# SPECIFICATION OF SAFETY REQUIREMENTS FOR WASTE PACKAGES WITH RESPECT TO PRACTICABLE QUALITY CONTROL MEASURES

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## ABSTRACT

Waste packages for disposal in a repository in the Federal Republic of Germany have to meet safety requirements derived from site specific safety analyses. The examination of the waste packages with regard to compliance with these requirements is the main objective of quality control measures.

With respect to quality control the requirements have to be specified in a way that practicable control measures can be applied.

This is dealt with for the quality control of the activity inventory and the quality control of the waste form. The paper discusses the determination of the activity of hard-to-measure radionuclides and the specification of safety related requirements for the waste form and the packaging using typical examples.

## INTRODUCTION

Within the scope of the licensing procedure for the repository in the KONRAD iron ore mine, acceptance requirements on waste packages are established in order to ensure safe operation of the repository (1,2). Connected with these requirements are quality control measures for demonstrating the compliance of the waste packages with these requirements.

On behalf of PTB (Physikalisch-Technische Bundesanstalt) which is responsible for the final disposal of radioactive waste in the Federal Republic of Germany, safety related requirements were specified to such an extent that practicable quality control measures can be developed.

## REQUIREMENTS FOR WASTE PACKAGES

Requirements for waste packages intended for disposal in the KONRAD repository are derived from the safety analyses and are essential criteria with regard to safety. A more detailed description of these requirements and their derivation is presented in Ref. 3.

The requirements for waste packages are related to the activity inventory, to the waste form as well as to the packaging. In order to consider different quality levels of various waste forms and packagings, six waste form groups and two waste classes were introduced. The requirements can be grouped together as basic requirements which are independent of the activity inventory and which have to be met by every waste package, and as additional requirements which apply to the particular waste form or the packaging, the latter both being dependent on the activity inventory.

Objectives of the basic requirements for waste disposal are the prevention of incidents, such as

- fire,
- explosion,
- criticality,

and the avoidance of contaminations. The essential basic requirements for waste products and packagings are presented in Ref. 3.

A waste packaging which meets only these basic requirements, is defined to belong to waste form group 01 and waste class I. For this waste form group and waste class the highest restriction exists regarding activity limits (see Table I, column "Incident" for representative activity limit values).

If the activity inventory exceeds the activity limits of waste form group 01, the waste form must meet the additional requirements of the subsequently higher waste form group. Thus, an activity increase of up to 3 orders of magnitude is possible within waste class I.

In order to further increase the activity inventory, it is necessary to proceed to waste class II. In waste class II all requirements are related to the packaging. The main design objective, here, is to ensure the integrity of the packaging in case of thermal loads.

By this measure - the use of waste class II - a further increase of the activity inventory is possible by a factor of about 20 over the maximum activity inventory of class I.

In this structure of waste classes and waste form groups, the maximum permissible activity inventories are directly related to the individual waste class or waste form group. This relationship is fixed in activity limitation criteria.

As a result of the incident analyses a limit value is specified for every nuclide, waste class and

TABLE I

Activity Limitation or Declaration values for Selected Radionuclides due to various Activity Limitation Criteria

| Nuclide | Incident 1) | Normal Operation 2) | Heating-up 3) |
|---|---|---|---|
| H 3 | -- | 2.3 E 08 | -- |
| C 14 | -- | 1.0 E 07 | -- |
| Fe 55 | -- | -- | 1.5 E 13 |
| Co 60 | 4.9 E 10 | -- | 2.6 E 13 |
| Ni 63 | -- | -- | 3.8 E 11 |
| Sr 90 | 1.9 E 09 | -- | -- |
| J 129 | 2.7 E 09 | -- | -- |
| Cs 137 | 4.0 E 10 | -- | 4.5 E 10 |
| $\Sigma$ $\beta/\gamma$ | 4.9 E 10 | 4.2 E 11 | 3.4 E 10 |
| $\Sigma$ $\alpha$ | 6.7 E 09 | 2.1 E 12 | 6.2 E 08 |

1) Activity Limits for Waste Class I, Waste Form Group 01 (for a nuclide mixture the contribution of each nuclide has to be taken into consideration)

2) Most Restrictive Declaration Values

3) Most Restrictive Declaration Values

waste form group. In the case of nuclide mixtures, the relative contribution of each nuclide has to be considered.

In order to limit the release of radionuclides during normal operation, special nuclides such as H-3 and C-14 have to be declared. For these nuclides, limit values are specified regarding their annual release; no limits are, however, specified for the individual waste package.

The activity limitation criteria with respect to the heating-up of the host rock in the post closure phase are similar to the criteria of the incident analysis.

If the waste packages delivered at the repository meet the described requirements, a safe operation of the repository in accordance with the German radiation protection regulations (4) is ensured.

## SCOPE AND QUESTIONS OF QUALITY CONTROL MEASURES

The examination that the waste packages comply with the safety requirements is carried out within the scope of quality control measures, (7). Two types of quality control measures can be applied to the control of waste packages:

- destructive or non destructive tests of waste packages.
  These measures are especially used for packages with "old wastes", which were produced and conditioned in the past.

- qualification and inspections of conditioning processes.
  This can be applied to packages with "recent wastes", which have to be conditioned in the future.

Due to radiation protection principles, quality control has to be restricted to spot checks; furthermore, the opening or destroying of waste packages should be avoided as far as possible.

These considerations lead to a system of quality control measures which is described in detail in Ref. 5.

In the case of "recent wastes", the conditioning process - i.e. the kind of fixation or packaging, etc. - is intended to be qualified in a way that the conditioned waste package meets the safety requirements. If this is not applied the compliance with the safety requirements has to be proved by examinations of the waste packages. During normal operation, this qualified process is examined by periodical audits of the conditioning process. Besides this, only minor additional measures are required in the repository, such as dose rate measurements and visual inspections.

If a sufficient documentation of the waste and the conditioning process exists and if the conditioning process is suitable for qualification, a later qualification of the conditioning process may be possible. In all other cases, the waste packages have to be examined. In order to reduce the volume of examinations, here also, spot checks are performed.

One major aspect regarding quality control is, specifying the safety requirements for waste packages in a way that practicable quality control measures can be developed. Thus, not only do the product properties such as "solid", "non-combustible" have to be quantified, but also, a lot of well defined quantities have to be proved by indirect methods because they are difficult to measure directly, e.g., the H-3 or C-14 activity.

Some qualitative requirements have to be quantified, e.g., the requirement that waste packages must not contain free liquids. In these cases, tolerance ranges have to be derived which are in accordance with the safety objectives of the acceptance requirements.

In the following chapters, some of our methods used to solve these questions are discussed using selected requirements and wastes.

## QUALITY CONTROL MEASURES RELATED TO ACTIVITY

Objective of quality control measures related to the activity inventory is to confirm the activity declarations of the waste producers. The problem is that in conditioned waste packages only nuclides with high-energy $\gamma$-radiation are easily detectable. In the case of $\alpha$- and $\beta$-emitting nuclides which are of major importance such as Pu, Sr-90, H-3 and C-14 (see Table II), the direct measurement requires opening of the packagings and destruction of the waste forms. Aside from being connected with a high expense of time and labor, this also entails a high radiation exposure of the personnel. Considering the activity concentrations of these nuclides - the activity values in Table I are presented for a waste package mass of 20,000 kg - the vastness of this problem becomes evident: proof of these nuclides is very difficult with technically feasible methods.

Thus, methods for indirect proofing should favourably be applied, e.g., the determination of the activity of hard-to-measure quantities by determining

their relationship to more easily measurable quantities, such as Co-60 and Cs-137 activity, mass or volume.

TABLE II

Exemplary Relationships of Resins of Nuclear Power Plants

BASIS : Co 60-activity = 4.1 $10^{11}$ Bq
Cs 137-activity = 2.0 $10^{12}$ Bq
mass [m] = 450 kg

| NUCLIDE | RELATIONSHIPS | ACTIVITY |
|---|---|---|
| H 3 | m 0.005 | 8.3 $10^{8}$ |
| C 14 | m 0.3 | 5.0 $10^{9}$ |
| Fe 55 | Co 60 1.5 | 6.2 $10^{11}$ |
| Ni 59 | Co 60 5 $10^{-3}$ | 2.1 $10^{9}$ |
| Ni 63 | Co 60 0.5 | 2.1 $10^{11}$ |
| Sr 90 | Cs 137 0.001 | 2.0 $10^{10}$ |
| J 129 | Cs 137 $10^{-6}$ | 2.0 $10^{6}$ |
| β/γ-nuclides | Co 60 0.2 + Cs 137 0.6 | 4.5 $10^{12}$ |
| α-nuclides | Cs 137 2 $10^{-4}$ | 4.0 $10^{8}$ |

In order to arrive at such relationships an intensive nuclide measurement program has to be carried out and the data has to be analysed statistically. This has been performed for waste from nuclear power plants.

Table II shows the results for relevant nuclides using PWR-resins as an example. In the same way the relevant nuclides for all kind of wastes from PWRs and BWRs were determined.

The relationships in Table II represent mean values for all German PWRs. However, using these mean relationships, the actual inventory of a waste package may be underestimated by a factor of up to five times. If, on the other hand as a conservative approach, the maximum values are used, a considerable overestimation of the actual inventory in the repository will result. Furthermore, using the maximum values is only applicable if the thereby overestimated activity inventory of existing waste packages does not pose any problems with regard to activity limit values.

In order to demonstrate that this procedure of using maximum values can be applied to all kinds of wastes, the activity limitation criteria were checked. Table III gives selected results of such a check for PWR-concentrates.

The upper part of Table III shows the characteristics of the waste. In the lower part, the column "Bq" shows, vertically, the nuclides to be checked. The next three columns present the results. Shown are the ratios of the calculated activity values (column 2) to the activity limit values resulting from the three activity limitation criteria, NORMAL OPERATION, INCIDENT, and HEATING-UP.

TABLE III

Examination of PWR Concentrates with Respect to Activity Limitation Criteria

| | | |
|---|---|---|
| waste | : | PWR concentrates |
| waste class | : | I |
| waste form group | : | 05 or 06 |
| packaging | : | VBA 400 (concrete container) |
| waste mass [kg] | : | 400 |
| Co 60-activity [Bq] | : | 1.6 E 10 |
| Cs 137-activity [Bq] | : | 3.0 E 10 |

ANALYSIS OF THE NUCLIDE SPECTRUM

| | | FRACTIONS OF LIMITATION VALUES | | |
|---|---|---|---|---|
| NUCLIDE | [Bq] | INCIDENT | NORMAL OPERATION | HEATING-UP |
| H 3 | 7.4 E 8 | | 1.7 E-1 | |
| C 14 | 8.0 E 7 | | 8.0 E-2 | |
| Fe 55 | 1.6 E 10 | | | 1.7 E-3 |
| Co 60 | 1.6 E 10 | 3.3 E-4 | | 6.2 E-3 |
| Ni 63 | 4.8 E 9 | | | 1.3 E-3 |
| Sr 90 | 9.0 E 7 | | | |
| J 129 | 1.8 E 4 | 6.7 E-6 | 8.6 E-3 | |
| Cs 137 | 3.0 E 10 | 7.5 E-4 | | 7.0 E-3 |
| Σ β/γ | 8.9 E 10 | 8.8 E-4 | 4.9 E-2 | 1.5 E-2 |
| Σ α | 9.0 E 6 | 1.3 E-6 | 1.7 E-6 | 6.0 E-4 |
| Σ | | 2.0 E-3 | | 3.2 E-2 |

With respect to INCIDENT and HEATING-UP, nuclides have to be declared only if the corresponding values (columns 3 and 5, respectively) exceed 0.01. These limitation criteria are not met if the actual values exceed 1. For NORMAL OPERATION, the corresponding values are 0.1 and 1. Thus, it can be seen from Table III that the H-3 activity from NORMAL OPERATION and the sum of the β/γ activity from HEATING-UP of the host rock are the only values that have to be declared. All other nuclides can be neglected.

Due to a lack of data and in order to cover all extremes, the approach of determining nuclides by indirect methods leads to a conservative overestimation. However, the survey carried out shows that the indirectly determined nuclide vectors of all wastes from nuclear power plants meet the activity limitation criteria. It can, thus, be stated that the derived nuclide relationships are a suitable basis for quality control measures related to the activity inventory of waste packages.

## QUALITY CONTROL MEASURES RELATED TO WASTE FORMS

The installation of quality control measures requires characteristics for each individual acceptance requirement which can be examined with quality control measures. In deriving these quantities both the safety objectives of the requirement and the practicability of the quality product control measures have to be considered.

Table IV illustrates these characteristics for selected requirements. In column "Requirements", selected requirements are presented related to their product properties. In column "Characteristics", the corresponding quantities are specified.

TABLE IV

Transformation of Requirements to Measurable Quantities
(Examples)

| PRODUCT PROPERTIES | REQUIREMENT[1)] | CHARACTERISTICS |
|---|---|---|
| activity inventory | nuclide-specific activity limits | key-nuclide activity |
| combustibility | waste form is not combustible | percentage of organic material < 1 % |
| solidity | waste form is a solid body or | percentage of powder < 1 % |
| | waste form is solid compacted | waste is compacted in cartridge with wall thickness > 0.75 mm and compacting pressure > 300 bar and free combustible percentage of activity < 1 % |
| purity of material | only metallic compounds | percentage of non-metallic compounds < 0.1 % |
| others | waste form contains no mobile liquids | percentage of liquids in the package < 1 % |
| | no putrefaction or fermentation | gas production rate < 1 ml/h |

1) These selected examples do not apply to every waste form groups

The following example may illustrate in which way safety considerations are taken into account. The requirement that "the waste form is a solid body" (3) primarily means that the waste form has to be of monolithic structure. However, as a result of the safety analyses it can be stipulated that even in the case of incidents assuming a total loss of packaging, a powder content in the waste form of up to one percent does not lead to any inadmissible radiological effect. Thus, the requirement "percentage of powder < 1 %" which could arise for example due to abrasion was proposed to be the corresponding characteristic.

The examination of waste packages properties is dependent on the waste form group and the packaging. Table V shows this relationship for selected requirements, waste form groups and waste classes.

TABLE V

Examinations of the Requirements for the Waste Form
(Examples)

| REQUIREMENT | WASTE CLASS I | | | | | | WASTE CLASS II |
|---|---|---|---|---|---|---|---|
| | 01 | 02 | 03 | 04 | 05 | 06 | 01 - 06 |
| activity inventory | X | X | X | X | X | X | X |
| combustibility | | X | | | | X | |
| solidity | | | | X | 0 | X | |
| purity of materials | | | 0 | | | 0 | |
| free liquids | X | X | A | X | X | A | (X) |
| putrifying and fermenting | X | X | A | X | A | A | (X) |

A : examination is or can be covered by examination marked 0

(X) : optional examination for leak-tight packages

Additionally, the extent of quality control measures is influenced by the treatment of the waste. E.g., if the waste is incinerated, the requirements regarding combustibility, free liquids, putrefaction and fermentation can be assumed to will be met.

Some typical waste treatments which positively influence safety related properties of the waste packages and which may be credited to reduce quality control measures are

- o shredding of crude waste,
- o incineration of crude waste,
- o cementation of drums in larger container.

Particulary the last mentioned cementation can be applied to nearly all kinds of old-waste packages, even to conditioned waste packages. On account of this treatment, the requirements regarding combustibility and free liquids may be cancelled, thereby achieving a practicable level of quality control measures for old waste packages.

## QUALITY CONTROL MEASURES RELATED TO PACKAGING

The compliance of the packaging with the acceptance requirements is mainly ensured, on one hand, by a type qualification comparable to that of the IAEA transport regulations (6) and, on the other, by

an adequate quality assurance system. With respect to type qualification, the quality control measures reduce to checking for proper type qualification of the delivered packagings.

Additional quality control measures pertain to visual inspections in the repository in order to ascertain that requirements such as "the packaging must not be damaged" are met. Table VI shows some typical requirements and the corresponding characteristics which allow the derivation of quality control measures and some proposals regarding tests and examinations.

TABLE VI

Selected Requirements for Packagings

| Requirements | Characteristics | Proposed for tests and examinations |
|---|---|---|
| undamaged packaging | deformation<br>decrease of wall thickness | visuell<br>inspection |
| specified leak tightness | leakage rate | He-leakage test |
| mechanical integrity*) for waste class I | size of opening due to mechanical impact; behavior of waste form during fire | drop test and fire test or theoretical analysis |
| mechanical integrity*) for waste class II | leakage rate | drop test with subsequent He-leakage test |

*) with respect to incidents

The described system of quality control related to the packaging ensures a high level of quality, same time minimizing the required expense time and labor and, above all, keeping radiation exposure of the inspection personnel at a minimum.

## CONCLUSION

Within the scope of safety analyses, requirements for waste packages were derived with respect to the activity inventory, the properties of the waste form and the packaging (1,2,3). As a number of requirements exist which require further specifications or which contain quantities which are very difficult to measure directly, every requirement existing was checked for both old and recent wastes for developing characteristics with respect to optimal proofing methods.

This resulted in a set of practicable quality control measures and suitable specifications which consider adequate safety objectives while avoiding high expenses and disproportionately high radiation exposure of the personnel.

## References


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