



# PREDIS

## Deliverable 2.6

# Guidance on waste form qualification

29.2.2024 Version Final

Dissemination level: Public

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This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945098.



Project acronym PREDIS	Project title PRE-DISposal management of radioactive waste	Grant agreement No. 945098
Deliverable No. D2.6	Deliverable title Guidance on waste form qualification	Version 1.0
Type Report	Dissemination level Public	Due date M42
Lead beneficiary CVRez /SURO, Czech Republic		WP No. 2
Main author Soňa Konopásková (SURO)	Reviewed by Liz Harvey (GSL) Lumir Nachmilner (CV Rez) Anthony Banford (NNL), WP2 Lead	Accepted by Maria Oksa (VTT), Coordinator
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**Abstract**

The deliverable D 2.6 provides direct guidance and practical advice for Member States regarding the waste form qualification (WFQ) in practice. WFQ process consists of a set of tests aimed at the identification of mechanisms of mobilisation of radionuclides and chemical species contained in radioactive waste (RW). Their goal is to quantify these release mechanisms, subject them to safety assessment for both operational and post closure periods of a repository lifetime, and – if necessary – to formulate requirements either on the design of a disposal system or on the process of waste conditioning for storage/disposal. The main objective of this output is to define guidance and methods or procedures to achieve this specific goal. Their description includes necessary specification and extent of test(s) performed in the course of the waste form qualification process and their justification. Examples of tests of paramount importance are provided as tests quantifying decisive safety related parameters such as mobility, leachability, stability, homogeneity, corrosion rate, and other work form properties. Deliverable 2.4 'International approaches to establishing a waste acceptance system' issued under sub-task 2.3.1, and Milestones No.11 (document M2.8) have been used as a basic source of the information for D2.6 preparation. The information has been retrieved and supplemented using other public sources, relevant national programmes and documents, knowledge and experience of task participants and colleagues from other technical work packages of the EURAD project, personal contacts with external experts, IAEA documents, and data from other EC projects dealing with WAC and waste form qualification approaches.

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**Notification**  
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**Acknowledgement**  
 This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945098.



## HISTORY OF CHANGES

Date	Version	Author	Comments
04.08.2023	0.1	Soňa Konopásková	First draft for review
05.10.2023	0.2	Soňa Konopásková	Second draft for review
29.02.2024	1.0	Soňa Konopásková	Issued

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## LIST OF ABBREVIATIONS

CAT A	Category A - short-lived, low-level and intermediate-level conditioned waste containing limited quantities of long-lived radionuclides (in Belgium)
COD	Chemical oxygen demand
COD <sub>Cr</sub>	Chemical oxygen demand, using K <sub>2</sub> Cr <sub>2</sub> O <sub>7</sub>
DRs	Dose rates
DTM	Difficult-to-measure
EBS	Engineered barriers system
EPA	United States Environmental Protection Agency
FEPs	Features, Events and Processes
HLW	High level waste
IER	Ion exchange resin
ILW	Intermediate level waste
LILW	Low and intermediate level waste
LLW	Low level waste
NPP	Nuclear power plant
NUREG	Nuclear Regulatory (NUREG) Group, Canada
QA	Quality assurance
RN	Radionuclide
RW	Radioactive waste
RWM	Radioactive waste management
SA	Safety assessment
SF	Spent fuel
WAC	Waste acceptance criteria
WA process	Waste acceptance process
WAS	Waste acceptance system
WF	Waste form
WFQ	Waste form qualification

## 1 Introduction

The waste acceptance system (WAS) consists of two basic elements: (i) the waste form qualification (WFQ) process which is understood as a proof that a selected waste form is compatible with its designated disposal system, and (ii) waste acceptance criteria (WAC) which is a set of parameters selected to check whether the generated waste form complies with the requirements established for the safe operation of a disposal facility. It also means that waste form qualification is a process determining how radioactive waste shall be processed so that it complies with operational and post closure repository conditions compatible with legislation and safety conditions.

Generally, waste form qualification should be performed in the context of a specific set of qualification requirements, usually based on nuclear safety and radiation protection directives. These conditions are set for a specific facility and specific waste form.

Waste form qualification may be accomplished in consecutive steps: for example, first, by the qualification during the predisposal stage, then, in a second step, by the qualification of the finally realized system, during the operational/pre-operational period of a repository sited in a defined host structure.

During the first step, qualification may rely on activities performed without knowledge of a repository design (this is called 'generic qualification' or 'prequalification'). Prequalification may significantly reduce the future effort in facility specific qualification; however, the application of specific qualification requirements must still be demonstrably met.

During the second step, qualification of the waste form uses already described conditions of the disposal system: the project solution including engineered barriers system (EBS), and the conditions and properties of the host rock. EBS system is usually composed of more barriers, which should provide one or more safety functions. By provisions of national legislation and/or regulators, final waste form can also be declared as a part of EBS, and it should provide safety functions as containment, isolation, stability, and potentially other features. These waste form safety functions should be described in detail in national legislation or in the quality documents developed by repository operator, and possibly approved by regulatory body before the waste form is used in operation.

*Note. The system of safety functions complies with the principle of defence in depth, which means that the application of more than one protective measure for a given safety objective, such that the objective is achieved even if one of the protective measures fails.*

Data collected in the initial phase of the project (see Deliverable 2.4) have provided little information on the application of waste form qualification methods in national programmes. Most of the countries concentrated on the waste acceptance criteria (WAC) definition and their control in the acceptance process. This approach does not exactly express the needs of the waste form qualification data. The process of waste qualification must detail how a waste form has been selected and verified for approval for disposal in a correctly described disposal system, i. e. for the repository design situated in special geological and hydrogeological host rock conditions. This Information has been retrieved and supplemented using public sources, relevant national programmes and documents, personal contacts, IAEA documents, and data from other EC projects dealing with WAC and waste form qualification approaches.

Typical waste form qualification approaches used in Member States' programmes to determine the suitability of the waste form for storage and disposal, especially of low- and intermediate-level radioactive waste, were described in Milestones No. 11 (document MS2.8). This document then served as the basis for the preparation of this WP 2.3.3 final output - Deliverable 2.6 Guidance on waste form qualification.

On the basis of the collected information, the expert team evaluated information on national approaches, practice and requirements for the process of qualifying the form of waste, and with this

in mind, guidance was proposed on how to provide evidence that the considered form of waste is suitable for disposal (Delivery 2.6).

## 2 Basis and outcomes

The WFQ process consists of a set of tests aiming at the identification of mechanisms of mobilisation of radionuclides and chemical species contained in radioactive waste (RW). Their goal is to quantify these release mechanisms, subject them to safety assessment for both operational and post closure periods of a repository lifetime, and – if necessary – to formulate requirements either on the design of a disposal system or on the process of waste conditioning for storage/disposal.

Typical mechanisms studied include the following effects:

- Water transport within and outside the waste form and its alterations as a consequence of different destructive effects, such as temperature changes (freeze-thaw cycling), long term contact with water, chemical degradation or reaction within WF, biodegradation, corrosion processes, etc.
- Gas generation and migration, either radioactive or explosive or inactive and their influence on containment functions of a repository
- Structural stability and its deterioration due to radiochemical, biological, or chemical processes (corrosion of WF and its container)
- Compatibility with engineered barrier system of a repository, such as the chemical interaction of a WF and migration through and interaction with particular elements of the barrier system

The WFQ process will be completed using the definitions of the WAC, i.e., parameters, to be verified in acceptance procedures at the storage or disposal facilities, and their values ensuring that the release of radionuclides, gasses, and chemical species will not exceed limits and conditions defined in the safety case of the facility.

WFQ conditions are defined as follows:

- Compliance with national waste management systems
- Compliance with waste disposal safety, i.e. radiation safety, nuclear safety, monitoring, emergency preparedness, physical protection, criticality prevention (not always relevant for LILW) for programs operating/constructing a repository
- Compliance with generic criteria for organizations, which do not operate/construct a repository
- Quality of final waste form production process (waste producer level)
- Control of final waste form production process (waste producer, repository operator, regulatory body)
- Compatibility of the waste form with expected radioactive waste streams and with the facility
- Related WAC complexity, relevance, traceability
- Quality of waste acceptance process (repository operator, regulator)
- Quality of disposal process (repository operator, regulator)

### 2.1 Waste form qualification

Waste form qualification is a process which leads to licensing of a waste form to be used in a particular disposal facility. Establishment of the WAC is an important part of the process.

### 2.2 WFQ relation to safety

Waste qualification parameters are the part of parameters basis serving as an input to safety assessment. Using this input, the safety assessment results have to meet legislative and regulatory safety conditions, namely in the sphere of radiation effects.

In a wider sense, safety is not only related to dose evaluated as a possible effect of radionuclides release in the post closure period. Safety is usually understood as:

- Nuclear safety, i.e. exclusion of the release of radionuclides, prevention of criticality and prevention from heat transfer. The concentration of materials which could lead to

accumulation of critical amount of fissile materials in post-closure repository environment is generally low because of the limited value of mass/volume activities.

- Radiation protection, i. e. limiting of doses of persons and radiation impact on environment using the radiation protection tools. It includes limiting of the doses to repository and waste management workers during operation in predisposal and disposal activities.
- Technical safety, i. e. safety related to safety systems and barriers performance.
- Monitoring, i. e. the control of a zero state, volume or mass activity in concerned environmental components, control of workplaces (dose rates, surface activity), implemented in operational, and closure/post closure periods.
- Emergency situations management, i. e. safe management of events that could lead to uncontrolled release of materials or to unacceptable doses.
- Physical protection of the facility.

A necessary condition for ensuring general safety at work is compliance with the above safety requirements, as well as high safety culture, etc. Considering the principle of defence in depths, evaluation of safety is closely related to:

- Radionuclide composition of the waste form (short lived, long lived, radiotoxicity)
- Waste origin (from nuclear power plants – operational waste, spent fuel, waste from decommissioning, from institutions – mixed waste, spent sealed sources, organic waste)
- Waste activity (VLLW, LLW, ILW, HLW, spent fuel)
- Intended or practised waste management activity (storage, disposal, both storage and disposal)
- Type of facility (surface trenches, surface, near surface, underground, and geological repositories)

*Note. Clearance is not considered as a waste management option in this document.*

All requirements specified in the process assuring safety could lead to formulations and/or supplements to WAC system, preceded by a WFQ procedure, implemented for each of expected waste stream. Safety concerns relating to emergency situations are not included to WFQ procedure: dose that would arise in the course of emergency scenarios, should not lead to restrictions in WAC derived from doses. In general case, the probability of an emergency scenario is considered as very low and it should not affect normal operating conditions.

### 2.2.1 Waste acceptance system

Waste acceptance systems will be created at different levels: nationally, approaches for its establishment should be defined, while their implementation would be specific for each storage/disposal facility. It should be noted that storage should be planned and integrated with future disposal options.

#### 2.2.1.1 Classification of RW for disposal

Classification of RW is proposed in the IAEA GSG-1 document 'Classification of radioactive waste' [1]. In principle, this document advises how to link specific waste categories to an adequate disposal option. In practice, national waste acceptance systems for disposal should include

- Disposal of very low-level waste (VLLW) which can be performed in surface trenches
- Disposal of low level (LLW) requires engineered structures which could be constructed at the near surface
- Disposal of intermediate level waste (ILW) is recommended at greater depths, of the order of tens of metres to a few hundred metres, and it requires a greater degree of containment and isolation than that provided by near surface disposal
- Disposal of high-level waste (HLW) and spent fuel (SF) if relevant), requires facilities in deeper geological formations with adequate engineered barrier system.

It should be noted that the selection of a proper disposal solution depends on climatic conditions (e.g., in arid areas LLW is disposed of in near surface trenches). Also, it is possible to build facilities allowing co-disposal of several waste categories (e.g., ILW & HLW), thus, the implementation of this classification scheme is subject to the national conditions of every country.

A national RW acceptance system has to encompass all RW generated in the country, it should consider old (legacy), current, and future waste arising.

RW inventory (i.e. volume, total activity, and specific or volume activities of important radionuclides in the waste) are key inputs for safety considerations that affect the selection of adequate disposal options. The effect of short-lived radionuclides will be nearly negligible in subsurface and geological disposal systems, but it may be crucial for waste storage and near surface disposal systems. The effect of long-lived radionuclides has to be assessed in all aspects of management, because of their higher radiotoxicity, especially in the case of radionuclides alpha. The RW inventory has often to be considered as a whole, in the case of operational waste from nuclear power plants; this type of RW may not distinguish between low level and high-level values of different radionuclides activities, and between various half time values.

Nevertheless, RW composition, form, and content of important radionuclides have to meet all requirements imposed by the conclusions of safety analyses.

### 2.2.1.2 Final waste form options

For disposal, solid or solidified RW is considered as the acceptable final waste form. Solidified final waste form should be compatible with the engineered system of the repository, if relevant, and with the host rock. This condition is usually not directly expressed as a special WAC, but the solidification matrix and the properties of the final waste form, such as stability, migration parameters, leachability, and strength resistance, are considered as inputs to repository safety assessment. Satisfactory results of safety assessment justify the use of the evaluated final waste form.

The final waste form in most cases consists of solid or solidified waste and a container. However, individual solid pieces could usually be disposed of without container under conditions set down by WAC.

Disposal of liquid waste is not desirable, even if it has been placed in containers with relatively long-life times, such as high integrity containers.

Containers used for disposal should explicitly be specified in the WAC. Properties of containers that are considered in repository safety assessment have to be defined and approved by a regulatory body and regularly checked in the QA and waste acceptance procedures. No other containers are permitted without explicit approval of repository operator supported by relevant safety assessment and/or the approval of the regulatory body, if they have an important impact on repository performance.

In the case of LILW and short-lived RW, not all container properties are safety relevant, e.g. a steel drum could be used just as a handling unit without any long-term safety function. However, container integrity, mass, volume, strength resistance, corrosion resistance and life time can be used as inputs into safety assessment. The sensitivity of repository system performance to container properties may affect the WAC definition. Safety functions of the container shall be assessed together with safety functions of other barriers (engineered and geological) according to the defence-in-depth principle.

Solidification media must guarantee long term stability of the final waste form by means of waste immobilisation, assuring transport parameters of the waste form, such as leachability, solubility, diffusivity and distribution coefficients, are as low as achievable. Not all solidification media are suitable for all types of waste: their improper use can lead to deterioration in waste properties during repository lifetime. Thus, the selection of an effective conditioning process is done on case-by-case bases while also considering WF performance during transport, storage and disposal.

Certain RW requires specific attention when selecting suitable conditioning process. Some examples of problem waste include sludge, chemically aggressive waste, spent ion exchange resins, organic waste, waste containing highly mobile radionuclides, such as isotopes of H, C, Cl, I. Potential interactions between solidification media and waste should be assessed as well as the effectiveness of available conditioning processes.

The solidification of some types of waste, such as spent sealed sources, may lack an immediate safety rationale. Although some types of this waste can be disposed in disposal systems for LLW, they remain an important hot spot of radioactivity: RW heterogeneity needs to be thoroughly considered in safety assessment and relevant conditions of disposal have to be set in WAC.

### 2.2.2 WAC definition

Final waste form properties that will be checked during waste processing and waste acceptance have to be set out in WAC, including requirements on frequency and the method of their control, specified for each of the waste forms that are approved to be disposed of in the repository.

In the case that the disposal site is available, or at least in the siting or construction phase, WAC definition should be formally developed in cooperation with the waste producer and the disposal site operator, while respecting national legislative provisions. Waste streams information including inventory, final waste form for disposal, and waste producer data are the principal inputs to the formulation of the WAC.

Countries, that do not operate a repository and may not have identified its final site, have to develop so called generic criteria, respecting national legislation, if it supports disposal as a waste management option. IAEA and NEA recommendations are to be taken into account in this process. Waste streams identification is helpful, if available

#### 2.2.2.1 Generic versus site specific WAC

WAC can usually be allocated to one of the following groups:

- *Administrative criteria*, which can be used to trace and define the waste on its way from producer to repository, or that impose general restrictions on the waste
- *Technical criteria*, which are usually related to repository technical components, but can be safety relevant as well
- *Safety criteria*, which are derived from radiation protection and/or nuclear safety requirements.

Administrative criteria are usually defined using legislative provisions or international recommendations, where future disposal site properties have a minimal impact on their wording.

Technical criteria can be developed using general technical and safety assumptions, however, without knowledge of parameters of the disposal facility or its site.

Site specific safety criteria cannot be derived without knowledge of site-specific data and facility design. Generic WAC are sometimes intended to be an independent or overarching set of general requirements applicable to multiple facilities, which will continue to be relevant even if site specific WAC are available.

#### 2.2.2.2 WAC quantification

Quantification is applied for both technical and site-specific safety criteria, defined as a value or as an extent of values. Typical examples are:

- Weight of waste package
- Dose rates on a waste package surface or at a defined distance from the surface
- Dose rates at the surface of closed vaults
- Mechanical strength

- Volume / mass / total activity of important radionuclides
- Waste package type and dimensions
- Durability of the waste package
- Homogeneity of waste expressed mainly by presumable deviations from calculated average volume activity in the final waste form, as well as by distribution of different types of waste (dimensions, materials, exposure rates, etc.)
- Leachability of final waste form
- Mobile activity in the repository
- Surface contamination of the container or individual piece of RW.

WAC need not to be derived from safety assessment in all cases and some criteria can hardly be quantified. Relevant conditions can be defined using general directives as prohibition of presence of free liquids, corrosive components, explosives, toxic materials, inflammable material, microbiological decay initiating components, and administrative criteria described above. Procedures involved in demonstration of the absence of these items include inspections at waste generators/processors as direct or indirect measurement of their values is practically very difficult.

### 2.2.2.3 WAC control methods

Each criterion principally consists of three elements: a parameter, its value, and a method of its demonstration. Thus, the definition of a control procedure is a necessary part of acceptance process. Control is employed as a tool for demonstrating compliance in the transition of the waste among stages of the whole process of waste management, starting with waste generation until its disposal. The control procedure shall be defined in accordance with the control system and operational instructions, typically in the context of QA programme. Typical examples of control methods are:

- Dose rate measurement
- Surface contamination measurement
- Direct activity measurement (gamma & beta spectrometry tomography, gamma scanning)
- Indirect measurement and calculation (gamma correlation, alpha measurement via neutron activity,)
- Calculation of average and maximum activities within the waste package
- Calculation of mobile activity
- Compressive strength determination
- Weighing
- Measurement of dimensions
- Leachability test and calculation of leached ratio
- Visual control
- Waste package integrity control
- Control of waste producer labelling and repository operator labelling
- Documentation of position in a disposal vault – records, photo documentation, schematic drawing.

### 2.2.2.4 WAC independent control

WAC independent control is provided by regulatory body, and usually consists of:

- Visual control (waste position, waste integrity, labelling)
- Control of administrative documents, certificates, and licences
- Records and waste self-conducted control
- Random measurement of dose rates or spectrometry
- Control of conformity of disposal procedures with operational instructions and limits and conditions, including WAC
- Control of the use of repository activity limits.



### 2.2.2.5 Generic WAC

Generic criteria are not based on site specific data.

For practical reasons, it is advantageous to formulate a set of generic criteria in the early phases of waste management, i.e., during waste generation, processing, and storage. This set of generic criteria can later be approved in the licensing process, usually in an optimized form, in the form of repository limits and conditions.

Generic WAC are usually developed using:

- Type of radioactive waste (NPPs, institutions)
- Characterisation of waste streams that are generated (operational waste of standard composition, individual sources, spent sealed sources, waste of significant volume, decommissioning, other...)
- Radionuclide composition (natural vs. artificial radionuclides, half-lives, content of long lived radionuclides, choice of important radionuclides, radiotoxicity).

In the WAC, it is not necessary to limit the content of non-safety relevant radionuclides present and / or declared in the RW. Activity, half-life, mobility, and radiotoxicity are radionuclide properties that should be used to determine if the radionuclide is to be included into WAC or not.

Other possible conditions for disposal refer to the facility status, potentially specified in the future licence and can be specified in generic WAC:

- RW containing artificial radionuclides in the case that the facility will be licensed as a nuclear installation
- RW containing natural radionuclides.

Repository specific criteria are usually specified as a part of licensing process starting in the construction period, and they are later updated during the licensing of operational and closure periods, as a component of repository limits and conditions.

#### Administrative criteria

These WAC should include:

- Origin of waste. It can be decided in the frame of the siting process that the disposal site will be used for a RW of a defined origin. This WAC can be developed from the data provided by the waste producer, and subsequently with the repository operator, and/or municipalities as a part of the operational licensing approval process. The regulatory body's role in the licensing process has to be considered
- Documentation of RW (passports)
- Unique identification of waste packages.

#### Technical criteria

Technical criteria that can be defined before the start of repository operation include the following items:

- Operability of technical devices needed for safe operation
- Operability of measurement devices including measurement of radiological values as dose rate, surface contamination, activity
- Meteorological conditions during the disposal process
- Operability of disposal vault / chamber / disposal site (generally), e. g. absence of free water, traceability of drainage system, correct collection of drainage waters, operability of drainage vault
- Operability of cranes and other handling devices
- Mass of a disposal unit
- Type and size of a disposal unit
- Stackability of waste packages
- The method for control of waste position in the disposal site

- Staff duties, commitments, and liabilities.

### Safety criteria

It is rather limited to formulate safety criteria without site/facility characteristics. Quantification of general scenarios based on RW handling, transport, as well as inadvertent intrusion during operation and/or after closure can be used to define limits on activities in disposal units, and to derive mass / volume activities in the waste. In practice only, the following ones can be determined:

- Dose rate on the surface of disposal unit and/or in a defined distance from the surface
- Surface contamination
- Volume / mass activity of important radionuclides in the final waste form
- Protection systems needed for safe handling of waste packages.

### 2.2.2.6 Site specific WAC

Site specific criteria can be developed as an output of the site selection process and can be specified as administrative, technical and safety criteria (radiological and non-radiological), as it is in the case of generic criteria. In practice, administrative and most of technical criteria are taken over from the set of generic WAC. However, it is advisable to revise those criteria that are linked to the facility design and operation to ensure their compatibility with the built disposal facility.

### Administrative criteria

The principal generic requirement is that the disposal facility has to be approved in the regulatory licensing process for each phase of its operation. The licensing process has multiple components that vary during siting, construction, operation and closure of the repository. The compliance with national legislative provisions should clearly be demonstrated, including the update of administrative activities and documentation.

### Technical criteria

Technical criteria are usually related to repository operation issues. But, with respect of site position, geometry and topographical situation, technical criteria related to handling, transport and connected devices shall be revised and updated. Special attention will be given to technical solutions that may relate to safety functions of the disposal facility, e.g., to designed (existing) barriers.

### Safety criteria

Safety criteria are formulated with respect to safety assessment results and recommendations. They include activity limits in waste packages, disposal vaults and the whole repository. However, for generic WAC these limits cannot be supported by safety analyses, therefore, their definition could be based on estimates and parallels with existing similar disposal facilities. Also, operational safety measures should be adequately addressed.

In the WAC, it is not necessary to limit the content of other than safety relevant radionuclides present and / or declared in the RW. Initial activity, half-life, mobility in the waste form and repository, and conversion dose factor are radionuclide properties that could determine whether the radionuclide is to be included into WAC or not.

## 2.2.3 Safety case

In the safety case, it must be proved that RW, repository performance, and host site capacity are in accordance with all safety legislative provisions. WAC developed in the safety assessment as a part of the safety case and other relevant WAC must address all waste that will be accepted in the repository, as it is described in the safety case.

Other waste can be accepted only after safety case re-evaluation, updating of WAC and their approval by the regulator, if relevant. Waste, which was found to be unacceptable (waste that does not meet WAC), cannot be accepted without change of its properties and disposal conditions

(inventory, waste form, container, geometry of disposal), and requires re-assessment in the safety case frame, and the relevant re-licensing procedure.

### 2.2.3.1 Position of safety assessment in the safety case

The Safety case is the complete set of arguments and documentation supporting the justification that a disposal system fulfils all requirements of radiation protection, nuclear safety, technical safety, emergency preparedness, monitoring and physical protection. The measure of compliance with these requirements is usually proved quantitatively, but also qualitatively, using safety analysis and safety assessment. Quantitative expression of safety is then formulated in WAC. The safety assessment is submitted to regulatory body as a part of safety case in the course of the permission process.

### 2.2.3.2 Safety assessment resources

Safety assessments must primarily use the known state of the disposal system and its components, as:

- Type of RW – waste category, origin of RW and half-lives of radionuclides present in the waste, waste radiotoxicity
- RW inventory to be disposed of in the repository: final waste form including waste matrix, containers and their properties, and radionuclide composition, RW volume to be disposed of, and available volume in the repository
- Repository type – surface (vault system, trench), subsurface (rock cavity, abandoned mine, drill etc.) underground, nuclear installation or not
- Location and geometry of the repository
- Repository engineered system (barriers, filling, sealing, drainages, retention systems for water collection)
- Hydrological, hydrogeological, geotechnical, geological properties of the host rock structure, its seismic potential, and hazard
- Radiological capacity of the site (anticipated total activity to be disposed of).

### 2.2.3.3 WAC derivation

Radiological properties of waste that can be disposed of in a potential site can be derived and/or justified using the means of safety assessment. Key inputs are collected by investigation of a waste form performance in the WFQ process. A procedure for WAC derivation for surface and subsurface repository system is explicitly described e. g. in IAEA TECDOC-1380 [2]. The possibility of derivation of radiological WAC in the case, that the RW inventory and the host site are uncertain, shall be described as well.

Input data for safety assessment procedure can be found in IAEA TECDOC 1380 [2], containing, among others, data for most of relevant materials and important radionuclides used for evaluation of scenarios included in the WAC derivation system. For other materials and radionuclides, data can be found in IAEA handbooks and documents, e.g., IAEA [Safety Reports](#) No. 19 [3]**Error! Reference source not found.**, TRS472 [4], and in documents from NUREG and the EPA. For most countries, the principal source of data will also be national legislation.

In principle, total activity in the disposal system (often referred as radiological capacity of the site) is derived from normal evolution scenario and it should not be set for a general site, i. e. if the hydrogeological properties of the site are not known.

The activity limit in lower volume units (vaults, chambers, drums) or mass activity of important radionuclides is usually derived from scenarios describing direct contact of personnel with the waste (such as residence on site, intrusion, worker activities).

Derivation of non-radiological criteria can rely on safety calculations, safety assessments, for example in the case of compressive strength and content of chemical components of the waste form. Repository impact on the properties of the waste package is taken into consideration as well. Exclusion of different substances, such as hazardous or explosive material is also a way of definition of the non-radiological components of the WAC.

### 2.2.3.4 Dose constraints

The total, volume and mass activities are derived by comparing calculated doses to dose limit set by regulator:

- For normal operational scenarios, a fraction of the general annual dose limit is set as a constraint, typically 0.3 mSv/yr, 0.25 mSv/yr, 0.10 mSv/yr – depending on the national approach.
- For non-standard scenarios as residence on-site, and intrusion, general dose rate limit is used, typically from 1 mSv/yr up to 10 mSv/yr – depends on national approach.

Emergency situations are evaluated in the safety assessment as well; the objective is to describe potential effects of incidents or accidents. But these estimated doses are not used for WAC derivation: dose limits by emergency situations are usually much higher than general dose limits for population and workers.

### 2.2.3.5 Recommended scenarios

The full set of inputs needed for the scenario description, model construction and calculation are provided by FEPs (features, events, processes) database. The FEPs database is a quality assurance and knowledge management tool as well. It can involve data from national research activities such as laboratory and field tests, existing national and international databases, and expert judgement. FEPs database should include all processes affecting safety applied in the scenario description, and to underpin and justify the choice of decisive events.

It should be noted that FEP's provide inputs for waste form qualification tests as they indicate critical parameters and processes leading to waste form deterioration or even destruction.

The following scenarios must be evaluated for derivation of WAC in subsurface disposal systems:

#### Normal evolution scenario – off site residence

Normal evolution scenario is also known as the project scenario or the off-site residence scenario. It describes the projected evolution of the disposal system as a whole, and usually serves as a tool for the derivation of the total radiological capacity of the site. The scenario describes disposal system evolution after repository closure, i. e. after application of all parts of EBS as sealing, backfill, and cover. All transport pathways such as drainage and water collection systems are assessed to minimise water infiltration.

Scenario processes are usually described in three subsystems: near field, far field and biosphere. All processes are described using the principle of projected or probable system and subsystems performance, with a reasonable measure of conservatism.

Near field processes evaluation has the objective of source term quantification, i.e. quantified release from repository (Bq/m<sup>3</sup>, Bq/y, Bq/(m<sup>2</sup>.y) for all important radionuclides. Far field processes evaluation leads to quantification of dilution and distribution of contaminants in the hydrogeological system of the host rock structure and the most important output from the analysis is the activity of important radionuclides in biosphere components, i. e. in groundwater, surface water and soil. Biosphere evaluations use these results to quantify the activity in other component of biosphere (plants, animals and their products, fish) and potential doses of representative person.

#### Alternative scenarios – off site residence

Alternative scenarios are developed to evaluate the case(s) of less probable system or subsystem performance. For ILW disposal systems, possible alternatives of system evolution are those that do not lead to incidents or accidents, but that still differ from the projected state, e. g.:

- Different type of radionuclide transport in the near field as a consequence of earlier drainage of water to the disposal position and / or shorter barrier lifetime

In most of modern disposal systems, in the after-closure period, radionuclide transport should be driven by diffusion at least for several thousands of years. This can be supported by preventing repository space from water penetration and excluding the possibility of creating fractures. The cover of the repository should have a long lifetime as well as the materials serving as solidification media, sealing and backfill. If these conditions are not guaranteed, the near field situation could change earlier than it was designed for and advection starts to be the primary process of migration in the near field. The effect of this is a higher concentration source term with an earlier maximum.

It is strongly desirable to evaluate the consequences of the situation described, as a part of the safety assessment.

### Intrusion scenario

Intrusion could lead to direct contact with waste, in some cases the radiation effects are decreased by shielding and distance, depending on situation described in the scenario. It usually leads to limiting volume / mass activity of important radionuclides, regarding doses induced by external irradiation and/or inhalation, in spite of the fact that dose limits used for judgement of scenario consequences are less restrictive than dose limits used for judgement of the normal evolution scenario consequences.

### On-site residence

On site residence is used for the derivation of volume activities in higher repository units, such as vault, trenches, and chambers. The scenario describes evaluation of all the usual pathways of irradiation, i. e. external irradiation, inhalation and ingestion assessing soil contamination, plants and animal product consumption. Usually, it leads to a limiting volume activity of important radionuclides. Dose limits used for scenario assessment are less restrictive than dose limits used for normal evolution scenario assessment.

### 2.2.3.6 Safety functions

Safety functions are usually defined for the system as a whole and/or for important subsystems (barriers). Even if they are mostly dealing with the repository barrier system, knowledge of their properties allows for the better definition of potentially destructive processes and for the assessment of negative impact of degraded waste forms. For robust systems, safety should not depend on a state of one subsystem providing the safety function (defence of depth). Typical safety functions are isolation, retention, dilution, stability, retardation, and mitigation. One subsystem can provide more safety functions to the system, e.g. a container is responsible for the stability and isolation of RW. The role of safety functions is considered in the safety assessment as an input, or it is evaluated as an output of the safety assessment procedure. Several WAC are derived using the knowledge of safety functions definition and role. Examples are final waste form stability and lifetime, activity of important radionuclides, etc.

Safety functions are described for a designed repository, but related properties of components responsible for safety should be defined much earlier, considering the role of the waste properties in the future disposal system and the need to take them into account during design.

### Barrier considerations

Safety functions are usually specified for engineered subsystems of the repository, because of quality and control conditions. Isolation is the principal safety function of a barrier system. In WAC, this function can be expressed as a measure of:

- Container lifetime
- Other engineered barriers lifetime.

Container isolation properties should be defined in the predisposal period, e. g. as the requirement for container integrity extending the necessary storage period needed before the waste is accepted for disposal.

### Hydrogeological conditions

In certain cases, safety function of a host structure could also be considered, e.g.:

- Retention
- Dilution
- Transport time for mobile radionuclides
- Heat dissipation.

From a quality and control point of view, properties of the host structure cannot be intentionally maintained or changed in the operational and post closure periods of the repository and/or affected using quality and control tools.

## 2.3 Sensitivity, uncertainty, optimization, justification

### Sensitivity issues

Those WAC that limit activity in the repository, need not necessarily be developed for all radionuclides present or declared in the RW. This would lead to difficulties in the waste processing and treatment process as well as to challenges during acceptance process. Some radionuclides can hardly be directly measured, and in addition, the activity of waste and the possibility of their detection can change during the period from waste processing to its disposal. But, regulatory provisions concerning inventory description have to be respected.

Safety relevant (important) radionuclides can be specified before the final waste form is produced using a screening method relevant for probable or existing type of repository. The screening method should be presented in the safety case, if relevant. The safety assessment can now be speeded up and simplified using computer codes. Radiation consequences can be quantified for all of the radionuclides present in RW, and it is possible to exclude those having minimal impact on operational and long-term safety. Generally, these could be radionuclides having most of the following properties: low initial activity, short half live, low mobility in host rock, and low dose factors.

### Uncertainty issues

Formulation of WAC can be affected by uncertainties of inputs of the safety case, which could be uncertainties in:

- Radionuclides inventory
- Final waste form properties
- Near field transport potential, i. e. barrier system performance
- Host structure uncertainties e.g., water transport pathways, fracture system, transport parameters of species in the host structure, type of transport (advection vs. diffusion)
- Description of the disposal system performance including subsystems, i. e. models
- Computer programmes used for model calculation

Uncertainty can be decreased using the appropriate research methods, such as laboratory and in situ experiments, relevant tracer tests, and analogue studies. In the case of models and computer

programs, verification and validation are the methods that could lead to model / program credibility and its standardization.

Confidence building can be improved using parallel calculations by independent bodies and using the principal of realism vs. conservatism. The measure of using a conservative approach is important. Conservatism is more defensible, but excessive conservatism could lead to underestimation of repository radiological capacity, which could have economic, public acceptance, and operational consequences.

### Optimization

WAC document is developed using a wide range of assumptions and input data. Not all of them can become a part of limiting requirements. For operational reasons, it is desirable to have a unified version list of criteria which could be approved on all levels of WAC development and approval: waste producer – repository operator – regulatory body.

WAC must be implemented in the national waste acceptance system respecting the limitation and conditions presented earlier.

WAC must follow regulatory directives, must be respected by a waste producer, and verified by the repository operator. Both, waste producer and repository operator are subject to regulatory body control. WAC must have the potential to be defended by the waste acceptance process. For that, the extent of criteria must be aligned to operational conditions of the waste producer, the repository operator, and to the regulator.

It is helpful, if waste producer can provide information on the waste generated before general criteria are developed. In co-operation with waste producer and/or bodies, which are licensed for waste processing, storage and disposal, WAC can be developed with all stakeholders to avoid future technological and acceptance challenges.

### Justification

In most cases, WAC are justified by the regulatory approval as a component of a facility licence. In earlier predisposal stages, WAC can be generally formulated with intention of later approval, considering the existing waste streams, potential host site capacity, technological resources of waste producer and repository operator, cost benefit issues, regulatory framework, public acceptance, and other.

## 3 WFQ parameters

### 3.1 National practice

Information from national overviews collected in starting phases of PREDIS project include a number of WFQ parameters. These parameter sets can be different for various proposed waste forms resulting from using different solidification media, which can be bitumen, cement, geopolymers, glass, and some others. Furthermore, non-solidified waste can be approved for disposal as well, under specific conditions, defined in WAC. Ashes, compacted waste, and single contaminated waste pieces (metal, concrete etc.) are usually considered as non-solidified waste.

The principal extent of parameter definition and checking arise from the requirements of nuclear safety, radiation safety, technical safety, and monitoring.

The design of future disposal systems has to include the system of engineered barriers: the EBS system is required to accomplish one or more safety functions, compiled with the principle of defence in depth, i. e. the application of more than one protective measure for a given safety objective, such that the objective is achieved even if one of the protective measures fails.

Waste package specifications provide a set of quantitative conditions to be satisfied by the waste package for handling, transport, storage and disposal.

Specifications for conditioned (usually solidified) radioactive waste should be established to ensure that the waste package satisfies the relevant acceptance criteria for storage or disposal, and the transport requirements. The radiological characteristics of the waste should be identified at an early stage of waste form qualification, as the principal data describing the waste form. Other specifications of the waste package may be divided into four main topics: chemical and physical properties, mechanical properties, containment capability, and stability or robustness.

Stability or robustness of the waste form concerns the capability of the waste package to retain radionuclides over extended periods of time.

The extent of parameters which should be assessed during the process of waste form qualification is listed below.

In national programmes, a range of methods may be used for waste form qualification, but procedures exist for assuring justification for special waste forms produced during the predisposal period, which can be defined as possible methods for waste qualification. These methods cover the wider system of waste form description so that it can be controlled in the waste acceptance process later applied in operated storage and/or disposal facilities.

Table 1 summarizes the state of national RWM programmes, including WAC existence, their extent and control methods, if available.



Table 1. Support of waste qualification 0

Country	Responsibility	Legislation	Facility	Treatment	WAC	Generic	Site specific	SA	WA process
<b>AUSTRALIA</b>	state	RWM Act	LILW repository	vitrified, cemented	approved in licensing procedure	physical, chemical, radiological (dose rates), labelling	not presented	nuclear safety	exists, not specified
<b>AUSTRIA</b>	government		long term storage	incineration, compaction, drying	none				gamma scanning, DRs
<b>BELGIUM</b>	Belgatom ONDRAF/NIRAS  • Current operator of a class II facility (remediation of a site of former nuclear activity)  • Future operator of a class I disposal facility (surface disposal)	in progress	CAT A, Dessel intended  cAt project in Dessel, i.e. the integrated project for surface disposal of Belgian category A waste	cemented	formal, minimum requirements  WAC are defined by ONDRAF/NIRAS based on General Rules.  These General Rules are laid down in formal legislation (Royal Decree)	revision in predisposal, physical, chemical and biological characteristics	63 radionuclides, activity, natural RN separately		compliance, monitoring, fully specified
<b>BULGARIA</b>	State Enterprise "Radioactive Waste" (SE RAW)	Act on the Safe Use of Nuclear Energy (ASUNE),  Strategy for spent fuel and radioactive waste management until 2030	National Disposal Facility (NDF) for the disposal of low - and intermediate level short-lived RAW - under construction  Centralized storage facility for RAW from nuclear	conditioned, packed in reinforced concrete packages, vitrified	exists, and under further improvement	general wording and physical, chemical and biological characteristics	available	nuclear safety, radiation protection, monitoring	waste preparation and characterisation, acceptance procedure

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			applications - in operation						
<b>CROATIA</b>			interim storage	none	for storage	extent available	extent available		dose rates, visual control
<b>CZECH REPUBLIC</b>	state, SÚRAO	Atomic Act	three in operation, one closed	cemented, bituminized, geopolymer matrix	exist	available	available	nuclear safety, radiation protection, monitoring	waste process, acceptance procedure
<b>FINLAND</b>	Fortum, TVO, Posiva		two in operation	exists	exist	available	available	not available	not available
<b>FRANCE</b>	ANDRA (French National Waste Management Agency)	Waste Act and other	VLLW – DF at at Morvilliers LL-IL/SLW – ANDRA DF at Soulaines LL/LLW - Disposal in project HL-IL/LLW - Project in Meuse / Haute – Marne - DGR	segregation, compaction, RAW is conditioned using mainly homogenous cement encapsulation or encapsulation by injection, placed in metal containers or concrete shells, (HLW- vitrification)	Exist Based on Fundamental Safety Rules (FSR) from authority (SNA) FSR relevant for disposals SNA updates them as guidelines: RFS I.2: about surface disposals (LL/IL-SL), RFS III.2.e: Solid waste package agreement conditions for surface disposal, RFS III.2.f: guideline relative to definitive disposal into deep geological facility	Exist General specifications (applicable to all packages) - General technical requirements (ACO SP ASRE 99-001)	radiological		Andra requirements e.g.: <ul style="list-style-type: none"> <li>• Dose rate at contact (&lt; 2 mGy·h<sup>-1</sup>)</li> <li>• Maximum mass – Specification drawings</li> <li>• Technical drop test</li> <li>• Fissile materials (&lt; 0.1 g-l-1 block of waste; &lt; 50 g fissile material/package)</li> </ul>
<b>GERMANY</b>		Repository Site Selection Act (StandAG)	KONRAD (Asse, Morsleben are closed to acceptance)	pre-disposal techniques	exist	available	available	SA related	available

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<b>HUNGARY</b>	Public Limited Company for Radioactive Waste Management - PURAM	Act on Radioactive Waste Management	LLW disposal facility – surface  LILW – LL disposal - geological	segregation, compaction, cementation	approved in licensing procedure		available for each site	operational and post closure safety	available
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### 3.2 Summary of national overviews

Various stages of waste management programmes are reflected in the responses to the questionnaire. Not many countries are experienced in managing real WAC and their applications, but in principle, all countries have a system that would lead to WAC development in the predisposal stage, i. e. in RW collection, treatment, conditioning, and storage activities.

A legislative and responsibility framework is available in all responding countries. Typically, operated facilities (storage, conditioning, and disposal) are mostly managed by state organizations, and the rules of the waste management processes are controlled by regulatory bodies. However, exceptions exist: private companies are responsible for managing several facilities as well.

Disposal facilities are available in some countries. The set of WAC is characterized by the existence of generic criteria such as recording, dangerous substances and free liquids prohibition, physical, chemical and biological characteristics of waste. Special criteria are set for containers, waste packages type and materials, dose rates on the waste packages surface, and radiological limits set for important radionuclides.

WFQ system is not directly specified in most of answers, but it is indicated, that it covers at least the extent of WAC specified with respect to the waste form approved for disposal.

WAC are usually approved in the framework of the licensing process of the repository, there is no need to have approved WAC at the predisposal stage of the process.

In the responses, the factual relationship of WAC to safety assessment has not been identified (the method for their derivation), in spite of the fact that safety assessment is usually meant as a supporting process for WAC development. The method is supposed to be specified for generic criteria, if relevant, and for site specific criteria, if possible. Simple examples of using existing procedures (e. g. specified in national documents and/or IAEA recommendations) will be helpful and could be shown in the course of project development.

Some of programmes seem to use only IAEA waste classification to connect WAC to a potential/existing disposal site. Dose rate on the waste surface is used as a measure of safety. On the other hand, radionuclides important for the assessment of long-term safety have mostly limited activity in the waste set by WAC. In the waste acceptance process, it is not very probable that it will be possible to run routine measurement of difficult to measure (DTM) radionuclides. The list of limited radionuclides is very extensive in some cases and the direct control of radionuclides activities cannot be routinely applied during the acceptance process (DTM radionuclides); the facility operator must verify the radionuclides content in the waste production process, and sometimes he has to rely on waste producer data.

The waste acceptance process is not usually described in much detail. The depth of its description varies among countries, and it is not directly linked to how advanced the national programme is.

The special question on WFQ is answered only in some of cases. But respondents do not directly assign their responses to WFQ. The understanding of WFQ and its necessary extent differs from country to country.

In principle, the rules of WFQ procedure are mostly understood as the following next steps:

1. Sampling and samples analysis
2. Derivation of the activity from the waste tracking system
3. Categorisation of the waste
4. Radioactivity measurements, including determining the amount of certain radionuclides.

In fact, there is minimal information on WFQ and it primarily relates to the process of WF choice, WF characterization, and evidence of compatibility of the WF with engineered barriers system, and the host site.

The characterisation techniques are not completely described. Methods used to verify compliance with WAC include both non-destructive methods, such as physical inspection, radiometric measurements, or gamma spectrometry, and destructive methods, such as radiochemical analysis, which are used to check waste package compliance with WAC either for storage or for disposal.

Dose rate measurement is the most widely used radiometric method for checking compliance with the radiation protection requirements. However, other methods such as specific nuclide vector / scaling factors presentation can complete dose rate measurements with the intention of deriving radionuclide activities. These methods require good knowledge of the origin and/or history of the waste including waste streams description and could be problematic for legacy waste. This is a very important task in predisposal activities.

Measurement of the masses of specific materials or chemical / toxic species in the waste, characterisation of challenging wastes, such as legacy wastes or heterogeneous wastes, and measurement of DTM radionuclides are indicated as principal challenges for checking the waste package compliance with WAC.

Waste disposal inspection process such as audits at predisposal operator, qualification of predisposal processes, and before the formulation of WAC, it also includes the control of waste data sheets, inspection and control findings, control of integrity of waste packages, dose rates, surface contamination, sample evaluation, acceptance of sampling processes, and other WAC specific controls.

For non-compliance(s) measures available before and after the waste packaging include, the following steps:

1. Any non-compliant waste transferred to the disposal site is quarantined and may require return to the waste producer
2. Treatment, re-packaging, and/or return to the producer are the responsibilities of the operator of the conditioning facility; in some cases, it might be done by repository operator provided that the cost of these operations is born by waste conditioner
3. Waste producers have responsibility for ensuring that any returned waste is appropriately received and managed
4. For waste that has been accepted and is subsequently identified to be non-compliant, the regulatory body or other relevant organization has to be informed and will make a decision on the appropriate remediation action.

In several national programmes, the qualification process is declared to be successful if the treatment and conditioning installation, a radiological characterization methodology, and/or primary waste packages meet all conditions set by WAC. This also implies in situ control carried out on the level of waste producer and quality control carried out by disposal facility operator and/or by regulatory body.

In national overviews, the role of licenses has not been emphasised for both waste producer and storage / repository operator.

In any case, it will be feasible to generalize the responses to general waste qualification rules, using support from other references as well.

### 3.3 Procedures and parameters followed in the WFQ process

For the parameters, usually used in waste form qualification, there are identified specific procedures used in the WFQ process and WAC control during the acceptance process. The range of WFQ development methods should be usually wider than that of WAC control methods: WAC parameters

are usually selected based on the result of the WFQ tests. WFQ results can be in a detailed form used as an input to the safety assessment of the facility, and they, as well as the results of safety analysis, form the bases for WAC formulation and control.

### 3.3.1 Radiological characteristics

It is advisable to measure the content of radionuclides in the raw waste (concentrate, sludges, ion exchangers, solid waste, etc.) as well as its volume and/or mass. Volume or mass activity are used to calculate the volume and/or mass activity of the final waste form using the concentration ratio derived from the solidification process parameters. Activity content of raw waste is usually measured by alpha, beta or gamma spectrometry. The content of DTM radionuclides can be determined using scaling factors often based on the comparison with the content of  $^{60}\text{Co}$  or  $^{137}\text{Cs}$  (this is possible only in the case of regular waste streams, such as waste from NPPs).

Activity concentration levels are derived from waste streams characterization.

Both of specified values (total activity, activity concentration) are compared to limit values derived from safety assessment, and usually become one of WAC items. The total activity limit is site specific and cannot be derived without basic knowledge of the site geological and hydrological system; it depends on the radiological capacity of the site, affected by expected infiltration dependent flows in the near field and far field, by distances of the facility from accessible components of environment, biosphere conditions, etc. Activity concentration limit is operated facility specific and can be derived using the information on the storage/repository requirements, such as container and other barrier properties, and probable use of the site after the termination of institutional control period, i. e. on site residence, transition, and possible intrusion.

The total activity content (alpha, beta and gamma) is derived from volume/mass activities with respect to anticipated waste processing technologies. The radiological capacity of the site must be considered. Other methods include justification, that all waste inventory intended to be disposed of can be emplaced to the repository in compliance to radiological limits. It is also possible to derive the total activity limit using safety assessment of the closed repository and regulatory dose restrictions.

Dose rate on the waste form surface and/or at defined distances from the waste form surface is usually measured in the acceptance process: its value need not to be directly bound to radionuclides content limited by WAC, but it is limited to assure radiation protection conditions of workers; measuring methods are easily available and can be different for various facilities (beta gamma measurement is a routine technique).

Surface contamination (alpha, beta and gamma) assessment is determined for two reasons: the first one is assuring the conditions of radiation protection for workers, the other one is determination of the quantity of mobile activity in the facility (besides leachable ratio and the activity of non-solidified waste); measuring methods are available and, usually, alpha and beta gamma measurement of surface activity is a routine matter, using the wiping test.

Evaluation and evidence of presence of individual activity sources is important for compliance with the requirement for waste homogeneity, usually applied in the surface and subsurface disposal sites. Presence of individual sources and proof of homogeneity are illustrated by calculation. In sophisticated systems, homogeneity can be documented by non-destructive scanning, and exceptionally, by destructive testing. Individual sources have to meet conditions set in the safety assessment, as a result of evaluation of on-site residence, transient, and intrusion scenarios.

Proper management of individual sources, results in avoiding the creation of hot spots, which may lead to higher dose effects in scenarios involving direct contact with waste; these requirements are relevant also for the post closure period of surface facilities and managing institutional waste.

Heat output is usually not relevant for surface storage/disposal facilities.

**WFQ parameters relevant to radiological characteristics of the waste form include determination and changes in relevant characteristics, typically:**

- ***structural stability,***
- ***leachability,***
- ***gas generation,***
- ***migration parameters of radionuclides and chemical contaminants.***

### 3.3.2 Chemical and physical properties

Chemical composition shall be described in the extent necessary for the evaluation of important waste properties as stability, potential for migration, speciation of radionuclides, etc.

Density, porosity, permeability to water and permeability to gases are measured to describe the waste performance with goal of recommending adequate alteration of the conditioning procedure and final waste form properties.

Production of gases by corrosion and biodegradation processes and their potential influence on the release of radionuclides is evaluated using laboratory methods and modelling tools.

Thermal, chemical, and radiation stability of a waste form is evaluated by laboratory tests (thermal and/or radiation loading and evaluation of stability of isolation or shielding properties, as well as strength resistance).

Homogeneity of the waste is tested by waste sampling, measuring the activity content, and subsequent calculation; and by non-destructive tests – tomography.

The waste compatibility with the matrix is studied by indirect methods, e.g., waste form leachability, strength resistance, and shape stability.

Dustiness is measured by an abrasion test, or indirectly by a drilling test. For safety assessment input, respirable particles ratio should be estimated.

Thermal stability is not usually evaluated for L/ILW, however, for HLW it is one of key parameters to be characterised.

Flammability is declared by a waste producer, it is assessed on the basis of waste composition.

Free liquid content in a WF is determined using the solidification process evaluation; other possible methods are water squeezing under compressive strength, shrinkage, and curing.

Leachability can be measured in a short term or long-term leachability test, or by measuring of leachability index.

Long term immersion tests are aimed at the assessment stability of the WF under disposal conditions.

Corrosion rate can be measured by the loss of mass and by other relevant methods, such as gas generation rates providing the information of corrosion rate and magnitude.

Materials and composition (e.g. metal alloy, glass, ceramic) of the waste form are described using standard laboratory analytical methods.

Porosity and its distribution can be measured by special methods using tension and pressure processes, as well as by the measurement of water/gas permeability.

Thermal conductivity is relevant mainly to HLW. But, the thermal conductivity is also measured for ILW waste forms to ensure that the waste forms are compatible with the repository design.

Solubility and corrosion rates in corrosive atmospheres or liquids such as water or brines can be measured using conductivity tests; using of available reference data is possible. Corrosion can definitely affect the lifetime of a container, and subsequently its isolation function.

Expected lifetime of the waste form can be derived from carbonation or corrosion rates, and from evaluation of possible destructive processes as strength, water penetration, freezing cycles, and creation of fractures. In fact, the description of most of these processes has to be extrapolated from short term laboratory tests.

**WFQ parameters describing chemical and physical characteristics are:**

- ***Chemical composition of the waste form***
- ***Compatibility of a WF with repository barrier elements***
- ***Distribution coefficients of important radionuclides***
- ***Solubility of important radionuclides***
- ***Diffusion coefficients or leaching potential of important radionuclides (total beta gamma and total alpha leaching ration may be sufficient)***
- ***Corrosion rate and its change after loading (thermal, radiological, microbiological)***
- ***Shielding capability and its change after loading (thermal, radiological, microbiological)***
- ***resistance and its change after loading (thermal, radiological, microbiological)***
- ***Possible release of respirable particles***
- ***WF homogeneity***
- ***Expected lifetime and its change after loading (thermal, radiological, microbiological, subjection to strength, freezing cycles).***

### 3.3.3 Physical properties of the waste package

Number / volume of voids in the waste package is controlled by the waste producer quality control. Minimum volume of voids is desired.

In the acceptance process, free volume in the container can be estimated by non-destructive X ray scanning.

Characteristics of the lidding and sealing arrangements is provided by waste producer and assured by quality control.

Weight, definition of dimensions, and calculation of waste package volume are used for waste description, as well as the description of its shape.

Sensitivity to changes in temperature can be investigated by freeze/thaw cycling.

***There are identified no independent WFQ parameters.***

### 3.3.4 Mechanical properties

Mechanical properties, such as tensile resistance and compressive resistance, are measured using different loading tests, e. g. static and impact loads, with the simultaneous control of dimensional stability of the waste form.

The performance under thermal loads is followed in the cases of thermal or freezing cycles.

Stackability of waste packages shall be determined to ensure safe handling procedures.

Drop tests are prescribed for waste containers to assess potential consequences of handling incidents.

**WFQ parameters describing mechanical characteristics are:**

- ***resistance and its change after loading (thermal, radiological, microbiological, subjection to strength, freezing cycles)***
- ***Dimensional stability and its change after loading (thermal, radiological, microbiological, subjection to strength, freezing cycles).***



### 3.3.5 Containment capability of a waste package

Diffusion potential is usually determined by a leaching test, or by measurement of diffusivity. Distribution coefficient in the final waste form can be measured in laboratory conditions; measurement of distribution coefficient in WF contact with water is often applied.

The potential for the release of gas under standard atmospheric conditions or under conditions in a repository is important for subsurface disposal systems.

The potential for the diffusion of tritium under standard atmospheric conditions or under conditions in a repository can be specified by measurement of tritium in the repository atmosphere in the case of underground systems, or by measurement of tritium in collected water (drainage system, outflow).

***There are identified no independent WFQ parameters.***

### 3.3.6 The stability or robustness of the waste package

Waste package stability is usually assessed as the stability of the unit composed of waste form and the container. Sometimes, container may be missing, e. g. in the case of individual solid pieces of waste such as large plant items, which are not containerised;

The behaviour under temperature cycling can be followed using the same methods as for the waste form, using stability and sensitivity in fire conditions, sensitivity to elevated temperatures and behaviour in a fire, behaviour under conditions of prolonged radiation exposure, sensitivity of the matrix to water contact, resistance to the action of microorganisms, corrosion resistance in a wet medium (for metal containers);

Other properties are followed as porosity and degree of gas tightness, the potential for swelling due to the internal build-up of gases.

***There are no identified independent WFQ parameters.***

### 3.3.7 WFQ and RWM programme

A waste management programme includes steps necessary for facility licensing as well as the description of activities, which are necessary for effective and safe waste management. Some of them affect the WFQ process, namely:

- A description of the radioactive waste streams and the efforts to be made to avoid and minimize them (time dependent waste mass or volume, waste form, evaluated activity and radionuclides composition, methods of sorting and conditioning, if available);
- The limits and conditions necessary for the waste to be managed safely (procedures used and their conditions approved by regulatory body);
- A comprehensive list of the waste categories and anticipated waste streams and inventories for the facility (type of waste, mass/volume of waste, activity, waste form(s));
- Identification of waste management options and associated steps, as well as identification of interdependences between waste management steps (sorting, release from control, treatment, conditioning, storage, disposal);
- Justification of the selection of appropriate management options on the basis of elements introduced above, and international good practices;
- The appropriate classification and segregation of waste, and maintenance of an accurate inventory for each radioactive waste stream, with account taken of the available options for clearance, storage, and disposal;
- The collection, characterization and safe storage of radioactive waste;
- The processing of radioactive waste to comply with waste acceptance criteria and to ensure safe storage, transport and disposal;
- Monitoring changes in the characteristics of radioactive waste by means of inspection and regular analysis, in particular in the case of long-term storage;

- Initiation, as necessary, of research and development activities for improving existing methods for processing radioactive waste or for developing new methods and techniques;
- Establishment of the location of the facility, to take into account safety and radioactive waste management aspects;
- Establishment or upgrading of a waste management inventory;
- Further development of the waste management inventory to incorporate potential secondary waste;
- Establishment of initial WAC, onward disposition criteria and storage criteria;
- Setting down of research and development requirements to determine the gaps in knowledge that need to be filled to achieve optimal waste management;
- Repetition of all activities through the concept design, development, detailed design and building sub-stages of design, in order to expand the database of information, develop future conditions for information and establish an auditable trail of decisions.

Waste form qualification for disposal in a defined final site is a special activity developed using the information on future waste streams and provision of evidence on their compatibility with the proposed waste form and the hosting site.

### 3.3.8 Impact of WFQ tests on waste management activities

Waste form qualification tests primarily assess the long-term waste form performance under repository conditions: the goal is to quantify potential releases of contaminants into the environment and set criteria which will guarantee keeping impacts of these leakages within prescribed safety limits. The implementation of these measures is twofold: either a disposal facility is sited and designed to ensure the required control of escaping contaminants or waste processing technologies are adopted to decrease contaminant release below the defined limit.

Thus, proper WFQ procedures can enhance the identification of critical mechanisms and pathways for the release of radioactive and chemical contaminants and – as such – serve as the source term for the development of alternative waste management solutions.

## 4 Guidance for WFQ methods implementation

The goal of this chapter is to provide guidance for Member States to implement of the WFQ process. The objective is to define how to achieve the goal, i. e. to defend a proposed waste form in a special disposal system. The waste form as a part of repository design has to be compatible with both, engineered and natural barriers system including host structure. That means that the waste form properties must not be negatively affected by the properties of barriers system, and on the other hand, barriers properties as well as potentially present other waste form must not be negatively affected by the waste form disposed in the repository.

The method described shall lead to the achievement of this goal, and it will necessarily include specification and extent of test performed in the course of the process, and their justification.

It is desirable that the waste form qualification process is supported by safety assessment using the properties of radioactive waste, waste form, and disposal system as relevant inputs. Generally, all changes in the repository that could affect the conditions of radiation protection and/or nuclear safety, are always subjected to safety assessment, in accordance with national legislation.

Examples of tests of significant importance are provided: tests quantifying decisive safety related parameters such as mobility, leachability, stability, homogeneity, corrosion rate, and relevant radiological properties.

In practice, the choice of tested parameters will be supported by analysis of relevant FEPs. The FEPs affecting this choice will be defined using the waste inventory and disposal system description

(features), processes specified as decisive in an uncertainty and/or sensitivity analyses, and in radiation protection conditions.

These conditions are principally represented by limiting of species mobility in the WF, stability of the WF for long term, and in operational period, also by complying with conditions of radiation protection.

- The choice of parameters should be justified by
  - FEPs analysis
  - The list of parameters supported by uncertainty and sensitivity analysis, considering anticipated effects of safety related processes as freezing-thaw cycle, microbial attack, potential of water infiltration, radiation effect, change of migration parameters in long term, ageing in special chemical conditions

The procedure is usually performed by waste producer in cooperation with a repository operator. The extent of testing and application of the methods are usually approved and controlled by regulatory body, within the licensing process and/or other decision makers.

#### 4.1 Mobility constraint and nuclear safety

Mobility of radionuclides in the WF should be minimized and its constraint can be justified by testing more parameters which can be both general and radionuclide specific. The limitation of radionuclides transfer to other parts of repository system as engineered barriers, and later to hydrogeology system in host rock formation and biosphere is actually one of goals of nuclear safety.

The procedure is usually performed by waste producer as a process approved and controlled by regulatory body.

The tested parameters and possibly used methods of tests are listed in Table 2.

*Table 2. Tested parameters related to radionuclides mobility constraint with respect to the nuclear safety and, possibly methods used for testing*

RN mobility constraint		
Parameter	Method	Note
Leachability	Leaching characteristics of radionuclides from different solidified waste and calculation of values suitable for comparison – e.g. leachability ratio test for bitumen, cement and geopolymer matrix; Co-60, Cs-137 measurement; correlation method can be used for uneasily measured radionuclides	ANSI/ANS-16.1-1986 in 5 days performance - index L(7), or in 90 days performance - index L(10). Dimensionless index does express the leachable ratio cause by diffusivity (decadent logarithm of inverse value of effective diffusivity [cm <sup>2</sup> /s] – activity increase in a sample of known volume and surface) [6]
Leachability of radionuclides alpha	leachability ratio test, analytical measurement	total beta gamma and total alpha leaching ratio may be sufficient some of the possible analytical methods are described e.g. in MS 2.6 [7]
Distribution coefficients (Kds) of important radionuclides in WF contact with water	laboratory test, field test	reference values can be used for uneasily measurable radionuclides
Solubility	solubility test	interpretation of leaching test can be used
Diffusion coefficient's of critical radionuclides from WF	leaching test	derivation using the results of leaching test

Gas generation	stress measurement, calculation	safety issues to be followed: the possibility of causing permanent damage, the amount of water displaced by the gas particularly important for WF with higher metal and/or organic content, as gas production is likely to be significantly higher
WF compatibility with repository barrier elements	subjection to conditions defined by barrier system	leaching test, solubility test, verification of distribution coefficients of important radionuclides
Expected lifetime and its change after loading	thermal loading, application of the dose $10^6$ Gy, microbiological tests, stress loading, freezing cycle	testing the durability of WF under conditions likely to be encountered in a repository setting (immersion testing, freeze-thaw cycles, compressive strength etc.). In order to predict WF performance, it is important to quantify the degradation-induced changes in leaching mechanisms, rates of changes, and mass transport properties of the degraded waste forms.[8]

## 4.2 Waste form stability

Integrity of the disposal system during all periods of its existence is in a wide range of measure ensured by stability of whole repository components, including waste form. Waste form stability has to be assured for long term period, at least during the periods of repository operation, closure, and subsequent institutional control. For surface repositories, the waste form stability should usually be guaranteed for some hundreds of years. Waste form stability is supported by compliance with a set of requirements; conformity with them has to be verified before the waste is accepted for disposal.

Some of the waste properties have to be also tested in extreme conditions, after the waste form is subjected to external and internal impact of processes as radiation and thermal loading, microbial attack, freezing cycles, and others. Stability of the waste form is then verified; the change of waste form properties affecting safety must not change in a measure that would lead to substantial loss of integrity and stability. The tested parameters that should not be negatively affected in the course of endurance test, are shielding capacity, leachability, corrosion rate (if relevant), strength resistance.

The tested parameters, and possibly used test methods are listed below in

Table 3.

Table 3. Tested parameters related to waste form stability and possibly used test methods

<b>Waste form stability parameters</b>		
<b>Parameter</b>	<b>Method</b>	<b>Note</b>
Waste package stackability	visual control	Determination of stackability to ensure safe handling procedures and optimization of the disposal system
Waste package integrity, dimensions	visual control, measurement	Loss of waste package integrity can contribute to instability of the disposal module and cover. Impact resistance and compressive strength are important properties that are tested to ensure that waste forms and packaging possess sufficient physical strength to maintain structural integrity. Typical R&D work includes: composition selection tests for immobilizing and coating materials; strength testing (e.g. tension, compression and impact); confinement properties (e.g. leach testing); thermal tests (e.g. freeze/thaw cycling and fire performance); gas generation (radioactive and/or inactive) and dispersion analysis (e.g. cracking trials); degradation tests (e.g. corrosion, microbial activity, radiation damage).[8]
Waste package mass	weighting	
Chemical composition of the waste form	analytical laboratory testing	The chemical and biological parameters may be crucial for stability and containment characteristics of the waste package (consider, for example, chemical or microbial generation of gas and/or heat, corrosion, etc.)
Corrosion rate	measurement of the weight loss, electrochemistry	calculation
Waste mass	calculation	$m_{\text{waste\_package}} - m_{\text{cover}}$ [kg]
Shielding capability	dose rate measurement, calculation	Investigations can be conducted using experimental data coupled with predictive mathematical model (e.g. MonteCarlo computer model).
Compressive strength, for cemented waste, and/or waste solidified by geopolymer	destructive or non-destructive testing	e. g. concrete test hammer
pH changes, immersion	laboratory tests	
<b>Endurance tests</b>		
<b>Test</b>	<b>Description</b>	<b>Note</b>
Radiation loading of the waste form	of the dose $10^6$ Gy	
Freeze-thaw test	-50°C and +60°C, or real extent of temperatures	

Thermal loading, heating test	heating test simulating real repository conditions	thermal response of the waste form system
Microbiological loading	test with nitrifying bacteria, test with sulphur oxidizing bacteria assessment of durability and long-term performance of a cement matrix subjected to heterotrophic microbial mediated degradation [9]	mainly in the case of cemented matrix  microbial corrosion represents the most significant uncertainty in terms of predicting the lifetime of waste form / packaging.

### 4.3 Radiation protection

Radiation protection conditions have to be assured by waste form properties. Surface activity shall be lowered to acceptable values valid for radiation protection; surface contamination shall be minimized as well as the presence of respirable ratio of particles present in the waste.

The tested parameters and possibly used methods of tests are listed below in Table 4.

*Table 4. Tested parameters regarding radiation protection and possibly used test methods*

Radiation protection		
Parameter	Method	Note
Shielding capability	dose rate measurement, calculation	attenuation capability
Surface contamination	measurement of summary surface activity alpha, and gamma/beta; non-direct measurement of abrasion	activity extent $3.10^{-1} \div 3.10^4$ Bq.cm <sup>-2</sup> , energy extent $2.10^5 \div 2.10^6$ eV
Possible release of respirable particles	reference values, dustiness measurement (mechanical or gas dispersion, weight loss measurement)	calculation of activity in air
Surface dose rate	control measurement of summary dose rate in radial direction	extent of dose rates $10^{-8} \div 1$ Sv/h, energy extent $4,8.10^4 \div 1,3.10^6$ eV
Radionuclides content	spectroscopy, radiochemistry	calculation of total activity in disposal units, verification

### 4.4 Choice of WAC for disposal

Waste acceptance criteria can generally be a part of waste acceptance conditions, which could include more technical, safety and administrative restrictions, limitations or requirements. Not all waste acceptance conditions are derived from safety analysis or from WFQ process. But in fact, WAC can be established on available and operative WFQ methods, with respect to the needs of waste acceptance procedure.

Long-term tests can be adopted in the process of WF generation, but in the acceptance process, these are usually used just as verification tools.

During a waste acceptance procedure, there exists a necessary extent of WAC, controlling radiological/safety related WF properties, and WF quality statements, as:

- radionuclides content, including checking activity values in disposal units, and in the repository, usually limited in accordance with the results of safety analysis; activities requiring

detailed calculations can be accomplished before within activities preceding the acceptance act

- surface contamination
- dose rate in defined measuring points (WF surface, defined distance, geometry options – radial, axial)
- WF and its physical properties as
  - type of container
  - WF dimensions and stackability
  - WF mass
  - WF integrity
  - strength resistance
- WF chemical properties
  - leachability

Migration parameters, as diffusion or distribution coefficients, and/or solubility are usually not easily measurable and their control cannot be carried out within the waste acceptance procedure. But, individual estimation of migration parameters may take place by waste production and as a part of research activities concentrated on WF performance.

Documentation of waste acceptance procedure including WF identification and description is provided by administrative WAC, in accordance with national legislation and/or regulatory requirements.

In some types of radioactive waste repositories, the WF may be regarded as a barrier providing one or more safety functions, e. g. isolation, retardation, and other. In such a case, a special set of tests has to be implemented, not necessarily implemented in the waste acceptance process.

During the siting process, it is desirable to check WF options compatibility with potential repository site properties.

The extent of WAC is subject to national legislative requirements, and additional regulator requests, if necessary.

### 4.5 WFQ - support of licensing during different project phases

During the design phase of the disposal facility/repository development project, WF proposal is an input for decision making process of engineered barriers system and its safety functions verification. At this period, the site hydrogeological conditions and geotechnical properties are usually available, and the compatibility of barriers system and site can be verified. This means at least to check the migration parameters of important radionuclides in the barrier system (EBS and host rock) and considering the accessible environment chemical conditions, with respect to existing potential water withdrawal points and other biosphere components.

In addition, there are more potential reasons for WFQ process within repository operational phase as well:

- newly arising waste streams (inventory, activity, form)
- another novel solidification technology availability (economy reasons, effectiveness, etc.) as a result of technological innovation
- legislative changes (limits, energy consumption, other, ...)

In such cases, WFQ process has to be subjected to revision, considering actual state of all inputs.

Newly arising waste should be characterized in as much detail as possible. Both pre-operational and operational phases can be considered as contexts for WFQ process revision. For already designed or operated repository, it is necessary to estimate future flows and total volumes of newly arising waste. It is desirable, that this waste complies with WAC, if they exist. That means that volume



activity of WF disposal should respect defined limits of volume activities in the specific components of disposal system. These limits are not usually dependent on the waste form used in the repository system.

The waste form planned or qualified for disposal in projected/operated repository has to be checked also for newly arising waste. In the case, that a new technology is proposed (e. g. waste form from a new waste producer, change of technology by waste producer, change of technology by a company providing the solidification for a waste producer, ...), the entire WFQ process is to be applied, including licensing, and upgrading of WAC.

### 4.6 Practical recommendations

For small inventory programmes, it will be beneficial to learn from existing programmes.

WFQ process can be minimized, if another RW producer dispose similar waste streams (e. g. standard waste form NPPs operation), and the technology for solidification of this waste is already available and used. Cemented waste form could be a good example of widely used WF. In some cases, it could be beneficial to transform the WFQ process used for cemented waste to a WFQ of similar WF (e.g. based on new type of cement matrix (cement mixtures), geopolymers, etc).

If the repository does not yet exist, it is possible to use existing technology, respecting the volume activity limits and conditions, which could be generally site independent. Then, the waste form can be qualified for a reference disposal system, and the waste can be processed and stored. The technology can be adapted to special national conditions even before the repository is available. In every case, the comparison with well described technology used in operated facilities (geopolymer, cement matrix, and other) will be desirable. The waste conditioned using this technology can be then stored in a suitable facility.

A helpful option of verification of the WFQ process is benchmarking, using the parameters of comparable disposal systems and comparable waste forms. It is usually not possible to compare hydrogeological conditions of the site, but in fact, in comparable disposal system design, radionuclide composition, activities and state of waste can be solidified into analogous waste form.

This possibility can lower the number of verification tests and/or straighten the extent of qualification procedure. Long term performance tests can be undertaken from technology operator and need not to be implemented in full measure. The number of short-term tests (leachability, strength resistance) can be limited to reasonable measure.

In the initial stages of development of new forms of waste, their WFQ process (and/or characterization) can be simplified by testing simulated waste form – so called "simulants". "Simulants" are materials designed to simulate specific types of chemical or physical behaviour of actual radioactive wastes. Simulated waste can be developed to exhibit only a limited set of important properties for a specific application or may be tailored to exhibit a broader range of chemical, physical, and rheological properties for a wide range of tests. Simulants testing is then used to reduce the costs and demands of the WFQ process itself, as well as to reduce the costs and hazards of developing treatment/conditioning processes designs, demonstrating that the proposed waste conditioning flowsheet is suitable and processable. However, final testing of actual waste forms produced from actual radioactive waste is essential to definitely demonstrate that the proposed treatment/conditioning process and products control strategies are robust enough to produce waste forms that meet all applicable waste acceptance criteria. The use of simulants in the WF qualification is described, for example, in IAEA-TECDOC-1537 [10].

In order to minimize the number of WF samples required for tests/determination of selected WFQ parameters, careful selection of a limited number of samples using a statistical analysis approach is recommended.

Acceptance of a waste form to a repository and/or its preparation to a future disposal system must be justified by safety assessment, and its upgrade, if relevant. Safety assessment inputs, relevant to waste form description, should preferably be obtained from WFQ procedure.

The technological production of the qualified waste form for disposal and its own disposal in a repository has to mandatory be licensed by a regulatory body.

It is obvious that close cooperation between waste producer, WMO and the regulator is essential for the successful implementation and progress of the WFQ process.

## 5 WFQ approach

WFQ is proposed to be implemented in several steps which compile and process available information to determine:

- The choice of a solidification process for the particular waste stream (e.g. cementation, bituminisation, geopolymers, vitrification...);
- A set of experiments aiming at the characterisation of the long-term waste form performance in a repository system;
- Experiments to evaluate the compatibility of the selected WF with the designed engineering system of the repository and with the host rock, usually demonstrated by safety assessment;
- To define parameters and their values to be checked during waste transition between waste management stages (waste acceptance criteria).

The WFQ process is built using principally the requirements of the radiation protection and the nuclear safety, as well as the needs of technical safety, monitoring, and emergency preparedness. Demonstrating containment of radionuclides through a set of experimental investigations (e.g., leachability of final waste form, solubility, diffusivity and distribution coefficients (Kd's) of principal radionuclides, degradation rate of the final waste form, durability of the container, etc.) is the principal goal to be achieved while documenting the nuclear safety requirements.

An example using combination of waste streams and solidification media is shown below in Figure 1.

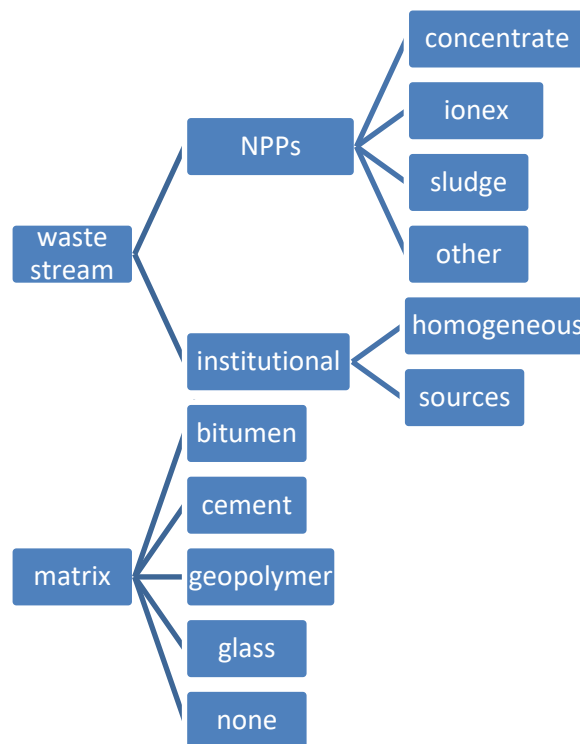
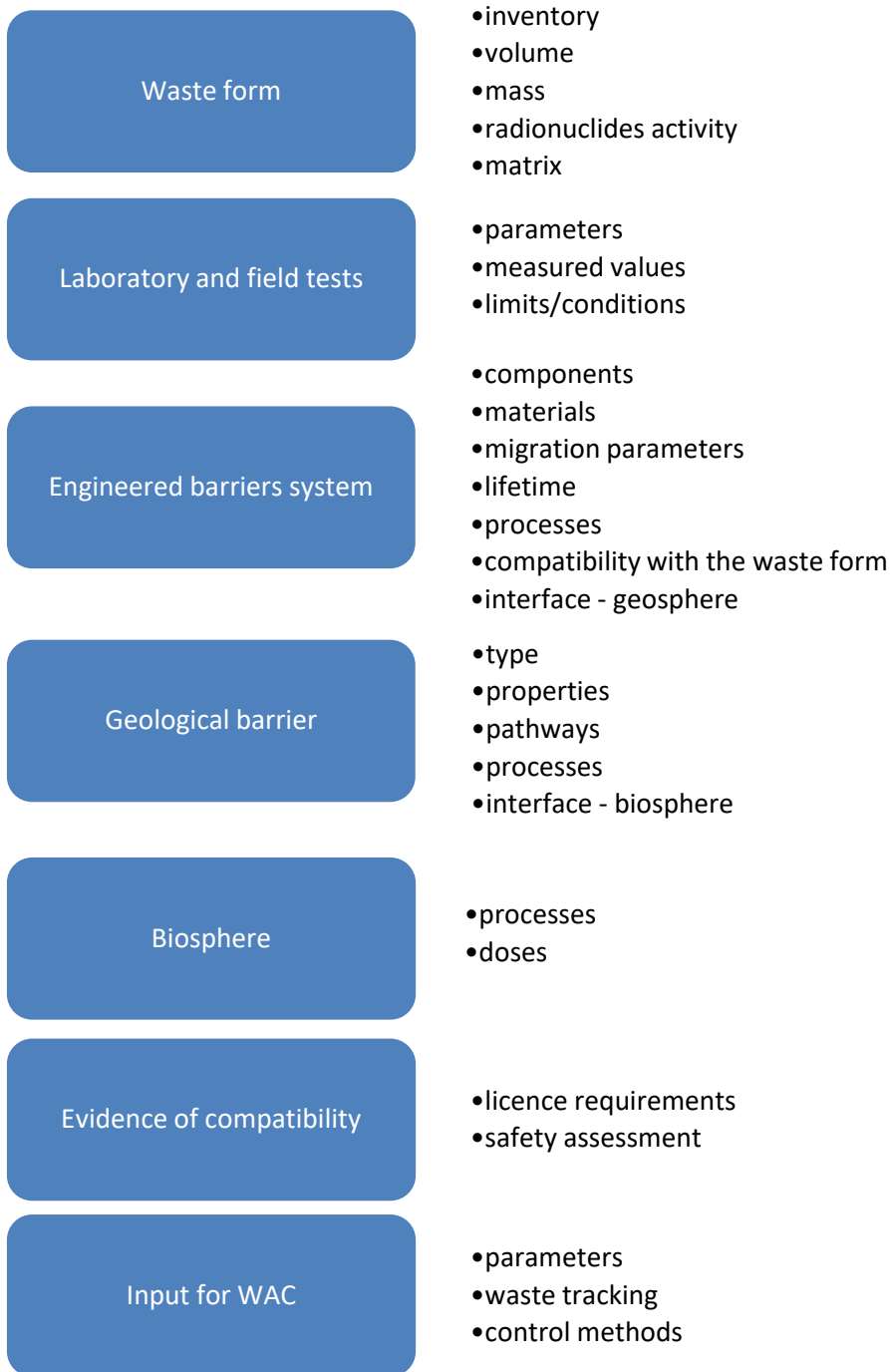


Figure 1. Example of using a combination of waste streams and solidification media

The goals of radiation protection are documented by the control of dose rates on the waste form / waste package, by measurement of surface contamination, and by the elaboration of emergency rules for accidental situations.

Both radiation protection and nuclear safety requirements should be demonstrated in the safety case. Waste processing (treatment, conditioning) is a licensed activity as well as waste storage and disposal. The limits and conditions of all these activities are defined prior to operation of these facilities as acceptance conditions/criteria. Acceptance criteria are to be defined as an output of waste form qualification process: they serve as a tool allowing checking whether waste has been processed as prescribed and whether it fulfils requirements determined for safe operation of a disposal facility.

For each waste stream type converted to a waste form, there is defined a set of tests and assessments, that will provide the evidence of waste form (including size) compatibility with the disposal system. This is summarised in Figure 2.



*Figure 2. Summary of assessment parameters for each phase of the WAC*

## 6 Summary

There is a number of common issues regarding waste form development and qualification for disposal. The goal of both activities is to develop a waste form that will meet safety provisions stated by national legislation, harmonised with European legislation, for RWM purposes.

Not all final waste form properties evaluated with respect to waste characteristics are later required as WAC because of limitations within controls carried out by waste acceptance procedure (matrix Kd's, porosity, lifetime etc.).

Similarly, parameters that needed to be tested in waste qualification process are not always considered in waste form characterisation process and for commonly used WAC.

There exist mutual links in relation to WAC, waste form development and qualification that could be relevant for various matrixes.

Sometimes, new RW streams and/or new RW forms can arise, e. g. after technological changes in waste producer processes. In such cases, the new waste form has to go through the WFQ process; it includes comparison with existing WAC, and in the case of non-compliance, development of new waste form specific set of WAC which has to be justified by new waste form oriented safety assessment.

The principal challenge is to identify a general set of experiments for WFQ, which would demonstrate and justify the choice of a particular waste form (including its dimensions) used for a defined waste stream intended to be disposed of in a disposal facility sited in a selected host structure.

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## Annex 1: An example of waste form qualification approach (Czech Republic)

Formerly considered methods for solidification of ion exchangers and sludge, previously stored in a semi-liquid form in NPP Dukovany, were based on cement and bitumen matrixes. Technology of solidification in geopolymer matrix enabled simple removal of waste from storage tanks and in situ solidification within waste containers. Several problems were solved such as transport over long distances and creating heterogeneities, as well as settling of particles, drying, and radiation stability information. The technology is mobile, and it does not need further investment. The waste load was about 20 weight %; to optimize the use of repository volume, while respecting existing limits of the specific volume activity.

The performed tests described below have provided evidence that the proposed geopolymeric waste form is compatible with the design and operational limits and conditions of the existing disposal facility URAO Dukovany. The results of performed analyses were approved by the regulatory body in the form of re-licensing of repository operations based on a safety assessment and a revision of WAC.

### Inputs for the safety assessment of URAO Dukovany

Complementing analyses were made to have other than reference values of migration parameters of important radionuclides.

Distribution coefficients were measured for following radionuclides:  $^{137}\text{Cs}$ ,  $^{94}\text{Nb}$ ,  $^{63}\text{Ni}$ ,  $^{14}\text{C}$ ,  $^{241}\text{Am}$  and  $^{90}\text{Sr}$ .

Distribution factors were measured only for the ratio of activity in water and in soil after equilibrium. Experimental measurement of distribution coefficients was carried for geopolymer matrix in demineralised water for important radionuclides, limited by WAC. Distribution coefficients in geopolymer matrix were not available; therefore, conservative reference values were used instead. In order to provide data on matrix performance, sequential leachability testing was performed on the matrix system.

### Resistance

Conservatively it was assumed, that the lifetime of the matrix shall be 300 years at least. The stability of newly developed matrix was subject to a testing during the maturation of the mixture (24 days). No changes have been observed in the matrix structure on laboratory samples and on container fillings as well.

### Radiation stability

Geopolymer matrix had been subjected to the dose of  $10^6$  Gy. No degradation processes have been identified.

### Other characteristics

Dustiness and the content of water were assessed as input data for the safety assessment.

### Comparison to cement matrix

The release mechanism from geopolymer matrix is usually driven by diffusion, similarly to cement matrix. There were no dependencies found on the built-in mode relevant to the matrix.

Geopolymer matrix does not necessarily need the pre-treatment of processed waste by dehydration, centrifugation, and/or drying.

## Types of tests

Tests were performed both for model and real waste product in the following extent:

- Resistance
- Leachability
- Diffusion
- Radiation stability
- Biodegradation
- Distribution coefficients
- Water bound in matrix
- Possible release of respirable particles
- Combustibility, explosiveness

Leachability and compressive strength were tested for samples with filling 17 - 25 weight % of dry product. Resistance is under 10 MPa for filling up 22 weight %. For filling around 20% the strength resistance is from 15 to 28 MPa, and it is about 50 MPa for a low filling. Maximum filling can be also limited by free water presence in the waste form.

Two independent methods were used for determining resistance (destructive, non-destructive).

Leachability tests were: 5 days for L(7) and/or 90 days tests for L(10).

*Note. Leachability index is the decimal logarithm of reverse value of effective diffusivity [ $\text{cm}^2/\text{s}$ ] specified as an increment of activity in leachate from the sample with known surface, in a given time.*

## Safety assessment assumptions

Conditions for WAC derivation were following:

- upper level of optimization is 250  $\mu\text{Sv}/\text{yr}$  for normal evolution scenario,
- personal dose limit 1  $\text{mSv}/\text{yr}$  for scenarios with low probability (on site residence, intrusion, transmission).

Waste disposed of is from NPPs Dukovany and Temelín. Filling of ion exchangers and sludge is 20% (optimum value for optimising the vault volume with respect to WAC). Possibility of combination with other types of matrixes has been assessed as well.

Real activities of waste compared to cemented waste from NPP Dukovany:

m	real RW in cement matrix [ $\text{Bq}/\text{m}^3$ ]		real RW in geopolymers matrix, 20% weight filling [ $\text{Bq}/\text{m}^3$ ]			
	ion exchangers	sludges	tank X1	tank X2	tank X3	tank X4
$^{14}\text{C}$			$6,62 \cdot 10^7$	$2,82 \cdot 10^7$	$2,08 \cdot 10^8$	$7,13 \cdot 10^5$
$^{41}\text{Ca}$	$1,80 \cdot 10^6$	$5,04 \cdot 10^5$	$< 7,69 \cdot 10^4$	$< 4,62 \cdot 10^4$	$< 1,30 \cdot 10^5$	$< 4,75 \cdot 10^4$
$^{59}\text{Ni}$	$7,20 \cdot 10^6$	$1,93 \cdot 10^6$	$4,15 \cdot 10^7$	$6,00 \cdot 10^6$	$1,82 \cdot 10^7$	$2,61 \cdot 10^7$
$^{63}\text{Ni}$	$1,03 \cdot 10^9$	$2,29 \cdot 10^9$	$2,12 \cdot 10^9$	$1,41 \cdot 10^9$	$1,43 \cdot 10^9$	$1,93 \cdot 10^9$
$^{90}\text{Sr}$	$1,10 \cdot 10^8$	$2,63 \cdot 10^7$	$1,23 \cdot 10^8$	$3,19 \cdot 10^8$	$1,61 \cdot 10^6$	$8,36 \cdot 10^6$
$^{94}\text{Nb}$	$2,30 \cdot 10^7$	$2,34 \cdot 10^7$	$< 1,62 \cdot 10^7$	$< 5,31 \cdot 10^6$	$< 1,16 \cdot 10^7$	$< 8,08 \cdot 10^6$
$^{99}\text{Tc}$	$2,06 \cdot 10^7$	$2,70 \cdot 10^6$	$< 5,38 \cdot 10^5$	$< 1,85 \cdot 10^5$	$< 3,90 \cdot 10^5$	$< 1,90 \cdot 10^5$
$^{129}\text{I}$	$3,30 \cdot 10^7$	$8,10 \cdot 10^6$	$< 7,69 \cdot 10^4$	$< 4,62 \cdot 10^4$	$< 1,30 \cdot 10^5$	$< 4,75 \cdot 10^4$
$^{137}\text{Cs}$	$1,1 \cdot 10^{10}$	$5,04 \cdot 10^8$	$4,85 \cdot 10^8$	$1,26 \cdot 10^9$	$1,31 \cdot 10^8$	$1,04 \cdot 10^9$
$^{239}\text{Pu}$	$4,30 \cdot 10^5$	$7,20 \cdot 10^3$	$9,62 \cdot 10^4$	$3,65 \cdot 10^5$	$3,77 \cdot 10^4$	$1,66 \cdot 10^4$
$^{241}\text{Am}$	$1,50 \cdot 10^5$	$2,52 \cdot 10^4$	$3,31 \cdot 10^5$	$6,24 \cdot 10^5$	$1,08 \cdot 10^5$	$7,60 \cdot 10^3$



The values of real waste were used to verify that the final waste form would meet the limit of radionuclides in 1 m<sup>3</sup>, in a vault, and in the repository.

Activities of non-limited radionuclides:

RN	real RW in geopolymers matrix, 20% weight filling [Bq/m <sup>3</sup> ]			
	tank X1	tank X2	tank X3	tank X4
<sup>54</sup> Mn	1,72.10 <sup>8</sup>	7,85.10 <sup>8</sup>	2,74.10 <sup>8</sup>	9,03.10 <sup>8</sup>
<sup>57</sup> Co	-	3,57.10 <sup>6</sup>	-	-
<sup>60</sup> Co	1,00.10 <sup>9</sup>	1,31.10 <sup>9</sup>	2,34.10 <sup>9</sup>	1,45.10 <sup>9</sup>
<sup>95</sup> Nb	-	1,26.10 <sup>7</sup>	-	-
<sup>110m</sup> Ag	2,77.10 <sup>7</sup>	4,71.10 <sup>7</sup>	-	-
<sup>134</sup> Cs	1,92.10 <sup>7</sup>	9,70.10 <sup>7</sup>	5,58.10 <sup>7</sup>	5,37.10 <sup>8</sup>
total	1,2.10 <sup>9</sup>	2,3.10 <sup>9</sup>	2,7.10 <sup>9</sup>	2,9.10 <sup>9</sup>

### Cement and geopolymers matrix comparison

#### Radiation stability

The dose applied was 1,027 MGy <sup>60</sup>Co (dose rate 2,5 kGy/hr). No significant change in compressive strength was identified:

sample	compressive strength before radiation	compressive strength after radiation	reduction of compressive strength [%]
	[MPa]	[MPa]	
matrix M7	57,4	48,4	16
waste in M7	23,1	25,1	0
matrix R4X	52,3	33,6	36
waste in R4X	21,1	22,6	0

#### Biodegradation

A 28 days test lead to the measured value of 3,58 % COD, the limit for non-degradability being 10 % COD<sub>Cr</sub>.

*Distribution coefficients [m<sup>3</sup>/kg]*

RN	K <sub>D</sub> – field experiment	K <sub>D</sub> – sequential leaching				cement
	soil	puzolan cement	portland. cement	geopolyme r+sludge	geopolymer + sludge	cement
<sup>14</sup> C	n.m.	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>41</sup> Ca	n.m.	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>59</sup> Ni	0,634	n.m.	n.m.	n.m.	n.m.	0,06 - 0,91
<sup>63</sup> Ni	0,634	n.m.	n.m.	n.m.	n.m.	0,06 - 0,91
<sup>90</sup> Sr	0,0266	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>94</sup> Nb	n.m.	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>99</sup> Tc	0,0001	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>129</sup> I	0,001	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>137</sup> Cs	2,9	1,64	0,25	1,48	1,46	(8-20)E-05
<sup>239</sup> Pu	1,95	n.m.	n.m.	n.m.	n.m.	n.m.
<sup>241</sup> Am	1,646	n.m.	n.m.	n.m.	n.m.	n.m.

n.m. the value was not measured

*Leachability and diffusion coefficients*

RN	diffusion coefficient, cement, [cm <sup>2</sup> /s]	leachability index, cement, L(7) *	leachability index, geopolymer, L(7) *
<sup>14</sup> C	5.10 <sup>-8</sup>	7.3	9-9,3 sludge 8,8-9,8 IER
<sup>41</sup> Ca	10 <sup>-8</sup>	8	
<sup>59</sup> Ni	5.10 <sup>-10</sup>	9.3	10,5-10,8 IER, Ni 10,5-10,7 sludge, Ni 10,5-12,1 sludge, Co 12,1-12,5 IER, Co
<sup>63</sup> Ni	5.10 <sup>-10</sup>	9.3	
<sup>90</sup> Sr	5.10 <sup>-11</sup>	10.3	12,8-15 sludge A1 12 IER 12,5 sludge
<sup>94</sup> Nb	10 <sup>-8</sup>	8	n.m.
<sup>99</sup> Tc	4.10 <sup>-11</sup> to 7.10 <sup>-8</sup>	10.4 to 7.15	n.m.
<sup>129</sup> I	5.10 <sup>-8</sup>	7.3	n.m.
<sup>137</sup> Cs	5.10 <sup>-10</sup>	9.3	8,5-10,7 sludge 9,4-9,6 sludge + IER
<sup>239</sup> Pu	5.10 <sup>-13</sup>	12.3	12,2- 18,5 sludge
<sup>241</sup> Am	5.10 <sup>-13</sup>	12.3	12,4-16,2 sludge

n.m. the value was not measured

*Compressive strength [MPa, N.mm<sup>-2</sup>]*

Results of laboratory tests, various geopolymer matrix formulations:

RW/geopolymer type	M7	R48 (R47)	R44	R49
sludge	15,6-27	NT	NT	
IER		18-25 (30-33)	21-30	n.m.
sludge + IER (IER < 5 %)	53-54	NT	20-30	16-24,5
sludge + IER (IER 5 – 20 weight %)	n.m.	18-21 (43-54)	21-35	
sludge + IER (IER 40 weight %)	n.m.	n.m.	5-8	n.m.

n.m. the value was not measured

*Water in matrix*

Water bound in cement represents 33 – 60 % of the matrix mass. No water is added to geopolymer process: free water of the waste origin enters the geopolymer reaction and is bound in the matrix. Water constitutes 28 - 30% of the matrix mass. After drying and setting processes, the mass of water decreases to about 20%. Drying and setting do not negatively affect resistance, which is usually increasing with time.

After drum closure, no additional loss of water is possible.

Loss of water and increase of resistance:

sample	time of drying [days]	0	9	21	29	370	490
14	loss of water, [%]	0	0	0,5	0,8	5,6	
	compressive strength, [MPa, n.mm <sup>-2</sup> ]	n.m.	n.m.	n.m.	24,00	26,44	27,08
21	loss of water, [%]	0	n.m.	n.m.	n.m.	6,3	n.m.
	compressive strength, [MPa, N.mm <sup>-2</sup> ]	n.m.	n.m.	n.m.	24,00	27,05	27,84

n.m. the value was not measured

*Freezing*

Tests were made for geopolymer with sludge (20 % weight filling), matrix was subject to frost from - 18 °C to - 26 °C for 24 hours, immediately after solidification. Other samples were subjected to frost after drying within 1 to 6 months. Samples solidified in normal conditions were not affected by frost, even repeated freeze-thaw cycles did not affect their quality. Samples frozen during the setting process did not reach prescribed degree of petrification.

### *Dustiness after destruction*

Simulation of this situation is difficult. Conservatively, the ratio of respirable particles has been used at the level of 10% for cemented matrix, about 20 % for geopolymer. Respirable particles were defined as particles that can hold themselves in the air, usually being of the size below 75  $\mu\text{m}$ . For geopolymer, the measured dustiness was conservatively 1,73 %, and it was higher than for cement (0,48 %). Measured values in air after drilling were 2,21  $\text{mg}/\text{m}^3$  for geopolymer, and 4,41  $\text{mg}/\text{m}^3$  for cement. The ratio of respirable particles was 9 % and 9,9 %, respectively.

### *Characteristics of typical RW*

No flammable substances and/or explosives were found in laboratory tests. No heavy metals, toxic substances, diluents, and corrosives were found as well.

## **Conclusions for safety assessment**

For safety assessment, following assumptions were used:

- Geopolymer contains lower mass of free water in the final waste form
- Final waste form shows good mechanical properties, preserved also under frozen thermal conditions
- This stability is justified by resistance
- The matrix provides good radiation and microbial attack resistance
- Migration parameters in near field and far field are better or equal to those of cement matrix
- The ratio of respirable particles in the air complies with the requirements of safety assessment
- Where it is necessary and justified, conservative values are used for safety assessment