MANAGEMENT AND DISPOSAL OF RADIOACTIVE WASTES AND SPENT FUEL

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LIST OF ABBREVIATIONS

Short name	Full name
ADR	European Agreement concerning the International Carriage of Dangerous Goods by Road
AK (AC)	Activity Concentration
Atv.	Act CXVI of 1996 on Nuclear Energy
BAF	Boda Aleurolit Formation, prospective high activity waste disposal site
BSS	Basic Safety Standards
DF	Decontamination Factor
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
KKÁT	Interim Spent Fuel Storage (ISFS) Facility
MEAK (EAC)	Exemption Activity Concentration
MVDS	Modular Vault Dry System
MF	Micro Filtration
IAEA	International Atomic Energy Agency
NBSz (NSC)	Nuclear Security Code (118/2011. (VII. 11.) Government Decree)
NRHT	National Radioactive Waste Storage Facility
OAH	Hungarian Atomic Energy Authority
OTH	National Public Health and Medical Officer Service National Public Health Office
RHK Kft.	Radioaktív Hulladékokat Kezelő Közhasznú Nonprofit Kft. a non-profit organization in charge of the treatment of radioactive waste
RID	Regulations concerning the International Carriage of Dangerous Goods by Rail
RO	Reverse Osmosis
UF	Ultra Filtration
VLLW	Very Low Level Waste
VVER	Vodo-Vodyanoi Energetichesky Reactor (Russian)

19 MANAGEMENT AND DISPOSAL OF RADIOACTIVE WASTES AND SPENT FUEL

19.1 LEGAL BACKGROUND

International regulations

IAEA International Basic Safety Standards for protection against ionizing radiation and for the safety of radiation sources (IBSS #115.)

IAEA Safety Standards, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards General Safety Requirements Part 3 No. GSR Part 3

IAEA Safety Standards, Predisposal Management of Radioactive Waste, General Safety Requirements Part 5, No. GSR Part 5

IAEA Safety Standards, Decommissioning of Facilities, General Safety Requirements Part 6, No. GSR Part 6

European Union law

Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionizing radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom

Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste

Acts of Law

Act LIII of 1995 on the General Rules of Environmental Protection

Act VXVI of 1996 on Nuclear Energy (Atv.)

Act I of 1997 on the promulgation of the Convention on Nuclear Safety concluded in Vienna on the 20th of September in 1994 under the umbrella of the International Atomic Energy Authority

Act LXXVI of 2001 on the promulgation of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management concluded under the International Atomic Energy Agency

Act II of 2014 on the promulgation of the Agreement between the Government of Hungary and the Government of the Russian Federation on cooperation in the peaceful utilization of nuclear energy

Government decrees

Government Decree 314/2005. (XII.25.) regarding the procedures of environmental impact assessment and the single procedure of authorization of utilization of the environment.

Government Decree 118/2011. (VII. 11.) on the nuclear safety requirements of nuclear facilities and related regulatory activities

Government Decree 246/2011. (XI. 24.) on the safety zone of the nuclear facility and radioactive waste storage facility

Government Decree 124/1997. (VII. 18.) on radioactive materials and equipment generating ionizing radiation not covered by Act CXVI of 1996 on Nuclear Energy

Government Decree 155/2014. (VI. 30.) on the safety requirements applying to storage facilities used for temporary storage or final disposal of radioactive waste and the related official activities

Government Decree 190/2011. (IX. 19.) on physical protection in the utilization of nuclear energy and the related authorization, reporting and controlling system

Ministerial decrees

Decree 23/1997. (VII. 18.) NM issued by the Ministry of Welfare on the exemption activity concentration of radionuclides and on the establishment on their exemption activity level

Decree 47/2003. (VIII. 8.) ESZCSM issued by the Ministry of Health, Social and Family Affairs on certain issues relating to the temporary storage and final disposal of radioactive waste and on radiation health issues of naturally occurring radioactive materials enriched in the course of industrial activities

Decree 16/2000. (VI. 8.) EüM issued by the Ministry of Health on the execution of certain provisions of Act CXVI of 1996 on Nuclear Energy

Decree 11/2010. (III. 4.) KHEM issued by the Ministry of Transport, Communications and Energy on the regime of the registration and controlling of radioactive materials and on the related data supplies

Decree 7/2007. (III. 6.) IRM issued by the Ministry of Justice and Law Enforcement on the rules applying to the registration and controlling of nuclear materials

Decree 51/2013. (IX. 6.) NFM issued by the Ministry of National Development on the transportation, forwarding and packaging of radioactive materials

Standard

MSZ 14344-1:2004 Radioactive wastes. Definitions and classification

19.2 LEGAL RULES APPLICABLE TO RADIOACTIVE WASTES AND SPENT FUEL

The system of regulation in Hungary is based on recommendations issued by various international organizations. Such recommendations are adopted and enshrined in its legislation by the European Union and then the domestic acts of law are drafted on the basis of the provisions contained in international norms and finally the expert bodies of the ministries concerned work out implementing decrees on the basis of such acts of law for those applying and enforcing legal regulations in Hungary. This is the legal background underlying and determining the technical and methodology principles ensuring the adequacy of the waste management system of the new units as well.

The following three organizations play dominant roles in the drafting and harmonization of the official regulations of the international codes applying to radioactive wastes:

- International Atomic Energy Agency
- OECD Nuclear Energy Agency (NEA)
- European Union

In addition to the above the International Commission on Radiological Protection (ICRP) is operating as an independent international committee making recommendations on the basis of the most recent results of scientific research and development.

The document containing the most comprehensive international recommendations is a document issued in 1995 by the International Atomic Energy Agency, entitled “International Basic Safety Standards for protection against ionizing radiation and for the safety of radiation sources (IBSS #115.)", worked out on the basis of the recommendation published by the ICRP in 1990. The volume published in 1995 is also the basis of the radiation protection regulation adopted by the European Union as well as the radiation protection regulation introduced in Hungary. This was then replaced by the IAEA Safety Standards, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards General Safety Requirements Part 3, published in 2014, as well as the Council’s 2013/59/EURATOM Directive of 5 December 2013 which will be transposed into the Hungarian regulations in the coming years.

The new BBS will entail a variety of changes in both the philosophy of the management of radioactive waste and the related licensing/authorization procedures and the relevant limit values. In general, the new recommendations are more pragmatic and contain more precise definitions of the requirements therefore, despite tightening some rules and requirements, on their whole they provide a wider scope for activities relating to radioactive wastes (exemption levels, very low activity waste category etc.).

According to the principles and guidelines applying to the preparation of the Environmental Impact Study regulation aligned to the current regulatory environment must be applied therefore currently effective acts of law and regulations in force at present, based on the provisions of the earlier BSS were taken into account. Wherever applicable in later stages, we will make references to amendments, for the information of the reader.

19.2.1 GENERAL LEGAL PROVISIONS RELEVANT TO RADIOACTIVE WASTES AND SPENT FUEL

The current modern legal foundations of radioactive waste management and disposal were laid down in Act CXVI of 1996 on Nuclear Energy (in Hungarian: Atv., for the purpose of this Chapter: the NE Act). The NE Act sets out the basic principles of using nuclear energy, including those applicable to radioactive wastes and spent fuel. The NE Act sets forth that whenever nuclear energy is used, safety concerns have priority over any and all other considerations.

The user of nuclear energy must ensure that the generation of radioactive wastes associated with pursuing its activities is kept to a reasonably practicable minimum. When using nuclear energy, the safe disposal of generated radioactive wastes and spent fuel must be ensured in compliance with the latest confirmed scientific achievements, international expectations and lessons learned such that no unacceptable burden is placed on future generations.

The NE Act also claims that in respect of managing the radioactive waste and spent fuel generated in Hungary, the ultimate liability is assumed and borne by the Hungarian State. Eventually the Hungarian State is held responsible for the safe and definitive final disposal of these materials, including the waste produced as by-product, if they are transported from Hungary to a Member State of the European Union or a third country for processing or re-processing.

The radioactive waste generated in Hungary must be disposed of in Hungary, unless at the time of transportation there is an agreement in effect with the country undertaking final disposal – in compliance with Article 16 (2) of Council Directive 2006/117/Euratom of 20 November 2006 on the supervision and control of shipments of radioactive waste and spent fuel observing the criteria defined by the European Committee – stating that the radioactive waste generated in Hungary may be delivered to a radioactive waste repository of the country concerned with final disposal. Prior to transportation to the country undertaking final disposal, Hungary must confirm to the greatest extent possible that the target country:

a) Has concluded an agreement relevant to spent fuel and radioactive waste management with the European Atomic Energy Community, or is a participating party to the joint agreement on the safe management of spent fuel elements and radioactive wastes,
b) Has programs in place with respect to the management and final disposal of radioactive waste with high level safety objectives equivalent to the goals set out in the NE Act, and
c) Has a radioactive waste repository with an operating permit scope including the radioactive waste to be transported, which was in use prior to the delivery in question, and is managed according to the requirements set out in the program aiming at the management and final disposal of radioactive waste.

19.2.1.1 National Policy and National Program

Upon a proposal submitted by the Government, the Parliament will approve a national policy with respect to the management of spent fuel and radioactive wastes (hereinafter: National Policy). In the development of the National Policy, the following principles must be enforced:

a) The generation of radioactive waste must be kept to the lowest possible level reasonably achieved in terms of both activity and quantity by suitable design measures, operating and dismantling procedures, in particular by reutilizing and reusing nuclear and other radioactive materials,
b) The interrelations between all steps of spent fuel and radioactive waste generation must be taken into account,
c) Spent fuel and radioactive waste must be managed in a safe way even in the long run, considering the criteria of passive safety,
d) The measures must be implemented observing the principle of gradual progress,
e) The costs arising from spent fuel and radioactive waste management must be borne by the entity having generated them, and
f) In all stages of spent fuel and radioactive waste management a documented and factual decision-making procedure must be applied.

The Government will approve a national program demonstrating the implementation of national policy objectives of spent fuel and radioactive waste management for all stages from generation to final disposal, including the dismantling of the nuclear facility (hereinafter: National Program). The National Program will include:

a) The general objectives of the National Policy,
b) The main implementation stages and the time scheduling of the performance thereof,
c) The inventory of all known existing spent fuel and radioactive waste,
d) The quantity estimate of spent fuel and radioactive waste to be generated in the future, including those originating from dismantling, too,
e) The concepts, plans and technical solutions regarding spent fuel and radioactive waste management, from generation to final disposal,
f) The concepts and plans relevant to the post-shutdown period of the existence of the final disposal facility, including the time period during which checks and controls must be maintained, plus the tools and devices necessary to preserve facility-related knowledge for a long period of time,
g) The description of the research, development and demonstration activities used to implement the solutions related to spent fuel and radioactive waste management,
h) The spheres of responsibility relevant to the execution of the National Program, the key performance indicators used to monitor progress,
i) The cost assessment of the National Program, the survey basis and assumptions, including the evolution of costs as a function of time,
j) The financing system in effect,
k) The tools and procedures promoting transparency and information supply, and
l) The agreements concluded with another Member State or a third country regarding spent fuel and radioactive waste management, including the use of final disposal facilities (repository).

19.2.1.2 Registration of radioactive and nuclear materials

The nuclear energy supervising body maintains a central register of radioactive materials – with a separate list of nuclear substances –. This central register includes:

a) The location, physical and chemical properties of radioactive materials,
b) For closed radiation sources – including those classified as radioactive waste –, the maximum authorized inventory, current inventory, type and activity of radioactive materials,
c) For radioactive wastes – not including closed radiation sources classified as radioactive waste – the type, quantity and inventory change of radioactive wastes by class, or, if known, by radionuclide.

In addition, for nuclear materials the central register also includes:

a) The activities and sites related to the nuclear fuel cycle,
b) The quality and quantity of nuclear materials at the disposal of individual organizations possessing nuclear materials by element (uranium, plutonium, thorium), including their fuel content,
c) The quality and cumulative quantity of nuclear materials at the disposal of all organizations managing nuclear materials by element (uranium, plutonium, thorium), including their total fuel content,
d) The circulation of nuclear materials between organizations in possession of nuclear materials,
e) The correctness of data supply used in international reports and date disclosure.

The user of nuclear energy must maintain a local register showing the location, physical and chemical properties of radioactive materials in its possession, including the related activities pursued, and supply data for the central register.

19.2.1.3 Storage and disposal of radioactive waste and spent fuel

The use of nuclear energy cannot be permitted (licensed) unless the safe disposal of the generated radioactive waste and spent fuel is ensured, in compliance with the latest confirmed scientific achievements, international expectations and lessons learned previously.

The disposal (temporary storage or final disposal) of radioactive waste and the disposal (temporary storage or closure of the nuclear fuel cycle) of spent fuel can be deemed safe if:

a) The protection of human health and the environment is ensured for the entire duration of these activities; and
b) The impact on human health and the environment beyond the country borders does not exceed the domestically accepted value.

The body assigned by the Government (at present the Public Agency of Radioactive Waste Management) manages the performance of tasks related to the final disposal of radioactive waste, the temporary storage of spent fuel, the termination of the nuclear fuel cycle, and the dismantling of nuclear facilities.

The Central Nuclear Monetary Fund is a dedicated state monetary fund in charge of financing the tasks related to the final disposal of radioactive waste, the temporary storage of spent fuel, the closure of the nuclear fuel cycle, and the dismantling of nuclear facilities.

The nuclear power plant must cover the expenses of the final disposal of radioactive waste, the temporary storage of spent fuel – including the dismantling of the storage facility -, the closure of the nuclear fuel cycle, and the dismantling of nuclear power plant itself, and the cost of support provided to controlling and information managing local government associations by executing payments to the Central Nuclear Monetary Fund.

For a nuclear power plant, the rate of contribution must be determined such that its payment is sufficient to fully cover the costs associated with:

a) The final disposal of radioactive waste generated during the entire operating period of the nuclear power plant and upon its dismantling, the temporary storage of spent fuel and the closure of the nuclear fuel cycle,
b) The dismantling of the power plant and the temporary storage facility of spent fuel, with the exception of the first preliminary dismantling plan,
c) The support provided to controlling and information managing local government associations, and
d) The storage of radioactive waste disposed of in nuclear waste repositories constructed prior to the creation of the Central Nuclear Monetary Fund and thus not covered from the payment liability, the operation and safety upgrade of the storage facility.

19.2.2 DETAILED RULES APPLICABLE TO RADIOACTIVE WASTES

19.2.2.1 Classification of radioactive wastes

The NE Act defines radioactive waste as: ”radioactive material for no further utilization that cannot be managed as common waste material on the basis of its radiation protection attributes”.

The classification of radioactive wastes is regulated by Annex 2 to Decree 47/2003. (VIII. 8.) ESZCSM issued by the Ministry of Health, Social and Family Affairs on certain issues relating to the temporary storage and final disposal of radioactive waste and on radiation health issues of naturally occurring radioactive materials enriched in the course of industrial activities.

Low and medium (intermediate) activity radioactive waste is:

- In which heat generation during disposal and (storage) can be neglected.
  - Short lifetime: the half-life of the dominant part of radionuclides in the isotope inventory is 30 years or less, and only contains a limited concentration of long lifetime α-radioactive radionuclides.
  - Long lifetime: the half-life of the radionuclides constituting a significant part of its contained isotopes and / or the concentration of α-radioactive radionuclides exceeds the short lifetime radioactive waste limit values.

High activity waste is:

- In which heat generation must be taken into account during the design of storage and disposal and in the course of repository operation.

The classification of radioactive waste into low and medium activity categories must be performed based on the activity concentration and exemption activity concentration (EAC; in Hungarian: MEAK) of the contained radioisotope.

Radioactive waste class	Activity concentration (Bq/g)	Activity concentration ratio in the presence of several isotopes
Low activity	1 MEAK - 10^3 MEAK	$\sum_i \frac{AK_i}{MEAK_i} \leq 10^3$
Medium activity	> 10^3 MEAK	$\sum_i \frac{AK_i}{MEAK_i} > 10^3$

where:

AK_i is the activity concentration of the ith radioisotope found in the radioactive waste [Bq/g], and

$MEAK_i$ is the exemption activity concentration of the ith radioisotope [Bq/g].

Table 19.2.2-1.: Low and medium category classification of radioactive waste based on activity concentration.

The EAC levels were defined in Decree 23/1997. (VII. 18.) NM issued by the Ministry of Welfare based on IAEA publication I BSS #115.

19.2.2.2 Exemption, clearance, release

19.2.2.2.1 Exemption

Based on the definition of Act CXVI of 1996 on the peaceful domestic utilization of nuclear energy: "radioactive material: naturally occurring or man-made material with one or more components emitting ionizing radiation". If this definition is contrasted with Section 1, paragraph (2) concerning the scope of the same Act, namely: " This Act shall not apply to activities relating to radioactive materials that do not qualify as dangerous to human life or health or to the animate or inanimate environment on account of the nature and level of the ionizing radiation that can be generated", it becomes apparent that there are radioactive materials that do not fall within the scope of the Act, that is, using terminology familiar from radiation protection, and as such are "exempted" from regulation.

The detailed regulation applicable to exemption is given is Government Decree 124/1997. (VII. 18.) and the related Decree 23/1997. (VII. 18.) NM issued by the Ministry of Welfare. The above exemption notion is specified as follows (1. § (1)):

> "The NE Act does not apply to radioactive materials a) in which the total radionuclide activity or b) in the course of activities relating to which the concentration of the activity per unit quantity of radionuclide content does not exceed the exemption level defined in specific other legislation".

Thus according to this definition a particular material can be regarded as exempted below activity or activity concentration levels – as it is, these depend on the radionuclides found in the material. Here the "or" relation must be stressed, as e.g. a low activity point source (which may have high activity concentration due to its small mass) can be exempted just the same as a material present in (relatively) large quantity but having low activity concentration.

As described in 19.2.2.1, the activity and activity concentration levels of the radionuclides of practical importance are given in Decree 23/1997. (VII. 18.) NM issued by the Ministry of Welfare.

The above decree also contains provisions for evaluating exemption for more than one radionuclide. In this case, the conditions of exemption are given as follows: " If the radioactive material contains more than one radionuclides or multiple different radioactive isotopes are used at a work place, the sum generated from the ratios of activity of each radionuclide or the concentration of activity to the relevant exemption level should not exceed 1".

In quantified terms exemption can be decided applied the formula below:

$$\sum_i \frac{AK_i}{MEAK_i} \leq 1$$

where:

AK_i : the activity concentration of the ith radioisotope in the material [Bq/g],

$MEAK_i$: the exemption activity concentration of the ith radionuclide [Bq/g].

It also follows from this that exempted materials actually contain radioactive isotopes, thus they are practically radioactive substances, but form a group of radioactive materials that are not regarded as radioactive from the legal perspective (due to their negligible radiation effect).

Further, it should be mentioned that the IAEA regulation (IBSS115) also provides for the case of disqualification from control, when it is the uncontrollability of certain radiation impacts (e.g., the internal radiation exposure caused by the ^{40}K isotope, with an average soil concentration of 370 Bq/kg, or cosmic radiation observed on the surface of Earth) lead to the cause of disqualification.

19.2.2.2.2 Clearance

The objective of clearance is to allow a radioactive material previously under the scope of control to be removed from the same, so e.g. it is no longer treated as radioactive waste.

The clearance of radioactive materials from the scope of health care radiology authority control is defined in Decree 16/2000. (VI. 8.) EüM issued by the Ministry of Health:

"a material containing radionuclides may be released from supervision by the authority if

a) the individual annual radiation load stemming from its reuse, recycling or treatment as non-radioactive waste does not exceed an effective dose of 30 µSv, and
b) the analysis finds clearance to be the best possible solution."

Clearance may be subject to conditions, which is called conditional clearance, and there is unconditional clearance, too. In the first case, the cleared substance can only be neutralized or used under the conditions established in the study justifying clearance or in the corresponding license. Unconditional clearance means that the substance removed from under authority control can subsequently be used or neutralized without any restrictions.

A clearance study must be developed to observe the 30 µSv annual committed population dose, wherein based on the physical and chemical characteristics of the material to be cleared, the technical and technological parameters of the planned location of use or disposal, and the data pertinent to organization measures it is possible to determine the irradiation pathways, which in turn allows the calculation of the additional radiation dose of the personnel working in the vicinity of radioactively contaminated materials or the nearby population, and finally the applicable clearance thresholds can be set.

In practice this means that it is invariably the so-called conditional clearance procedure that is applied. The cleared material can only be neutralized or used applying the method put forward in the analysis. E.g., the cleared quantity has an imposed maximum, and so has the range of the contained radioisotopes and their respective activity concentration; the waste can only be disposed of in specific repositories, the operations required for transportation and disposal are also precisely outlined, etc. The practice of clearance thus requires complex and adequately quality assured measurement and transport processes.

In conclusion, clearance can only be granted in particular (and carefully analyzed) cases. Finally, the basis of clearance is not the level of activity, but rather the population radiation exposure (dose) caused by cleared waste or other substances.

19.2.2.3 NSC directives

The main provisions of Government Decree 118/2011. (VII. 11.) on the nuclear safety requirements of nuclear facilities and related regulatory activities related to radioactive waste and spent fuel govern the design and operating requirements of radioactive waste and spent fuel management and storage systems.

19.2.2.3.1 General directives

During the operation and dismantling of nuclear facilities, the transportation of radioactive materials related to these activities, and the management of radioactive waste, the following safety objectives must be enforced:

1. One general nuclear safety objective is that individuals and groups of society as well as the environment be protected against the danger of ionizing radiation.
2. The goal of radiation protection is to make sure that the exposure of the operators and residents in the course of the operation of the nuclear facility is always kept below the prescribed maximum limit levels, indeed, at the reasonably possible minimum level. This must be ensured in the case of radiation loads resulting from malfunctions relating to the design basis and – to the extent reasonably possible – from malfunctions and/or accidents beyond the design basis as well.
3. The technical safety objectives include that the it should be possible to prevent or obstruct the occurrence of operational malfunctions or disruptions at a high degree of security, that in the case of all assumed initial events taken into account in the design of the nuclear facility the possible consequences should be kept within acceptable limits and that the likelihood of accidents should be adequately law.

The permit holder must provide for the safe and secure controlling of all radioactive materials used, produced, stored or transported and of the entire radioactive waste output. The output of radioactive waste must be minimized in terms of both activity and quantity.

The requirements applying to the management, treatment and storage of nuclear fuels and radioactive wastes within the site and the adequate storage capacity within the site must be determined in accordance with the national strategy applying to the management, treatment and final disposal of spent fuel and radioactive wastes.

19.2.2.3.2 Management and storage of radioactive waste

The systems, techniques and procedures applied in the management of radioactive wastes must be so designed that will ensure that the final waste output is in line with the requirements applying to transport and temporary storage as well as to the requirements applying to take-over and acceptance for final deposition (to the extent such requirements are known).

To ensure efficient waste management the different components of the radioactive waste output must be separated by physical state and categorized by activity content. Their physical and chemical properties must be taken into account in processing them.

The radioactive wastes produced must be qualified. The representative isotope composition and activity of the radioactive wastes must be established by nuclide specific measurements.

Storage in the site of the nuclear power plant must be so organized that will make it possible to check and identify all radioactive waste package and recover them if necessary.

All waste output of the operation of the facility must be treated as radioactive wastes as long as the contrary is established by documented control measurements.

The generation of any and all radioactive waste that is not compatible with the available storage and processing technology or does not meet the requirements of final disposal (deposition) must be avoided.

The sites and attributes of the radioactive waste package in the storage facility must be recorded and kept registered.

19.2.2.3.3 Management and temporary storage of spent nuclear fuel

The systems and system elements used for the management, transportation and temporary storage of irradiated nuclear fuel must meet the following requirements:

a) To ensure the drain of residual heat in every operating mode,
b) To prevent heavy objects from falling on the fuel assemblies,
c) To ensure the option of visual inspection of the irradiated fuel assemblies, and the possibility to check and certify the airtight coat of the fuel elements (bundles), and
d) To ensure the storage of fuel elements or assemblies with assumed or apparent defects suitable for their conditions.

When determining the storage capacity required for irradiated fuel assemblies, it must be taken into account that, in conformity with the planned treatment procedure of the fuel assemblies in the nuclear reactor, the removal of the necessary quantity of fuel assemblies from the nuclear reactor must be feasible and practicable at all times.

The underwater storage system of the irradiated fuel assemblies must ensure the following:

a) The option to check the irradiated fuel elements, whenever necessary,
b) To make available the devices required for the water chemistry / radiology control of the storage medium,
c) To provide water purification, leakage collector and leak-off control systems, and
d) To provide the storage (decay) pool level and temperature control and monitoring systems.

19.2.2.3.4 Decommissioning

During nuclear facility design, the requirements applicable to the final shut-down and eventual dismantling of the facility must be taken into account as well.

Both the radiation exposure of the population and the persons staying within the nuclear power plant area, and radioactive release must be kept to the reasonably achievable minimum level, and the radioactive contamination of the environment must be avoided during dismantling, too. To this end, the design solutions applied should be such that they allow the optimization of radiation doses expected to occur during dismantling, and attempt to constrain the quantities and activities of the generated radioactive wastes to a reasonable low level.

The permit holder develops a dismantling strategy for all of its sites or site networks. Should there be several nuclear facilities with different permits in a particular site, then the interactions and relations between nuclear facilities must also be considered in all nuclear facility specific preliminary dismantling plans. The strategy must be aligned with the effective national strategies related to dismantling and radioactive waste management and disposal, and also with other national strategies and international obligations influencing the dismantling strategy.

The permit holder must develop the Preliminary Dismantling Plan of the facility as early as the construction permit issuing stage, and update it every five years.

19.2.2.4 Storage and final disposal of radioactive wastes and spent fuel

Government Decree 155/2014. (VI. 30.) Korm. regulates the safety requirements of storage facilities ensuring the temporary storage or final disposal of radioactive wastes. The safety objective of the temporary storage and final disposal of radioactive wastes in a storage facility is to isolate the radioactive isotopes present in radioactive waste and carrying a hazard to people and the environment from the biosphere and the environmental elements having an impact on it, thereby protecting the present and future generations and the environment.

It is a general safety target that human beings and the environment must be protected against the harmful effects of ionizing radiation.

A goal set for radiation protection is to confine the radiation exposure of the employees concerned and the population below the published limit values at all times, to a reasonably achieved lowest level. This must be ensured during operational failures included in the design basis and – to a reasonably practicable extent – for radiation doses occurring during accidents of operational failures beyond the design scope.

A goal set for technical safety is to maintain the possible consequences of all events, incidents potentially assumed in the design of a storage facility within acceptable ranges, and to adequately suppress the probability of accidents.

Prior to the takeover of radioactive wastes in a storage facility the acceptance criteria of radioactive wastes must be defined. Limits must be found and applied as necessary among others for isotope content and activity concentration in each waste package, and also for all other parameters essential from the safety angle. These waste acceptance criteria must define as a minimum the following:

a) Restrictions on waste composition,
b) Restrictions on waste form,
c) Restrictions on waste package storage containers, and
d) If necessary, restrictions on waste packages.

Storage facilities (repositories) may only store radioactive waste that was put into a form and package during pre-acceptance treatment which is in compliance with the waste acceptance requirements.

19.2.2.5 Transportation of radioactive materials

The transportation of hazardous materials – in particular radioactive wastes – is regulated in international agreements. Although in Europe separate agreements apply to different (marine, road, river, air and railway) transportation modes, these show significant similarities. Among these transportation modes, road transport plays the most prominent role, as in all other cases the only way to deliver radioactive materials to the place of loading (e.g. railway station) is by road.

Concerning the transportation of radioactive materials, the most important protection objective is to prevent contamination, and to protect people, material assets and the environment against radiation effects. The choice, material and dimensions of packaging depend on the properties of the radioactive material to be transported. The selection of suitable packaging is dictated by the nature of the radioactive material, plus the type, activity and physical-chemical characteristics (e.g. state of matter) of the radioactive isotope to be transported.

The safe application of radioactive materials emitting ionizing radiation and materials capable or potentially capable of self-sustaining nuclear chain reaction is supervised by a strict authority system. According to the NE Act, the sphere of competence of the Hungarian Atomic Energy Authority (HAEA) includes among others licensing the transportation of radioactive materials in compliance with the legal provisions regarding the transportation of hazardous goods, or the control and approval of radioactive material packaging. This means that the HAEA issues licenses for international transports and also the domestic transport of very high activity radioactive materials. Other HAEA tasks are to issue and receive the related international notifications, and to initiate the operative measures prompted by extra-ordinary events potentially occurring in international transportation projects.

Decree 51/2013. (IX. 6.) NFM issued by the Ministry of National Development on the transportation, forwarding and packaging of radioactive materials essentially regulates the licensing and control competence of individual authority offices.

Government Decree 190/2011. (IX. 19.) on physical protection in the utilization of nuclear energy and the related authorization, reporting and controlling system sets the conditions of physical protection (guarding, security) required during transportation.

Concerning issues of content, in the context of road transportation the provisions of ADR (European Agreement concerning the International Carriage of Dangerous Goods by Road) relevant to Class 7 must be applied.

For railway transportation, the provisions of the RID (Regulations concerning the International Carriage of Dangerous Goods by Rail) Regulations apply. The text of its version in effect (RID 2011) and harmonized with ADR is included in Act LXXX of 2011.

The requirements related to packaging types are based on the activity values inducing radiology hazard, defined for each isotope. In the current regulation, separate limit values apply to water-insoluble, non-dispersible, so-called ”special form” (A1), and water soluble or dispersive radioactive materials (A2). To define these values, those potential irradiation pathways were analyzed that can produce significant internal and / or external dose exposure to individuals in a transportation accident.

Depending on the characteristics of the material to be transported, the packaging can be one of the following types:

- Type "A" packaging: This is the classic packaging form for radioactive materials. This packaging type must contain its radioactive material content and preserve its radiation shielding features under normal transportation circumstances, including minor loading anomalies (e.g. sudden cargo imbalance and tipping). The maximum activities to transport in type "A" packaging are set by the A1 / A2 values.
- Type "B" packaging: To be used to transport radioactive materials exceeding the A1 / A2 values – that is, with high inherent radiology hazard. Consequently, this package type must retain its radiation shielding features even in case of accidents, permitting only a limited release of radioactive materials. In terms of authority approval, the packaging can have either the (unilateral) "B(U)" marking, exclusively approved by the authority of the given country, or the (multilateral) "B(M)" marking affixed, whenever authorities in more than one country approve of the packaging form. This can be meaningful for international transports.
- Type "C" packaging: To be used when some high activity radioactive material must be transported by air. Type "C" delivery items must be prepared in a way making it improbable to release radioactive materials and lose the continued ability to shield radiation due to in-transit incidents, including circumstances prone to air traffic accidents.

- Fissile material packaging: Can be any of the already listed packaging types depending on activity. When transporting such materials, the packaging must also guarantee to sustain sub-critical status under both normal delivery or accident-related conditions during the entire trip duration.

19.2.2.6 Registration of nuclear materials and radioactive wastes

Decree 11/2010. (III. 4.) KHEM issued by the Ministry of Transport, Communications and Energy regulates the registration and control agenda of radioactive materials and wastes. For radioactive wastes, the local register must have the following data entered:

a) Class of radioactive waste;
b) Basis of classification;
c) Nature of storage;
d) Form of waste;
e) Packaging;
f) Quantity, unit of measure;
g) Radionuclide element and mass (nucleon)number (if known);
h) Activity, date of activity (if known);
i) Activity determination method;
j) Date of classification as waste, test report number, name and address of MoM keeper.

Decree 7/2007. (III. 6.) IRM issued by the Ministry of Justice and Law Enforcement regulates the registration and control of nuclear materials. The objective of maintaining the central register is to fulfill the obligations undertaken by Hungary in the Safeguards Agreement and the Additional Protocol, and to comply with the liabilities defined for Hungary in the Euratom Agreement and the Euratom Directive.

Any organization in possession of nuclear materials must keep a local register of the nuclear materials at its disposal. This local register must meet the requirements set forth in the Safeguards Agreement.

19.3 MANAGEMENT OF RADIOACTIVE WASTES

The definition of radioactive waste fits any material generated as a result of some planned nuclear activity, with no demand or method for further use, containing radioisotopes with concentrations in excess of the limit values to be safety released into the environment (regarded safe) or to be disposed.

In terms of management / treatment, radioactive waste is:

- Such gaseous, liquid or solid state radioactive material for which no future use is planned, and which is subject to control by some authority as being radioactive waste;
- Such radioactive materials that have no further use, which cannot be treated as ordinary waste due to its radiation protection properties.

19.3.1 CLASSIFICATION AND GROUPING OF RADIOACTIVE WASTES

Radioactive wastes can be classified along several criteria:

- By origin (nuclear cycle, non-nuclear cycle)
- By radiological features:
  - Radiation type (low-, high penetrating power)
  - Activity concentration (low, medium, high)
  - Dose rate (low, medium, high)
  - Surface contamination (fixed, not fixed)
  - Half-life (short, medium, long)

- By physical properties:
    - State of matter: (solid, liquid, gaseous)
    - Heat generation
    - Criticality
    - Size or volume
- By chemical properties (toxic, corrosive, combustible, flammable, organic)
- By waste management method (can be pressed, incinerated, cemented etc.)

Irrespective of their type, in practice the internationally accepted method to categories radioactive wastes is to use the so-called waste index – calculated using the sum of the activity concentration (AK_i) and exemption activity concentration ($MEAK_i$) quotients of the radioactive isotopes contained - as a basis according to the parameters given in table .

In addition to the table showing activity concentrations, the Hungarian standard includes a similar supplement of express practical importance: a determination of waste classes based on the gamma-dose rate value measured on the surface of the packaging materials enclosing the waste. This classification method is more intimately connected to practical waste management activities, as the dose rate can be measured easily and accurately with some hand-held manual instrument, whereas the measurement of isotope selective activity concentration (i.e., to find the AK_i values in the formula) entails lengthy and cumbersome procedures. Here the values corresponding to individual categories were again summarized in table 19.3.1-1.

Radioactive waste	$\sum \frac{AK_i}{MEAK_i}$	Surface dose rate [μGy/h]
Low activity	1,0 - 1 000	< 300
Intermediate activity	1 000 - 1 000 000	300 - 10 000
High activity	> 1 000 000	> 10 000

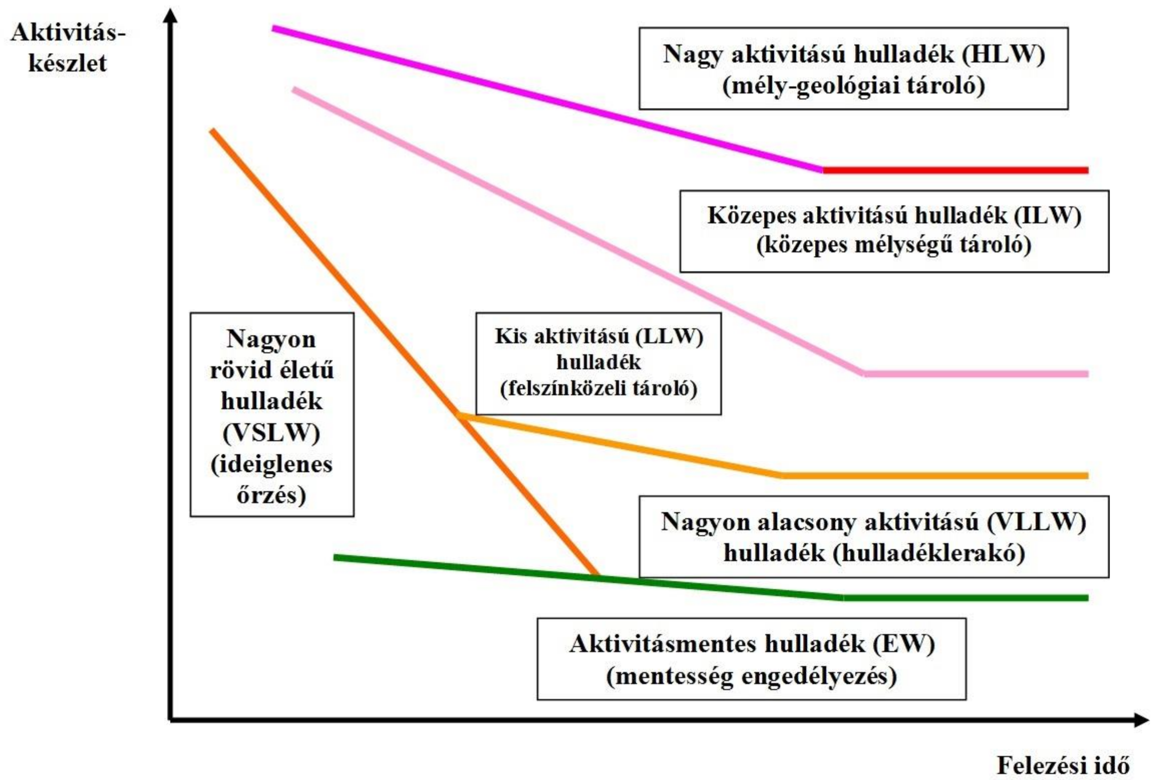
Table 19.3.1-1.: Categorisation of radioactive wastes based on waste indices and surface dose rates derived from isotope composition. [19-2]

In recent years, driven by efforts getting more and more widespread in international practice, aiming at disposing of very low activity wastes, the very low-level activity radioactive waste (VLLW) category was included in the new IAEA safety guidelines defining the basis of the waste management system [19-20] laying down the foundations of the VLLW waste management system. According to the IAEA definition, very low activity radioactive waste (VLLW) is waste that does not necessarily meet the waste clearance / exemption criteria, but does not require efficient confinement either.

Based on the actual state of matter, solid state, liquid and gaseous radioactive wastes are distinguished. Of these, the present subchapter only considers the environmental effects and impacts of solid and liquid wastes, as the effects of wastes in gaseous phase were described among the airborne releases of the nuclear power plant. In fact solid and liquid phases can co-exist within a particular form of waste, e.g. mud or sludges may contain fluids and dispersed solid particles in them, or for instance the solid phase of wet wastes may contain a significant amount of porewater. Due to this, in practice radioactive wastes are classified as solid or liquid based on their management options.

Based on waste **processability**, the following classes are distinguished:

- Solid, compactable (plastic, paper, textile-fabric, rubber etc.)
- Solid, non-compactable (demolition waste, scrap metal, machine parts, tools etc.)
- Solid, bulky (steam generator, tank, heat exchanger etc.)
- Solid, wet (sludges, muds, ion exchange resins, filters etc.)
- Solid, incinerable (paper, plastic, wood, organic materials etc.)
- Contaminated technological solutions (different boric acid solutions, decontamination solutions etc.)
- Contaminated solvents and oils (flammables, organic materials)
- Metals (can be cleared or reused etc after decontamination)



Legend: Aktivitás készlet - Activity content; Felezési idő - Half-life; nagy aktivitású hulladék (mély geológiai tároló)- High-level waste (HLW) (deep geological disposal), Közepes aktivitású hulladék (közepes mélységű tároló) - Intermediate-level waste (ILW) (intermediate depth geological disposal), Kis aktivitású hulladék (felszínközeli tároló) - Low-level waste (LLW) (near surface disposal), Nagyon alacsony aktivitású hulladék (hulladéklerakó) - Very low level waste (VLLW) (landfill disposal), Aktivitásmentes hulladék (mentesség engedélyezés) - Exempt waste (EW) (licensing of exemption), Nagyon rövid életű hulladék (ideiglenes őrzés) - Very short lived waste (VSLW) (decay storage)

Figure 19.3.1-1: Schematic concept of radioactive waste classification.

Based on their place of generation, radioactive wastes can be characterized by the following origins:

- After commissioning and parallel connecting the facility, from daily cleaning and maintenance work mainly primary loop-linked generation can be assumed (e.g. technological cleaning during each shift, for the entire inner containment space, contaminated boric acid solution)
- From the replacement of activated part or equipment, plus the tools, wiping cloth pieces, cleaning / absorbing substances used for replacement,
- From (planned or non-planned) primary loop coolant leakage.

There is no need to set up additional radioactive waste categories to express the time sequence of generation, as the wastes generated during shut-down / maintenance / transfer only differ from those produced during power generation mode to some smaller (in quality) or greater (in quantity) degree, and even these quantity increments can only be expected to occur intermittently – every 18 months – due to the nature of planned shut-downs.

When describing a grouping based on **disposability**, classes of conventional waste types must also be invoked (flammable, toxic, unstable, soluble, organic, decomposing etc.), however, the waste categories defined along the lines of radioactive waste disposal take into account the following characteristics:

- Level of radioactivity,
- Degree of heat generation,
- Lifetime of radioactive waste.

These features and factors determine the time period over which the level of radioactivity drops to some acceptable level, the knowledge of which proves to be indispensable for waste storage installation design.

19.3.2 SOURCES OF RADIOACTIVE WASTE IN NUCLEAR ENERGY PRODUCTION

Various types of waste will be generated during the operation and maintenance of the planned nuclear power plant. A certain part of these, generated in the reactors or in systems directly connected to them, may become contaminated by the radioisotopes escaping the fuel elements, or by the radionuclides deriving from substances activated by neutron radiation in the reactor zone. Radioactive wastes are inevitable by-products of nuclear-based electric energy generation, therefore their management, temporary and final storage must be addressed and ensured. Solid and liquid radioactive wastes are generated during normal operation and maintenance, and during dismantling the nuclear facility.

The following **solid radioactive wastes** can be generated during normal operation and maintenance activities:

- Activated equipment and parts removed from the reactor,
- Surface contaminated fittings, equipment, piping, heat insulation etc. from maintenance,
- Construction materials from reconstruction work (concrete rubble, wood, glass etc.), or other contaminated scrap metal, cabling etc.
- Waste metal, worn-out tools, chippings from maintenance workshops,
- Clothing, other personal protective equipment, filter inserts, wiping rags, sheet films/foils discarded from operations and maintenance etc.
- Radioactive sludges and other secondary waste generated from liquid radioactive waste management (ion exchange resins, sorbents, filter inserts, etc.).

The following **liquid radioactive wastes** can be generated during normal operation and maintenance activities:

- Boric acid wastes of primary loop non-organized leakages, drainages, air ventings
- Regeneration wastes and loosening waters of primary loop water purifiers
- Used ion exchange resins of primary loop water purifiers
- Chemical contaminated wastes from equipment decontamination (steam generator)
- Liquid wastes and other floor waters from premise decontamination
- Primary loop laboratory and laundry wastes
- Contaminated shower waters
- Contaminated organic solvents (petroleum benzene, alcohol etc.) and oils.

19.3.3 BASIC PRINCIPLES OF RADIOACTIVE WASTE MANAGEMENT

The fundamental objective of radioactive waste management is to establish protection for the health and environment of today's humanity and to safeguard future generations. In the light of this, the basic principles of the IAEA waste management guidelines [19-19] are those listed below:

- The protection of human health and the environment in a given country and even beyond its borders,
- The protection of the future generation,
- The creation of a national regulatory system,
- The control of waste generation,
- The mutual investigation of waste generation and management / manageability, and
- The expression of safety issues, concerns and safe conduct.

Recapitulating the relevant literature leads to the summary conclusion that the existing international recommendations, guidelines, general principles and – last but not least – experience concerning radioactive waste management delineate the boundaries, and by doing so create a framework wherein each country may set up its own waste management practice considering its domestic industries, applied technologies, emission directives, disposal opportunities and budgetary constraints.

Observations and findings confirm that, in addition to using the best technologies available – adherence to general waste management principles is influenced to some degree by the economic (financial) background, culture, mindset of the countries, and the ownership relations of power plants, waste management facilities and final disposal repositories, too. There is a need for collaboration among the various groups and stakeholders involved in this process – those generating waste, those determining official regulations and emission limits, those taking over waste and independent laboratories – as well as for flows of information among stakeholders, process controlling, feedback and an overarching quality assurance regime.

Waste management is a complex system made up of a multitude of different elements and variables. Its subject is always the same however, that is, waste itself, the most important general attributes of which include physical state, density, form, radionuclide content, dose rate, manageability (possibility to compress, flammability or the lack of it etc.). The problem stems from the fact that a great variety of different wastes are produced while in general there are three different possibilities for disposal/deposition (surface/near surface, deep geological or in the case of releasable waste the non-radioactive waste depots). Therefore it is completely impossible to prescribe and introduce a standardized waste management system even for similar forms and packages of waste.

From the aspect of the entity producing waste, in view of the life cycle of radioactive waste, the key pillars of the strategy include (quantitative and qualitative) design, generation (and selective collection), treatment, conditioning, internal storage, transport and deposition possibilities. One of the most important steps between generation and treatment/conditioning is as precise identification as possible of the waste output, qualification and labeling of the waste packages, in order to ensure traceability. The applicable processing and conditioning technologies are affected by the requirements of the storage facilities and the possibility to deposit waste.

19.3.3.1 Waste management during nuclear power plant operation

Activities related to radioactive waste management must be appreciated in a unified system from generation to final disposal, and the measures necessary in any particular phase must be defined accordingly. Waste management is nothing but the umbrella system of activities linked to waste, existing within a single organization or split into several of them. In what follows, these activities linked to waste and the requirements imposed on them are listed.

- Within the scope of general strategic planning, the operating organization should develop a radioactive waste management program, including measures to address in particular, but not limited to, the following:
    - Apply a suitable technology to keep radioactive waste generation at some practicable minimum level in terms of both activity and volume,
    - Reuse and re-utilize materials as efficiently as possible,
    - Classify and select wastes adequately, maintain their accurate inventory for each waste stream, not forgetting the options of clearance and final disposal,
    - Collect, qualify and store radioactive waste in an acceptable, safe and secure way,
    - Create storage capacity to match assumed radioactive waste generation trends,
    - Ensure waste recoverability by the end of the storage period,
    - Treat and condition radioactive waste in compliance with the requirements of safe storage and final disposal,
    - Transport, haul and forward radioactive waste in a safe manner,
    - Control releases (emissions, discharges etc.) to the environment,
    - Monitor waste at source and in environment to check conformity,
    - Monitor radioactive waste packaging status at storage location,
    - Monitor changes in radioactive waste characteristics by visual inspection and periodic analysis, in particular during prolonged storage periods.
- The generation of liquid radioactive wastes must be limited to the practicable minimum level.
- Radioactive liquid waste – especially water-based – must be classified along processing avenues, based on specific activity and chemical element content.

- The generation of radioactive solid wastes must be minimized by in turn minimizing the production of gaseous and liquid wastes, thereby reducing the processed waste quantity applying among others good selection practices, including clearance at the place of waste generation.
- Radioactive solid wastes must be classified on the basis of activity and type.

19.3.3.2 Strategy and method of waste qualification

The adequate control of the physical, chemical and radiological parameters of radioactive waste throughout the entire waste lifecycle, and the comprehensive testing of final waste forms and waste packages are part and parcel of every waste management strategy. An error in the control process in any step may lead to far-reaching consequences, not only regarding the subsequent step, but it can result in the generation of a waste package that does not meet the waste acceptance criteria defined for long-term storage or final disposal.

The objectives of waste qualification are listed hereunder:

- Create a process for radioactive waste qualification for the entire lifecycle;
- Contribute to safety during different phases of the radioactive waste lifecycle, that is: generation, processing, storage and final disposal;
- Demonstrate that the acceptance criteria (takeover, processing, storage, final disposal) - which were determined based on the evaluation of the final disposal concept - are met; and
- Generate technical data for the following:
  - identification of necessary and sufficient (adapted and cost-efficient) waste qualification requirements;
  - identification of accuracy requirements (in particular for the regulatory authority);
  - waste categorization, identification of related regulatory requirements (demand, selection process);
  - necessary quality assurance and related activities;
  - contribution to the selection of optimum waste forms (in terms of final disposal); and
  - demonstration of conformity of the relevant waste forms with specifications.

19.3.3.3 Radioactive waste inventory

The most important requirement of a radioactive waste qualification strategy is an inventory updated on an on-going basis. The main reason underlying this is that with environmental and radiation protection criteria becoming more and more stringent the management of radioactive wastes also took on a different form, and at present there are strict expectations in each of its operations, in fact, the principle of radiation protection must be enforced in every phase of waste management. This rule applies not only to employees but also to members of the population (including the future generation) and the environment, so the management of radioactive wastes is inconceivable without radiation protection planning and control.

Thus in nuclear power plants – apart from waste collection and the operation of waste management technologies – the key task is to confirm the later disposability of radioactive wastes.

In view of this, the following must be known with the highest accuracy:

- The quantity, physical, chemical form and activity of each radioactive waste type in the nuclear power plant,
- The rate of generation of different waste types, and
- The quantity of waste to be disposed in the future and its activity composition.

According to the description of the preceding chapter, together with the waste management-related strategic goals of the power plant these information are indispensable to define the risks related to the full nuclear power plant lifecycle.

19.3.4 WASTE MANAGEMENT TECHNOLOGIES

19.3.4.1 Expectations imposed on processing technologies

Radioactive waste processing technologies in the nuclear power plant must meet the following basic requirements:

- The generated radioactive waste should not restrict the operation of the nuclear power plant at any time during its full operating period.
- The wastes generated in the area of the nuclear power plant should be stored temporarily in the smallest optimum quantity and volume achievable, made available for recovery.
- The required investments should be minimized, without any superfluous storage capacity constructed.
- The technologies should be adaptable to the existing systems, ensuring optimum utilization and cost-efficiency.
- Only a small amount of secondary wastes should be produced as a result of processing.
- The waste packages obtained by processing and conditioning should meet all criteria imposed by final disposal, too (packaging, composition, physical, chemical and biological characteristics).

19.3.4.2 Central radioactive waste processing facilities

There are still relatively few central radioactive waste processing facilities world-wide. According to document [19-23], in Europe there is only one such central facility in Bohunice, Slovakia, whereas considering all nuclear power plant units in the United States (104 operating, 9 being dismantled) there are more than 15 central or regional waste management and conditioning facilities available. The Slovak waste processing unit was constructed on the site of the nuclear power plant units, and also processes waste delivered there. The exceedingly high investment costs only produce positive return if considerable waste quantities are processed. The expert committee preparing the technical documentation concluded that a central facility was not viable unless there are at least 10 operating units, e.g. in Russia or the Ukraine. The option of a cross-border regional facility was also raised for East and Central Europe (Czech Republic, Hungary, Slovakia), but as it is the current legal and economic circumstances exclude this possibility.

Considering practical issues, joint waste management for the Paks NPP and Paks II appears to be a non-viable solution too, as:

- The primary loop systems of VVER 440 and the planned units manifest major differences, resulting in different waste streams, both in terms of waste type and composition and the rate of generation.
- The operating periods of the two nuclear power plants will have a relatively short overlap only, and waste management technologies should be essentially adapted to them.

Due to these, it is justified to establish a stand-alone radioactive waste management technology for the new units.

It goes without saying that the later shared processing of minor quantities of specific wastes cannot be excluded, but in the design phase management capacities should better be adjusted to the waste streams of the units to build.

19.3.5 PHASES OF RADIOACTIVE WASTE MANAGEMENT

The management of radioactive wastes generated during nuclear power plant operation depends on their types, which in turn are determined by their physical, chemical and radiological properties. The main phases of this treatment are shown in *missing reference* based on [19-22].

After processing, a part of wastes becomes subject to clearance, or can be sent for reuse. Other classified radioactive wastes find different processing options in order to establish the desired storage and disposal conditions.

The pre-processing of radioactive wastes starts readily after their generation, and it sets out the limits of subsequent processing, too. Some wastes are qualified as inactive, while others are separated to allow processing and storage.

It the aim of temporary storage is to allow radionuclides to decay to make later processing simpler and safer, then the time span involved can be fairly long.

The processing of radioactive wastes involves the alteration of waste properties to enhance safety and increase profitability. Methods are often combined to improve the efficiency of processing.

Conditioning radioactive wastes involves operations that convert processed waste to some stable form manifesting chemical, thermal and radiological resistance, fit for transportation, storage and disposal.

The process of solidifying waste, embedding it into some matrix and incorporating it into an airtight coat is called immobilization. Immobilized radioactive waste can be placed into conventional 200 dm^3 steel barrels or thick walled containers of complex structure. The selection of the matrix material (actually the solidification technology) hinges on waste properties and characteristics, storage / disposal conditions, transportation circumstances, etc. The material binding the waste must have good isolating properties (anti-leaching resistance), show compatibility with waste components – ensuring high radioactive waste packing rate and small volume -, and demonstrate suitable strength. These can be organic (bitumen, polymers), inorganic (cement, glass, ceramics, glass ceramics), metallic and mixed materials. The most resistant items are homogeneous form, conditioned radioactive wastes, in which the components enter the structure at molecular level, e.g. in glass and ceramics. Heterogeneous wastes, with particles mixed mechanically, are less resistant.

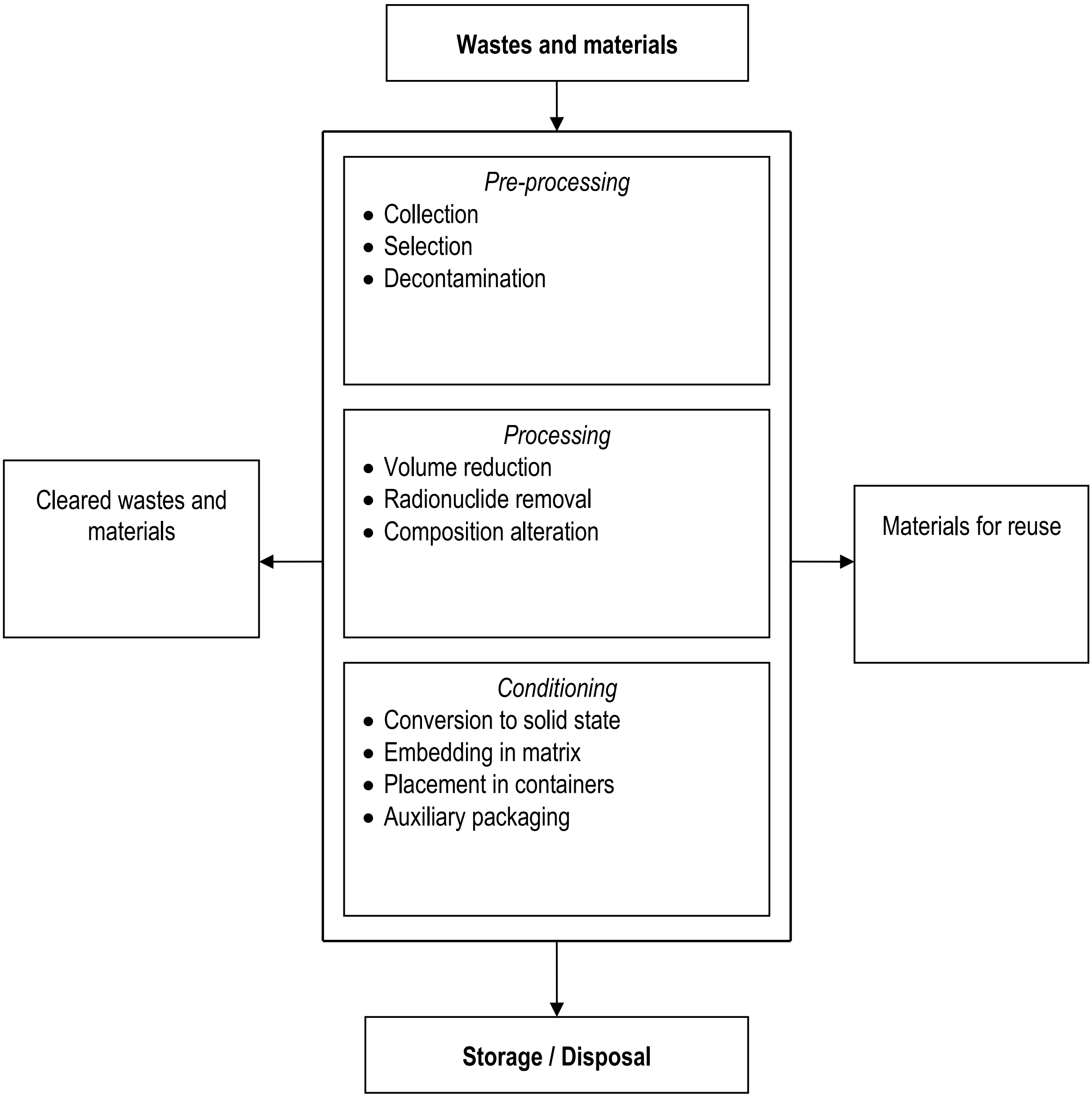


Figure 19.3.5-1: The main phases of radioactive waste management. [19-22]

The current approach applied to solidify low and medium activity liquid radioactive wastes uses most frequently cement and bitumen, glass being used for high activity items. Recently, the method of vitrification appears to spread to low and intermediate activity wastes, too. Processing and conditioning are often done in close connection. Waste storage and transportation may occur between phases or even within a single phase.

The last phase of processing is final disposal, which translates into long-term, safe and monitored waste storage (repository). The currently accepted approach to safe storage is the creation of multi-barrier isolation.

19.3.5.1 Pre-treatment

Both liquid and solid wastes require pre-treatment prior to processing. Pre-treatment comprises the following main elements:

- Separation of wastes into active and inactive (clearable) waste streams,
- Shaping the components found in the separated active waste stream into a form allowing maximum ease of treatment, conditioning and packaging for storage,
- Reusing all recoverable products.

Pre-treatment requires the selective grouping of waste according to the main points listed below:

- Radiation protection standards and objectives,
- Waste minimization,
- Applicability of pre-treatment technologies,
- Environmental protection considerations,
- Post-treatment waste management, conditioning, storage, transportation and final disposal requirements.

Table 19.3.5-1.recapitulates the main pre-treatment technologies, their features, and the factors limiting their application.

Pre-treatment mode	Features	Limitations
Collection and selection	Precedes packaging and waste qualification Separates different waste types with different physical and chemical properties Minimizes radioactive waste quantity Eventual waste processing and utilization only effective subject to separation	Extra personnel dose limits the activity and quantity of waste to be pre-treated. Higher dose rate wastes require special technologies (selection boxes, vaults, manipulators etc.) The feasibility and time constraint of sampling and continuous control may hit technical and work organization barriers
Chemical treatment	Prepares liquid wastes for conditioning, embedding, by pre-setting their chemical properties Decomposition of organic and corrosive materials, pH value setting etc.	Technological complexity, not always profitable due to high investment and operating costs Chemicals required Generated secondary wastes (precipitated salts, sludges etc.) require treatment
Physical treatment of liquids	Settling, filtering, centrifugal action etc. Separates sludges, suspended (dispersed or colloidal) particles, organic matter etc. from liquid waste	Manageability of generated secondary waste Separated phases produce new waste streams that must suit the existing technologies Not directly applicable to dissolved materials (e.g. only after chemical treatment) Separability of dispersed and colloidal particles limited as a function of technology
Physical treatment of solid wastes	Reduces dimensions, by chopping Adjusts waste size to container dimensions Improves space filling, in practice reduced overall volume translates to diminished waste quantity Containers reduce transportation hazards	Extra personnel dose requires pre-treatment of high activity wastes (e.g. decontamination) Not applicable for small quantities (e.g. a few pieces do not justify the installation of a chopping technology)

Pre-treatment mode	Features	Limitations
Packaging	Blocks the spreading of contamination (surface contaminations cannot escape) Avoids decontamination if coupled with proper packaging Prepares for transportation, treatment, conditioning, final packaging, or final packaging for temporary and/or final storage	Extra personnel dose Availability of packaging suitable for physical and chemical waste properties (e.g. hard wastes can tear apart bags, organic or corrosive materials degrade packaging fabrics) Issues above justify more expensive packaging (e.g steel barrel) or waste conditioning (e.g. embedding)
Decontamination	Reduces waste volume by enabling reuse Prevents spreading of non-fixed surface contamination (environment polluted by radioisotopes, radionuclides emitted into the atmosphere) Reduces external dose by removing surface activity Reduces internal radiation exposure hazard	Manual decontamination (wiping) not always efficient Industrial methods (immersion, US, electrochemical procedures etc.) require complex technologies Management of generated secondary radioactive wastes (aggressive chemicals, toxic solvents etc.) requires solution Conditioned quantity of secondary wastes exceeds that of treated materials. Thus a cost-benefit analysis is needed: Waste quantity reduced? Collective dose reduced? Simpler technology safer? More economic?
Waste classification	Allows the definition of isotope inventory, enabling planning of future processing and disposal Isotope composition investigated as function of diverse properties can be used to separate radioactive wastes for differing processing and disposal (e.g half life, activity, dangerous character index, radiation energy etc.)	Required instruments (e.g. segmented gamma scanner) and analytical methods (e.g. alpha spectroscopy) very expensive Certain isotopes first require lengthy preparation methods then long measuring times to determine. Buffer storage may be required as early as in selection Particular waste types and certain isotopes may require unique sampling and measurement methods developed. Radioactivity content may vary during later treatments (e.g. compaction increases activity concentration, filtering may reduce/increase it in some phase or another etc.) Maintaining the isotope inventory requires IT background (due to radioactive decay, daughter element build-up, mixing with other fluids, physical-chemical treatment etc)

Table 19.3.5-1: Features and limitations of the main pre-treatment technologies. [19-30], [19-19]

Of the pre-treatment modes described in the table above, the following will be applied in the nuclear power plant to be constructed:

- Collection and separation based on waste minimization and manageability,
- Determination of isotope selective activity content in view of clearance,
- Chemical treatment to increase the volume reduction of leachate waters,
- Chopping bulky solid waste pieces,
- Packaging (collection in different empties & returnables to avoid the spreading of contamination),
- Decontamination of low contamination metal wastes in view of clearance, rinsing PPE and washing clothing,
- Leachate water settling and filtering.

19.3.5.2 Treatment of solid radioactive wastes

The treatment of radioactive solid wastes can be split into two main activities: waste processing to reduce volume, and direct conditioning by placement in containers.

19.3.5.2.1 Pre-processing solid radioactive wastes

Pre-treatment activities promote and enhance the effectiveness of radioactive waste management. The steps involved in pre-processing the generated waste are: collection, selection and temporary storage.

Selection of radioactive solid wastes

Selection is determined by the method of further treatment / management and waste acceptance criteria.

Subsequent to radiochemical selection, flammable – non-flammable and compressible – non-compressible grouping follows, as a function of the planned technology to be applied: incineration or compaction.

Radioactive wastes requiring chopping (mostly bulky or oversized contaminated equipment) form a separate group, to be embedded directly into some matrix after chopping.

Metals and castings potentially eligible for reuse after decontamination are collected separately.

Selective collection into suitable containers accelerates the subsequent selection process.

Figure 19.3.5-2.: Waste collection area in the Paks NPP.

Temporary storage

Temporary (interim) storage is usually performed on the site of the nuclear facility. One of its objectives is to wait until short half-life radionuclides decay. On the other hand, in the case of management during campaigns the necessary waste quantities must be piled up between two successive campaigns. Here temporary storage should be minimized, as conditions may prove to be more favorable at the place of processing.

Figure 19.3.5-3: Temporary storage of solid radioactive wastes. [19-3]

19.3.5.2.2 Mechanical processing technologies of solid radioactive wastes

Cutting (Chopping)

Contaminated large, bulky equipment and machinery, construction industry units made of a number of materials (e.g. concrete) are unwieldy, cumbersome to handle, thus such waste must be cut into smaller bits and pieces.

International practices list the following practical methods for chopping: plasma arc - ”arc cutting” method, diamond strap conventional cutting method etc. For concrete / RI concrete elements, the application of drilling, splitting, flame or heat cutting, diamond strap cutting, and high pressure water jet methods are added to the previous list.

Conventional processes do not result in difficult-to-process secondary wastes. These operations involve knives, saws, hydraulic cutters etc. made of high strength steels and alloyed materials. These can be used underwater, too.

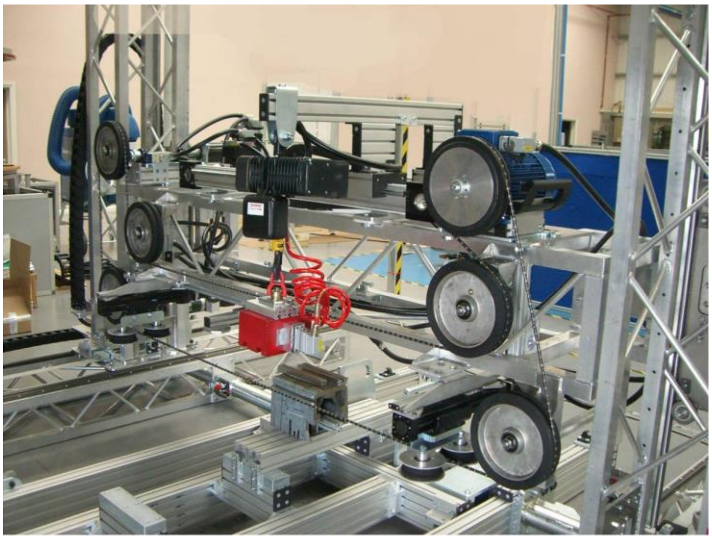


Figure 19.3.5-4.: Dimanod rope cutting apparatus. [19-4]

Compaction (pressing)

A widely used technology, the simplest and economic method of reducing the volume of solid wastes. Pressing is always preceded by waste preparation, as sizeable metal elements, concrete chunks and other non-compactable materials may damage or ruin the pressing machine. Residual liquid wastes in the volume to be processed may cause equipment contamination; nevertheless, the machine must be equipped with a liquid collector tank. The entry of toxic, chemically aggressive, explosive and flammable materials into the materials to be pressed must be excluded; however, even so radioactive gaseous aerosols may escape from the waste due to pressing, hence the need to fit the machines with gas filters. The composition of waste must be checked prior to pressing.

Presses are either hydraulically or pneumatically operated, but threaded presses can be used as well, these can exert compressive forces in 5-500 t range depending on design and layout. Compressed waste can take the form of bricks, packs or bundles.

In respect of some materials pressing is not efficient, as these can restore their original form (e.g. foils, rubber etc.). These can only be pressed into steel barrels or together with other wastes that resist expansion (resilience). Depending on material characteristics, the volume reduction factor achieved here falls between 2-5. The values depend on the prior selection and separation of wastes. Due to mixed material composition, it is rather difficult to define the required compressive force and the degree of compaction.

Due to compaction – a larger quantity of radioactive waste is present in a given volume – the specific activity of the radioactive waste package can increase, requiring the use of radiation protection devices in extreme cases.

The most widespread method is pressing into standard barrels. The standard barrels used are of 200 dm^3 size. Depending on storage and final disposal conditions, barrels are put in containers and are cemented. Due to its good efficiency, the reliability of the machinery used, and its availability on the market, the pressing method is quite frequently used.

Figure 19.3.5-5: 50 kN automatic waste compaction press. [19-5]

Supercompaction

Super- (or high pressure) compaction uses 1,000-1,500 t compressive force. Thus, in addition to regular materials (paper, fabric, etc.) this method is also applicable to metals, concrete waste, glassware, wood, sand etc. The technology generally achieves 10-12-fold volume reduction. Eliminating all gaps, cracks etc. present in radioactive waste requires considerable pressure, therefore 70-90% compaction is an acceptable compromise. Pressed waste is usually cemented in the end.

Supercompactors are expensive machines, so their use requires an economic rationale and a financial case. They can be used best where a large amount of radioactive solid waste is generated.

Figure 19.3.5-6: Supercompaction machine and pellet storing unit. [19-6]

Figure 19.3.5-7: Low activity solid waste before and after supercompaction. [19-7]

19.3.5.2.3 Thermal processing technologies of solid radioactive wastes

The thermal processing of flammable solid wastes (amounting to 70% of all solid wastes) is encouraged by both economic and ecological arguments. As a result of such processing, waste volumes can be reduced considerably, which in turn cuts storage and disposal costs and at the same time underpins safety.

The most efficient process known is incineration, reducing waste volume by a factor of 50-100. The procedures involved are based on physical and chemical processes taking place at high (elevated) temperatures.

There are several familiar thermal (caloric) procedures, including – but not limited to, the following:

- Burning by adding excess air,
- Oxidation in wet air,
- Incineration in melted salt,
- Incineration in molten glass,
- Burning by plasma arc,
- Cyclone firing,
- Pyrolitic decomposition,
- Plasma pyrolisis,
- Metal meltdown.

The investment and operating costs engendered by thermal radioactive waste management technologies are high. Given that the quantity of incinerable wastes generated in the units to be constructed will not attain the threshold amount justifying the profitable operation of thermal technologies (e.g. 5,000 m^3/yr for a conventional incinerator unit), these will not be considered for application.

Figure 19.3.5-8: Plasma incinerator equipment. [19-8]

Figure 19.3.5-9: Melting down metals. [19-9]

19.3.5.3 Treatment of liquid radioactive wastes

To select the radioactive liquid waste management system, the following points will be considered:

- Separation of generated wastes by property,
- Discharge (release) requirements of treated liquids,
- Potential technologies and associated costs,
- Conditioning condensates issued from treatment,
- Storage and final disposal of conditioned condensates,
- Quantity of secondary wastes.

19.3.5.3.1 Treatment of water-based liquid radioactive wastes

The available technologies of water-based radioactive liquid waste management are listed below (their key features and the factors limiting their application are summarized and presented in table 19.3.5-2.

Precipitation generation

Chemical precipitation generation is a well-established theory to extract radioactive contaminants. The aim of chemical precipitation generation is to remove radioactive nuclides from liquid state waste in tiny, insoluble particulate form. The principal chemical precipitation-inducing materials are metal hydroxides, used to extract radionuclides from neutral or alkaline (basic) media. In this process, a number of radionuclide are hydratised, while there are others that are absorbed on the flakes so formed (co-precipitation). It is possible to add different adsorbent agents to liquid waste that are capable of adsorbing additional radionuclides from the waste.

Chemical precipitation comprises the following steps:

- Adding coagulant and / or adjustment,
- Flocculation,
- Settling,
- Solid – liquid separation.

Ion exchange

Ion exchange is a water treatment technology known and used for many years, one that used to be well-developed even prior to its first application in nuclear power plants. Ion exchange technologies are incorporated into various nuclear power plant techniques.

Organic ion exchange resins

Organic ion exchange resins are custom-developed materials that find anion and cation exchange uses in a wide range of applications. Ion exchangers can be put to use to treat highly and weakly acidic (corrosive), as well as alkaline (caustic) materials. By carefully selecting the function groups in the resins sometimes significant ion selectivity becomes possible. These solutions are widely used to treat the water of spent fuel storage pools.

Organic ion exchangers are prone to decomposition under thermal and irradiation effects, plus certain chemical treatments may also induce resin degradation, which in turn implies the wash-out of the already bound activity.

Figure 19.3.5-10: Inorganic ion exchange resin. [19-10]

Inorganic ion exchange resins

Inorganic sorbents are widely used in the treatment of water-based radioactive wastes. These substances are generally reasonably resistant to radiation and chemical action, and tend to be compatible with potential embedding matrix constituents. Inorganic ion exchangers can be natural minerals or synthetic materials.

Minerals found in nature include, among others, alumino-silicates, like

- zeolites,
- clays, and
- feldspars.

As a rule, synthetic ion exchangers can be put into one of the following categories:

- hydrated metal oxides, e.g. hydrated titanium oxide, polyantimonite acid,
- insoluble salts of multivalent metals, e.g. titanium phosphate,
- insoluble salts of variable polarity acids, e.g. ammonium molybdenum phosphate,
- complex salts in insoluble ferricyanides,
- synthetic zeolites.

Evaporation (Boiling, Concentration)

Evaporation is widely used as a waste treatment procedure due to its favorable decontamination and volume reduction factors. The aqueous phase exits the waste as steam, which can be reused after repeated filtering and ion exchange, or can be released subsequent to proper control. Evaporation residues can be cemented, or embedded in some other material for long-term storage, or evacuated and delivered for final disposal.

Figure 19.3.5-11: Evaporation equipment. [19-11]

Membrane technology

With the aid of membrane technology, the selective separation of certain radioactive components present in waste water becomes possible. Pressure drives materials to pass through the membrane. The following membrane technologies are known:

- Reverse osmosis (RO),
- Ultrafilering (UF),
- Microfiltering (MF),

depending on membrane pore size.

The following pore sizes are established:

- RO < 1 nm,
- UF 1-100 nm,
- MF 0.1-1 µm.

Membrane technology is used to treat the mixture of nuclear power plant laundry waste waters and laboratory waste waters, and to treat boric acid solutions in view of reuse.

Figure 19.3.5-12: Reverse osmosis (RO) equipment. [19-12]

Electrochemical technologies

This technology can be applied well mainly to treat secondary wastes.

The following technologies are used:

- Electrochemical or electroflotation technology,
- Electro-osmotic drainage,

Used mainly to drain water from low conductivity muds and sludges. The solid matter content is 2-5 wt%,

- Electrodialysis,
- Electrochemical ion exchange.

Table 19-3.5-2. presents a summary of water-based radioactive liquid waste treatment technologies and their limitations.

Treatment technology	**Features**	**Limitations**
Chemical precipitation generation and separation	Optimum result for large quantities of high salt content wastes Reduces the radioactive contamination and chemical concentration of the solution Relatively simple industrial operation Low specific operating cost	Typically lower decontamination factor (DF) than for other technologies (10 <DF <10^2 (ß,γ) 10^3 (α)) After discharge chemical constituents may be deposited or accumulated in nature Efficiency depends on solid – liquid separation step Secondary waste is generated
Ion exchange with organic resin	Good DF for low salt content wastes > 10 to 10^3, on average 10^2 Good mechanical properties and relative ease of treatment Can be regenerated as necessary	Radiological and chemical effects may damage resin Some chemical pollutants may block sorption or wash the previously bound radionuclides (e.g. Na^+, K^+ ions the radioactive Cs^+ ions). Resin price is a decision factor It conditioning requires chemical treatment Secondary waste is generated
Ion exchange with inorganic sorbents	Better chemical, radiological and thermal stability than for organic ion exchangers Relatively simple final disposal Broad product range ensures high selectivity DF > 10 - > 10^4, average 10^2 - 10^3	Influenced by high salt content and complexes, but to a lesser degree than organic ion exchangers Filter bed clogging issues Prices may be elevated Often difficult regeneration and reuse Secondary waste is generated
Evaporation	Can achieve high decontamination: DF10^4 – 10^6 Well-established technology High volume reduction factor possible	Technology application limited by circumstances (deposition, foaming, corrosion, volatile isotopes) High operating costs High investment costs
Reverse osmosis, RO	Extraction of dissolved salts DF 100 - 1000 Volume reduction factor (VRF) 100 - 1000 Economic Accepted for high purity operation	High pressure system Expensive Not rewashabe Subject to clogging Secondary waste is generated
Ultrafilter, UF	Separates dispersed and colloidal particles in saline solutions Organic and inorganic membranes are used. Inorganic ones have better chemical and radiological stability, and tolerate higher operating temperatures Operating pressure < 1MPa	Clogging may prompt chemical cleaning or rewash from time to time Organic membranes are subject to radiological decay Secondary waste is generated

Treatment technology	Features	Limitations
Microfilter, MF	Low operating pressure (100 – 150 kPa) High separation rate (99 %) Pre-treatment technology for RO Inorganic membranes can be procured Clogging can be removed by air	Rewash frequency may become too high, depends on solid matter content of waste stream Secondary waste is generated
Electrochemistry	Supplementary technological parameter Helps to minimize reagent use Low energy consumption Increases reaction efficiency	Sensitive to contaminants present in the waste Waste ionic strength affects performance Risk of gas generation Clogging issues above 10 g/l solid matter content

Table 19.3.5-2: Water-based radioactive liquid waste treatment technologies. [19-30]

Of the technologies shown in table 19.3.5-2., due to their simplicity, high decontamination factor and good volume reduction factor the following ones will be used:

- Ion exchange with organic and inorganic resin,
- evaporation.

19.3.5.3.2 Liquid organic wastes

Within the range of radioactive nuclear power plant liquid wastes, the quantity of organic liquid waste is far smaller that that of water-based radioactive liquid waste. However, no matter how small the quantity, incorrect treatment may increase its hazard potential. Water-based wastes can be discharged to the environment between certain limits subsequent to proper treatment, whereas radioactive organic wastes may pose a risk to both health and the environment not only due to their nuclide content but also due to other organic matter present in them. The time-honored practice of thinning and dissolution well-suited to water-based wastes usually fails for most liquid organic wastes.

In general, the following treatments are used for liquid organic wastes:

- Conversion into solid form,
- Quantity reduction,
- Cleaning and reuse,
- Conversion into some form compatible with cementing.

The simplest local treatment of liquid organic wastes is the so-called absorption procedure. Here the liquid waste is added to the absorbent placed in the transport container, and the agent eventually absorbs every liquid. This technology is applied routinely for the solidification of radioactively contaminated turbine and pump oils.

Usually the following absorbents are used:

- Natural fibers (cotton, sawdust),
- Synthetic fibers (polypropylene),
- Vermiculite (mica),
- Clay,
- Kieselguhr (rock meal).

The main treatment technologies are compared in table 19.3.5-3. below.

Treatment technology	Features	Limitations
Incineration	Organic matter contained in waste decomposes Considerable quantity reduction Can be used mixed with other waste too Eliminates biological activity (infection hazard)	Treatment of secondary wastes Complete decomposition requires high temperature Exiting flue gases require scrubbing and control before being emitted to the atmosphere
Emulsification	Enables embedding liquid organic waste into a cement matrix	Low liquid-cement factor (restriction imposed by emulsified liquid content threshold value in cement matrix)
Absorption	Binding organic liquids on solid adsorber Simple and cheap	Due to sorption characteristics (solid phase can correctly fix a little liquid only) ideal for relatively small waste volume only The properties of such waste not always meet the acceptance criteria of the repository
Phase separation (e.g. extraction)	Removal of liquids and toxic substances from the waste Solvents can be cleaned	Special technology, rarely applicable Relatively expensive technology for industrial size waste quantities
Wet oxidation	Low temperature Simpler than incineration Suitable for biological wastes	Oxidation agent storage hazardous Complicated processing technology Treated materials require post-treatment care

Table 19.3.5-3: Organic radioactive liquid waste treatment technologies. [19-30]

Of the technologies shown in table 19.3.5-3., only absorption will be used, due to presumably insignificant radioactive liquid waste generation and ease of implementation.

19.3.5.4 Waste conditioning technologies

19.3.5.4.1 Radioactive waste solidification methods

Conditioning, a post-processing activity, is an important part of radioactive waste treatment and management, as it reduces the escape potential of radionuclides during subsequent treatment, storage, transportation and disposal.

There are four solidification methods employed in practice:

- cementing,
- bituminization,
- polymer embedding, and
- vitrification.

The solidification of liquid wastes is necessary because in liquid state safe storage is only possible for several decades. The method of solidification depends on several factors such as: physical-chemical waste properties, radionuclide composition, deep final repository acceptance criteria, requirements imposed on end product, availability of technology and materials, binding material compatibility with waste components, etc.

At present, the most widespread waste treatment method applied to low and intermediate radioactive waste is cementing and bituminization. Vitrification and geopolymerization tend to gain acceptance as alternatives to cementing.

Solidified wastes must meet the following requirements:

- high chemical, thermal and radiation resistance,
- homogeneous structure,
- high mechanical strength.

The conditioning techniques widely used in nuclear industry practices are listed below:

Cementing

Cementing is one of the most widely used conditioning and solidification method applied to low and intermediate activity radioactive wastes, due to the simplicity and availability of the technology, the inexpensive equipment used, and finally the non-combustible and non-plasticity of the end product. Liquid wastes make use of the ability of cement to bind water, while the same cement can be used as a supplementary protective barrier for solid wastes.

The defining parameters of cements are: refining grade, solidification / setting time, volume reduction degree, heat generation on water contact, and the strength parameters of set cement. The most often used type is portland cement, due to its manifest high strength.

Cements prepared to different recipes possess varying physical and chemical properties, and the latter have an impact on waste solidification, therefore composition is matched to the type of waste to be solidified.

Types of liquid waste suitable for cementing:

- evaporation residues,
- concentrates of radioactive liquid waste processing,
- decontamination solutions,
- regenerating solutions, etc.

Types of solid waste suitable for cementing:

- ion exchange resins,
- sorbents,
- ashes produced by incineration,
- different pre-treated equipment.

The properties most intimately determining the quality of cemented wastes are the following:

- mechanical strength,
- chemical durability,
- resistance to the storage environment.

The technology is well-developed and does not require complex machinery, however, the reasonable choice of process components is important.

Borates are considered as cement poisons, as they have an adverse effect on the end product properties (in extremis cement does not even set), so only a little borate-containing waste water can be mixed to the cement.

The drawback of cementing is the increase of waste volume.

Geopolymerization

This technology can be regarded as a special type of cementing. While cement is formed by calcium silicate, geopolymers are amorphous aluminum silicates with inorganic polymer structure.

The great advantage of this technology is that the borate content of the boric acid solutions constituting a significant part of radioactive liquid wastes is incorporated into the crystal lattice, therefore the attainable waste water – binding material rate is much more favorable than that for cementing.

Figure 19.3.5-13: Cemented waste. [19-31]

Bituminization of radioactive wastes

One of the methods used to process liquid and wet radioactive wastes is embedding them into bituminous materials. The reason why bituminization became so widespread is its associated thermoplasticity and waterproofing ability, together producing a stable, homogeneous and watertight end product. The base material (feedstock) is quite cheap and can be found everywhere, making it easily accessible. Given that water is evaporated during the process, unlike cementing, this type of processing does not involve volume incerase. The complexity and associated expenses, the reduction of primary waste volume and the migration of radionuclides into the end product place this technology somewhere between cementing and vitrification.

Vitrifying radioactive wastes

This method is primarily used for processing high activity radioactive wastes due to its cost.

Technological advantages include: sufficiently high corrosion resistance ability in aqueous environment, high thermal and radiation resistance, good mechanical durability, low sensitivity to waste composition, maximum volume reduction, high degree of technological development and reliability.

Technological drawbacks include: actual glass composition is basically precisely determined by the waste mix. When defining this composition, an effort must be made in all cases to avoid uneven material distribution. It is important to avoid phase separation in glazed waste, as phase properties are different and sensitive to structure. The formation of the end product in terms of different phases leads to the degradation of strength indicators.

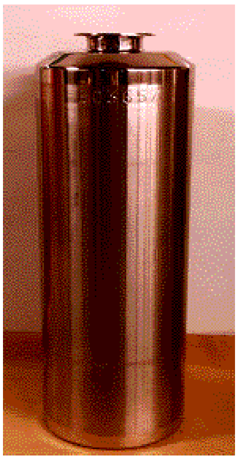


Figure 19.3.5-14: Container holding vitrified waste. [19-32]

Table 19.3.5-4 below summarizes the general features and limitations of solidification technologies.

Matrix	**Benefits**	**Drawbacks**
Cement	Readily available, inexpensive material Compatible with a broad range of materials Excellent radiation stability Inflammable High pH value product, shows good chemical retention for most radionuclides	Swelling organic resin coats may induce matrix cracking Low rate of admixable radioactive waste and cement End product volume exceeds that of original waste Moderate anti-leaching resistance for some radionuclides (e.g. Cs, K etc.)
Geopolymer	Relatively cheap Inflammable Stable end product Volume increase less than for cement	Works well for solutions containing boric acid only Relatively recent method, few lessons learned about its aging
Bitumen	Good anti-leaching resistance All water removed, resulting in favorable waste-bitumen ratio	End product softens as temperature is raised Structural mechanical stability requires container use Organic resin may swell if water enters the matrix Organic waste form flammable, subject to biodegradation Lower radiation stability than cement Expensive and dangerous technology
Polymers	Relatively many polymers can be used Good anti-leaching resistance	Usually more expensive than cement or bitumen Waste trace elements influence polymerization reactions Lower radiation stability than cement
High integrity container	Simple, cheap, easy to use Steel containers have excellent radiation resistance	Only container integrity blocks escape Does not comply with regulatory environment in several countries Radiation resistance of plastic containers limited
Vitrification	Vitrified form has good anti-leaching resistance Good radiation tolerance Good volume reduction can be achieved	High temperature process High investment and operating costs Alkaline ions (K, Cs etc.) may leach out

Table 19.3.5-4: Comparative summary of radioactive waste solidification (immobilisation) processes.

Of the technologies indicated in table 19.3.5-4., taking into account the physical and chemical properties of radioactive liquid wastes, plus the advantages detailed above, cementing will be used. The cementing procedure is a generally accepted and applied method in the nuclear industry.

19.3.5.5 Radioactive waste treatment options

In our day many highly efficient treatment and conditioning technologies are available, which, if used, can significantly improve the utilization of the existing storage capacity, dispenses with the necessity of re-packaging subsequent to temporary storage, and minimizes or reduces seepage from the package during the storage period.

The applicability of different technologies in respect of certain particular radioactive waste types are recapitulated in 19.3.5-5.

During establishment, the task and responsibility of far-sighted design & planning, coupled with adequate control is to ensure the application of sophisticated conditioning techniques as early as during the temporary storage period, as the optimum approach to waste management includes the consideration of profitability and like financial issues. The economic priority of suitable waste treatment technologies is very often low, especially when sufficient temporary or final storage capacity is available for the disposal of the generated waste even subject to using low efficiency volume reduction technologies. In those countries where the temporary on-site storage of waste is tolerated up to dismantling, or until the final waste disposal facility is implemented, this opportunity has a negative effect on the use of volume reducing and conditioning technologies. In fact, this *status quo* encourages the operator to store waste as it is generated, awaiting the advent of future, more efficient technologies. Thus the formulation of an adequate waste management strategy must start already in the design & planning phase.

Technology	**Radioactive waste types**							
	Rubber / plastic	**Cellulose**	**Ion exchange resins**	**Biological substances**	**Mixed solid substances**	**Lubricants**	**Organic solvents**	**Other liquids**
Non-destructive technologies								
Drying and evaporation	N	K	K	K	K		K	K
Distillation	N	N	N	N	N		K	
Physical conditioning			N	N	K	N	N	N
Decontamination	K	K	N	N	K	K	K	
Absorption	N	N	N	N	N	K	K	K
Compaction	K	K	K		K	N	N	N
Direct immobilization	K	K	K		K	N	N	N
Destructive technologies								
Incineration	K	K	K	K	K	K	K	K
Pyrolysis / steam conversion	K	K	K	K	K	K	K	K
Alkali-hydrolysis (TBP / OK)	N	N	N	N	N		K	
Vitrification	K	K	K	K	K			
Plasma treatment	K	K	K	K	K	K	K	K
Melted salt oxidation	K	K	K	N	K		K	
Electrochemical treatment	N	N	N	N	N		K	K
Direct chemical oxidation							K	
Acidic decomposition	K	K		K	K		K	
Wet oxidation			K			K	K	K
Supercritical aqueous oxidation	K	K	K		K		K	
Thermochemical treatment			K	K		N	N	N
Microwave treatment			K	K				K

Legend:
K – Known, or probably applicable technology,
N – Not applicable technology,
Blank field – Not known, or possible.

Table 19.3.5-5: Possible applications of radioactive waste management technologies. [19-25]

19.3.6 FINAL DISPOSAL

The final disposal of low and intermediate activity wastes can take place in surface, near surface and subsurface storage facilities.

Figure 19.3.6-1. shows an example of surface storage. In this case, the repository was constructed at the surface, then after reaching its capacity it will be covered by multiple layers and the area will be recultivated. Repositories constructed down to ca. 30 m below ground level are called near surface facilities, an example is given in Figure 19.3.6-1. Subsurface storage facilities are constructed from some 10 to some 100 m depth underground, one example in Hungary being NRHT.

In many countries, it is an accepted strategy that the health of the entire society can be protected against the risk and hazard represented by short lifespan, low and intermediate activity wastes by surface repositories alone. However, if such wastes are disposed of in carefully selected sites, in technically suitably designed subsurface repositories, then extra, additional protection can be ensured to counter dangers of both human and natural origin. Subsurface repositories are designed and constructed to guarantee the required protection for a long time even without supervision, in contrast to near surface facilities where usually prolonged – stretching several centuries – future institutional control is assumed. Consequently, a technically well-constructed subsurface repository offers a higher degree of protection and safety for society than a similar surface installation. International experience reveals that the construction of subsurface repositories for the disposal of low and intermediate activity wastes is by now a ripe technology.

Over the past few years, it became more and more accepted in international practice that the disposal of very low level activity wastes does not require such efficient shut-off and isolation as that of low and intermediate activity wastes, therefore surface repositories may be suitable for their disposal provided limited authority monitoring is given. Currently known foreign cases prove that VLLW repositories tend(ed) to be established in the proximity of an already operating surface low and intermediate activity waste repository, either in its protection zone or directly in the operating area of the nuclear power plant. Studying international examples it can also be stated that these storage installations bear close resemblance with conventional hazardous waste landfills.

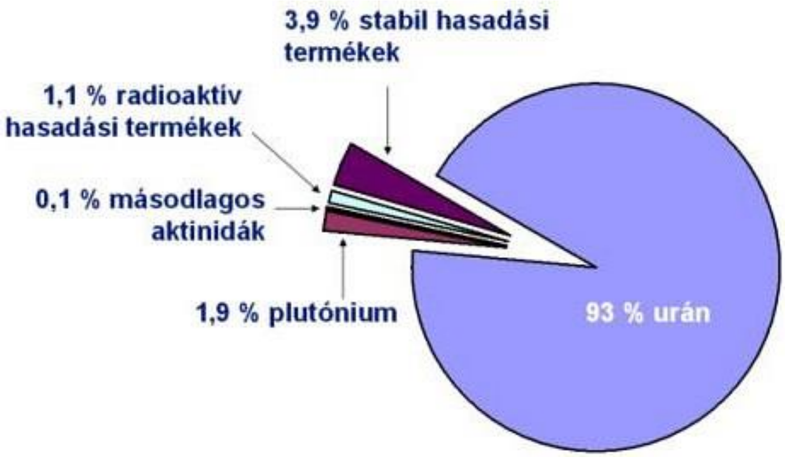
Figure 19.3.6-1: Surface low and intermediate activity radioactive waste landfill. [19-26]

Figure 19.3.6-2: VLLW landfill, Morvilliers, France. [19-27]

19.4 MANAGEMENT OF SPENT NUCLEAR FUEL

Taking into account [19-14], in accordance with Act No. CXVI of 1996 on Nuclear Energy Section 2 Paragraph 14: "*spent nuclear fuel:* nuclear fuel irradiated in a nuclear reactor and permanently removed from the reactor. *Spent nuclear fuel* **cannot be classified as waste** because of its ability of being reprocessed or when it is yet classified as waste its final repository has to be ensured".

The nuclear fuel being used in most of the currently operating nuclear power plants consists of uranium dioxide pellets. Nuclear fuel can be characterized by using the burnup. Burnup is a measure of how much energy is extracted from a primary nuclear fuel source containing unit of mass of uranium (or uranium and plutonium) during irradiation in the nuclear reactor. A typical composition of nuclear fuel with a burnup of 50 MWd / kgU can be seen in Figure 19.4-1.



Legend: 3,9% stabil hasadási termékek – 3.9% stable fission product; 1,1% radioaktív hasadási termékek – 1.1% radioactive fission product; 0,1% másodlagos aktinidák – 0.1% secondary actinide; 1,9% plutonium – 1.9% plutonium, 93% urán – 93% uranium

Figure 19.4-1: Typical composition of nuclear fuel having a burnup of 50 MWd / kgU.

It can be seen that the remaining uranium represents 93% of the mass of spent nuclear fuel in which the ^{235}U content is slightly higher than in natural uranium. The total mass of the ^{238}U isotope decreases during the irradiation. This decrease is primarily not due to fission but rather to neutron capture. The plutonium content of this spent fuel is 1.9% but the secondary actinide content is only 0.1%. The ratio of the fission products can be valued at 5% whereof 3% is the ratio of the stable isotopes. By way of the decay of the radioactive isotopes even more stable isotopes are produced so the ratio of the radioactive isotopes becomes lower and lower during storage.

In the process of radioactive decay, the decay energy is released and a significant amount of heat is produced in the spent nuclear fuel, therefore this heat should be removed during the storage.

In reference to the final disposal and the recycling of spent nuclear fuel, the important factors are the mass and the activity of the spent fuel, the heat produced by decay, and finally radiotoxicity (which is a measure of the biological damaging effects of spent fuel).

The **quantity** of spent nuclear fuel produced by a nuclear power plant is primarily determined by the power and the type of the operating nuclear reactor(s). In general, the higher power a nuclear reactor has, the more spent fuel is produced. The burnup (i.e. how much energy is extracted from a unit mass of nuclear fuel during the operation) also has an effect on the quantity of spent nuclear fuel. Using fuel with higher enrichment (depending on the technology) higher burnup can be reached because the nuclear fuel having higher enrichment contains more fissile isotopes.

The mass of spent nuclear fuel can be reduced by reprocessing, and the recovered fissile material can be recycled in fuel fabrication. In the recycling process of the spent nuclear fuel high activity waste is produced. The developers of the nuclear reactor types have sufficient fuel manufacturing and production capacity to operate that type of reactor. Some fuel suppliers take care of the reprocessing of the spent fuel returned to them after use, and they also manage the useless high activity waste produced by reprocessing.

The **activity** of spent nuclear fuel initially originates from the short-lived fission products and, after a couple of hundred years, the activity of the plutonium, uranium and the other actinides becomes dominant. Amongst fission products, the following isotopes have extraordinary importance: cesium (^{135}Cs – 2.3 million years, ^{137}Cs – 30 years), strontium (^{90}Sr – 29 years) and also technetium (^{99}Tc – 211 thousand years), iodine (^{129}I – 16 million years) and zirconium (^{93}Zr – 1.5 million years). The specific activity of nuclear fuel having an average burnup at the end of usage is 10^7 TBq/kg, while 600 years later it becomes 100 TBq/kg which is a reduction by a factor of one hundred thousand. The rate of decrease is highest in the first period, the activity of the off-loaded fuel becomes 1/10 during the first ten years of storage.

The quantity and composition of isotopes in spent nuclear fuel are influenced by burnup and the type of fuel. Fuel produced using reprocessed fissile material contains more plutonium at the beginning of re-use therefore such type of nuclear fuel contains more actinides than the UO_2 fuel with the same burnup. A unit of mass of fuel with higher burnup contains more fission products but less spent fuel, therefore the total amount of fission products remains almost the same.

The rate of the **heat production** in spent fuel diminishes with time just like activity. After five years of storage, the heat production rate in spent fuel becomes 1/10,000 of the rate during normal operation and 1/500 of the rate at the shutdown of the reactor.

The **radiotoxicity** of spent nuclear fuel measures what potential health damaging effect the radioactive isotopes of the spent fuel could have when they are incorporated in humans. Mathematically radiotoxicity is the weighted sum of the activity of radioactive materials in spent fuel. It can be calculated from the activity of the isotopes using the appropriate conversion factor (the dose conversion factor) for each isotope. The dose conversion factors of the actinides are much higher than those of the most important fission products. Most of the radiotoxicity of spent nuclear fuel originates from these isotopes already a few decades after the operation. The radiotoxicity of spent nuclear fuel is ten thousand times higher initially than the radiotoxicity of the natural uranium used for production. The radiotoxicity level of natural uranium is reached by the spent nuclear fuel after a few hundred thousand years.

There are significant differences in the way of managing spent nuclear fuel between the countries. The strategies are generally congruent; they agree to store spent fuel under water in a pool near the reactor for a few years, as the fuel needs intensive cooling because of its high heat production rate. In practice in Hungary and abroad as well, removed from this pool the fuel is placed in a storage facility for temporary storage that can be wet or dry depending on the chosen technology. During the decades of this storage the heat production rate of the fuel elements becomes low enough and the spent fuel can be placed in a permanent storage facility.

Concerning the management of spent nuclear fuel some countries have adopted the ”*Wait and see*” policy. According to this, the fuel elements are stored until the decision on their permanent disposal is made. This strategy was replaced by the *”Do and See”* policy in the last few years. This policy is briefly about to consider that every process of program could contain consecutive phases and between these phases there could be branching. At these branching points decisions have to be made concerning the program according to the appropriate deliberations. This policy implies both activity (progression) and prudence (deliberation) and therefore is better than the strategy of waiting which occasionally leads to passivity.

Concerning the management of spent nuclear fuel there is no unified practice. The main tendencies are the following:

- reprocessing of the fuel, re-using of the fissile material, permanent disposal of the high-level waste produced in the processes;
- reprocessing of the fuel abroad, returning of the fissile material and the high-level waste;
- transporting the spent fuel back to the country of manufacture;
- direct disposal of the spent fuel in an appropriate geological repository.

19.4.1 DECAY OF SPENT FUEL

The fuel elements removed from the reactor produce a lot of heat after usage, therefore the fuel assemblies are stored for years in a spent fuel decay pool situated at the reactor site. During this storage, the level of the activity and the decay heat of the short-lived isotopes significantly decrease. In case of longer campaigns the burnup reaches higher values and there are more fission products in the spent fuel and therefore longer storage time and thicker shielding are needed before any further handling and transport.

19.4.2 INTERIM STORAGE OF SPENT FUEL ASSEMBLIES

Before any further processing, the spent fuel removed from the decay pool is moved to a facility for temporary storage. The residual heat has still to be removed after the storage in the pool, but at that point the heat can be removed less intensively (using air of natural draft).

There are some options for the temporary storage of spent nuclear fuel:

- wet storage,
- dry storage,
- modular vault dry store (MVDS, for example the “ISFS+ Facility)
- dry cask storage.

19.4.2.1 Wet storage

During wet storage, the fuel assemblies are stored under the technical conditions similar to that in the temporary storage pool but the heat production rate of these assemblies is much lower than the original one, therefore residual heat removal is no longer crucial.

Benefits:

- lower space requirements (no case between the assemblies);
- connected to the temporary storage pool the assemblies can be continuously underwater during manipulation and transfer (no need for transport cask).

Drawbacks:

- need for being continuously monitored and treated, need for technology (quality and quantity of the cooling water);
- little domestic experience (experience only in working with a pool situated directly at the reactor site);
- uneven distribution of investment costs (not modular, the whole facility has be built up for the first assemblies);
- extra costs of the water treatment (laboratory);
- earthquake risk (the coolant can be lost when the pool becomes damaged);
- the implementation area of Paks II is not large enough for such a facility.

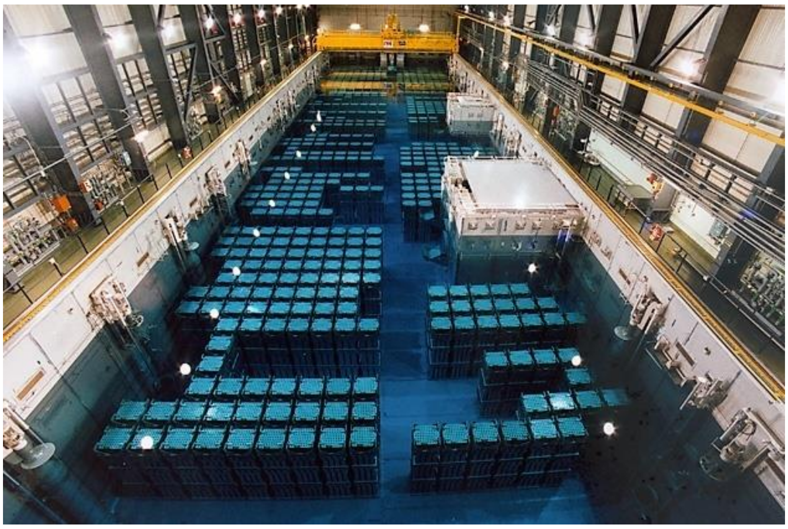


Figure 19.4.2-1: Example for wet storage, AREVA, LaHague, France. [19-15]

19.4.2.2 Dry storage, modular vault

The modular vault dry storage system is based on the Interim Spent Fuel Storage Facility (KKÁT) connected to the Paks NPP and is dimensioned according to the technical parameters of the VVER-1200 fuel assemblies.

Benefit:

- more economic operation on account of passive heat removal.

Drawbacks:

- need for being continuously monitored (as opposed to the dry cask storage, for example);
- uneven distribution of investment costs (can be built modularly, but the first module must be finished for the first assemblies);
- the extension of the Interim Spent Fuel Storage Facility being in operation cannot be arranged;
- there is no appropriate site on the planned implementation area.

Figure 19.4.2-2: MVDS type storage, ISFS Facility Paks. [19-33]

19.4.2.3 Dry cask storage

In the storage pool the fuel assemblies are placed underwater in a cask / sheath for long-term storing / transporting and further shielding is installed to the cask depending on whether the assemblies will be stored or transported.

The commercially available containers have to be chosen considering the dimensions (especially the length) of VVER-1200 fuel assemblies. The average burnup (MWd/kg U) of the fuel assemblies is also an important parameter related to biological shielding and heat removal.

Before placing the assemblies in the storage place, the sheath is shielded using various metal and composite layers. The outer layer that has an important role in shielding can be made of steel (corrosion resistant steel or carbon steel) or pre-stressed concrete. The containers can be placed upright or lying depending on which layout is preferred by the technology supplier.

The ready-to-place containers can be stored as follows:

- The containers having a metal outer case are placed in a storage hall in which there are no weather effects. The walls of the hall have moderate shielding compared to the MVDS system because each of the containers already has its own shielding.
- The surface layer of the containers having thicker concrete jacket is basically exposed concrete. These containers are placed on an appropriately dimensioned concrete case with lattice parameters specified by the safety technology. The containers having concrete jacket and placed on open surface have a shielding thick enough and therefore no additional shielding is needed. The multi-layer structure of the storage system makes it possible to install air ducts in the jacket of the containers to ensure the cooling by air or natural draft (keeping in mind biological protection).

Benefits:

- Only a little supervision is required (physical protection and monitoring).
- The investment costs have uniform distribution (in the case of open surface storage the use of reinforced concrete for making the plate is economically favorable. The acquisition of the containers for storing can be done parallel with the capacity reduction of the storage pool.
- With no lightweight storage hall there is no need for special preparations on the site.
- Minimum infrastructural needs (no factory siding track is required, the storage place can be access on road).
- Higher protection against natural forces and crime (10 assemblies can be placed in a single container. This means significant separation. In case of plane crash there is negligible probability of equal damage for every cask at the same time.)

Drawbacks:

- Need for place inside the controlled area of the power plant for the equipment used to fill and seal up the containers.
- When not a multifunctional container is chosen, additional operations have to be carried out to place the spent nuclear fuel elements in the transport cask before transport to the permanent storage facility.

Figure 19.4.2-3: Dry cask storage in a storage hall (CASTOR casks, Gorleben, Germany).

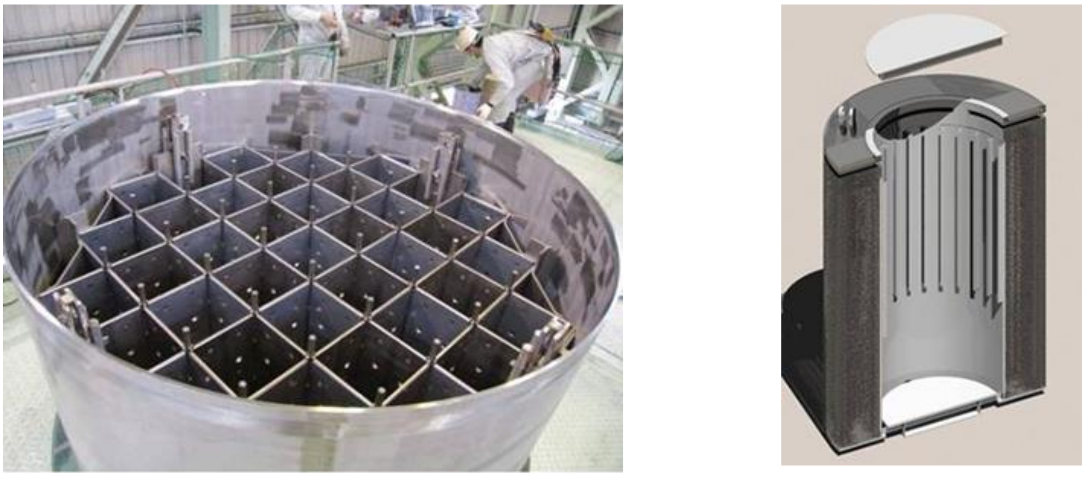


Figure 19.4.2-4: MAGNASTOR type storage cask and reinforced concrete structural shell. [19-34]

Open surface storage and the degree of the biological protection can be seen in Figure 19.4.2-5. It can be seen in the picture that the outer cladding of the containers is exposed aggregate concrete.

Figure 19.4.2-5: Dry cask storage, upright layout. [19-35], [19-36]

The storage system having lying layout is shown in Figure 19.4.2-6. There are two ways to place a storage cask in a single module. There is a vent on the bottom part and there are air outlets for the warm air on the upper part of the module.

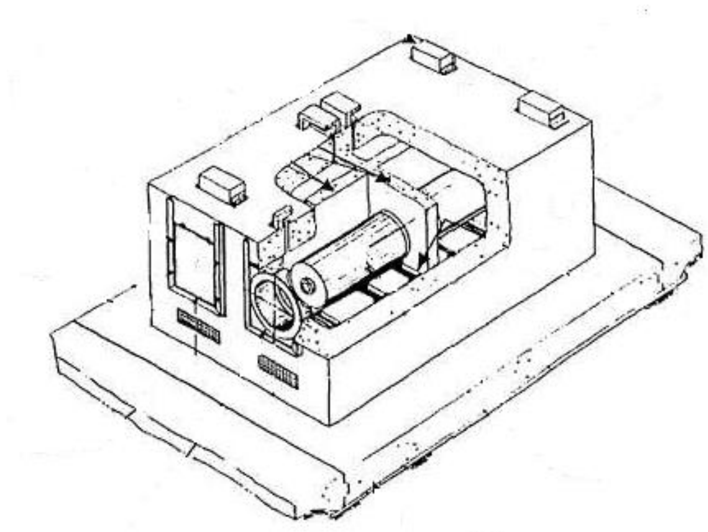


Figure 19.4.2-6.: Filling the cask for dry storage, lying layout. [19-37], [19-38]

Knowing the Magnastor type storage system and knowing that 3 135 assemblies containing spent nuclear fuel is produced per unit for the whole operation time – see Chapter EIA, Characteristics and basic data of the new units planned to be implemented in Paks – and calculating with the container having 19 baskets for the assemblies gives 330 containers for two units and for 60 years.

The area for storage is a ca. 22 ×70 m large concrete surface on which 48 of the containers (see picture) can be placed. The 330 containers from the example above require a seven times larger area and, in addition, there is the need for a place for loading and for the safety margin area. Making a conservative estimation in can be calculated that this needs an approximately 115 × 100 m large area (as a completely independent facility). Inside the site of the Paks NPP it needs a lower degree of self-security measures with lower area demand.

The typical layout of a dry cask storage site can be seen in Figure 19.4.2-7.

Figure 19.4.2-7: Typical layout of dry cask storage site. [19-39]

19.4.3 OPTIONS FOR CLOSING THE NUCLEAR FUEL CYCLE

There are several options for the final part of the nuclear fuel cycle. The differences between them lie in what kind of nuclear waste is produced and permanently stored, and the place and the conditions of the repository. Between the options discussed below another difference is how feasible they are according to our best current knowledge.

19.4.3.1 Final disposal of spent fuel

Choosing the permanent disposal of spent fuel elements there is no need for complicated chemical treatment. The spent fuel removed from the reactor can be permanently stored after a few decades of storage without processing. This fuel cycle is referred to as an open fuel cycle (or a once-through fuel cycle). The waste being permanently stored without processing has high activity level and high heat production rate. The radiotoxicity level of the spent fuel reaches the radiotoxicity level of natural uranium only after a couple of hundred thousand years.

The benefit of permanent isolation and storing is that no tasks and problems will arise for the next generations and there is no need for intervention in the future. However, this way there is no opportunity to apply the latest scientific and technological results and findings and the material placed permanently can no more be recovered or re-used.

The most important requirement of permanent disposal is the perfect isolation of the waste from nature, which can be ensured using a multi-barrier system (geological and technical). There should be many supplementary natural and artificial barriers between the waste and the biosphere. Such barriers are, for instance, the conditioning of waste, the multi-layered container, the surrounding goaf, the secure closure of containers and the repository facility, as well as the appropriate geological environment.

In many countries, research is in progress on the permanent repository of the high-level waste. There is unanimous agreement: high-level waste having long half-life can only be placed safely in a stable geologic formation (i.e. in deep geological disposal facilities).

19.4.3.2 Processing spent nuclear fuel

During partial reprocessing:

- the metal parts of the spent fuel elements are converted into intermediate-level waste that should be placed in an appropriate intermediate depth geological disposal facility;
- the fission products and the accumulated transuranics contained in the spent fuel are converted into high-level waste that should be placed in an appropriate deep geological disposal facility;
- the uranium and the plutonium are chemically separated and recovered from the irradiated nuclear fuel.

The fissile materials (that can be used to manufacture nuclear fuel) produced during reprocessing can also be re-used. Nowadays, partial reprocessing is frequent, and is implemented as industrial practice. The radiotoxicity level of the nuclear waste intended to be permanently stored will reach the radiotoxicity level of natural uranium after a couple of ten thousands of years.

Based on economic and technical considerations, to construct a reprocessing plant is worth only in international co-operation and in a country having many operating nuclear power plants. It has very likely no sense to build such a reprocessing plant in Hungary, therefore when there will be any need for reprocessing of the spent fuel then it should be arranged to be carried out abroad.

Besides the above mentioned realistically feasible options, remarkable efforts are being made recently to find a cycle-ending way with which besides the re-use of the remaining fissile material even most of the transuranics can be converted (transmuted) into isotopes having much better properties (for example shorter half-life).

The main point of complete reprocessing, i.e., partitioning and transmutation (abbreviated as P/T) is to separate not only the uranium and the plutonium but also the other actinides and the long-life fission products. The uranium, the plutonium and the long-life isotopes can be converted into isotopes having shorter half-life in the transmutation technology using fast neutron reactors. This way it can be achieved that the radiotoxicity of the waste for permanent disposal remains notably high only for a few hundred years. After transmutation the waste to be permanently disposed contains fission products only with short half-life. Closing the nuclear fuel cycle this way with the implementation of transmutation can only be realized and brought to bear after an intense spell of research & development work, after political and economic decisions and in regional and international co-operation. Therefore the closing of the nuclear fuel cycle cannot yet be based on this solution.

Using fourth generation reactors, the separated uranium isotopes together with plutonium and transuranics can be used as fuel for breeder reactors having fast neutron spectrum. Using these isotopes as fuel in breeder reactors a very large amount of energy is released and the isotopes mentioned above can be converted into waste having more favorable properties. Such reactors are subjects of on-going development, but the operation of them can only start decades later. To our current knowledge, the fleet of conventional nuclear power plants and the proposed fast reactors will be able to fully convert the uranium, plutonium and transuranics that came from conventional nuclear power plants. The composition of the reactor fleet should be optimized to ensure that the waste production rate of conventional nuclear power plants just meets the fuel requirements of the fast reactors. It is hard for small countries to develop such a fleet of the reactors. In their case, regional cooperation and external helping hands are needed.

These angles also need to be taken into consideration in the making of the policy regarding the closure of nuclear fuel cycle:

- In the long run, recyclable fissile materials can become valuable, their ownership needs to be made clear. This has to be envisaged in the National Programs.
- The spent fuel and the vitrified high-level waste can only be disposed in deep geological repositories.

19.5 DISMANTLING NUCLEAR POWER PLANT UNITS

Several methods of dismantling of a nuclear power plant (dismantling policy) are known internationally.

Immediate dismantling: It begins very shortly following shutting down the nuclear power plant (within 5 years or earlier).

Safe storage (conservation) or deferred dismantling: Will be realized only with an important delay following the shutting down of the nuclear power plant. Some parts of the facility will stay under regulatory control. This condition may persist for decades. Then the final dismantling of the facility will take place and regulatory control of the facility ends. The safe confinement option is chosen if:

- no developed national waste management policies exist;
- the necessary funds for dismantling are not readily available; or
- in the dismantled facility different nuclear plants are being dismantled (in this case it is possible to allocate optimally the human resources and tools necessary during dismantling).

Encapsulation: A specific variant of dismantling, during which the so-called ‘nuclear islands’ are determined as confined as possible, and all remaining radioactive stuff is concentrated there. This space is filled out with concrete: the radioactive materials are encapsulated in the remaining building if the plant.

Mixed dismantling possibilities: From the above basic versions – considering also the possible final states – diverse variations can be elaborated in the practice. Thus the dismantling process can be realized in a time interval of 10 – 20 years.

The international expectations regarding the dismantling of nuclear power plants are getting more and more severe, permitting continuously shorter running time.

The choice of dismantling strategy is a far-reaching and complex task. With regard to the selection of the first approximation, the following important questions can be formulated:

- Which is the final goal set?
- What should be done to achieve this final goal?
- When are the most favorable conditions to carry out these activities?

Moving to higher resolution, going into details, the following practical questions can be asked:

- Are the planned operations executable?
- Are enough trained people available?
- Is the protection of human beings and nature assured?
- How much does the operation of dismantling cost?
- Is funding for the execution secured?
- What are the effects of dismantling exerted on society and the surrounding settlements?
- How do we ensure the support of the affected?

The two new nuclear power plant units to be established in the facility of Paks may start functioning in the middle of the 2020’s and are scheduled for 60 years of operation, thus their shut-down can be expected in the middle of the 2080’s. Policy for the dismantling of the new nuclear power units should be designed taking into account the dismantling policy of the already existing Paks nuclear facility, the time schedule for the programs for the management of spent fuel and radioactive waste management.

In the present EIS, the immediately dismantling option is assumed, taking in consideration the international trends and following standpoints:

- the current legislation ensures the costs of dismantling being available till the end of service life
- final disposal of the radioactive waste arising from dismantling is possible to be ensured in the available time frame
- the knowledge required for dismantling is not expected to be lost.

Based on Government decree 314/2005. (XII. 25.) Korm. on environmental impact assessment and the integrated environmental authorization process, the dismantling activities are independently supposing an environmental impact assessment and RHK Ltd. is competent to do it. According to legislation, a preliminary investigation should be done regarding the dismantling activities and an impact assessment is to be accomplished. The impact study should be done prior to shut-down among the so-called preparatory activities and should be submitted for approval to the authority. In addition, a dismantling permit shall be required from the Hungarian Atomic Energy Authority regarding the dismantling activities.

Pursuant to legislation, a Preliminary Dismantling Plan should be prepared during the authorization of the establishment of a nuclear power plant. It needs to be updated every 5 years. The preliminary dismantling plan and its updates serve multiple purposes:

- show that at a given point of time, the available technical options allow the safe dismantling of the facility,
- provide the opportunity of comparing the different available options and technologies for dismantling, to optimize the technical and economical solutions,
- the cost of the dismantling activities can be estimated according the plans,
- over the updates the base of the authorization of the dismantling activities becomes more and more detailed.

19.6 BASELINE PRESENTATION

The planned facility will be realized on the area functioning as the staging area of the currently operating Paks Nuclear Power Plant as a brownfield investment / development. In the installation area marked on the site layout **actually** there is neither radioactive waste nor **spent fuel** management.

The financial contribution to the management of spent fuel and radioactive waste (including the final disposal of radioactive waste, and the temporary storage of spent fuel and the closing of the nuclear fuel cycle, furthermore the dismantling of nuclear power plants) is ensured by the allocated state funds created by the Atomic Energy Act, the Central Nuclear Financial Fund. Regarding the new power plant units, the above activities and cost estimation, taking into account the existing facilities, should be worked out, and the future authorization holder should begin with the payment to the Central Nuclear Financial Fund the year following the one the new unit is put into work.

In Hungary, two storage facilities designed for the accommodation of low and intermediate level activity waste of short lifetime are in operation:

- The Radioactive Waste Processing and Containing Facility working in Püspökszilágy is near surface type and serves basically the placement of short lifetime, low and intermediate level activity waste from non-nuclear plant origin. The temporary storage facility receives the long lifetime, high level activity waste from non-nuclear plant origin which should be disposed finally in the high level activity disposal facility implemented in the future in order to comply with the security principles.
- The National Radioactive Waste Repository (NRWR) working in Bátaapáti is subsurface, but not a repository installed in a deep geological formation; the low and intermediate level activity, solid or solidified radioactive waste is disposed there.

For choosing the location for the final disposal of high level activity radioactive waste, of long lifetime low and intermediate level activity radioactive waste, of the spent fuel or the residue from the processed spent fuel an adequate – deep geological – high activity level radioactive waste repository, year-long studies were conducted in the area of the Western Mecsek hills, on the territory of the Lower Permian Boda Siltstone Formation which is the most suitable for disposing radioactive waste of high activity. In 2004, a conceptual plan regarding the placement of high activity level radioactive waste was completed, while in 2008 the conceptual plan of the long-term program of the study was completed. The 1st period of study phase 1 was closed in 2010. In 2014, the studies will restart in the Western Mecsek where they are expected to designate the exact location of the underground laboratory till 2030. First a deep geological laboratory will be built in the clay soil, than the repository will also be worked up, which will serve to place the spent fuel and high activity level waste produced in Paks Nuclear Power Plant."

During the operation of the new power plant units built, spent fuel will be produced, which will rest for maximum 10 years in the spent fuel decay pool placed near the reactors. Paragraph 2 of Article 7 of the Convention promulgated in Act II of 2004 ensures the possibility – following storage in the spent fuel pool – to transport the spent fuel produced during the operation of the new power plant units to Russia for technological storage, or for technological storage and recycling. According to the referenced convention in the case of technological storage and recycling, the time frame of the storage of high activity level radioactive waste can be prolonged up to 20 years.

According to international trends, the claim to introduce the category of very low activity level waste (VLLW) appears also in Hungary. During the planned service life of the new units, it is plausible that the regulatory environment will change in a way permitting the implementation of a facility making possible the final disposal of this kind of waste.

19.7 IMPACT OF PAKS II CONSTRUCTION

19.7.1 DIRECT IMPACTS

Direct impacts due to the generation, collecting, handling or disposal of radioactive waste are expected during the installation. During the approximately 60 months of the installation period, mostly closed radiation sources will be used for material testing purpose. During their application, there is no need to deal with the generation of radioactive waste. In the case where closed radiation sources will no longer be needed, they will be disposed of according to the applicable legislation. The first load will arrive to the plant 1 year prior to finishing the installation.

19.7.2 INDIRECT IMPACTS

During implementation no direct impacts are expected regarding radioactive waste, thus the development of indirect effects is not expected either.

19.7.3 IMPACT AREAS OF PAKS II CONSTRUCTION

During implementation, the emission of radioactive isotopes is not expected, thus the indirect effects, and the notion of impact area are not relevant, indirect effects are not expected (in the absence of relevant factors).

19.7.4 TRANSBOUNDARY ENVIRONMENTAL IMPACTS

According to the previous two subchapters, no effect under the threshold value set in the legislation or effects inside the facility are expected to form, thus in the absence of relevant factors the impact area due to collecting, storing radioactive waste, the transboundary environmental effects cannot be fixed.

19.8 EXPECTED IMPACTS OF PAKS II OPERATION

19.8.1 NORMAL OPERATION

During the design of the unit type, particular attention was paid to the fact that compared to the previous technological solutions a smaller amount of radioactive waste should be generated. Due to the development of primary-loop systems and because now technology is more compact, the amount of radioactive waste will be significantly less than in the case of the actual working unit of Paks.

The frequency with which the types of material belonging to different categories, tools and equipment are produced and their amount is largely dependent on the campaign length of the reactor, the number of planned and unplanned stops, maintenance and the systems affected by the repair/maintenance works. The quantitative distribution of the produced mostly low and intermediate activity level waste will slightly depend on the changes and modifications in reactor performance.

The systems of the new unit are designed to be capable to process the radioactive waste produced during the lifetime such that the emission of solid, liquid and gaseous materials remains as low as reasonably achievable. During planning the accumulated experience, the changes of costs during life time and the changes in the available storage capacity were also taken into account. If necessary, the existing systems can be completed, modified.

According to our current knowledge, steps of collecting, handling, temporary and final storage of radioactive waste will generally follow the existing procedures applied in the Paks Nuclear Power Plant for waste of similar type. Of course in the case of special treatment equipment (supercompactor, high activity level waste container) and for certain individual waste forms (dustlike ion-selective sorbents, solvents) they will be differences in the related sub-processes, but collection, treatment, and the entire process of conditioning will be based on the accumulated experience and requirements (waste acceptance requirements, radiation safety standards, their control, etc.).

19.8.1.1 Expected quantity and distribution of radioactive wastes

19.8.1.1.1 Low, intermediate and high activity solid radioactive wastes

Solid radioactive waste is formed during normal operation in the equipment used to clean and handle liquid and gaseous radioactive waste, during maintenance and during eventual malfunctions.

The formation of two types of solid radioactive waste is expected:

1. solid radioactive waste:
   - equipment removed from the reactor and their parts (the parts form the handling equipment of the control rod, thermometers, transducers from ionizing chambers and their leads, etc.)

- contaminated, disassembled tools beyond repair, pipe sections, valves
- contaminated tools and parts,
- exhausted aerosol and iodine filters from the gas cleaning and ventilation technique
- contaminated work clothing, shoes, disposable personal protective equipment, which are not washed (for decontamination)
- contaminated building elements, insulation.

2. solidified liquid radioactive waste.

The average amount of annually generated solid radioactive waste for one unit before treatment is shown in Table 19.8.1-1.

Type of solid waste	Amount of waste [m^3/year]
Low activity level	70
Intermediate activity level	11
High activity level	0,5
Large-scale, not manageable (formed during maintenance / repairing)	5

Table 19.8.1-*1: Estimated annual quantity of generated solid radioactive wastes per unit. [19-28]*

The average amount of annually generated solidified radioactive waste (also taking in account the effect of volume reducing technologies):

- 20 m^3 cemented evaporation residue;
- 8 m3 cemented used ion exchange resin.

19.8.1.1.2 Liquid radioactive wastes

Liquid radioactive waste is produced during the entire service life of the power plant; it can contain organic and inorganic compounds. During the operation of the power plant, the following liquid radioactive waste is produced:

- waters from equipment, fittings, pipelines and decontamination of rooms
- regeneration, loosening and transport waters from the filters of special water treatment systems, water coming from draining systems and potential leakage
- washing and rinsing water from the liquid waste evaporation systems
- liquid waste from sampling and laboratories
- liquids from draining equipment, pipe sections, fittings and potential leakage.
- shower water from primary loop dressing rooms and waters from special laundry.

During the cleaning and handling of liquid radioactive waste, secondary liquid radioactive waste is also produced:

- evaporation residue
- used ion exchange resin
- mud and inorganic isotope-selective sorbents.

The average amount of the annually generated liquid radioactive waste for one unit under normal operating conditions is the following:

- evaporation residue 25 m^3/year
- ion exchange resin 10 m^3/year
- sludge-slurry filter 0.1 m^3/year
- leachate slurry 0.5 m3/year

19.8.1.1.3 Waste quantity for final disposal

The estimated average amount of low, intermediate and high activity level radioactive solid wastes produced during the operation of the reactor units of Paks II to be implemented is shown in Table 19.8.1-2. During the estimation of the amount of waste to be disposed, the effect of the new waste treatment and conditioning technologies implemented together with the new units was also taken into consideration.

Waste	Amount of waste [m^3/year]	Amount of waste after treatment (solidification, crushing, etc.) [m^3/year]	Number of units to be handled / stored
Low activity level solid	70	28	140 barrels
Intermediate activity level solid	11	4	20 barrels
High activity level solid	0,5	-	5 assemblies
Large-scale, not manageable (formed during maintenance / repairing)	5	-	-
Cemented evaporation residue	25	20	100 barrels
Cemented ion exchange resin	10	8	40 barrels
Cemented mud	0,6	0,5	3 barrels

Table 19.8.1-2*: Estimated annual quantity of generated soild radioactive waste per unit. [19-28]*

The planned amount of waste from different activities generated during the estimated service life of minimum of 60 years for 2 units is shown in Table 19.8.1-3.

Type of waste	Amount of waste [m^3/60 year]	Amount of waste after treatment (solidification, crushing, etc.) [m^3/year]	Number of units to be handled / stored
Low activity level	8 400	3 360	16 800 barrels
Intermediate activity level	1 320	480	2400 barrels
High activity level	60	-	600 assemblies
Large-scale, not manageable (formed during maintenance / repairing)	600	-	-
Cemented evaporation residue	3 000	2 400	12 000 barrels
Cemented ion exchange resin	1 200	960	4 800 barrels
Cemented mud	72	60	300 barrels

Table 19.8.1-*3: Estimated annual quantity of generated soild radioactive waste per 2 units during lifetime.*

19.8.1.2 Estimated quantity of spent fuel

According to supplier data, this means 3 135 and 6 270 spent fuel assemblies, respectively in the case of one or two units. During the 18-month campaigns 76 spent fuel assemblies will be produced per unit.

Based on the known data of the units, the amount of the spent fuel during the entire service period can be estimated. Calculating with UO_2 fuel and a service period of 60 years, according the given data 1 674 t spent fuel will be produced in one reactor, calculating with two units this gives 3 348 t.

19.8.1.3 Collection, treatment and storage of low, intermediate and high activity wastes

Low and intermediate activity solid radioactive wastes

The solid wastes produced in the controlled zone of the new nuclear power plant will be **collected in a selective manner already** on the place of production. The **selection** of waste will be done based on radiological parameters, taking also into account the following waste types:

- potentially inactive waste
- low activity level waste for compaction
- low activity level large-scale, not cuttable
- intermediate activity level waste for compaction
- intermediate activity level large-scale metallic waste
- intermediate activity level not treatable waste.

The potentially inactive waste will be **cleared** after **radiological classification**, and it will be managed as conventional waste.

The part of low activity level waste for which the isotope content has - according radiological qualification - the clearance limits of maximum 10 years is separated and temporarily stored after packaging. If the isotope content of the waste reaches due to radioactive decay the clearance limit, following a new radiological qualification, it will be cleared.

The waste suitable to be compressed will be compacted, this way the volume of waste for final disposal decreases. The compacted waste is temporarily stored, then if needed it is conditioned (to form waste which can be stored in the NRWR, for instance placed in a container and filled out with cement).

After a short temporary storage the conditioned waste will be transferred for **final storage** to the NRWR to RHK Ltd.

Low activity level large-scale waste not fit for cutting is separated for temporary storage. Following the 10 years long temporary storage period the waste will be treated as a function of its radiological nature:

- clearance
- decontamination, followed by clearance
- cutting, followed by compaction or embedding in cement matrix.

High activity solid wastes

High activity level solid wastes produced during maintenance activities must be **packaged.** If the nature of such high activity level solid waste allows it, its volume must be reduced.

The **temporary storage** of high activity level waste packages is done until the dismantling of the unit or the opening of the NRWR in a repository built for this purpose.

Following temporary storage, high activity level radioactive waste is transferred for final disposal to RHK Ltd.

Liquid radioactive wastes

During the design of primary loop systems, it is a primary concern to optimize the **amount** of liquid radioactive waste produced during the operation of the power plant with the aim to minimize the amount of radioactive waste for final disposal. To decrease the amount of radioactive waste at the source site, the following can be done:

- **selective collecting and handling** of the produced liquid radioactive waste according to its composition and activity level
- minimization of the use of chemicals (ion-exchange filters with low salt concentration, but high activity level are not regenerated)
- use of decontamination methods resulting in a low amount of liquid waste
- elimination of radioisotopes from potentially inactive or low activity level liquid waste with ion-selective sorbents and the issuance of the purified liquid.

Due to the above, in the case of normal operation the annual amount of leachate to handle will be significantly lower compared to the previously designed VVER440 units.

The drains, ventings, controlled leakage containing boric acid will be collected, handled and recycled separately. This way only a minimal amount boric acid enters to the leachate, thus again minimizing the volume of liquid radioactive waste.

Ion-exchange resin regenerating, loosening solution resulting from steam generator blowdown, the waters from the special laundry and the shower room of the primary loop locker room are released depending on their activity concentration either to the environment without treatment or after treatment with selective sorbents.

Following **volume reduction,** radioactive leachate is **conditioned** (solidification) such that the end product meets the requirements regarding final disposal at the NRWR.

If the chemical and radiological compositions of the condensates produced during volume reduction of the radioactive leachate are satisfactory, they will be reused in the primary loop, or released to the environment as extra water volume.

During the cleaning and treatment of liquid radioactive wastes, secondary liquid radioactive waste is produced, which is also conditioned in the manner such that the end product meets the requirements imposed on final disposal.

After **temporary** storage, the conditioned waste is transferred to RHK Ltd. for **final disposal** in the NRWR.

19.8.1.3.1 Solid radioactive waste management system

The task of the solid radioactive waste handling system is to handle the solid radioactive waste produced during normal and emergency operations, and the solid waste produced in the liquid waste solidification system. The essential role of the system is to collect, classify in an appropriate manner the solid waste, and in order to optimize storage capacity, to reduce its volume as much as possible. The treatment of solid radioactive wastes ensures that according to the standards set in legislation, no radioactive materials are released to the environment. This procedure prepares solid radioactive wastes for the transfer to the final repository.

During planning the following security and other criteria were observed:

- During the operation and maintenance of the system, the dose limits set by the standards and prescriptions regarding the personal shall be guaranteed.
- The activity concentration in the atmosphere of the premises and environmental elements must remain under the approved limits ensuring an acceptable exposure dose due to incorporation regarding the operating staff and the population.
- The probable failures of the system and the associated troubleshooting should not lead to the impossibility of fulfilling the above criteria.
- The waste media must be classified in terms of processing waste according to activity level, state (of matter), flammability, toxicity and other properties.
- When designing the treatment of solid radioactive waste, a technical barrier should be incorporated to ensure that the collected solid radioactive waste can in no way be released to the environment.
- The radioactive waste treatment systems and the applied technologies should be designed in the way to ensure that the end product radioactive waste corresponds to the requirements of transport, temporary storage and transfer for final disposal.
- The treatment of solid waste should attempt to store the waste in an optimum smallest volume.

Solid waste collection

The basic technical and organizational principles of the waste collecting systems that evolved along the above criteria are as follows:

- When collecting and selecting low and intermediate activity level radioactive waste the radiation level at source should be also taken into account, and further treatment should also depend on these levels.
- The radiation emitted by the waste should be controlled by the control organization with mobile devices which are able to detect also alpha, beta and gamma radiation, if necessary.
- For dust management, the waste should be filled into disposable containers / bags. Then the container / bag should be provided with an identification label for further handling.
- On the identification label of the bag /package containing low activity level waste the methods (for example: compaction, for clearance, etc.) of further handling should be indicated. The package is to be collected in separate white containers.
- The dose level of the gamma-radiation is 0.4 mGy at 0.1 m from the surface. In this case the activity level is also to be indicated.
- Packages with a dose level of gamma-radiation of 0.4 mGy at 0.1 m from the surface should be collected without biological protection in metal containers of 200 dm^3 volume; these should be also suitable for further temporary storage on the site of the power plant. Care should be taken to separate and lock up the metal containers, so that due to the security measures applied to the storage environment the exposure does not exceed the allowable level. If necessary, the container should be coated with a shielding layer in order to reduce the dose rate.
- The large-scale, low activity level waste which is not fit for cutting or cannot be reduced in size should be stored in a separate premise taking appropriate security measures. These items should be covered with foil for transporting.
- The high activity level solid radioactive waste from the primary loop should be placed prior to transport into shielded capsules / containers, and transported in this manner to the building for handling with an appropriate vehicle.

Solid waste clearance

The potentially inactive waste is collected separately. Following the radiological qualification of the waste (measurement of activity concentration with gamma spectrometry, measurement of surface contamination) in case of values under the limit, the waste will be cleared. Otherwise, the waste is to be treated as radioactive waste.

Clearance – according to the characteristics of the waste – can be without condition or conditional. In case of clearance without condition, further treatment, recycling, disposal can be done without additional constraints, arbitrarily. In case of conditional clearance, the waste can be recycled, or transferred for disposal according to the conditions stated in the authorization. The cleared waste is transferred outside the controlled zone.

Solid waste packaging

In case of low and intermediate activity level waste, the most common method is packaging into metal barrels. The management principles associated with this method are as follows:

- The primary classification of the waste packages is done according the dose level of gamma-radiation. It is decided on the basis of the measured value if the waste piece can be temporary stored for the purpose of decreasing its activity.
- As the Hungarian legislation is based on the isotopic composition, the determination of the precise isotope inventory is performed with a selective waste qualification system (for example segmented gamma scanner). Prior to further treatment the waste stored on the basis of surface gamma in the expectation of activity reduction, will be rated by gamma spectrometry.
- The high level activity radioactive waste deriving from the reactor shaft and the primary loop will be placed with special tools belonging to the reactor building in capsules with biological protection and transported in transport containers to the building for treatment, where the capsules can be moved safely to the shielded storage space.

Solid waste storage to reduce activity concentration

The knowledge of the precise isotope inventory makes possible to estimate, before the further volume reducing steps, if the given waste package will indeed meet after 10 years of temporary storage the limits permitting its clearance. If yes, the waste package will be stored without any further treatment separately to reduce the activity concentration and to be cleared in the future.

Those waste packages whose isotope content reaches - according the calculations - the clearance level, can be cleared following a repeated qualification.

The knowledge of the precise isotope inventory makes also possible to estimate if the given waste will reach during temporary storage the level related to very low level activity wastes. If yes, choosing the appropriate time frame for temporary storage can make it possible for the waste to be transferred to the VLLW repository – if it will be available at that time – instead of to the NRWR. If this will be possible, the waste package will be stored separately without any further treatment. After temporary storage, following a repeated classification the final disposal could be in the VLLW repository.

Solid waste volume reduction

For the volume reduction of solid waste the supercompaction technology was chosen based on the factor of possible volume reduction, the favorable characteristics of the end product and earlier operational experience. The supercompacted waste is placed into barrels of 165 dm^3 volume. The pellets produced during supercompacting are placed into barrels of 200 dm^3 volume as tightly as possible (to fill out totally the barrels). Then the barrels are sealed. The sealed barrels are transferred following a radiological classification for temporary storage.

Large-scale metal waste pieces not suitable for supercompacting can be cut and put in an optimum arrangement saving maximum room in barrels of 200 dm^3 volume. The conditioning of waste placed into the barrels happens with cement; this can be achieved with the solidification system for liquid radioactive waste.

Temporary storage of solid wastes

The storage of the packages prepared for temporary storage is performed as follows:

- The appropriately identified, volume reduced and compacted low and intermediate activity level waste can be stored for 10 years on the site of the power plant before being transferred to the radioactive repository.
- The large-scale, low activity level waste is stored in separate premises, foiled, or with individual casing. This way until the beginning of further processing, the storage can be ensured (maximum for 10 years).
- The high level activity radioactive waste deriving from the reactor shaft and the primary loop can be stored during the entire lifetime of the power plant, till dismantling.
- The condition, radiological characteristics of the stored waste can be controlled periodically by visual inspection / personal perception and instrumental measurements.

Final disposal

The supercompacted or cemented waste package can be made compliant to the criteria concerning the transfer for final repository. If necessary, it will be possible to place the waste in thin-walled steel containers (each for 4 pieces of 200 dm^3 barrels) of good shape to be placed in the NRWR geometry, and to fill out the gap between the containers with active or inactive cement paste.

19.8.1.3.2 Liquid radioactive waste management system

For collecting, handling, storing and conditioning the liquid waste produced during the operation of the planned power plant, the following system will be established:

- leachate collecting and treatment system
- treatment system for low activity level wastewater
- liquid radioactive waste storage system
- liquid radioactive waste solidification system

Leachate collection and treatment system

The aim of the system is to collect and treat the liquid radioactive medium produced during the operation of the power plant. The tasks of this system are:

- collecting radioactive leachate
- evaporation of leachate
- releasing condensates as extra water

The role of the system is to collect the waste water from the sewage system of the following primary loop buildings:

- auxiliary building
- reactor building
- health care building
- building of security systems

The leachates merging in the sump are raised by a pump to a two-cell sedimentation tank, where most of the sludge settles. The liquid phase is then fed to a hydrocyclone, which completely separates the solid and liquid phases.

The muddy phase is drained from the sedimentation tank of the mud collecting cell and the hydrocyclone from time to time to the collector tank of the waste solidification system.

The liquid phase is drained into a collector tank. The chemical treatment (adding of NaOH) previous evaporation is done in the tank, then the liquid comes to the evaporator. As the solubility of boric acid is relative low, by increasing the pH level the sodium-borate solution can be evaporated to a concentration of 400 g/dm^3 of the total salt concentration; this results in a significant decrease of the volume of the liquid radioactive waste.

The concentrate solution is drained from the evaporator through a mechanical filter to the collector tank.

The heads of the evaporator (practically distilled water) is chilled in the post-condenser and drained through ion exchange filters to the release tank. The non-condensating gases / vapor are drained to the gas cleaning system.

Low activity waste water treatment systems

The system is collecting the “probably clean”, presumably very low radioactivity level liquids produced in the shower rooms of the health care buildings and the special laundry. The “probably clean” waste waters (regenerating solutions, loosening waters, etc.) produced in the auxiliary buildings also arrive there.

Following the analysis of the water sample taken from the tank, the content of the tank is conditionally released to a control tank; depending on the analysis results, if the specific activity:

- do not reach the release threshold value, it can be released to the sewerage, or
- if the content of the control tank is over the threshold value, it is drained for additional cleaning to the selective ion exchangers.

The selective ion exchangers are suitable for a high degree of purification of waste waters. The applied inorganic cation exchange cartridges are suitable to bind cesium, cobalt ions and ions with similar sorption capacity with a decontamination factor between 10^2-10^4.

The purified water depending on the measurement following sampling is:

- released to the environment if the activity concentration is within the limits,
- if not, it is drained to the primary loop leachate treatment system.

The tanks are sized to be capable to store liquids for three months; during this period the short-lived radionuclides are degraded in a substantial part, in the case of a radioactive contamination (for example ^{131}I) due to an unexpected leakage.

Liquid waste storage system

The radioactive waste waters produced during the operation of the nuclear power plant (and the waste treatment systems) are collected according to the method of further treatment.

The produced waste is classified as follows:

- concentrates (concentrated boric acid solutions)
- ion exchange resin and ion selective sorbents
- muds
- special inactive ion exchange resins from the primary loop water cleaning systems.

The tasks of the liquid radioactive waste storage system are as follows:

- the receipt and transfer of the condensation residue from the liquid radioactive waste treatment system to the solidification system
- the receipt and transfer of the spent ion exchange resin and ion selective sorbents to the solidification system
- receipt and control of low activity level spent resins
- receipt, dewatering and removing after release of non radioactive ion exchange resins.

The tanks for the storage of the ion exchange resins and their transport water are sized to be capable to store liquids for three months; during this period the short-lived radionuclides are degraded in a substantial part. The system is suitable to separate the transported resin from the transport water. The transport water from the tanks of the system is transferred to the leachate treatment system.

Liquid waste solidification system

The task of the system is to process the liquid radioactive waste (cementation) and form it to unit volume (barrel).

The system established in the auxiliary building is able to condition the waste during normal operation and as well as in an emergency situation.

The technology is suitable to produce end products meeting the waste transfer requirements from the following liquid waste:

- concentrates of 400 g/l concentration
- ion exchange resin
- mud and inorganic sorbents.

The used barrels of 200 dm^3 are also suitable to place solid radioactive waste, with active and inactive cement gap filling.

To minimize the volume of solidified liquid radioactive waste, the solidification system contains an additional system to degrade the ion exchange resins by heat and an extra condenser to further concentrate the evaporation residue before cementing it.

19.8.1.4 Storage and treatment of spent fuel

After being discarded form the reactor, the spent fuel is placed into the spent fuel decay pool where the removal of remanent heat is assured till its value decreases to a value which allows the dry temporary storage of it. Spent fuel can spend a maximum of 10 years in the spent fuel pool.

Following storage in the spent fuel pool, the spent fuel is transferred for temporary storage. There are currently two options available:

- the spent fuel cartridges are transported to the Russian Federation for temporary technological storage or for technological storage and reprocessing. The spent fuel, or in case of reprocessing the nuclear waste, is stored in the territory of the Russian Federation for the same time period which is prescribed by the Article 7 Paragraph 1 of the agreement (contract) regarding the nuclear fuel supply (20 years), then it will be retransported to Hungary for
- the domestic temporary storage of spent fuel.

In terms of the planned lifetime of the new units and the time fixed in the interstate agreement for the domestic storage of spent fuel, we took into account the domestic temporary storage in the facilities of the units or in the immediate vicinity. Temporary storage will be maintained until the indirect final disposal of the fuel, or the high activity level radioactive waste produced during reprocessing is not guaranteed under domestic conditions.

Following the temporary storage we expect the domestic storage of spent fuel, subject to the following:

- according to the NE Act for the final disposal of the waste produced in Hungary one of the requirements – the operation of the radioactive waste repository was authorized to accept this special waste and was already operating before the production of the waste – is not fulfilled
- because of the scheduled length of the operation time the realization of other long-term possibilities is questionable, there are significant risks

Decay

The task of the system storing and handling nuclear fuel / fuel elements is to ensure transport and storage, in summary: the control of the fresh and spent fuel from receipt to delivery. This system incorporates all tools and procedures, which are needed to ensure the tasks related to the fuel inside the plant and the containment.

The functions performed by the system in relation to handling and storing spent fuel are as follows:

- transferring spent fuel to the spent fuel decay pool,
- storage in the spent fuel pool, cooling (for 10 years),
- following the decay time transferring the assemblies to the spent fuel repository,
- moving the spent cartridges to outside of the site.

The spent fuel pool is positioned inside the containment, directly near the reactor. 732 fuel assemblies and 24 sealed capsules can be placed there. The system includes equipment detecting the faulty assemblies.

Interim storage

From the temporary storage possibilities mentioned in Chapter 19.4.2 In the EIS the area for storing temporarily the spent fuel is envisaged between Unit 4 of the Paks Nuclear Power Plant and Unit 1 of Paks II.

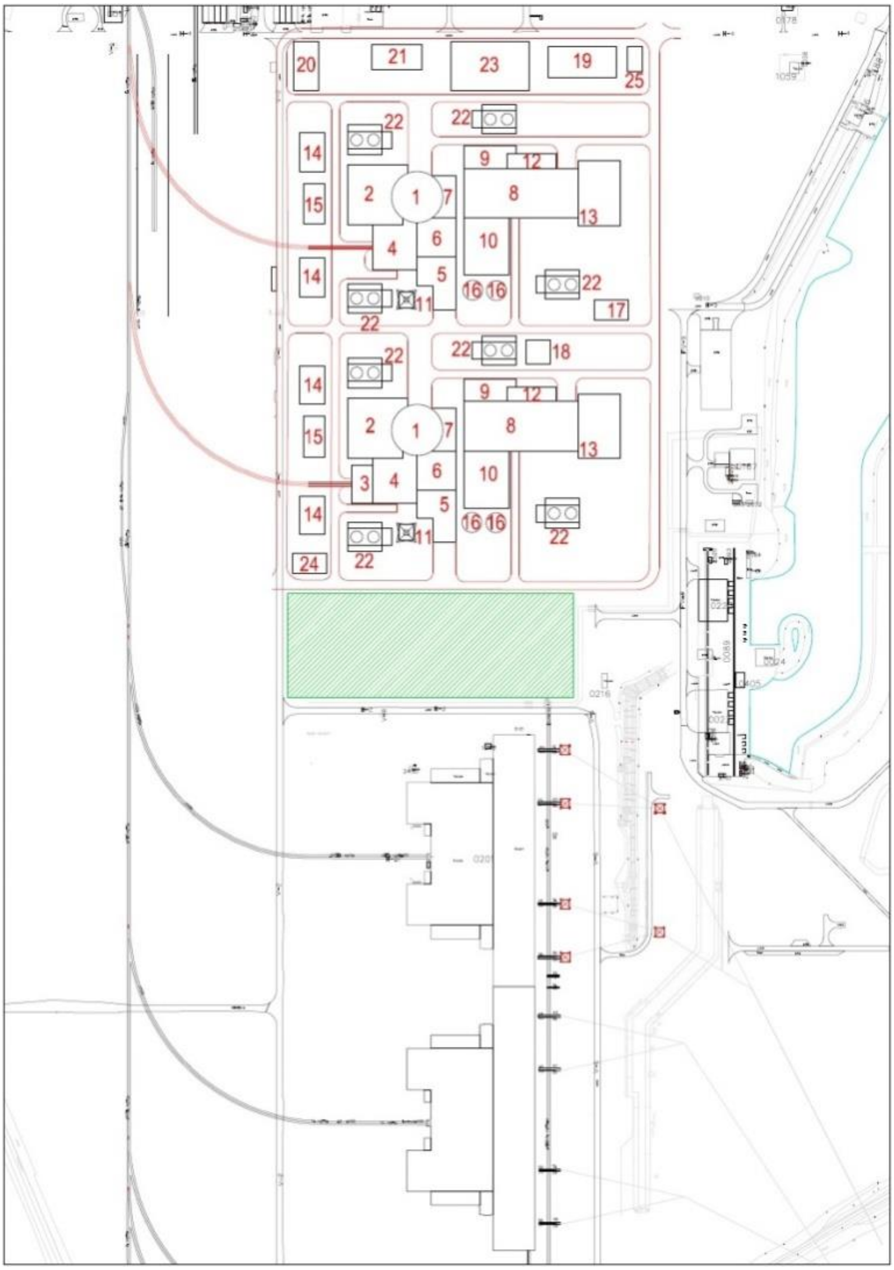


Figure 19.8.1-*1: Area proposed for the installation of the dry container storage facility.*

Spent fuel can stay for 10 years in the decay pool. The spent fuel is placed for dry storage in containers with biological protection. The containers are decontaminated outside, then dried and controlled for surface contamination. After the control of the proper sealing of the containers they are transferred from the reactor building to the temporary storage place for spent fuel.

Final disposal

The choice of the storage place of spent fuel / or its residual recovery is currently in progress. Till the final repository is chosen and implemented as necessary, the above described dry container method is applied.

Following temporary storage, spent fuel will finally be disposed of in the high activity level waste repository which will be established in Hungary. The container type chosen for the temporary storage will be also suitable for the transport of the spent fuel, thus no other operations will be needed.

19.8.2 IMPACT AREAS OF PAKS II OPERATIONS

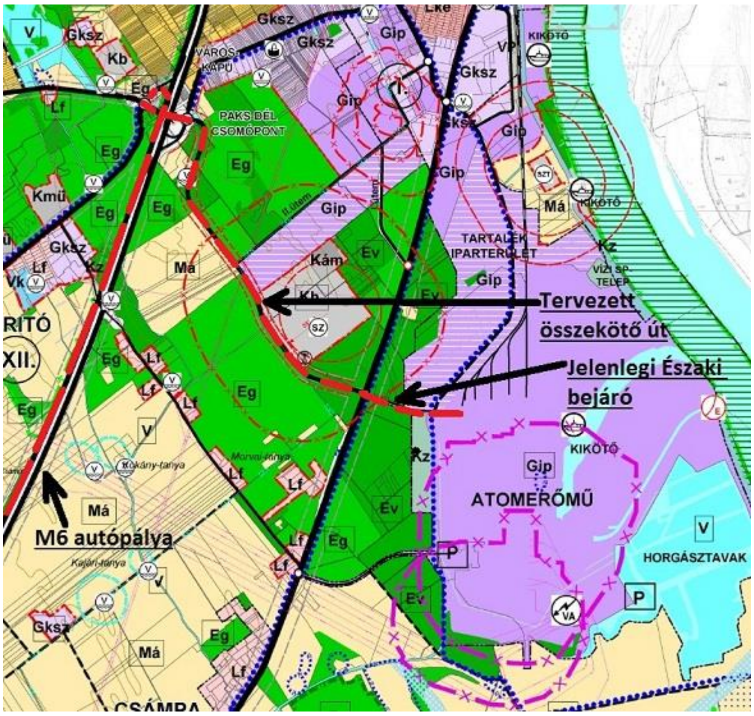
When determining the expected impact area estimated based on the environmental impacts of radioactive wastes, an important factor is the place of treatment of wastes having different activity concentrations (on or off site, and surface or sub-surface) as well as the duration of the given technological step (taking into account the decrease in the activity concentration due to the half-life of the individual radionuclides).

19.8.2.1 Area of direct impacts

19.8.2.1.1 Impact area of the management /storage of low- and intermediate-level wastes

After having been put into barrels (handling, transportation and storage units) as a result of the necessary volume reduction, conditioning, solidification, the storage of low- and intermediate-level wastes generated during operation of the nuclear power plant will be realized in the storage area and hall established at the site. These wastes contain only fixated radioisotopes, hence, during their storage, based on the dose rates that can be measured on their surface, typically only construction of the physical protection is necessary; the impact area of the environmental impacts examined from the radiological point of view is limited to the power plant site and, within that, to the area of the storage hall.

Conditioned wastes are disposed of at NRHT. The vehicle carrying waste packages, according to the regulations of ADR7, shall not make a stop in addition to traffic rules and the value of the external gamma dose rates induced in the vicinity of the cargo is subject to restrictions, too. The route of transportation to NRHT is 64 km long. 49 km of the route goes on Motorway M6, hence, in that section, the radiation exposure to a person standing supposedly on the edge of the carriageway does not need to be considered, since pedestrian traffic is prohibited along motorways; furthermore, a person present in the service areas and the territory of the filling stations that are part of the motorway can be only so close to the centre line of the lane where the effect of the radiation is negligible. In the first section of the transport route a feeder road between the present northern access road to the Paks Nuclear Power Plant and M6 is included in the master plan of the town of Paks, which extends from the intersection of the northern access road and Road 6 to Paks M6 southbound exit, without going into populated areas. The red dashed line in Figure 19.8.2-1. below indicates the first section of the transport route, the planned approach of Motorway M6 as well as the access.



Legend: M6 autópálya - M6 Motorway, Tervezett összekötő út - Planned shortcut, Jelenlegi Északi bejáró - Current North Gate, NPP. Paks Dél Csomópont – Paks Southern junction, horgásztavak – fish ponds, atomerőmű – nuclear power plant, kikötő – quay, tartalék iparterület – reserve industrial zone

Figure 19.8.2-1: Transport route of low- and intermediate-level wastes from the site to Motorway M6. [19-40].

After leaving the motorway at Exit 164 (Figure 19.8.2-2.) there are another 11 km of public road; in this section (based on a conservative approach) the person standing on the edge of the carriageway is to be taken into account. Within the municipality boundaries of Bátaapáti a bypass has been constructed to protect the residential buildings next to the transport route (and their occupants) in order to reduce the conventional and radiological burden.

Legend: 56103 bekötő út - 56103 accessway, 5603-as összekötő út - 5603 shortcut, M6 autópálya - M6 Motorway.

Figure 19.8.2-2: Transport route of low- and intermediate-level wastes from Motorway M6 to NRHT. [19-41]

The impact area for NRHT can be interpreted within its premises assuming normal operating procedure.

The section on Environmental radioactivity – radiation exposure to the population living in the vicinity of the site addresses the determination of the degree of radiation exposure induced along the transport route during transportation to the disposal facility in detail. Based on the results the annual radiation exposure to the people, even with conservative estimation, is by orders of magnitude below the public dose limit, or rather the dose constraint defined for the nuclear power plant, hence, during transportation to the disposal facility, the transport route, or rather the edge of the carriageway can be considered as the border of the impact area with the very conservative assumption that it is always the same person standing on the edge of the road when the transport vehicle passes by.

In case of realization of the storage facility for very low level wastes, the radiation exposure to the population induced by the transport of these wastes will be also negligible considering the significantly lower activity content of the waste.

19.8.2.1.2 Impact area of the management / storage of high-level activity wastes

High-level wastes, after their production, are placed directly in the auxiliary building of the primary loop in containers with appropriate shielding. Since, during the process, the waste is enclosed in capsule, and then placed in a shielded storage container until the end of the service-life, it cannot get into contact with the environmental components. Therefore, the radiation exposure induced by this waste form with respect to environmental components is limited to the area of the site, or rather within the impact area identical to the 500-meter boundary of the formally not yet officially determined safety zone. According to Section 4 § (1) of Government Decree 246/2011. (XI. 24.) Korm., the designer of the nuclear installation makes a proposal for the boundary of the safety zone of Paks II in the supporting documentation of the construction permit application. Based on the information currently available we consider, conservatively, the boundary line shown in green in the figure below as the boundary of the impact area of radioactive wastes in respect of the proposed facility. (Figure 19.8.2-3.)

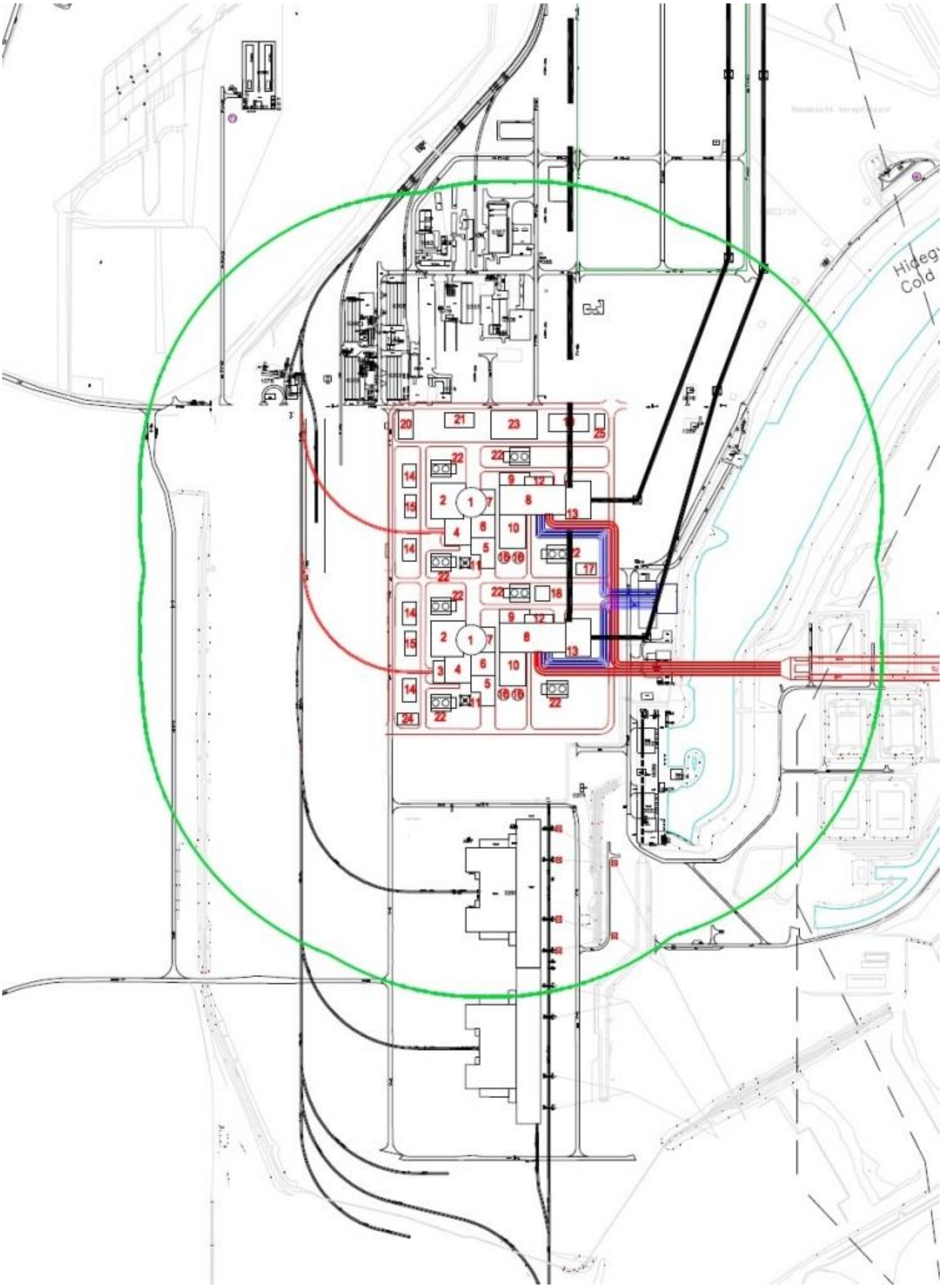


Figure 19.8.2-3: Boundary of the impact area of the high-level wastes of Paks II.

19.8.2.1.3 Impact area of the management /storage of spent fuel assemblies

After the period spent in the cooling (decay) pool, spent fuel assemblies are transported to the temporary storage facility, where they are temporarily stored with surface storage container technology for decades. Thanks to the technology, the containers will be approachable without any special protective equipment on the open, coated surface for the purpose of continuous monitoring measurements. The containers will spend, expectedly, decades on the storage area, and then they are transported to either a reprocessing plant or a disposal facility without any further manipulation, since the surface storage containers provide adequate protection even during transport operations.

The environmental radiation exposure due to the containers on the surface storage area does not exceed the dose constraint even on the boundary of the impact area identical to the boundary of the safety zone (Figure 19.8.2-3.).

In the case of transportation to a reprocessing facility the section of the assigned railway line to the border is taken into account. When planning the route it is also considered that the train passes by the minimum number of populated areas possible based on the existing rail network, it has priority and it is secured, thus, including the planned stops, too, the length of stay is also minimal.

At the moment there is not even a storage facility for disposal after the temporary storage within the national borders; the investigation of the area that is considered to be suitable (Boda Claystone Formation) is ongoing.

19.8.2.2 Area of direct impacts

19.8.2.2.1 Indirect impact area of low- and intermediate-level wastes

Regarding the indirect impacts of the temporary storage and the subsequent disposal affecting the environmental components, it should be noted that the radioactive content of the waste cannot get out during the estimated life-span of the procedure of volume reduction and solidification (conditioning) of solid and liquid wastes as well as the realization of the package enabling the transportation to the storage area, and the life-span of the packaging; therefore the area of indirect impacts will be identical to the site boundary of storage area.

19.8.2.2.2 Indirect impact area of high-level wastes

The impact area of the indirect impacts of high-level wastes depends on the management and storage technology of the wastes mentioned. After their production the on-site storage lasts at least until the end of the service-life of the nuclear power plant; meanwhile, along with the proportion of the short half-life isotopes the heat generation also decreases. What follows is the transportation to and the final placement in the disposal facility. The BAF, currently investigated with high priority, may serve as a disposal facility within the borders. So far as the area has been given the required permits and the deep geological repository has been built, in accordance with the preliminary plans, it will allow, besides high-level wastes, the disposal of spent fuels.

The direct and the indirect impacts that may originate from such facilities basically depend on the functioning and the operation of the engineered barriers implemented in accordance with the specifications. Deep geological repositories may retain radioactive isotopes safely for several tens of thousands of years. The typical storage technology is the waste packages provided with the engineering protection specified, the repository chambers separated by a thick watertight concrete layer from the natural rock then filling up the chambers full of containers with rock gob and sealing with a concrete layer. From the data of the geophysical monitoring system installed in the repository chamber before filling up and closing, one can infer a leakage, which might occur as a direct impact related to the immediate environment of the deep geological storage area, however, the probability of this is practically negligible.

19.8.2.2.3 Indirect impact area of spent fuels

The temporary storage of spent fuels was discussed in detail above. Containers, as a means of temporary storage, allow for temporary storage for a few decades to several hundreds of years.

The determination of the indirect impact area also depends on the handling after the temporary storage. So far as the fuel is subject to reprocessing and recycling, i.e. the isotopes usable as fissile material are extracted for the purpose of further energy production, the transport route from the temporary storage area to the reprocessing facility and the environment of the utilizing facility shall be also taken into account. However, during the processing, a part of the isotope inventory is transformed to high-level waste, which is adequately conditioned (usually by means of vitrification) in the reprocessing facility. This high-level waste (in accordance with the actual legal environment) is returned to the nuclear power plant, from where it gets to the BAF deep geological repository in the way as it has been described for high-level wastes.

Spent fuels that are not recycled after the temporary storage are finally classified as high-level wastes, with further treatments, accordingly.

19.8.2.3 Area of transboundary environmental impacts

In the case of compliance with the strict instructions and process descriptions (procedures) during nominal operations with respect to the management of radioactive wastes, the environmental impacts originating from the management of radioactive wastes of different levels shall not reach or go beyond national borders. The same applies to the spent fuel assemblies.

19.8.3 IMPACT OF THE JOINT OPERATION OF PAKS II AND THE PAKS NUCLEAR POWER PLANT

The Paks Nuclear Power Plant and Paks II, amongst others, in accordance with the legislative provisions and the safety requirements, are completely independent facilities. Accordingly, their solutions for radioactive waste management are implemented and realized separately. The emission of radioactive wastes from one facility in normal operation has no effect on the operation of the other one. In the case of non-nominal operations the wastes produced, and rather the emissions might have an impact on the other facility, nevertheless, to the design-basis incidents strict safety scenarios are applied.

During the joint operation, the reactors of Paks II will be in the first decade of their operation; the on-site emplacement and the temporary storage of the radioactive waste and the spent fuels produced during this period will be solved in the auxiliary building of the primary loop located next to the containment and in the cooling pond; i.e. no dispatch of the radioactive wastes from the site and no manipulation of spent fuel assemblies used outside the containment are expected in the case of Paks II. Even if there will be dispatch for the purpose of disposal, the amount will be probably smaller compared to the dispatches from the Paks Nuclear Power Plant. Regarding radioactive wastes and spent fuels, the impacts expected during the joint operation will come forward almost exclusively on the units of the Paks Nuclear Power Plant due to the incidental environmental impacts of such wastes produced during the technical actions required and due at the end of the operation of the units.

19.8.4 IMPACT AREAS OF THE JOINT OPERATION OF PAKS II AND THE PAKS NUCLEAR POWER PLANT

The operation, the safety system and the waste management of the Paks Nuclear Power Plant and Paks II are done independently. This is true for the collection, the management, the storage, and the transportation of radioactive wastes too. Investigation of the common impacts of the joint operation is, therefore, justified solely by the localization; however, the radioactive waste management and the environmental impact(s) induced by it can be interpreted individually, and presented together.

What are common in the joint operation are the dispatch of the radioactive waste from the site by road, and the spent fuel dispatch by rail. In the joint operation, during the preparations for the 20 years of protected preservation, the transportation of low- and intermediate-level waste to NRHT is expected from the units of the Paks Nuclear Power Plant closed one after the other. The spent fuel assemblies removed from the closed units of the Paks Nuclear Power Plant are taken to KKÁT. The exact scheduling of the dispatch of the spent fuels to the ISFS Facility after the 50 years of cooling is currently not known; however, the delivery times of the wastes from the two facilities as well as the transport routes involved should be aligned in order to avoid the additive impacts.

The impact area of the radioactive waste emission during normal operation as well as the management and temporary storage of the wastes of the two facilities can be considered to be identical to the boundary of the safety zone.

Radiological impact above the limits that can affect environmental components can occur in respect with the radioactive wastes dispatched during the dismantling of the two facilities.

Transboundary environmental impacts of the joint operation of the Paks Nuclear Power Plant and Paks II can be excluded during normal operation.

19.8.5 IMPACT OF DESIGN-BASIS EVENTS

Note: Design-basis events will be named design-basis conditions by the expected revision of the NSC after its entry into force

19.8.5.1 Design-basis events

Pursuant to paragraph 3.2.2.3100 of the NSC, in the design of the nuclear power plant at least the internal events listed in Table 19.8.5-1. shall be taken into account; however, hereinafter, only those items will be discussed in more detail, in the event of which it can be expected that radioactive waste will be produced in conjunction with the event.

NSC 3.2.2.3100.	**Radioactive waste produced expectedly when the event occurs**
a) loss-of-coolant accident;	contaminated boric acid
b) break in the main steam system and the main feedwater system;	-
c) uncontrollable drop in the mass flow of the primary loop coolant;	contaminated boric acid
d) uncontrollable drop or increase in the mass flow of the main feedwater;	-
e) uncontrollable drop or increase in the mass flow of the main steam;	-
f) unintended opening of the valves of the pressurizer;	-
g) unintended operation of the emergency core-cooling system;	contaminated boric acid
h) unintended opening of the safety valves of the steam generator;	-
i) unintended closure of the main steam isolation valves;	-
j) steam generator tube rupture;	contaminated water, ion-exchange resin
k) unintended movement of the control rods;	damaged fuel assembly
l) uncontrolled pulling-up and ejection of the control rods;	damaged fuel assembly
m) instability of the active zone;	-
n) malfunctioning of the chemical and volume control system	contaminated boric acid, ion-exchange resin
o) rupture of a tube connected to the primary coolant loop of the nuclear reactor and partially located inside the containment or damage in a heat exchanger tube;	contaminated boric acid, tube material
p) incidents related to the management, transportation and storage of nuclear fuel;	damaged fuel assembly
q) dropping of heavy load when using heavy lifting equipment;	damaged tool, equipment
r) effects of fire, detonation and internal flooding and the initiating events they induce and	liquid and solid waste due to detonation
s) processes potentially inducing initiating event, especially flying objects including loose parts of the turbine, dangerous material leaving faulty systems, vibration, whipping movement of broken pipeline, effects of jets.	complex series of events → several wastes

Note: paragraph 3.2.2.3100 of the NSC, which the table is based on, will significantly change after the expected amendment of decree; furthermore the list of internal events to be taken into account will be considerably extended.

Table 19.8.5-1: Radioactive wastes produced in the course of internal events in off-normal operation.

Radioactive wastes of different level shall be meant by the right side of table 19.8.5-1., which, after collection, are taken to the auxiliary building of the primary loop for storage before further treatment. Besides the wastes listed in the table, a considerable amount of solid waste is also produced in the course of the work related to the recovery of the events but this does not exceed the capacity of the temporary storage facility.

In the case of pressurized water reactors planned leakages of the primary coolant loop are collected in a closed system. The amount of radioactive waste produced can be significantly reduced by the continuous monitoring of the radionuclides getting potentially into the secondary loop through the heat exchanger tubes of the steam generator due to non-hermetical sealing, by the extraction of the radionuclides getting into the secondary loop with water cleaning systems, and furthermore by the recirculation of the contaminated water into the waste treatment system of the primary loop. From the foregoing it follows, that in the course of design-basis operation events but in off-normal conditions, in the case of pressurized water reactors, the location and the time of the waste production is considerably limited.

19.8.5.2 Direct, indirect and transboundary impact

The collection and management of those types of radioactive wastes that are produced in the course of design-basis operation events but in off-normal conditions can be performed in the auxiliary building of the primary loop; hence the impact area of the direct environmental impacts of these wastes will be expectedly within the boundary of the safety zone of the site, and therefore analysis of the indirect as well as the transboundary environmental effects is not justified.

19.9 IMPACTS OF DECOMMISSIONING PAKS II

When investigating the decommissioning of Paks II, the indications in the life-time extension permit of the Paks Nuclear Power Plant shall be taken into account. The permit, amongst others, contains the possible scenarios for the decommissioning of the units with extended lifetime.

For the Paks Nuclear Power Plant, the accepted and optimum solution is the ”Deferred dismantling with 20 years of protected preservation”, for Paks II it is the immediate dismantling.

In relation to the dismantling of Paks II it can be fixed that the objective is the dismantling of the units and a ”clean-up” of the site which makes its further use possible.

19.9.1 DECOMMISSIONING WASTES

In the course of dismantling, especially that of the technological equipment and buildings inside the controlled area, a big amount of radioactive wastes is also produced. The radioactive wastes to be expected are as follows:

- reactor vessel and its internal components
- concrete biological shield surrounding the reactor vessel
- decontamination solution originating from the autonomous decontamination of the technological systems
- steel wastes of the technological systems
- contaminated building structures, claddings
- cables, cable covers
- thermal insulation materials
- etc.

The majority of the wastes produced in the course of the commissioning are of the very low-, low-, or intermediate-level category. Most of the high-level wastes come from the reactor vessel and its internal components as well as from the concrete biological shield surrounding the vessel.

Systems contaminated with radioactive isotopes are autonomously decontaminated depending on the level of contamination for the mitigation of dismantling worker exposure. This dismantling step reduces not only the dose level of the workspace but also the activity level of the solid waste produced.

Following the radiological classification of pieces of radioactive wastes dismantled, decision is made based on the results. Metallic wastes can be processed by the following technologies:

- decontamination for the purpose of clearance (followed by clearance)
- decontamination for the purpose of reclassification as VLLW
- melting
- supercompaction
- cutting and embedding into cement matrix.

After chemical treatment, decontamination solutions are solidified by means of cementing. The slag produced during melting is embedded into cement matrix.

Protective clothing, individual protection devices contaminated in the course of dismantling and other compactable wastes are supercompacted.

After cutting into pieces contaminated building structures are embedded into cement matrix.

The handling and the conditioning of radioactive wastes results in waste packages that meet the criteria for disposal.

Low- and intermediate-level radioactive wastes are taken to NRHT, high level wastes to deep geological repository for disposal.

19.9.2 DIRECT IMPACT AREA OF DECOMMISSIONING

The permanent shut-down of the nuclear power plant and the corresponding technological steps takes years. Demolition causes effects similar to that of the construction with the difference that, contrary to construction, a significant amount of mainly low- and intermediate-level waste is produced, which waste steam shall be managed on-site.

19.9.3 IMPACT AREA OF INDIRECT AND TRANSBOUNDARY IMPACTS OF DECOMMISSIONING

A large amount of low- and intermediate-level waste will be produced at decommissioning. The disposal of this large amount of waste requires significant mining and transport activity, however the impact area of the impacts originating from this will remain expectedly within the national borders.

Decommissioning takes place inside the country, at a significant distance from the borders. This means, that by taking the site into account, the occurrence of transboundary impacts may be possible only in very extreme cases. In the course of dismantling activities performed as planned, transboundary radiological consequences do not need to be considered.

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