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Quality assurance requirements and methods for high level waste package acceptability

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FOREWORD

For many years, the management of high level radioactive waste has been subject to extensive research, development and industrial scale programmes at national and international levels. Technologies and processes are in hand for treatment, conditioning, handling, transport and storage of radioactive wastes. Important projects, including underground experimental test facilities, are being implemented with a view to build deep underground disposal facilities from the year 2000 onwards.

Radioactive waste management requires, as any other industrial activity, planned and systematic actions to provide adequate confidence that the entire system, the processes and the products involved will satisfy given requirements for quality. The primary responsibility for the quality in terms of the achievement of specified requirements rests with the waste producer. To achieve the overall protection goals set by the regulatory authorities, quality assurance has to take place in every phase of waste management. At present, efforts to implement such measures are, to a certain extent, concentrated on waste conditioning.

The present document was prepared as part of the IAEA's programme on quality assurance of radioactive waste packages. It outlines the quality assurance requirements and methods for the processes of conditioning high level liquid waste and spent fuel, which may be used in the determination of those characteristics specified in the waste acceptance criteria for radioactive waste. Although quantitative waste acceptance criteria for disposal of high level waste are not yet available, the application of the quality assurance requirements and methods in the waste conditioning processes will contribute to a high probability of compliance with future final waste acceptance requirements. The IAEA document relevant to the subject of the present report is "Qualitative Acceptance Criteria for Radioactive Wastes to be Disposed of in Deep Geological Formations", IAEA-TECDOC-560, Vienna (1990).

A draft of this report was prepared by three consultants, S. Bouchardy of France, P. Carter of the United Kingdom and E. Warnecke of Germany with the assistance of V. Tsyplenkov of the IAEA Division of Nuclear Fuel Cycle and Waste Management. After the draft of the report had been reviewed by seventeen experts from ten Member States and one international organization, comments were incorporated and final revisions were made by the same group of consultants.

The IAEA would like to express its thanks to all those who took part in the development of the report, especially to E. Warnecke who also acted as Chairman of the Technical Committee Meeting held in Vienna, 6-10 May 1991.

EDITORIAL NOTE

In preparing this material for the press, staff of the International Atomic Energy Agency have mounted and paginated the original manuscripts and given some attention to presentation.

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1. INTRODUCTION

1.1. Background

In the field of nuclear power generation the subject of radioactive waste management is of great importance. In all types of nuclear facilities the minimization, handling, storage and treatment of radioactive waste should be managed to the required quality. The subsequent conditioning, transport, interim storage and disposal of the waste should also be managed in such a way as to protect man and the environment in accordance with the ICRP recommendations and national regulatory standards and minimize any burdens placed on future generations. In addition to the latest technology and skills, appropriate methods of quality management shall also be applied in all these areas, from the point at which the waste arises through to the point of disposal. Quality assurance (QA) as an essential management tool is being strongly applied in the area of pre-disposal high level waste management and will equally be necessary in the planning, design, construction, operation and closure of a repository. Quality control is now considered as a part of the planned and systematic actions of quality assurance and is not identified separately in this document.

The objective of a radioactive waste repository system is to ensure that man and the environment will not suffer unacceptable detriment from the disposed wastes at present and in the future.

With the aim of providing IAEA Member States with basic guidance on the protection of humans from hazards associated with the disposal of radioactive wastes in deep geological repositories, an internationally agreed set of safety principles and technical criteria for the underground disposal of high level radioactive wastes has been published as an Agency's Safety Standard [1]. The intention was to provide a basis for specifying more detailed standards and criteria for use in the subsequent stages of development of the waste disposal concepts.

Generally, the efficiency of a disposal system may depend upon a number of components - the waste form, the container, the overpack, the backfill material, the geological formation and the surrounding environment. These components should be selected and/or designed to be compatible and to achieve an acceptable overall degree of isolation. In the long term, the isolation capability of a waste disposal system will depend on the combined performance of its components.

Since the waste form and the container are important components in the stages of transportation, interim storage and handling in a repository, their performance shall be specified in waste acceptance requirements.

To ensure compliance with the requirements, and to assure the envisaged behaviour of the waste package and the disposal system, a quality assurance programme for all stages of waste management, including waste conditioning, is required. The IAEA has already published the principles and objectives of a quality assurance programme for nuclear power plants in a Safety Standard as part of the Nuclear Safety Standards for Nuclear Power Plants [2]. Ten supporting Safety Guides include an overall quality assurance programme for a nuclear power plant and also separate quality assurance programmes for each of the constituent areas of nuclear power generation activity (e.g. design, manufacturing, construction, commissioning and operation).

It is recognized by the IAEA that guidance on the quality assurance for radioactive waste packages should be given to the operators of waste conditioning facilities and repositories, to the consignees and consignor for the transport of radioactive waste, as well as to the national authorities in charge of the licensing and/or surveillance of conditioning processes or repositories, taking into account the existing principles and procedures established for nuclear power plants.

This Technical Document is issued within a series of publications on the quality assurance of waste packages by the IAEA. It is a companion document to the IAEA-TECDOC-560 "Qualitative Acceptance Criteria for Radioactive Wastes to be Disposed of in Deep Geological Formations" [3] and outlines the quality assurance requirements and methods for the processes of conditioning high level radioactive wastes, which may be used in the determination of those characteristics specified in the qualitative waste acceptance criteria for radioactive wastes. Quantitative waste acceptance criteria for the disposal of high level radioactive wastes are not universally available, however, generic criteria have been developed in the USA [4] and in France [5]. The application of the quality assurance requirements and methods in the waste conditioning processes will contribute to a high probability of compliance with future waste acceptance requirements being achieved or will at least enable the parties involved to judge the acceptability of the waste packages already produced.

However, it must be recognized that the basic responsibility for achieving quality in performing a particular task (e.g. in design, in manufacturing, or in the commissioning and operation of a conditioning plant for high level radioactive wastes) rests with those assigned the task and not with those seeking to ensure, by means of technical audit, that it has been achieved.

1.2. Objectives

This document should serve as guidance for assigning the necessary items to control the conditioning process in such a way that waste packages are produced in compliance with the waste acceptance requirements. It is also provided to promote the exchange of information on quality assurance requirements and on the application of quality assurance methods associated with the production of high level waste packages, to ensure that these waste packages comply with the requirements for transportation, interim storage and waste disposal in deep geological formations.

The document is intended to assist both the operators of conditioning facilities and repositories as well as national authorities and regulatory bodies, involved in the licensing of the conditioning of high level radioactive wastes or in the development of deep underground disposal systems.

1.3. Scope

Quality assurance measures shall be applied to ensure the compliance of waste packages with the projected waste acceptance requirements and methods of disposal in deep geological formations. Although this principle applies to all types of radioactive wastes, the document addresses only high level radioactive wastes, i.e. vitrified high level liquid wastes from reprocessing of spent fuel and conditioned spent fuel if it is declared a waste. This limitation is justified because of the unique combination of properties of the high level radioactive wastes, i.e. radiological relevance, presence of long lived radionuclides and heat generation.

The document recommends the quality assurance requirements and methods which are necessary to generate data for those parameters identified in IAEA-TECDOC-560 on qualitative acceptance criteria, and indicates where and when the control methods can be applied, e.g. in the operation or commissioning of a process or in the development of a waste package design. Emphasis is on the control of the process and little reliance is placed on non-destructive or destructive testing.

In addition, the document mainly addresses qualitative criteria and proposes quality assurance requirements and methods which should be used in order to contribute to the objective of safe disposal of high level, heat-generating waste. Quantitative criteria, relevant to disposal of high level waste, are repository dependent and are not addressed here.

2. ROLE OF QUALITY ASSURANCE IN WASTE CONDITIONING

The aim of quality assurance measures in radioactive waste conditioning is to ensure that the waste packages actually produced comply with the waste acceptance requirements.

As long as waste acceptance requirements for high level radioactive waste packages have not been fixed by a licence for a repository, the quality assurance efforts may be directed either towards preliminary waste acceptance requirements (if they exist) or towards the waste package specifications of the producer. It is a task of the responsible national authorities to decide this matter by judging the degree of confidence that high level radioactive wastes can be disposed of in their own State.

2.1. Quality assurance

Quality assurance of the waste packages is an important part of the overall quality assurance programme for radioactive waste management. Provision of a quality assurance programme has been specified as a waste acceptance criterion in IAEA-TECDOC-560. The quality assurance programme should be applied to all activities that affect compliance with waste specifications during waste package production, handling, storage, transportation and disposal. Such a programme should be developed in accordance with national and/or international legislation, codes and standards, and include the following elements: organization and responsibilities, procedures and instructions, document control, control of purchased materials, process control, non-conformance control and corrective actions, records, audits and training of personnel.

The management of quality assurance shall be independent of the management of plant operation. All activities shall be under the surveillance of the assigned national authorities or institutions.

2.2. Responsibilities

The quality control of radioactive waste packages involves the waste generator and the repository operator who can make use of third parties (e.g. a waste conditioner) to fulfill their duties. Both are supervised by the responsible national authorities.

The waste generator shall ensure and demonstrate the proper conditioning of his radioactive waste. Conditioning processes should only be applied, if they are sufficiently instrumented and if it has been shown that they are able to produce waste packages of the required quality under the specified operating conditions. The organizational structures should include a control of the conditioning process independent of the plant operation. All relevant data of the commissioning and operation of the conditioning process should be recorded. The responsibility for the delivery of a proper waste package to a repository lies principally with the waste generator. This responsibility is not changed if the duties are carried out by a contractor (e.g. a waste conditioner) who is nevertheless responsible to the waste generator for his activities.

The repository operator should check the quality of the waste packages before acceptance for disposal, to ensure compliance with the waste acceptance requirements. This should mainly be performed by controls within the conditioning process by an independent control institute, in agreement with the waste conditioner and the responsible national authority. Checks on waste packages may be partially performed at the repository, in the storage facility, or at an independent laboratory. In accepting the radioactive wastes package, the responsibility for its safe disposal is taken over by the repository operator.

2.3. Process control

IAEA-TECDOC-560 has specified as a waste acceptance criterion that control of the conditioning process should assure the required quality of the waste form and other parts of the waste package. This requires that sufficient development and testing be performed to determine that the important parameters for product quality are identified and quantified and means of control established. This means that a combination of results from the development and characterization of the waste package, the commissioning of the conditioning process and the process operation must lead to the availability of sufficient information to confirm that the quality of the waste package is in compliance with the waste acceptance requirements and the "Regulations for the Safe Transport of Radioactive Material" [6]. The conditioning process should be sufficiently instrumented to achieve this. If necessary, product sampling may be performed to confirm the proper performance of the process control.

2.4. Sampling and testing

An alternative to waste package quality control by conditioning process control might, in principle, be the non-destructive and/or destructive testing of the waste forms or waste packages or a combination of both. This concept requires hot cells with the specific equipment for statistically representative sampling, sample analysis and reconditioning of the waste package. It may produce secondary wastes and may lead to additional radiation doses to personnel. Destructive testing of waste packages should only be applied if the required data are not or cannot be obtained by other means including process control.

2.5. Records

The data obtained from the application of the overall QA programme shall be recorded (e.g. with the help of electronic data processing) by the waste generator, conditioner and repository operator. It should be established which records shall be permanent or non-permanent and which data may be exchanged between the parties involved. The record system (e.g. receipt, accessibility or modification of records), and the storage, safekeeping and preservation of the records has to be planned and controlled in such a way that the records cannot be lost or destroyed. Modifications shall only be made by adding of data by authorized personnel.

2.6. Non-conformances

It is recognized that some individual waste packages may not comply with every requirement. In this situation, the conditioner shall identify the non-conformances and propose remedial measures. Decisions shall be made by the repository operator and/or responsible national authority on a case-by-case basis. The resolution of these non-conformances will permit disposition of such waste packages.

3. WASTE PACKAGE ACCEPTANCE REQUIREMENTS

3.1. General approach

Waste acceptance criteria define a number of principles or standards which have to be met by a waste package to be acceptable for disposal in deep geological formations. Waste acceptance requirements will be site specific and of a quantitative or qualitative nature depending upon the requirement being formulated. They describe those properties which are necessary for the waste disposal in deep geological formations. In addition the requirements from other waste management steps (e.g. transport and interim storage) shall be applied.

The development of waste acceptance requirements for a repository and waste package specifications by the waste producer is an interactive process. Finally, waste package specifications should conform with waste acceptance requirements.

Waste acceptance requirements are derived from a safety assessment for the respective repository, taking into account the types of waste packages, the layout of the repository and the site specific overall geological situation. This means that a final assessment can only be made if:

- the site has been sufficiently explored,
- the layout of the repository has been made, and
- the waste package specifications have been formulated by the waste conditioner on the basis of development and characterization of a waste package.

The safety assessment results and the waste acceptance requirements are subject to the licensing procedure of a repository. Therefore, it is only possible to finalize waste acceptance requirements for a licensed repository.

3.2. Current situation

Since no such repository for high level waste disposal has been licensed so far, only qualitative waste acceptance criteria for radioactive wastes to be disposed of in deep geological formations have been formulated [3]. These criteria are a basis for the formulation of the respective waste acceptance requirements for a particular repository.

Quantitative waste acceptance requirements for interim storage exist in countries with operational vitrified waste stores (e.g. France, United Kingdom). For transport, international regulations exist [6] which have been adopted into national legislations. On such a basis, quantitative specifications have been elaborated for waste packages of vitrified high level liquid waste by the conditioner to obtain approval for transportation and interim storage (e.g. France, United Kingdom). Furthermore, preliminary specifications have been formulated in order to provide the basis for developing design specifications for the repository and the waste package [7, 8].

Existing safety assessments for repositories, which are based partially on model assumptions, show, in principle, the suitability of vitrified high level waste for disposal [9, 10]. According to the site specific requirements, a corrosion resistant container or an overpack may be applied.

For spent fuel conditioning, principal evaluations in the R&D phase are being carried out, which include provisions to deal with the high content of fissile material and the hazardous (e.g. volatile) radionuclides associated with the spent fuel. Further efforts are directed to the development of dissolution models based on spent fuel studies, and to the engineering developments necessary to bring spent fuel disposal technology to a stage of development required for licensing a repository. The safety evaluations of the geological disposal of spent fuel support the view that it can be feasible in accordance with the radiation protection requirements [11-13]. This subject is covered in IAEA Technical Reports Series No. 320 [14].

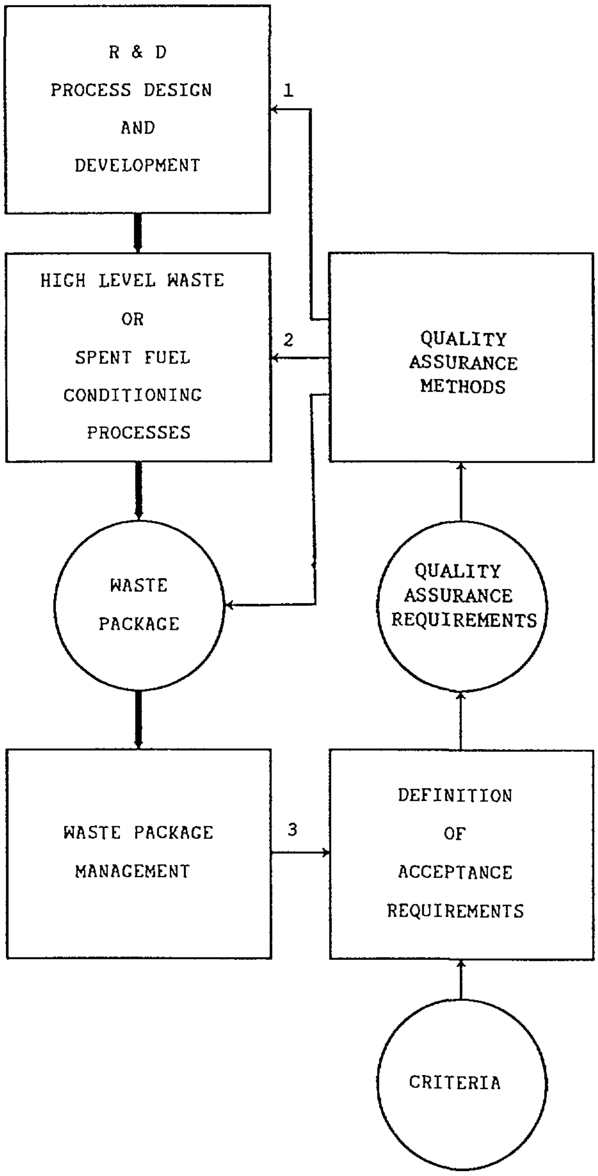
3.3. Development of waste acceptance requirements

IAEA-TECDOC-560 addresses and discusses the following waste acceptance criteria:

- Radionuclide inventory
- Thermal power, thermal loads and thermal effects
- Nuclear criticality
- Radiation effects, radiation damage and contamination control
- Mechanical stability, mechanical strength and stress resistance
- Combustibility and thermal resistance
- Gas generation
- Free liquids
- Explosive and pyrophoric materials, fire and explosion hazards
- Compressed gases
- Toxic and corrosive materials
- Chemical durability
- Physical dimensions and weights
- Unique identification
- Responsibilities and organization
- Quality control
- Records
- Compliance with codes and standards.

These criteria form a basis for the formulation of waste acceptance requirements. In waste acceptance requirements quantitative values have to be attributed to specific properties of a waste package (e.g. leak tightness of a container, homogeneity of the waste form, specific surface area or dose rate).

Development of waste acceptance requirements is a stepwise process. It is interactive with the development of a disposal system, from the first generic approach to the finally licensed site and approved procedures.



1. Before, outside conditioning process
2. During, inside and outside conditioning process
3. After, outside conditioning process

FIG. 1. Conceptual approach to quality assurance and development of HLW package acceptance requirements.

Three stages can be identified in the development of waste acceptance requirements:

- at the initial planning stage requirements are defined in a very general form on the basis of waste acceptance criteria and the overall waste disposal strategy which includes general information on the types and quantities of wastes that have arisen or are expected to be generated. The availability of certain host rocks for the construction of a repository may be recognized at this stage and disposal options identified. The development of preliminary qualitative and very general requirements may result from an initial safety assessment.

- after the area survey and preliminary site selection has been started, the general characteristics of potential disposal sites are known and a more detailed safety assessment of the disposal system can be made. At this stage an iteration between waste package characteristics and repository design may lead to the formulation of more specific preliminary requirements.

- at the site confirmation stage the final characteristics of the disposal system are established and a final safety assessment will lead to the firmer definition of waste acceptance requirements.

After licensing and the start of disposal operations, the waste acceptance requirements may be modified within the given licence or by an amendment to reflect operating experience, technical development and scientific progress.

TABLE I. CRITERIA REQUIRING A QUANTITATIVE APPROACH IN HLW SPECIFICATIONS

	Waste Acceptance Criteria	Waste form	Container	Waste Package
1.	Radionuclide inventory	X	o	X
2.	Thermal power, thermal loads and thermal effects	X	o	X
3.	Nuclear criticality	X	o	X
4.	Radiation effects, radiation damage and contamination control	X	o	X
5.	Mechanical behaviour	X	X	X
6.	Combustibility and thermal resistance	X	o	X
7.	Gas generation	X	o	X
8.	Free liquids	o	o	o
9.	Explosive and pyrophoric materials, fire and explosion hazards	o	o	o
10.	Compressed gases	o	o	o
11.	Toxic and corrosive materials	X	o	X
12.	Chemical durability	X	X	X
13.	Physical dimensions and weights	o	X	X
14.	Unique identification	o	X	X
15.	Compliance with codes and standards	X	X	X

X = Quantitative approach in HLW specifications is necessary.
o = Not applicable to HLW or qualitative/general data are sufficient.

Figure 1 illustrates the derivation of waste acceptance requirements: its dependence on the waste package management (here: waste disposal) and the criteria are expressed as well as the development of quality assurance requirements and the performance of quality assurance methods for the waste package. It is important to note that quality assurance, although mainly focused on process control during the conditioning stage, also encompasses the design and post-conditioning stages.

3.4. Quantitative requirements

Analysis of the qualitative criteria given in [3] must show which criteria need a quantitative approach. The specifications of the waste packages should contain at least quantitative requirements as shown in the Table I. It is not possible to assign actual limiting values to these quantitative requirements because they are highly dependent on many variable parameters, for example, the engineering design of the waste package, site specific repository conditions and compliance with the licensing criteria of the different national authorities and regulatory bodies.

4. DESCRIPTION OF CONDITIONING PROCESSES

There are two particular processes presently identified for the conditioning of high level waste:

- vitrification and packaging of high level liquid waste
- conditioning of spent fuel.

Vitrification processes have been developed to an industrial scale and have been successfully operated in fully active conditions since 1978 whereas spent fuel conditioning processes are presently in the development stage.

4.1. Vitrification processes

These processes may be described as a number of linked process steps and associated support processes (Fig. 2). The main process steps in a typical vitrification process may include:

(i) Temporary storage of the liquid waste at the vitrification plant where confirmatory analysis for selected radioactive isotopes and possible feed adjustments may take place. The activity of other radionuclides may be calculated by scaling factors from the isotopes analysed.

(ii) Feeding of the liquid waste into the main vitrification step where, in the case of reprocessing waste which has arisen from a nitric acid based dissolution process, denitration takes place (one stage or two stage process). The main purpose of the vitrification process is to convert the liquid waste into a solidified form by the addition of suitable glass making materials and fusing these materials at high temperature, typically greater than 1000°C.

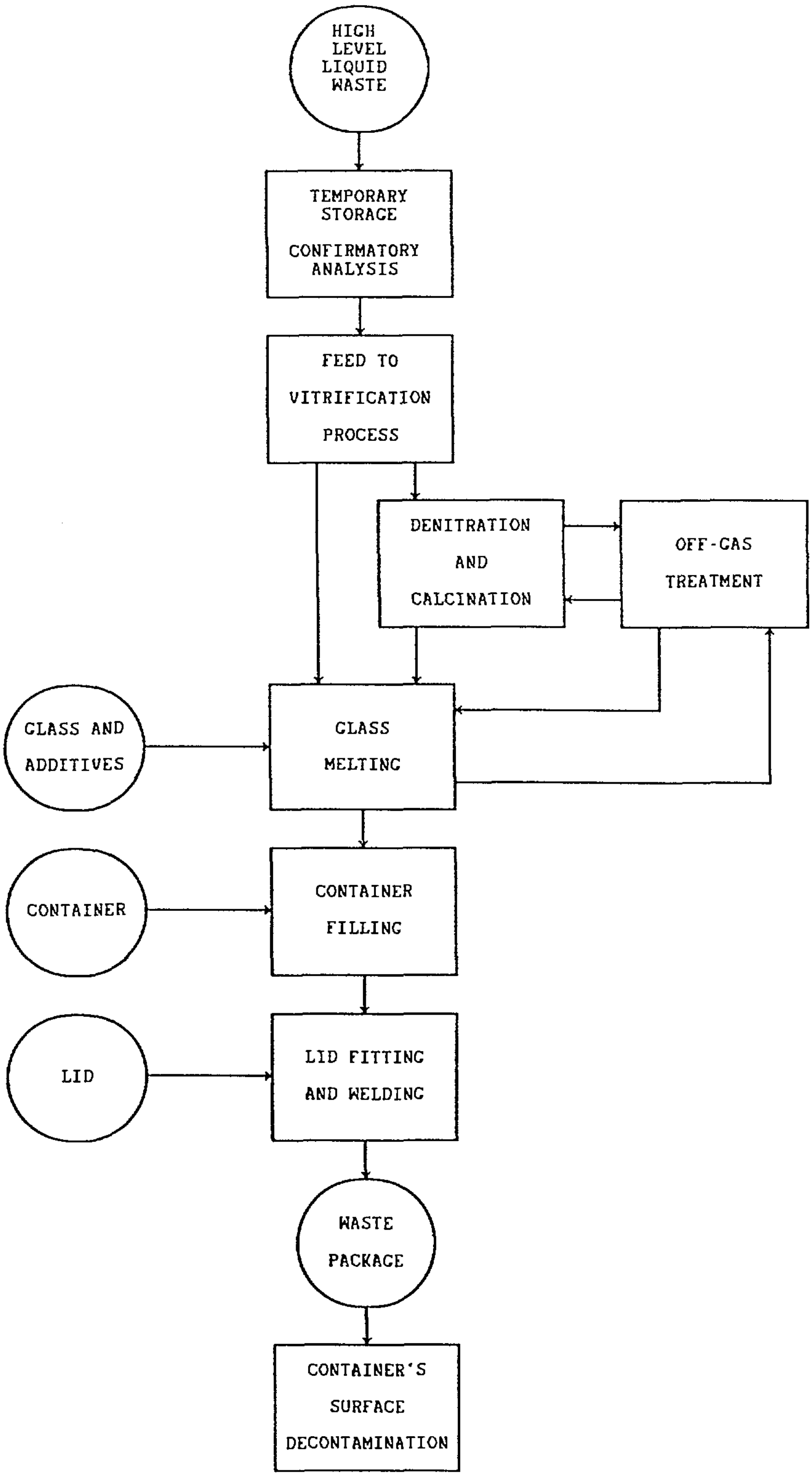


FIG. 2. HLW vitrification and packaging processes.

(iii) Mixing of the waste, glass making additives and other additives in the correct proportions, to guarantee the suitable composition of the final product, may take place in the melter or within the pre-melting stages. Melting of this mixture under acceptable conditions (temperature/residence time) gives a vitrified waste which is then poured into a preheated container where it cools down and solidifies.

(iv) The off-gas from the vitrification process typically includes air, water vapour, carbon dioxide, nitrous oxides from the denitration process (in the case of nitric acid based waste) together with entrained particulate and volatile waste components from the liquid feed. The off-gas treatment system is designed to remove radioactive material from the off-gas. This material may then be recycled to the vitrification process in order to reduce discharges from the vitrification process to acceptable levels.

(v) Lid fitting and welding by one of the well known welding techniques (i.e. plasma arc).

(vi) Decontamination of the external surfaces of the closed waste package. This may be necessary if surface contamination levels are found to be above acceptable limits.

The waste package is then transferred to an interim store for further cooling prior to transport to a repository. Intersite transfers would be made in shielded casks of type B(U) [7] construction.

Some specific vitrification processes have been described in several published documents [15-27].

4.2. Spent fuel conditioning processes

Concepts for the conditioning of spent fuel for direct disposal have been described in an IAEA Technical Report [28]. This document covers the major factors in the development of waste package designs and pre-conditioning processes, and reviews conditioning concepts from ten Member States.

The basic concept for spent fuel conditioning processes may be described as a number of linked process steps (Fig. 3) in the following way:

(i) Pre-conditioning - this consists of a number of alternative methods, to prepare the spent fuel assemblies for the main conditioning steps, which belong to the following 3 groups:

- the shape of the fuel assembly is not altered;
- the fuel assembly is dismantled, but the shape of the fuel rod is not altered; and
- the fuel rods are mechanically worked by penetration to allow degassing and folding, chopping or compaction to reduce volume.

Other possible process options are still under investigation, but are not considered in that document.

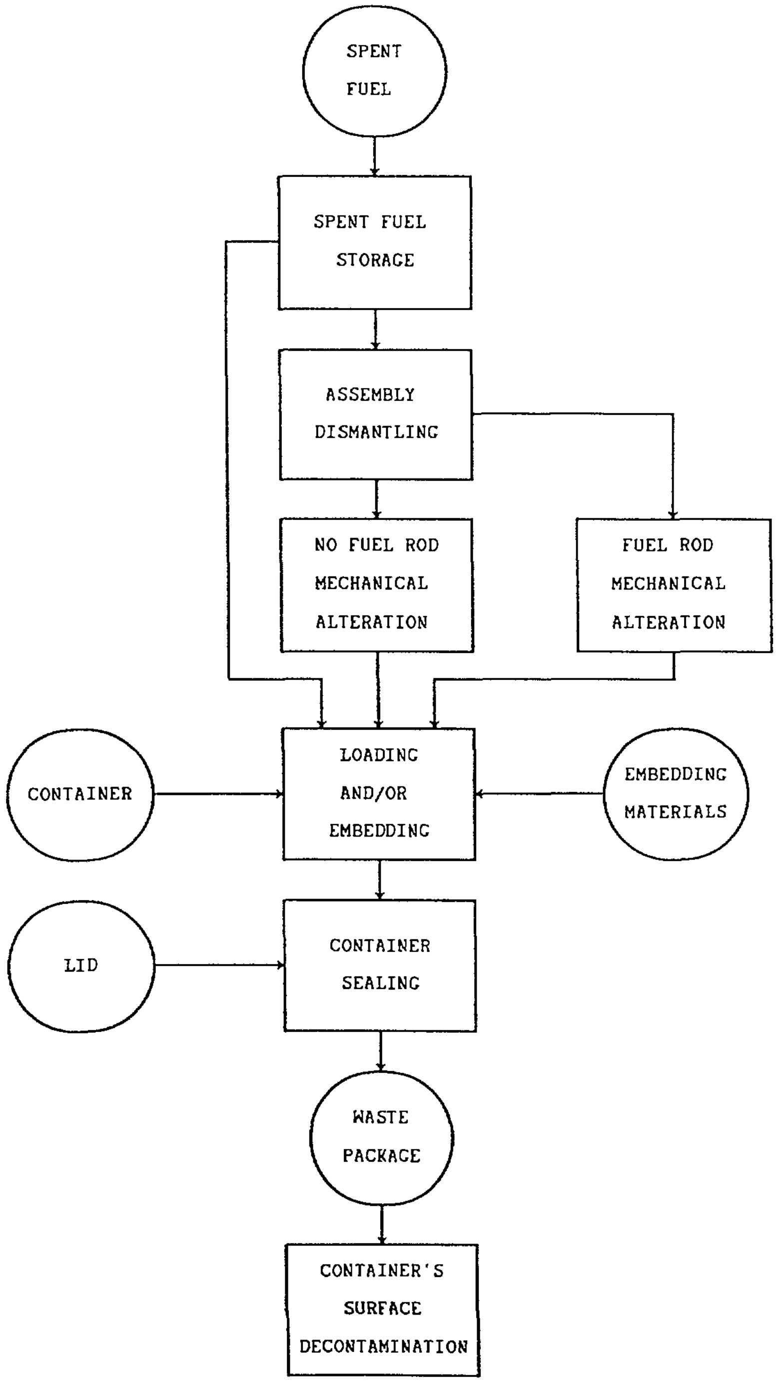


FIG. 3. Spent fuel conditioning process.

(ii) Placing the fuel into the container and/or embedding within the container in a suitable matrix such as copper by hot isostatic pressing or lead in cast or shot form.

(iii) Container closure using a number of well known welding techniques, such as electron beam or plasma arc. Other techniques are being considered.

(iv) Decontamination of the external surface of the sealed waste package. This may be necessary if surface contamination levels are found to be above acceptable limits.

The waste package is then transferred to interim store prior to transport to the repository. Intersite transfers would be also made in shielded casks of type B(U) [7] construction.

5. QUALITY ASSURANCE REQUIREMENTS AND METHODS

5.1. Basic considerations

Quality assurance methods to ensure the acceptability of the waste package will need to be applied at various stages during the design development and production of the waste package. These methods are required for existing processes to facilitate the provisions of reliable and sufficiently accurate data to satisfy requirements for transport and interim storage. In the future, these methods should also provide means of checking the acceptability of the waste package for disposal. In Section 5.2, the qualitative criteria provided in IAEA-TECDOC-560 are addressed individually, and a number of quality assurance requirements and methods are recommended which will provide information and data to satisfy those criteria. In making these recommendations it is recognized that alternative methods may be possible or that more advanced methods may be developed in the future. A basic objective in employing these methods is to minimize the necessity for sampling the highly active vitrified waste or for destructive testing of the waste packages in order to avoid generation of secondary wastes and additional irradiation of personnel.

The information and data obtained by the application of the quality assurance methods during the waste conditioning processes will fall into four distinct approaches:

- Direct measurement of the waste package or its constituent parts during or at the end of the process;

- Calculation, by suitable computational methods, from direct measurement of the raw waste or the waste package;

- Determination, by correlation with measured values, which cannot be directly calculated as described in Section 5.2.1;

- Estimation or calculation by correlation to basic research, development or commissioning information and data.

These approaches may be used either individually or in combination.

The precision with which measurements can be taken and the accuracy of determination of parameters are important factors in assessing the acceptability of determined values. It should also be noted that as experience is gained during commissioning and operation of the process and as technological improvements become available such precisions and accuracies can be expected to improve. Current development or testing away from the operating process and consequential improvements to equipment and/or the process may also lead to improvements in the determined information.

5.2. Requirements and methods

5.2.1. Radionuclide inventory

CRITERION: The type, characteristics and contents of radionuclides in the waste package should be known with sufficient accuracy to ensure compliance with authorized limits and should be documented accordingly.

Requirement: A detailed time-dependent inventory and the expected range of variations due to process variations of all those radionuclides which are relevant for the transport, storage and disposal of the waste package shall be determined by the time of the waste package production.

Method 1: Chemical and physical analytical methods can be applied to a representative sample of the raw waste prior to conditioning, in order to obtain direct measurements of radionuclide content, followed by decay calculations. The characteristics of the conditioning process should also be taken into account in determining the inventory of the waste package.

Method 2: Radionuclide inventories can be calculated using validated fuel inventory codes which include decay calculations, taking as input data the original composition of the fuel in, or associated with, the waste package and the irradiation histories.

Inventory estimates for each waste package are required to confirm that each waste package falls within ranges considered in licensing, safety and disposal assessments, and for estimates of releases under anticipated processes and events, and accident scenario conditions. Expected variations in radionuclide inventories are necessary to adequately quantify uncertainties in radionuclide release estimates for repository performance assessments.

Initial radionuclide inventory projections may be made without including process variations because it is currently estimated that they will have much less effect than the other sources of uncertainty.

If the process varibility studies indicate that this is not true, then the projections will be modified to include process variability.

For vitrified high level waste a combination of methods 1 and 2 may be necessary. In cases where a particular nuclide is either difficult to measure or measurement would not be to the desired accuracy then method 2 may be used to obtain the missing data. From the known behaviour of reprocessing and vitrification processes it may also be possible to calculate the inventory of certain nuclides from measured key nuclides by correlation factors or with the data determined using methods 1 and 2. A similar combination could be applied to spent fuel, however method 2 would be used as the main method as the codes mentioned have been specifically developed for the purpose of accurately predicting radionuclide inventories and are recognised as standard methods throughout the nuclear industries of the world.

The accuracy with which values may be determined by any method may be improved as a result of basic research and development; therefore the selection of which method should be used will depend upon the state-of-the-art in analytical methods and computational techniques at the time of determination of the data required.

5.2.2. Thermal power, thermal loads and thermal effects

CRITERION 1: The thermal power output of the waste packages shall comply with limits applicable for storage, transportation and handling prior to emplacement.

Requirement: Actual values for the expected thermal power output of each waste package and its range due to expected process variation shall be determined for an appropriate point in time.

Method: Thermal power of each waste package can be calculated through the radionuclide inventory by use of validated codes.

CRITERION 2: The thermal power output of the waste forms shall be limited such that any associated changes to physical, chemical and mechanical properties of the waste form, waste package components, other engineered barriers and repository components and the host rock do not adversely affect the safety of the overall disposal system.

Requirement: A heat generation limit with a specified accuracy shall be set to ensure that the temperature reached in the waste package components or the host rock do not impact their expected performance capabilities. The waste package shall not exceed this limit.

Method 1: Thermal power of each package can be limited by limiting the waste loading within the container through design of the container taking account of basic characteristics of the waste.

Method 2: Thermal power of each package can be limited by limiting the waste loading into the package during the conditioning process.

Method 3: Thermal power of each package can be limited by extending the storing time period of the raw waste (prior to conditioning) or of the conditioned waste (prior to disposal).

Method 1 may be achieved by testing the temperature related characteristics of the waste form and the temperature resistance of the container. The results of such development work should then be used to specify process flow sheet conditions related to the control of high level waste incorporation level for each waste package as mentioned in method 2.

It is anticipated that the producer shall provide the following data on the vitrified waste form:

- The transition temperature where the slope of the thermal expansion versus temperature curve shows a sharp increase.
- A time temperature transformation diagram that identifies temperatures and the duration of exposure at the temperature that causes significant changes in either the phase structure or the phase composition of the waste form. The requested data analysis and appropriate technical support shall be provided by the producer. The method used to produce these data shall be described.

In addition, the producer shall certify that after the initial cooldown, the waste packages to be shipped have been handled and stored in a manner such that the maximum temperature of the waste form has not exceeded the specified transition temperature.

The maximum temperature determined to be reached by the waste package in the repository shall not be such that significant alteration of the glass matrix or spent fuel will take place. Such deterioration for waste glass would include excessive crystallization or phase separation such that the chemical durability and physical integrity of the glass under repository conditions may significantly be reduced. For spent fuel, deterioration may include potential for oxidation to U_3O_8 with an attendent significant volumetric increase and tendency to powder. The extension of storage time for high level liquid waste and spent fuel prior to conditioning (method 3) may be used to reduce thermal power.

5.2.3. Nuclear criticality

CRITERION:	Waste packages shall be designed to preclude nuclear criticality of a single waste package. Handling, storage and disposal systems shall be designed and operated to ensure that criticality of arrays of waste packages cannot occur.
Requirement:	It shall be demonstrated that the waste package and arrays of waste packages will remain subcritical under conditions likely to be encountered from production through receipt and emplacement at the repository.
Method:	The inventory of fissile materials in the waste package shall be determined (see Section 5.2.1, Radionuclide inventory). Criticality calculations shall be made in order to validate the waste package design and the arrays of waste packages. Calculations of an effective multiplication factor K_{eff} may be used.

A K_{eff} calculation should be included in the nuclear safety analysis of the conditioning facility prior to finalization of its design. K_{eff} shall be shown to be sufficiently less than 1, after allowance for the bias in the method of calculation, the burnup of the spent fuel and the uncertainty in the experiments used to evaluate the method of calculation.

As the fissile material contained in the spent fuel is almost fully extracted by the reprocessing and boron (a neutron absorber) is a constituent of the glass, criticality is of minor importance for vitrified high level waste. Nevertheless, care should be taken to avoid the concentration of remaining fissile material in the conditioning process such that uncontrolled quantities are inadvertently fed to individual containers. For spent fuel the avoidance of criticality in the waste package is of particular importance. The conditioning process may include volume reduction and concentration of fissile material to a greater extent than in the fuel assemblies as discharged from the reactor.

Subcriticality of multiple waste package arrays must be ensured through all stages of waste management by the responsible facility operator. Sufficient information on the characteristics of the waste and the waste packages should be provided by the waste producer to enable the designer to confirm subcriticality under handling, storage and disposal conditions. In particular, subcriticality in a repository shall be ensured if water intrudes into the disposal area. Near and far field effects such as leaching of the waste form, transport of radionuclides in groundwaters, etc. will have to be studied to assure subcriticality under disposal conditions.

5.2.4. Radiation effects, radiation damage and contamination control

CRITERION 1: External radiation dose rates of conditioned waste packages shall be in compliance with the limits for facilities and equipment in which they will be handled, stored and transported prior to emplacement in a repository.

Requirement: A limit on beta-gamma and neutron dose rate levels shall be established. Dose rates shall be determined for an appropriate point in time.

Method: Expected dose rate and the range of expected variation can be calculated from radionuclide inventory. The gamma and neutron dose rates will be measured prior to shipment from the conditioning facility.

Both gamma and neutron dose rate calculations will be performed during the waste package design by independent means to verify the calculations. Experimental verification of the dose rate calculations is required.

During production, dose rates will be measured just before the waste package is transferred from the conditioning building to the interim storage facility. At any time of shipment, these may be measured again in the shipping and receiving facility.

CRITERION 2: Transferable radioactive contamination on the exterior of waste packages should be maintained within limits established for storage, transportation, and packaging facilities where wastes are to be handled.

Requirement: A limit on the level of removable radioactive contamination on all external surfaces of each waste package shall be established. Actual values for removable radioactive contamination on the external surfaces of each waste package shall be determined.

Method: Determination of removable radioactive contamination shall be performed by direct measurement, adapted to the nature of the surface. Visual inspection of the waste package should be provided additionally to ensure that no visible amounts of material remain adhered to the external surface of the waste package. If contamination levels exceed the established level the waste package shall be decontaminated.

Contamination limits are used extensively in nuclear industry practice to demonstrate that surfaces are free of removable contamination. A method of decontaminating the waste package and a technique of removing visible material from the surface should be developed. The waste glass container can be decontaminated, e.g. by air-injected frit slurry blasting [29-31] or by electropolishing [32]. It effectively cleans the container external surfaces, and generates no wastes which cannot be recycled to the vitrification process. Assurance that the waste packages do not exceed the specified contamination may be provided by a smear test of the container's external surfaces in the conditioning facility before transfer to interim storage.

CRITERION 3: Radiation dose rates shall be controlled to levels sufficient to ensure that radiation-induced processes (such as radiolysis) and degradation of material properties of the waste package, repository components and the host rock do not occur to an unacceptable degree.

Requirement: Radiation resistance of the matrix and near field materials shall be assessed. The composition of the waste form shall be within the limits of guaranteed values of an approved specification.

Method: Performance of the waste forms and near field materials shall be investigated under specific conditions to assess their radiation resistance.

If necessary, a limit in the activity content of the waste package is to be established to reduce the dose rate and probable degradations. Extended storage and additional shielding by an overpack are other possible solutions to reduce the external dose rate.

Borosilicate glass is rather insensitive to alpha recoil damage [33, 34]. The effect of beta-gamma irradiation is negligible. In the case of spent fuel, because of the extensive damage to the fuel caused by fission in the reactor, additional damage due to alpha recoil is insignificant [35].

5.2.5. Mechanical stability, mechanical strength and stress resistance

CRITERION 1: The waste packages must be able to withstand stresses arising without unacceptable deterioration in their ability to accomplish safety related functions.

Requirement: The mechanical stability of waste packages shall be assessed for the expected temperatures taking lithostatic or hydrostatic pressure and internal gas generation into account.

Method 1: The waste container should be designed to adequately withstand stresses induced during fabrication, transport and storage.

Method 2: Gas generation over time may be assessed by R&D work and the anticipated internal pressure evaluated during the design of the container.

Method 3: If, within the repository design, pressure effects such as lithostatic, hydrostatic and swelling pressure are considered to be important, then such effects on the waste package may be evaluated by experimental tests or models.

The mechanical stability of a waste container during fabrication, transport and storage of a waste package may be demonstrated during design, development and testing of a conditioning process. The conditioning of a high temperature waste form in a thin-walled container, as it is carried out within the vitrification of high level liquid reprocessing waste, requires investigations of reactions of the waste form and the container during pouring and cooling. Possible thermally induced alterations of the mechanical properties of the container should be taken into account, as well as interactions of the glass and the container (shrinking, surface increase) and reactions within the glass (crystallization, segregation of particles). The period and temperature of storage of a waste package should be reflected in the development of a waste container. The transportation cask will bear the load of transportation.

Internal gas pressure is not anticipated to be a problem, neither for high level waste nor for spent fuel. However, in the case of spent fuel the pressure effects of gases contained within the fuel rods should be included during the execution of method 2 above.

CRITERION 2: Waste packages shall be designed such that, in conjunction with handling systems, releases due to mechanical impact under foreseeable incidents are limited to acceptable values.

Requirement: An appropriate demonstration of the integrity of the waste package under impact conditions shall be performed.

Method: An appropriate drop test according to the handling operation configuration should be defined. Drop tests with prototypic non-active filled containers or qualified mathematical modelling can be carried out. The test results shall include information on measured container leak rates and container deformation after the drop test.

This demonstration may be made during design, development and inactive testing of the waste package. Requiring that the waste package passes a drop test obviates the need for establishing detailed specifications on the material properties of the container and the waste form. This is necessary to demonstrate the integrity of a container after a drop test or to limit the radionuclide source term associated with a possible leak of a container as a result of a drop accident to acceptable levels.

For transportation incidents, the transportation cask will bear the load.

5.2.6. Combustibility and thermal resistance

CRITERION 1: HLW packages shall not contain combustible materials which could burn under normal repository conditions or foreseeable incidents.

Requirement: The prevention of the ingress of combustible materials into the waste package shall be demonstrated by use of appropriate physical and administrative measures.

Method 1: The conditioning process should be designed and administratively controlled so as to prevent the introduction of combustible materials into the waste package.

Method 2: An evaluation of the waste package should be performed to demonstrate that for the range of material compositions, it remains non-combustible after having been subjected to anticipated temperatures.

The waste glass does not contain combustible materials (including any organic materials) because all components of the glass have already been exposed to high temperatures (>1000°C). The presence of glass in the container is sufficient evidence of the exposure of the glass to high temperature, because glass can only be poured into a container while molten. However, a spent fuel waste container may contain some materials potentially burnable under some circumstances. To prevent an ingress of combustible material into the waste package, procedures to control procurement, processing steps and physical barriers should be applied.

CRITERION 2: Waste packages shall be designed so that, in conjunction with handling systems, release due to thermal impact under foreseeable incidents are limited to acceptable values.

Requirement: Thermal resistance of the waste package under foreseeable incident conditions shall be demonstrated.

Method: Appropriate thermal tests according to handling operation configurations shall be defined. Thermal tests with non-active containers, or qualified mathematical modelling shall be carried out. Test results should include radionuclide releases, pressure tightness and alterations to the waste form.

This demonstration may be performed during design, development and inactive testing of the waste package. Requiring that the waste package passes a thermal impact test obviates the need for establishing detailed specifications on the thermal properties of the container, the closure system and the waste form. This is necessary to limit the radionuclide source term associated with a possible leak of a container as a result of a thermal impact to acceptable levels. If necessary, a thermally resistant overpack may be employed. Temperature increase due to radioactive decay shall be taken into account.

5.2.7. Gas generation

CRITERION: Gas generation in the waste package or in surrounding media should not jeopardize the performance of the overall disposal system.

Requirement: The volatile radionuclides and active gas generation associated with the waste package shall be determined. Interaction of those gases with the components of the overall disposal system shall be assessed in order to demonstrate the adequate performance of the overall disposal system.

Method: The gas content of a waste package and the future gas generation from radioactive decay should be calculated, its consequences assessed and taken into account in the design of the container and its closure system. It shall be ensured that the waste package does not contain free gas other than cover and existing radiogenic gases that are included at the time of production of the waste package.

Spent fuel will contain the gaseous radionuclides generated during reactor operation if the fuel rods remain intact. The vitrified waste is free of such gases due to its exposure to high temperatures (>1000^{o}C). In both waste forms gaseous radionuclides will be generated as a function of time and as a consequence of fission and decay of radionuclides. These amounts are small compared to the generation of gaseous radionuclides in a nuclear power plant.

Specific investigations for a given site and the waste package shall be carried out to assess the mechanisms and consequences of gas generation and release due to radiolysis and/or chemical reactions in the near field. These mechanisms could include the time dependance of the release of gaseous radionuclides from the waste package, the retention of gaseous radionuclides along the pathway and the possible generation of new pathways. Releases of gaseous radionuclides and the resulting dose to man can be assessed. (See also Section 5.2.4, Radiation effects).

5.2.8. Free liquids

CRITERION: The quantity of free liquids in waste packages should be sufficiently low to ensure that the performance of the overall disposal system is not jeopardized.

Requirement: It shall be demonstrated that after closure the waste package will not contain free liquids that could be drained from the container.

Method: The conditioning process should be designed and administratively controlled as to prevent free liquids in the waste package. The waste package shall be designed in such a way that free liquids are not generated and are prevented from entering the waste package.

For high level vitrified waste, conditioning methods eliminate free liquids. Damaged fuel rods may contain small amounts of water. Although this has to be taken into account, it is not expected to present a significant problem in practice.

A possible source of free liquids in the container is water in the case of a pond storage or the water/frit slurry used to decontaminate the container. Although liquids might be evaporated depending on the actual temperatures of the waste package, special means should be developed to ensure that liquids do not enter the container. This may be ensured by an appropriate closure of the container by a weld or seal. Suitable procedures should be used to prevent the introduction of any free liquids into the containers before or after waste loading. Tests should be performed in the facility to demonstrate the effectiveness of the procedures.

5.2.9. Explosive and pyrophoric materials, fire and explosion hazards

CRITERION: Waste packages containing materials that might cause explosion or ignition hazards shall not be accepted for disposal.

Requirement: It shall be demonstrated that no materials within a waste package are in a form which could cause an explosion or a pyrophoric reaction.

Method: Administrative controls and other measures that prevent the introduction of explosive and pyrophoric materials into the waste package shall be applied. An evaluation of the waste package shall be made such that for the range of material compositions, no hazards from explosion or ignition occur over the range of expected temperatures.

High level vitrified wastes and spent fuel are non-explosive as well as non-combustible (See Section 5.2.6, Combustibility and thermal resistance).

The specifications for container procurement should require that the container manufacturer cleans and degreases the container, and covers the nozzle opening with a gasketed metal cap for shipment. This will prevent the introduction of such materials into the empty container. Each container should be inspected before it is introduced into the conditioning facility to ensure that there are no visible prohibited materials in the container. After loading, the temporary container closure will prevent unwanted materials from entering the container before the final container closure is made. The final closure will then prevent prohibited materials from entering the container.

Possible hydrogen generation due to radiolysis may be expected outside the waste package in the near-field (See Section 5.2.7, Gas generation).

5.2.10. Compressed gases (pressurized gas containers)

CRITERION: Containers of compressed gases should not be accepted for disposal.

This criterion is not applicable for high level waste packages. However, see Section 5.2.5, Mechanical stability, and Section 5.2.7, Gas generation.

5.2.11. Toxic and corrosive materials

CRITERION 1: The contents of non-radioactive toxic and hazardous materials should be known with sufficient accuracy to ensure compliance with authorized limits.

Requirement: The quantity of non-radioactive chemically toxic and hazardous materials within a waste package shall be evaluated with an established accuracy.

Method: Quantities of non-radioactive toxic materials in the waste packages should be estimated and their accumulation in the repository assessed.

For vitrified high level waste it is very unlikely that the toxicity of non-radioactive materials would be significant compared with the toxicity of radioactive materials. However, special consideration should be given to container materials and embedding materials for spent fuel, taking into account the time dependent in growth of non-radioactive toxic material by radioactive decay.

CRITERION 2: Waste forms should not contain materials which will corrode the waste containers or other barriers in the disposal system.

Requirement: It shall be demonstrated that the contents of the waste form shall not lead to significant internal corrosion of the container such that there will be an adverse effect on normal handling during storage, transport and repository operation.

Method: The extent of chemical reactions within the waste package should be determined from either available sources or new experimental evidence, as necessary. Long term integrity data should be provided. The results of such an investigation should be used to show that the predicted performance of the waste container and other barriers can be achieved.

Long term corrosion testing indicates that the waste form will not lead to significant internal corrosion of the container, as long as liquid waste is excluded from the waste package. Control to be implemented to prevent liquid water from entering the waste package is described in Section 5.2.8, Free liquids.

5.2.12. Chemical durability

CRITERION: The chemical durability of the waste package should be sufficient to provide the required containment of radionuclides in the disposal environment.

Requirement: The release of radionuclides from the waste package shall be assessed if the waste package has a major contribution to the long term safety of the repository.

Method : The release of radionuclides by chemical mechanisms from a waste package may be assessed by the use of suitable testing on inactive and active samples under simulated repository conditions and modelling methods. The effects of pressure, radiolysis and temperature shall be taken into account.

The chemical durability of a waste package may be an important factor in the long term safety of a repository. The experimental results obtained on inactive and active samples may be incorporated into a predictive model which expresses the expected releases of radionuclides from waste packages under the repository conditions as a function of time. Chemical durability contributes to the source term for the release of radionuclides by possibly intruding groundwaters in the long term behaviour of the repository. The importance of the waste package barrier function, in comparison with the other barriers in the overall repository system, might be different depending on the site specific situation.

Chemical durability can be determined during the development of a waste package and the conditioning process. It should include the investigation of the waste form, the container and their interdependencies.

The characterization of a waste package may be made with standard test methods [36] which have been proposed for measuring the release of radionuclides from the waste forms and the container materials in contact with the envisaged host rock or intruding groundwater. Such tests should overestimate the predicted release of radionuclides under site specific conditions of the repository. This method is applicable if the calculated dose uptake for release of radionuclides from a repository in the long term safety assessment is considerably below the required limit, even under the conservative circumstances of this approach.

If such a simplified approach is not sufficient to evaluate the long term safety of the repository, a more realistic and sophisticated method can be applied by determination of the chemical durability under the conditions of the repository. This implies that in the complex system of a repository, consideration has to be given to a large number of parameters which can influence the chemical durability of the waste package.

The chemical composition of intruding groundwater should be determined. It can be modified, e.g. by the backfill, by the chemical interaction with the waste package, by radiolysis and temperature. This can cause alteration of oxidation states of radionuclides. The waste package can also undergo alterations, such as chemical interaction with intruding groundwater which can cause local corrosion, formation of protective layers or new surfaces as well as gas generation. The hydrostatic pressure of the intruding waters or the lithostatic pressure of the host rock or the swelling pressure of the backfill can cause mechanical alteration to the container and/or the waste form (see also Section 5.2.5).

Such a realistic system is characterized when the radionuclide release mechanisms can be described and modelled. The radionuclide release rates can then be expressed as a function of time.

National authorities may decide if it is necessary to confirm the scientific basis for the predicted release rates of the models by investigating active samples from the conditioning process.

5.2.13. Physical dimensions and weights

CRITERION: The physical dimensions and weights of waste packages should be compatible with provisions for transport, handling and emplacement.

Requirement: Configuration, dimensions and weight of the waste packages shall be controlled. Handling and lifting features shall be provided.

Method 1: The weight of a waste package may be measured directly.

Method 2: The weight of a waste package may be calculated by the control of the filling/loading process.

Method 3: The physical dimensions of a container may be checked against its design specification during its fabrication process.

The weight of a waste package is an important safety factor which shall be determined to avoid overloads on the handling equipment. The package weight can be detected in the shipping facility by placing the container on a balance after the filling/loading or by including a balance in the lifting device for a waste package.

The weight of a waste package can also be calculated if the weight of the empty container is known and the filling/loading is controlled (e.g. control of the glass level or accounting for the fuel rods/assemblies and embedding material).

The overall dimensions of the waste package shall be such that no forcing is required to place it in the disposal container, and there is compatibility with container geometry.

The dimensions of a container should be controlled by the container manufacturer before delivery to the waste generator and any effects of the conditioning process on the length, diameter, or wall thickness should be determined from measurements of non-radioactive waste packages produced under conditions representing the range of those expected in the plant.

5.2.14. Unique identification

CRITERION: Each waste package for emplacement in a repository should be marked with a unique identification.

Requirement: It shall be demonstrated that the identification method will not impair the integrity of the waste package and that it is legible until the emplacement in a repository. Each waste package identification shall be demonstrated to be consistent with the waste package's permanent written records.

Method: The waste package labelling system design and application method should be assessed in order to ensure that its durability and compatibility with the integrity of the container are ensured at least until the emplacement of the package in the repository. The consistency of each package label with the permanent records should be checked.

The adequate and unique identification of waste packages is necessary to attach the correct inventory to each waste package and to assure correct heat loading, e.g. of the interim store and the repository. The marking may be made before delivery of the container to the waste generator. It shall be controlled to ensure the adequate marking of the waste package and to avoid duplication of an identification number. The adequate recording of waste packages should be performed within the conditioning process.

TABLE II. SUMMARY OF QA REQUIREMENTS AND METHODS

QA Requirement (R)	QA Methods (M)	Time/Place HLW	SF
1. Radionuclide Inventory			
R-Determine radionuclide inventory	M 1 - Chemical and physical analysis	B/O	B/O
	M 2 - Calculations	B/O	B/O
2. Thermal Power, Loads, Effects			
R1 - Determine and limit thermal power for storage, transportation and handling	M 1 - Calculation	B/O	B/O
R2 Determine and limit thermal power for disposal	M 1 - By design of container	B/O	B/O
	M 2 - Process control	D/I	B/O
	M 3 - Extended storage	A/O	A/O
3. Nuclear Criticality			
R - Ensure subcriticality	M - Assessment of fissile material	B/O	B/O
4. Radiation effects and contamination			
R1 - Limit dose rates	M - Measurement and/or calculations	A/O	A/O
R2 - Limit contamination	M - Measurement/Inspection/Decontamination	A/O	A/O
R3 - Limit matrix degradation	M - Assessment of radiation resistance of		
	waste package and	B/O	B/O
	near field	A/O	A/O
5. Mechanical stability, strength and stress resistance			
R1 - Pressure related waste package stability	M 1 - Container design against stress	B/O	B/O
	M 2 - R & D work on gas generation	B/O	B/O
	M 3 - Testing and modelling for static pressure	B/O	B/O
R2 - Design for drop/impact resistance	M 1 - Waste package design and qualification	B/O	B/O

QA Requirement (R)	QA Methods (M)	Time/Place HLW	Time/Place SF
6. Combustibility and thermal resistance			
R1 - Exclude combustible materials	M 1 - Physical and administrative controls	D/I	D/I
	M 2 - Non-combustible waste package	B/O	B/O
R2 - Design for thermal impact resistance	M - Waste package design and qualification	B/O	B/O
7. Gas generation			
R1 - Determine volatile nuclides and inactive gas generated	M - Process design and control, calculations	B/O and D/I	B/O and D/I
8. Free liquids			
R - Limit free liquids	M - Process design and administrative control	B/O and D/I	B/O and D/I
9. Explosive and pyrophoric materials, fire and explosion hazards			
R - Exclude explosive and pyrophoric materials	M - Administrative control and waste package evaluation	D/I and B/O	D/O
10. Compressed gases (Pressurised containers)			
R - Not applicable	M - Not applicable	-	-
11. Toxic and corrosive materials			
R1 - Evaluate quantity of non-radioactive toxic material	M - Estimation of quantities	B/O	B/O
R2 - Compatibility of waste form with container	M - Determination of chemical reactions in waste packages	B/O	B/O

TABLE II. (cont.)

QA Requirement (R)	QA Methods (M)	Time/Place HLW	SF
12. Chemical durability			
R – Assess radionuclide release	M – Testing and modelling of radionuclide releases from waste packages under simulated repository conditions	B/O	B/O
13. Physical dimensions and weights			
R – Determine overall dimensions and weight	M 1 – Direct measurement of weight	D/I	D/I
	M 2 – Calculations of weight from process data	D/I	D/I
	M 3 – Dimensional control during container fabrication	B/O	B/O
14. Unique identification			
R – Provide unique and durable identifier	M – Labelling system design and container manufacture	B/O	B/O
15. Compliance with codes and standards			

Legend for Time & Place

Time: B = Before the conditioning process
D = During the conditioning process
A = After the conditioning process

Place: O = Outside the conditioning process
I = Inside the conditioning process

Waste: HLW = Vitrified high level reprocessing waste
SF = Spent fuel

5.2.15. Compliance with codes and standards

CRITERION: Waste forms and waste packages shall be certified as being in compliance with applicable codes and standards prior to acceptance for disposal.

Requirement: The waste acceptance requirements and the quality assurance arrangements shall be demonstrated to be in conformity with the applicable national legislation, codes and standards.

The competent national authorities will determine the applicable legislation, codes and standards for the conditioning of radioactive wastes for interim storage and disposal. The waste acceptance requirements and the quality assurance arrangements shall be formulated against the applicable national legislation, codes and standards.

5.3. Implementation of quality assurance methods

Section 3 introduced the concept of interactive development of quality assurance requirements between the waste package producer and the repository designer/operator. In the case of waste being produced at the present time, perhaps several decades before the repository operates, this process is inhibited by the lack of specific repository data upon which to base the development of the requirements. However, such a process should commence at the earliest opportunity in order that the quality assurance methods presently being applied may be tailored, as far as possible, to take account of repository requirements. This interactive process will also assist the repository designers. The selected quality assurance methods suggested in the previous section may be categorized generally by place and time as shown in Fig. 1, i.e.

(i) Before, outside conditioning process (B/O);
(ii) During, inside or outside conditioning process (D/I or D/O);
(iii) After, outside conditioning process (A/O).

Table II summarizes the main technical areas, the quality assurance requirements and the quality assurance methods. It also indicates where and when these methods should be applied. The time and location of such methods is important particularly when planning a new waste package conditioning method.

6. CONCLUSIONS

Quantitative waste acceptance requirements for disposal cannot be finally formulated until a repository site has been selected and characterized. Development of these requirements will also need to take into account the characteristics of those high level waste packages which have already been produced. However, qualitative waste acceptance criteria have been developed and this document recommends the quality assurance measures which are necessary to meet these criteria for high level waste packages. The application of the recommended quality assurance requirements and methods will contribute to a high probability of compliance with future quantitative waste acceptance requirements. Quality assurance of the waste packages is an important part of the overall quality assurance programme for radioactive waste management.

Because of the difficulties associated with sampling and testing of high level waste packages, it is preferable to assure the quality of the waste form by other means (e.g. basic research, process and waste package design/development, in-process control and data acquisition and post-conditioning non-destructive checks) and this is reflected in the recommended quality assurance methods. Destructive testing of waste packages should only be applied if the required data are not or cannot be obtained by other means.

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DEFINITIONS

The following definitions are intended to assure proper understanding and interpretation of the key terms used in this Technical Document. They have been taken from IAEA-TECDOC-447, "Radioactive Waste Management Glossary" [37] or from IAEA-TECDOC-560, "Qualitative Acceptance Criteria for Radioactive Wastes to be Disposed of in Deep Geological Formations" [3] and Safety Series No. 50-C-QA "Code on the Safety of Nuclear Power Plants" [2]. Some terms which are not contained in these documents have been defined in a sense in which they are commonly used in the radioactive waste management field, others should be considered as an explanation in the context of this document only.

audit: A documented activity performed to determine by investigation, examination and evaluation of objective evidence and adequacy of, and adherence to, established procedures, instructions, specifications, codes, standards, administrative or operational programmes and other applicable documents, and the effectiveness of implementation.

chemical durability: The ability to withstand the effects of chemically induced processes such as corrosion, dissolution, phase transformations, etc.

conditioning of waste: Those operations that transform waste into a form suitable for transport and/or storage and/or disposal. The operations may include converting the waste to another form, enclosing the waste in containers, and providing additional packaging.

container, waste: The vessel into which waste is placed for final disposal, conversely the final barrier protecting the waste from external intrusions. For example, molten HLW glass would be poured into a container where it would cool and solidify. In a multibarrier system the sealed container would then become the final barrier protecting the waste form against intrusion by water.

criteria: Principles or standards on which a decision or judgement can be based. They may be qualitative or quantitative. Acceptance criteria are set by a regulatory authority. (Some Member States use terms such as 'protection goals' instead of 'acceptance criteria'.)

deep geologic repository: A repository constructed, usually in consolidated rock, at a depth of several hundred meters or more in a continental formation.

design: The process and the result of developing a concept, detailed plans, supporting calculation and specifications for a nuclear facility and its parts.

heat generating waste: Waste which is sufficiently radioactive that the energy of its decay significantly increases the temperature of its surroundings.

high level waste:

(i) The highly radioactive liquid, containing mainly fission products, as well as some actinides, which is separated during chemical reprocessing of irradiated fuel (aqueous waste from the first solvent extraction cycle and those waste streams combined with it).

(ii) Spent reactor fuel, if it is declared a waste.

(iii) Any other waste with a radioactivity level comparable to (i) or (ii).

incident: An occurrence of an action or situation likely to lead to adverse consequences.

long term: In waste management, refers to periods of time which exceed the time during which administrative controls can be expected to last.

non-conformance: A deficiency in characteristics, documentation or procedure which renders the quality of an item unacceptable or indeterminate.

operation: All activities performed to achieve the purpose for which the plant was constructed, including maintenance, refuelling, in-service inspection and other associated activities.

overpack (used as a noun): Secondary (or additional) external containment for packaged radioactive waste.

overpack (used as a verb): Application of a component as described above.

quality: The totality of features and characteristics of an item or service that bear on its ability to satisfy a defined requirement.

quality assurance: All those planned and systematic actions necessary to provide adequate confidence that an item or service will satisfy given requirements for quality.

radioactive waste: Any material that contains or is contaminated with radionuclides at concentrations or radioactivity levels greater than the 'exempt quantities' established by the competent authorities and for which no use is foreseen.

records: Documents which furnish objective evidence of the quality of items or services and of activities affecting quality.

regulatory authority (national): An authority or system of authorities designated by the Government of a Member State as having the legal authority for conducting the licensing process, for issuing of licenses and thereby for regulating the siting, design, construction, commissioning, operation, shutdown, decommissioning and subsequent control of nuclear facilities (e.g. waste repositories) or specific aspects thereof. This authority could be a body (existing or to be established) in the field of nuclear-related health and safety or mining safety or environmental protection, vested with such legal or environmental protection, vested with such legal authority, or it could be the Government or a department of the Government, or it could be an international agency.

requirement: A condition defined as necessary to meet product, material, or process criteria.

repository: A facility or designated site for storage or disposal of radioactive waste.

repository operator: The organization responsible for the operation of the repository.

site: The area containing a nuclear installation (e.g. a conditioning plant or a waste repository) that is defined by a boundary and which is under effective control of the implementing organization.

siting: The process of selecting a suitable site for a nuclear facility, including appropriate assessment and definition of the related design bases.

specification: A written statement of requirements to be satisfied by a product, a service, a material or process, indicating the procedure by means of which it may be determined whether the specified requirements are satisfied.

storage: The placement of waste in a facility with the intent that it will be retrieved at a later time.

testing: The determination or verification of the capability of an item to meet specified requirements by subjecting the item to a set of physical, chemical, environmental or operational conditions.

underground disposal: Disposal of waste at an appropriate depth below the ground surface.

waste acceptance criteria: Those criteria relevant to the acceptance of waste (packages) for storage, transport and disposal.

waste acceptance requirements: Those requirements relevant to the acceptance of waste packages for storage, transport, and disposal.

waste conditioner: The organization responsible for the facility where the waste is conditioned. The waste can be subsequently conditioned at many places, e.g. for transportation and storage at the site of generation and for disposal at the disposal facility.

waste disposal system: a combination of geological environment, a repository and waste packages emplaced in the repository.

waste form: The physical and chemical form of the waste (e.g. liquid, in concrete, in glass, etc.) without its packaging.

waste management: All activities, administrative and operational, that are involved in the handling, treatment, conditioning, transportation, storage and disposal of waste.

waste package: The waste form and any container(s) as prepared for handling, transport, treatment, conditioning, storage and disposal of waste. A cask or overpack may be a permanent part of the waste package or it may be re-usable for any waste management step. The waste package may vary for the different steps in waste management.

waste package specifications: These specifications define the requirements to be satisfied by the waste package for transport and storage and the projected requirements for disposal.

waste packing: Any component or assembly of components which is applied to a waste form during conditioning to prepare it for disposal.

waste producer: The responsible organization for the facility where the waste is generated.

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