system is used to recycle selected products with appropriate treatment (filtration, distillation, etc.) and to minimise the generation of waste.

A suitable effluent monitoring programme allows fuel cycle facilities to measure and monitor the concentrations of radioactive materials in airborne and liquid effluents and to establish that the protection of the
public and the environment is optimised and in any case below regulatory limits established by the national authority. Airborne effluents from all routine and non-routine operations are usually continuously sampled. It is also usual to have an “environmental sampling programme” using a combination of fixed and ‘random” sampling locations around the facility to ensure that any releases from monitored or unmonitored sources are adequately accounted for and corrective action taken where necessary.

### 3.7 Other safety issues

There are several other important aspects of safety in nuclear fuel cycle facilities. Naturally, there are specifics for different types of installations but nevertheless these safety aspects have common grounds for all types of facilities, including nuclear power plants. Examples of these “other” safety aspects include:

- Deterministic and/or probabilistic safety assessment (described in more detail in Modules 6 and 7)
- Siting considerations and Environmental Impact Assessment (described in more detail in Module 9)
- Maintenance (described in more detail in Module 13)
- Human performance (described in more detail in Module 22)

### 3.8 Questions

1. What is criticality safety and how is it achieved?
2. Which factors affect criticality?
3. How is the uncontrolled dispersion of radioactive substances to the environment prevented in a fuel cycle installation?
4. What chemicals present the most important chemotoxic or explosion hazards in the conversion of UO$_2$ to UF$_6$?
Module XVII: Fuel cycle, spent fuel management and safety transport of radioactive materials

4 IAEA FUEL CYCLE RELATED PROGRAMS

Learning objectives
After completing this chapter, the trainee will be able to:
1. Classify the nuclear fuel cycle safety standards;
2. Describe the Safety evaluation of fuel cycle facilities during operation (SEDO);
3. Describe the Fuel incident notification and analyses system (FINAS);
4. Describe the Integrated nuclear fuel cycle information systems (INFCIS);
5. Describe the fuel bank.

4.1 Nuclear fuel cycle safety standards

IAEA’s Nuclear Fuel Cycle and Materials Section covers the whole nuclear fuel cycle from uranium mining to final disposal, including:
- Production of nuclear-grade uranium;
- Fabrication and in-reactor performance of nuclear fuel;
- Management of spent nuclear fuel;
- Advanced fuel cycles including recycling.

Uranium production

To increase the capability of interested Member States for planning and policy making on uranium production, the IAEA works together with the OECD Nuclear Energy Agency (NEA) to collect and provide information on uranium resources, production and demand. The cooperation results in a publication entitled Uranium - Resources, Production and Demand, commonly known as the Red Book [22]. It has been published since mid-1960 and is now being published at two-year intervals. The Red Book covers the following topics:
- Estimates of uranium resources in several categories of assurance based on existence and economic attractiveness;
- Production capability;
- Nuclear capacity;
- Related reactor requirements.

IAEA Publications on the Uranium Production Cycle are provided on the web site:
http://www.iaea.org/OurWork/ST/NE/NEFW/nfcms_rawmaterials_pu
blications.html
Fabrication and in-reactor performance of nuclear fuel

IAEA programmes cover the following topics of expertise:
- Production of nuclear-grade uranium;
- Development, design and engineering;
- Fabrication: manufacturing techniques, nuclear safety and radiation protection in fuel fabrication;
- Fuel behaviour, analysis and modelling;
- Utilization and management;
- MOX, alternative fuels and advanced fuel technologies and materials;
- Economic and other aspects, e.g. environmental issues;
- Quality assurance and control.

The related main IAEA safety standards and requirements are provided on the web site:

Management of spent nuclear fuel

This IAEA programme supports Member States through two projects:
- Promoting technologies and strategies for spent fuel management; and
- Providing technical guidance on good practices for long term storage of spent fuel.

The related main IAEA publications are provided on the web site:

Advanced fuel cycles including recycling

In the area of Advanced Nuclear Fuel and Fuel Cycles, the IAEA's supports Member States through the following projects:
- Supporting emerging nuclear fuel cycle technologies for advanced and innovative reactors; and
- Supporting development of proliferation resistant fuel cycles.

The related main IAEA safety standards and requirements are provided on the web site:

4.2 SEDO

The main purpose of the Safety Evaluation of Fuel Cycle Facilities during Operation (SEDO) is to assist requesting Member States to enhance the operational safety of their fuel cycle facilities and to promote the continuous development of operational safety in all Member States operating fuel cycle facilities by disseminating information on good safety practices.
SEDO includes assessment of: conversion facilities, enrichment facilities, fuel fabrication facilities, spent fuel storage facilities and reprocessing and associated waste treatment facilities as well as fuel cycle R&D facilities.

The objectives of the SEDO are to:
- Provide useful information on opportunities for improving operational safety.
- Identify good practices.
- Broaden the experience of facility staff through informal exchange of information.
- Instruct the facility staff in the use of the SEDO methodology which could be used for conducting future self-assessments.

SEDO is intended to be a peer review conducted by a team of international experts with experience in the operational and technical areas being evaluated. Judgments on the safety performance of the facility are based on IAEA FCF Safety Standards. It should be emphasized that SEDO is not a regulatory inspection which assesses fuel cycle safety against national regulatory requirements. It does not rank the operational safety performance of different facilities.

A full SEDO mission will review the following areas:
- Management, organization and administration
- Training and qualification
- Operation
- Maintenance and periodic tests
- Modifications
- Other technical support
  - Radiochemical & chemical analytical services
  - Decontamination services
  - Information technology
- Management, organization and administration
- Criticality safety
- Radiation protection
- Waste management
- Fire, chemical and industrial safety management
- Emergency planning and preparedness
- Effluent management and environmental protection

Preparation for a mission begins 12 months before the mission and the duration of the mission is up to 2 weeks. A follow up mission takes place 12 to 18 months after the main mission to assess implementation.

The main safety requirements and guides relevant for SEDO are
4.3 FINAS

The Fuel Incident Notification and Analysis System (FINAS) is a web-based system for the voluntary exchange of lessons learned from operating experience gained in fuel cycle facilities (FCFs). The main objective of FINAS is to provide timely feedback on safety related events, to help to prevent the occurrence or recurrence of such incidents or accidents at other facilities.

Fuel cycle facilities relevant for FINAS are: uranium and thorium mines and milling, refining facilities, conversion facilities, enrichment facilities, fuel fabrication facilities, radioisotope production facilities, waste treatment and conditioning facilities, fuel handling and intermediate storage facilities, reprocessing facilities, and fuel cycle research and development laboratories.

FINAS activities include the collection, evaluation and dissemination of event reports, and the organization of meetings and workshops for participating Member States.

For each safety significant event, its description, cause analysis, lessons learned including the implemented corrective actions provide valuable information to organizations professionally involved in the nuclear fuel cycle industry, such as regulators and their technical support, operating organizations, vendor companies such as design firms, engineering contractors, manufacturers, and research establishments working in the fuel cycle field.

4.4 INFCIS

Comprehensive information on worldwide nuclear fuel cycle activities is available through the IAEA’s Integrated Nuclear Fuel Cycle Information System (INFCIS).

The on-line information system includes:
- Nuclear Fuel Cycle Information System (NFCIS)
- Post Irradiation Examination Facilities Database (PIE)
- World Distribution of Uranium Deposits Database (UDEPO)
- Nuclear Fuel Cycle Simulation System (NFCSS)
- World Thorium Deposits and Resources (ThDEPO)
- Minor Actinide Property Database (MADB)

NFCIS

Nuclear Fuel Cycle Information System (NFCIS) covers civilian
nuclear fuel cycle facilities around the world. It contains information on operational and non-operational, planned, and cancelled facilities. All stages of the nuclear fuel cycle are covered, starting from uranium ore production to spent fuel storage facilities.

**UDEPO**

The *World Distribution of Uranium Deposits Database* (UDEPO) covers uranium deposits around the world, drawing on reports to IAEA technical meetings and other sources. It includes classification of deposits, technical information about the deposits, detailed geological information about regions, districts and deposits.

**ThDEPO**

*World Thorium Deposits and Resources* (ThDEPO) covers thorium deposits around the world based on preliminary data as in IAEA (2013) *World Thorium Occurrences, Deposits and Resources* (under preparation). Details of individual deposits and occurrences are incomplete in many respects due to non-availability of data. More details will be included as they are made available in future.

**PIE**

The *Post Irradiation Examination Facilities Database* (PIE) is derived from a catalogue of such facilities worldwide that the IAEA issued in the 1990s. It includes a complete survey of the main characteristics of hot cells and their PIE capabilities.

**NFCSS**

The *Nuclear Fuel Cycle Simulation System* (NFCSS) is a scenario-based simulation system to estimate long-term nuclear fuel cycle material and service requirements as well as material arisings. The code uses simplified approaches to make estimates.

**MADB**

*Minor Actinide Property Database* (MADB) is a bibliographic database on physical and chemical properties of selected Minor Actinide compounds and alloys. The materials and properties are selected based on their importance in advanced nuclear fuel cycles.

### 4.5 Fuel Bank

On 3 December 2010, the IAEA Board of Governors authorized the IAEA Director General to establish a reserve of low enriched uranium (LEU), or an IAEA LEU bank. Owned and managed by the IAEA, the IAEA LEU bank will help to assure a supply of LEU for power generation.

In the case that an IAEA Member State’s LEU supply to a nuclear power plant is disrupted and cannot otherwise be restored, it may call
upon the IAEA LEU bank to secure LEU supplies. This initiative does not diminish in any way States’ rights to establish or expand their own nuclear fuel production.

The IAEA LEU bank will be sited in Kazakhstan. The LEU will be made available to an eligible IAEA Member State at the market prices prevailing at the time of supply. The proceeds will then be utilized to replenish the stock of LEU in the IAEA LEU bank.

**Requirements for supply**
LEU from the bank will only be supplied upon advance payment, as a mechanism of last resort, to a Member State which fulfils the following eligibility criteria:

- The Member State is experiencing a supply disruption of LEU to a nuclear power plant and is unable to secure LEU from the commercial market, or through State-to-State arrangements, or by any other such means;
- The IAEA has made a conclusion that there has been no diversion of declared nuclear material and no issues relating to safeguards implementation in the requesting State are under consideration by the IAEA Board of Governors; and
- The Member State has brought into force a comprehensive safeguards agreement requiring the application of IAEA safeguards to all its peaceful nuclear activities.

**Recipient State’s obligations**
The Recipient State shall conclude a Supply Agreement with the Agency and through it shall undertake that:

- The LEU from the IAEA LEU bank can only be used for fuel fabrication for the generation of energy at a nuclear power plant;
- The LEU may not be used to manufacture any nuclear weapon or nuclear explosive device, nor for any other military purpose;
- It shall not further enrich, reprocess, retransfer or re-export the LEU unless the IAEA agrees;
- It shall apply the applicable IAEA safeguards, safety standards and physical protection measures to the LEU; and
- It shall take responsibility for all liability for any nuclear damage that may be caused by a nuclear incident associated with the use, handling, storage or transport of the LEU supplied under the Agreement.

**4.6 Questions**

1. What is the purpose of the “Red book”?
2. What are the objectives of SEDO (Safety Evaluation of Fuel Cycle Facilities during Operation)?
3. What is the main objective of the Fuel Incident Notification and Analysis System (FINAS)?
4. Which information is available through the IAEA’s Integrated Nuclear Fuel Cycle Information System (INFCIS)?
5. What is the LEU Bank?
5 REFERENCES

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Advisory Material for the IAEA Regulations for the Safe Transport of


The views expressed in this document do not necessarily reflect the views of the European Commission.