Milestone 11
Overview of waste qualification approaches

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Abstract

The milestones document (delivered by Task 2.3 Waste acceptance systems) describes typical waste form qualification approaches used in the Member States programmes in order to determine the suitability of a waste form for storage and disposal in particular of low and intermediate level radioactive waste. This document is considered to provide the input to the PREDIS Deliverable 2.6 Guidance on waste form qualification (M42, February 2024).

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Milestone Description

Milestone No. 11 associated with Work package 2 ‘Strategic Implementation’, Task 2.3 ‘Waste acceptance systems’ has been completed on 04.08.2022.

The justification for the readiness is described below and complies with the Grant Agreement Description of Action noting verification by M2.8 document issued.

The readiness of the milestone was reviewed and agreed upon by Lumír Nachmilner (CVRez) as T2.3 leader and Anthony Banford (NNL) as WP 2 Leader.

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List of Abbreviations

COD	Chemical oxygen demand
COD_{Cr}	Chemical oxygen demand, using $K_2Cr_2O_7$
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
RW	Radioactive waste
SF	Spent fuel
WAC	Waste acceptance criteria
WF	Waste form
WFQ	Waste form qualification

1 Introduction

The waste acceptance system 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 requirements.

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 requirements 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 requirements 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 in detail described in national legislation or in 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 depths, what 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 application of waste form qualification methods in national programmes. Most of countries concentrated on the waste acceptance criteria (WAC) definition and their control in the acceptance process. This approach does not exactly express the need of waste form qualification data. The process of waste qualification has to imply 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.

2 Basis and outcomes

2.1 Introduction

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 leakages, 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 creation 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 containers)
- Compatibility with engineered barrier system of a repository, such as chemical interaction of a WF and leakages with particular elements of the barrier system

WFQ process shall be completed by the definition of WAC, i.e., parameters, to be verified in acceptance procedures at the storage or disposal facilities, and their values ensuring that escape 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 system
- Compliance with waste disposal safety, i .e. radiation safety, nuclear safety, monitoring, emergency preparedness, physical protection, criticality prevention (not relevant for LILW) for programs operating/constructing repository
- Compliance with generic criteria for organizations, which do not still 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)

2.2 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. Establishing of WAC is an important part of the process.

2.2.1 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 related only 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 non advertent release of radionuclides, prevention from critical state and prevention from heat transfer. The last two points are usually not considered in the process of disposal in surface and subsurface repositories.
- Radiation protection, i.e. limiting of doses of persons and environment using the radiation protection tools. It includes limiting of doses of repository and waste management workers during operation in predisposal and disposal activities.
- Technical safety, i.e. safety related to barrier performance.
- Monitoring, i.e. the control of a zero state, volume or mass activity in concerned environmental components, control of workplaces (does 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.

General occupational safety is ensured, if the above mentioned conditions are satisfied.

Considering the principle of defence in depths, evaluation of safety is closely bound with

- Radionuclide composition of the waste form (short lived, long lived, radiotoxicity)
- Waste origin (nuclear power plants – operational waste, spent fuel, waste from decommissioning, 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 concern of emergency situations is 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; the emergency scenarios probability is very low and should not affect normal operational conditions.

2.2.2 Waste acceptance system

Waste acceptance system shall 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 has to be planned and solved with respect to future disposal options.

2.2.2.1 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 system 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 built in near surface
- Disposal of intermediate level waste (ILW) is recommended in the depth below 30m, and
- 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 subjected to 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 decisive 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 by waste storage and in surface disposal systems. The effect of long lived radionuclides has to be followed in all ways of management, regarding their higher radiotoxicity, especially in the case of radionuclides alpha. RW inventory has often to be considered as a whole, definitely in the case of operational waste from nuclear power plants; this type of RW does not allow distinguishing between low level and high level values of different radionuclides activities, and between various half time values.

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

2.2.2.2 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 requirement is usually not directly expressed as a special WAC, but the solidification matrix and the properties of 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.

Final waste form in prevailing 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, as high integrity containers.

Containers used for disposal should explicitly be specified in WAC. Properties of containers that are considered in repository safety assessment have to be set and approved by 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, namely, if having 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 container shall be assessed together with safety functions of other barriers (engineered and geological) according to defence-in-depth principle.

Solidification media have to 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, as low as achievable. Actually, not all solidification media are suitable for all types of waste: their improper use can lead to deterioration in desirable waste properties during repository lifetime. Thus, the selection of effective conditioning process shall be done on case-by-case bases while considering also 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, as isotopes of H, C, Cl, I. Potential interactions between solidification media and waste should be assessed as well as effectiveness of available conditioning processes.

Solidification of some types of waste, such as spent sealed sources, may lack safety rationale. Although some types of this waste can be disposed in disposal systems for LLW, they stay an important spot of point irradiation: RW heterogeneity shall be thoroughly considered in safety assessment and relevant conditions of disposal have to be set in WAC.

2.2.3 WAC definition

Final waste form properties that shall be checked during waste processing and waste acceptance have to be set out in WAC, including requirements on frequency and method of their control, specified for each of the waste form that is 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 waste producer and disposal site operator, while respecting national legislative requirements. Waste streams knowledge including inventory, final waste form for disposal, and waste producer data are the principal inputs to the formulation of WAC.

Countries, that do not operate a repository and even have not 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 as well. Waste streams identification is helpful, if it exists.

2.2.3.1 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 requirements or international recommendation, and in fact, future disposal site properties have minimum 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.

2.2.3.2 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 the following ones:

- Weight of waste package
- Dose rates on a waste package surface or in a defined distance from the surface
- Dose rates on the surface of closed vaults
- Strength resistance
- Volume / mass / total activity of important radionuclides
- Waste package type and dimensions
- Homogeneity of waste expressed by mathematical equation
- Leachability of final waste form
- Mobile activity in the repository
- Surface contamination of the container or individual piece RW.

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

2.2.3.3 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 transition 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 control system and operational instructions. Typical examples of control methods are the following:

- 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 examination
- Weighting
- 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 – record, photo documentation, schematic drawing.

2.2.3.4 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-conduct 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.

2.2.3.5 Generic WAC

Generic criteria are not based on site specific data.

For practical reasons, it will be 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 WAC, it is not necessary to limit the content of other than safety relevant radionuclides present and / or declared in the RW. Activity, half-life, mobility, and radiotoxicity are radionuclide properties that should decide if the radionuclide is to be included into WAC or not.

Other possible conditions of disposal referring to the facility status potentially specified in the future licence 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 siting process that the disposal site will be used for a RW of a defined origin. This WAC can be developed from the negotiation results with the waste producer, consequently with the repository operator, and/or municipalities as a part of operational licensing approval process. Regulatory body role in the licensing process has to be taken into account
- 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 in the course of disposal process
- Operability of disposal vault / chamber / disposal site (generally), e. g. non presence 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 way of control of waste position in the disposal site
- Staff duties, commitments, and liabilities.

Safety criteria

Possibility to formulate safety criteria without site/facility characteristics is rather limited. Quantification of general scenarios based on RW handling, transport, inadvertent intrusion during operation and/or after closure can enable to limit activities in disposal units, and to derive mass / volume activities in the waste.. Actually, only the following ones could 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.

2.2.3.6 Site specific WAC

Site specific criteria can be developed as an output of site selection process and they 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 form 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 of lifetime periods of its existence. Licensing process has more components that vary during siting, construction, operation and closure of the repository. The compliance with national legislative requirements 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 shall be put on 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 namely 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 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 host structure, and conversion dose factor are radionuclide properties that could decide if the radionuclide is to be included into WAC or not.

2.2.4 Safety case

In the safety case, it has to be proved that RW, repository performance, and host site capacity are in accordance with all safety legislative requirements. WAC developed in the safety assessment as a part of the safety case and other relevant WAC have to 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 regulator, if relevant. Waste, which was found 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), without re-assessment in the safety case frame, and without relevant re-licensing procedure.

2.2.4.1 Position of safety assessment in the safety case

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 the requirements of radiation protection, nuclear safety, and technical safety is proved usually quantitatively, but also qualitatively using safety analysis and safety assessment. Quantitative expression of safety is then formulated in WAC. Safety assessment is submitted to regulatory body as a part of safety case in the frame of permission process.

2.2.4.2 Safety assessment resources

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

- Type of RW – waste category, origin of RW prevailing 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 off, and available volume in the repository
- Repository type – surface (vault system, trench), subsurface (rock cavity, abandoned mine, drill etc.) underground, nuclear installation or not
- 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, and its seismic potential
- Radiological capacity of the site (anticipated total activity to be disposed off).

2.2.4.3 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 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 decisive for WAC derivation. For other materials and radionuclides data can be found in IAEA handbooks and documents, e.g., IAEA Safety Reports No. 19 [3], TRS472 [4], and in more documents of NUREG and EPA. For most countries, the principal source of data will be also 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.

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

2.2.4.4 Dose constraints

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

- For normal evolution scenario, a fraction of general annual dose limit is set as a constraint, typically 0,3 mSv/yr, 0,25 mSv/yr, 0,10 mSv/yr – depends on national approach
- For non-probable 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 of population and workers.

2.2.4.5 Recommended scenarios

Apparently, full set of inputs needed for the scenario description, model construction and calculation are provided by FEPs (features, events, processes) database. FEPs database is a quality assurance and knowledge management tool as well. It can involve data from national research activities as laboratory and field tests, existing national and international databases, and expert judgement. FEPs help to consider all processes affecting safety to be involved in scenario description, and to defend 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 have to be evaluated for derivation of WAC in subsurface disposal systems:

Normal evolution scenario – off site residence

Normal evolution scenario is known also as project scenario or off-site residence scenario. It describes projected evolution of the disposal system as a whole, and usually serves as a tool for derivation 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 preference transport pathways as drainages and water collection systems are treated against 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 reasonable measure of conservatism.

Near field processes evaluation has an objective in source term quantification, i.e. quantified release from repository (Bq/m^3, Bq/y, Bq/(m^2.y) for all important radionuclides. Far field processes evaluation leads to quantification of dilution and distribution of contaminant 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 evaluation 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 such 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. Cover of 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 and advection starts to be the leading process of migration in near field. The effect of this state is higher source term with earlier maximum.

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

Intrusion scenario

Non advertent intrusion could lead to direct contact with waste, in some cases the radiation effects are decreased by possible 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 that dose limits used for judgement of 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 usual pathways of irradiation, i.e. external irradiation, inhalation and ingestion assessing soil contamination, plants and animal product consumption. Usually, it leads to limiting volume activity of important radionuclides. Dose limits used for scenario assessment are less restrictive that dose limits used for normal evolution scenario assessment.

2.2.4.6 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 repository barrier system, their knowledge allows for the better definition of potentially destructive processes and for the assessment of negative impact of destructed waste forms. For robust systems, safety should not depend on a state of one subsystem providing safety function (defence of depth). Typical safety functions are isolation, retention, dilution, stability, retardation. One subsystem can provide more safety functions to the system, e.g. a container is responsible for 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 safety assessment procedure. Several WAC are derived using the knowledge on safety functions definition and role. The examples are final waste form stability and lifetime, activity of important radionuclides, etc.

Safety functions are described for a designed repository, but their system should be defined much earlier, considering the role of the waste properties in the future disposal system.

Barrier considerations

Safety functions are usually specified to engineered subsystems of the repository, because of quality and control requirements. Isolation is the principal safety function of 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 quality and control point of view, properties of 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.

2.2.5 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 RW. This would lead to difficulties in the waste processing and treatment process as well as to troubles during acceptance process. Some radionuclides can hardly be directly measured, and in addition, activity of waste and the possibility of their detection can change during the process from waste processing to its disposal. But, regulatory requirements 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. Screening method should be presented in the safety case, if relevant. By now, safety assessment can be speeded up and simplified using computer codes. Radiation consequences can be quantified for all of radionuclides present in RW, and it is possible to exclude those having minimum 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 host structure, type of transport (advection vs. diffusion)
- Description of the disposal system performance including subsystems, i.e. used 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 non-dependent bodies and using the principal realism vs. conservatism as well. The measure of conservative approach is important. Conservatism is better defendable, but excessive conservatism could lead to underestimation of repository radiological capacity, which could have economical, 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 of list of criteria which could be respected and approved on all levels of WAC development and approval: waste producer – repository operator – regulatory body.

WAC have to be implemented in the national waste acceptance system respecting the limitation and requirements presented earlier.

WAC have to be in compliance with 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 waste acceptance process. For that, the extent of criteria has to be adapted 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 disconformities.

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 to be later approved, respecting 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, geopolymer, 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 also 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 on nuclear safety, radiation safety, technical safety, and monitoring.

The project of future disposal system has to set the system of engineered barriers: EBS system is proposed to accomplish one or more safety functions, compiled with the principle of defence in depths, 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 mean the set of quantitative requirements 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 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 followed during the process of waste form qualification is listed below.

In national programmes, there are usually no exact methods 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 wider system of waste form description than 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 [4]

Country	Responsibility	Legislation	Facility	Treatment	WAC	Generic	Site specific	SA	WA process
AUS	state	RWM Act	LILW repository	vitrified, cemented	approved in licensing procedure	physical, chemical, radiological (DRs), labelling	not presented	nuclear safety	exists, not specified
AUT	government		long term storage	incineration, compaction, drying	none				gamma scanning, DRs
BEL	Belgatom ONDRAF/NIRAS	in progress	CAT A, Dessel intended	cemented	formal, minimum requirements	revision in predisposal, physical, chemical and biological characteristics	63 RN, activity, natural RN separately		compliance, monitoring, fully specified
BUL	state, SE RW	Act on the Safe Use of Nuclear Energy	in construction, temporary storage		under development	general wording			under development
CRO			interim storage	none	for storage	extent available	extent available		DRs, visual
CZE	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
FIN	Fortum, TVO, Posiva		two in operation	exists	exist	available	available	not available	not available
FRA		Waste Act and other				exist	radiological		
GER		Repository Site Selection Act (StandAG)	KONRAD (Asse, Morsleben are closed to acceptance)	predisposal techniques	exist	available	available	SA related	available

3.2 Summary of national overviews

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

Legislation and responsibility frame is available in all responding countries. Typically, operated facilities (storage, conditioning, and disposal) are mostly managed by state organizations, and the rules of 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 existence of generic criteria as recording, dangerous substances and free liquids prohibition, more 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 frame of licensing process of the repository, there is no need to have approved WAC in the stage of predisposal process.

In the responses, factual relation of WAC to safety assessment has not been identified (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 course of project solution.

Some of programmes seem to use only IAEA waste classification to connect WAC to a potential/existing disposal site. Dose rate on 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 method of limiting them seems to be rather formalized.

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.

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 that would just concentrate on 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 to derive 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 waste package compliance with WAC.

Waste disposal inspection process as such includes 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 before and after the waste packaging the following steps are usually done:

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, regulatory body or other relevant organization has to be informed and should made a decision on the appropriate remediation action.

In several national programmes, qualification process is declared to be successful if the treatment and conditioning installation, a radiological characterization methodology, and/or primary waste packages meet all requirements set by WAC. This implies also 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 of 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}Co or ^{137}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 to be compared to limit values derived from safety assessment, and usually become one of WAC items. 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.

Total activity content (alpha, beta and gamma) is derived from volume/mass activities with respect to anticipated wastes processing technologies. The radiological capacity of the site must be taken into account. Other methods is justification, that all waste inventory intended to be disposed of can be emplaced to the repository with 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 in 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 r 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 on waste homogeneity, usually applied in the surface and subsurface disposal sites. Presence of individual sources and prove 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 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 evaluating direct contact with waste; these requirements are relevant also for post closure period of surface facilities, managing institutional waste, and suchlike.

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 to recommend 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;

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.

Explosiveness 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 leachability short term or long term test, or by measuring of leachability index.

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

Corrosion rate can be measured by the loss of mass and by other relevant methods providing the information of corrosion rate and intensity.

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 the measurement of water/gas permeability.

Thermal conductivity is relevant to HLW only.

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.

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)***
- ***Dustiness***
- ***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 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.

Weighting, 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 not identified any 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 set 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 usually 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 a 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;

Behaviour under temperature cycling can be followed using the same methods as for 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 not identified any independent WFQ parameters.

3.3.7 WFQ and RWM programme

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 (used procedures 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 conception, development, detailed design and building sub-stages of design, in order to expand the database of information, develop future requirements 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 of future waste streams and proving 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 aim at the assessment of 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 to required level.

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 proposal of alteration of waste management solutions.

4 WFQ approach

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

- The choice of a solidification process for the particular waste stream (e.g. cementation, bituminisation, geopolymerisation, 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).

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 of to be achieved while documenting the nuclear safety requirements.

Example of using convenient combination of waste streams and solidification media is shown below in Figure 1.

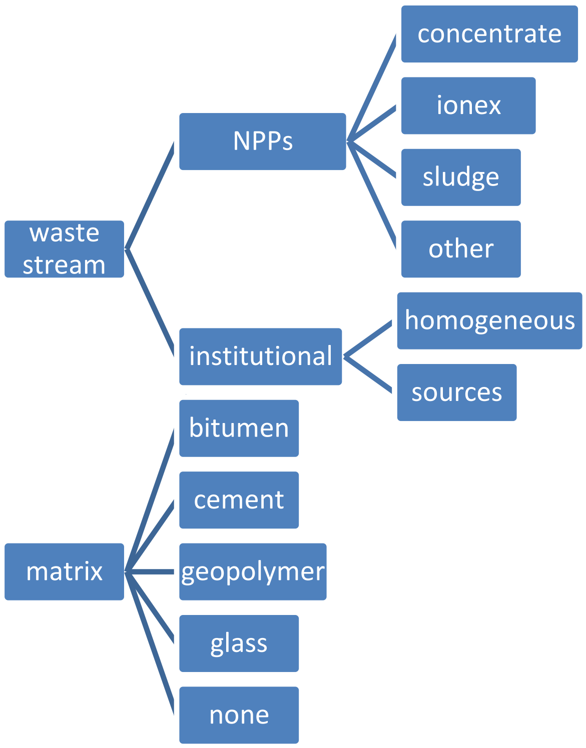

Figure 1 Example of using convenient combination of waste streams and solidification media

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 subject to wording of special limits and conditions of operation of these facilities and 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 a defined waste form, it is defined the set of tests and assessments, that will provide the evidence of waste form compatibility with the disposal system. The implementation of the waste form into the system assessed as a detail serving for WAC definition is represented in the following diagram (Figure 2).

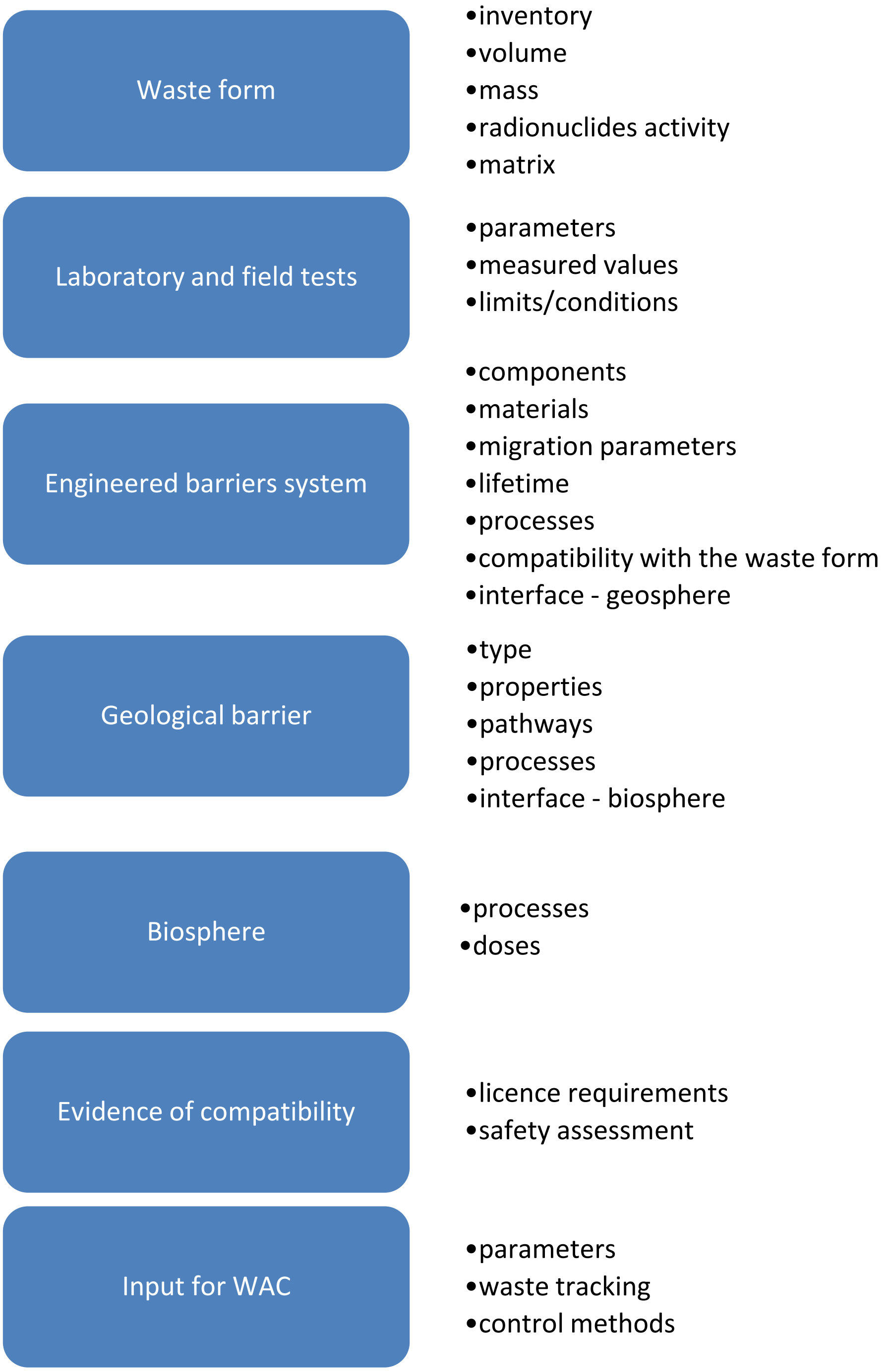

Figure 2 Diagram of implementation of the waste form into the system assessed as a detail serving for WAC definition.

5 Challenges

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 requirements stated by national legislation for RWM purposes.

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

Similarly, parameters needed to be tested in waste qualification process are not always considered in waste form characterisation process and 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 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 used for a defined waste stream intended to be disposed of in a disposal facility sited in a selected host structure.

6 References

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[4] Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Terrestrial and Freshwater Environments, TRS472, IAEA 2010
[5] Country input – PREDIS questionnaires (National Analysis) – T2.3.1: Collection of existing information on waste acceptance systems worldwide.

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

Formerly considered methods for solidification of ion exchangers and sludge, earlier stored in a semi-liquid form in NPP Dukovany, and 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. Several problems were solved such as transport on long distances and creating heterogeneities, as well as setting of particles, drying, and radiation information credibility. The technology is mobile, and it does not need further investment. The waste load was about 20 %; this enabled to optimize the use of repository volume respecting existing limits of the specific volume activity.

The performed tests described below have brought evidence that the proposed geopolymeric waste form sounds 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 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}Cs, ^{94}Nb, ^{63}Ni, ^{14}C, ^{241}Am and ^{90}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 demi water for important radionuclides, limited by WAC. Distribution coefficients in geopolymer matrix were not available; therefore, conservative reference values were used instead.

For real description of matrix performance, sequence leachability was set in the matrix system.

For real description of matrix performance, sequence leachability was set in 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 long term tested. 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 followed as a support of input data file for safety assessment.

Comparison to cement matrix

The species in geopolymer are bound in 3D (volume) structure, what is the difference from cement matrix (2D planar structure). The release mechanism is usually driven by diffusion, similarly to cement matrix.

Geopolymer matrix does not necessarily need the former 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
- Dustiness
- Combustibility, explosiveness

Leachability and resistance were tested for samples with filling 17 - 25 % of dry product. Resistance is under 10 MPa for filling up 22 %. 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 setting of 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 revers value of effective diffusivity [cm^2/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 μSv/yr for normal evolution scenario,
- personal dose limit 1 mSv/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 spending 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 [Bq/m^3]		real RW in alusil, 20% filling [Bq/m^3]			
	ion exchangers	sludge	tank X1	tank X2	tank X3	tank X4
^{14}C			6,62.10^{7}	2,82.10^{7}	2,08.10^{8}	7,13.10^{5}
^{41}Ca	1,80.10^{6}	5,04.10^{5}	< 7,69.10^{4}	< 4,62.10^{4}	< 1,30.10^{5}	< 4,75.10^{4}
^{59}Ni	7,20.10^{6}	1,93.10^{6}	4,15.10^{7}	6,00.10^{6}	1,82.10^{7}	2,61.10^{7}
^{63}Ni	1,03.10^{9}	2,29.10^{9}	2,12.10^{9}	1,41.10^{9}	1,43.10^{9}	1,93.10^{9}
^{90}Sr	1,10.10^{8}	2,63.10^{7}	1,23.10^{8}	3,19.10^{8}	1,61.10^{6}	8,36.10^{6}
^{94}Nb	2,30.10^{7}	2,34.10^{7}	< 1,62.10^{7}	< 5,31.10^{6}	< 1,16.10^{7}	< 8,08.10^{6}
^{99}Tc	2,06.10^{7}	2,70.10^{6}	< 5,38.10^{5}	< 1,85.10^{5}	< 3,90.10^{5}	< 1,90.10^{5}
^{129}I	3,30.10^{7}	8,10.10^{6}	< 7,69.10^{4}	< 4,62.10^{4}	< 1,30.10^{5}	< 4,75.10^{4}
^{137}Cs	1,1.10^{10}	5,04.10^{8}	4,85.10^{8}	1,26.10^{9}	1,31.10^{8}	1,04.10^{9}
^{239}Pu	4,30.10^{5}	7,20.10^{3}	9,62.10^{4}	3,65.10^{5}	3,77.10^{4}	1,66.10^{4}
^{241}Am	1,50.10^{5}	2,52.10^{4}	3,31.10^{5}	6,24.10^{5}	1,08.10^{5}	7,60.10^{3}

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

Activities of non-limited radionuclides:

RN	real RW in alusil, 20% filling [Bq/m3]			
	tank X1	tank X2	tank X3	tank X4
^{54}Mn	$1,72.10^{8}$	$7,85.10^{8}$	$2,74.10^{8}$	$9,03.10^{8}$
^{57}Co	-	$3,57.10^{6}$	-	-
^{60}Co	$1,00.10^{9}$	$1,31.10^{9}$	$2,34.10^{9}$	$1,45.10^{9}$
^{95}Nb	-	$1,26.10^{7}$	-	-
^{110m}Ag	$2,77.10^{7}$	$4,71.10^{7}$	-	-
^{134}Cs	$1,92.10^{7}$	$9,70.10^{7}$	$5,58.10^{7}$	$5,37.10^{8}$
total	$1,2.10^{9}$	$2,3.10^{9}$	$2,7.10^{9}$	$2,9.10^{9}$

Cement and geopolymer matrix comparison

Radiation stability

The dose applied was 1,027 MGy ^{60}Co (dose rate 2,5 kGy/hr). No change in resistance was identified:

sample	strength resistance before radiation [MPa]	strength resistance after radiation [MPa]	reduction of strength resistance [%]
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_{Cr}.

Distribution coefficients [m3/kg]

RN	K_D – field experiment	K_D – sequential leaching				cement
	soil	puzolan cement	portland. cement	alusil +sludge	alusil + sludge	cement
^{14}C						
^{41}Ca						
^{59}Ni	0,634					0,06 - 0,91
^{63}Ni	0,634					0,06 - 0,91
^{90}Sr	0,0266					
^{94}Nb						
^{99}Tc	0,0001					
^{129}I	0,001					
^{137}Cs	2,9	1,64	0,25	1,48	1,46	(8-20)E-05
^{239}Pu	1,95					
^{241}Am	1,646					

Leachability and diffusion coefficients

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

Strength resistance [MPa]

Results of laboratory tests:

<table>
<tr><th>RW/alusil type</th><th>M7</th><th>R48 (R47)</th><th>R44</th><th>R49</th></tr>
<tr><td>sludge</td><td>15,6-27</td><td>NT</td><td>NT</td><td></td></tr>
<tr><td>IER</td><td></td><td>18-25
(30-33)</td><td>21-30</td><td></td></tr>
<tr><td>sludge + IER
(IER < 5 %)</td><td>53-54</td><td>NT</td><td>20-30</td><td rowspan="2">16-24,5</td></tr>
<tr><td>sludge +IER
(IER 5 – 20 %)</td><td></td><td>18-21
(43-54)</td><td>21-35</td></tr>
<tr><td>sludge + IER
(IER 40 %)</td><td></td><td></td><td>5-8</td><td></td></tr>
</table>

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:

<table>
<tr><th>sample</th><th>time of drying [days]</th><th>0</th><th>9</th><th>21</th><th>29</th><th>370</th><th>490</th></tr>
<tr><td rowspan="2">14</td><td>loss of water, [%]</td><td>0</td><td>0</td><td>0,5</td><td>0,8</td><td>5,6</td><td></td></tr>
<tr><td>strength resistance, [MPa]</td><td></td><td></td><td></td><td>24,00</td><td>26,44</td><td>27,08</td></tr>
<tr><td rowspan="2">21</td><td>loss of water, [%]</td><td>0</td><td></td><td></td><td></td><td>6,3</td><td></td></tr>
<tr><td>resistance, [MPa]</td><td></td><td></td><td></td><td>24,00</td><td>27,05</td><td>27,84</td></tr>
</table>

Freezing

Tests were made for alusil with sludge (20 % filling), matrix was subject to frost from -18 ^{o}C to - 26 ^{o}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.

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 alusil.

For alusil, 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 mg/m^3 for alusil, and 4,41 mg/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 alusil contains lower mass of free water in the final waste form
- Final waste form shows good mechanical properties, preserved also in adverse 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
- Dustiness complies with presumptions of safety assessment
- Where it is necessary and justified, conservative values are used for safety assessment