nagra National Cooperative for the Disposal of Radioactive Waste

TECHNICAL REPORT 93-21

Model Radioactive Waste Inventory for Swiss Waste Disposal Projects

Volume 1: Main report
Volume 2: Database

June 1994

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ABSTRACT

A model radioactive waste inventory has been developed for Swiss waste disposal projects. It describes the conditioned and packaged waste expected to arise from i) the 5 operational NPP's (3 GW(e) over 40 years), i.e. operational, decommissioning, reprocessing and non-fuel reactor waste, ii) medicine, industry and research and iii) decommissioning of a high power beam (1.5 mA, 600 MeV p) accelerator facility. The characterisation data of 80 waste package types with yearly package productions are reported in the form of a database in Volume 2. The waste types are provisionally allocated to and summed for the 3 waste categories to be disposed of in the 2 Swiss repositories, the first for short-lived LLW and ILW and the second for HLW including long-lived ILW. Uncertainties in the projected repository inventories are discussed.

SUMMARY

A model radioactive waste inventory (MIRA) has been developed for Swiss waste disposal projects (repository planning, safety assessment).

The model inventory describes the conditioned and packaged wastes that are expected to be produced by the 5 operational Swiss NPP's (3GW(e) over 40 years, with the last NPP assumed to be shut-down in 2024). The waste categories assumed are: operational waste, exchangeable non-fuel reactor core components, decommissioning waste and reprocessing waste (with the assumption of complete reprocessing of spent fuel). The model inventory also details the wastes arising from medicine, industry and research between 1984 (previous waste arisings were sea-dumped) and 2053. The decommissioning wastes of the PSI 600 MeV proton (1.5 mA) accelerator facility, assumed to be shut-down in 2034, are also included.

The waste arisings are described by 80 waste sorts each defined using an average waste package description linked to the number of packages arising with time. Radionuclide activities (overall 117 nuclides with $t_{1/2} > 60$ days), other radiological characteristics, material content, specific properties (such as surface to mass ratios for metals), package internal drawing and predicted maximal values of the package are provided in a database of the waste sorts given in Volume 2. This data is presented and discussed in Volume 1, along with a short description of the origin of the raw waste.

Within Volume 1 a provisional allocation of the waste sorts to the 2 repository types "SMA" and "LMA/HAA" has been made. This results in 3 waste sort classifications: SMA (short-lived LLW + ILW), LMA (long-lived ILW) and HAA (HLW). The summation of the waste volumes, activities, radiotoxicities and materials gives an overview of the repository waste contents. Uncertainties associated with the repository inventories data have been analysed. The results provide an indication of the waste volume reserves to be considered for the repositories and of the relative accuracy of the activities for the safety analyses.

ZUSAMMENFASSUNG

Im Rahmen der schweizerischen Endlagerprojektierung (Endlagerplanung, Performance Assessment Studien) wurde ein modellhaftes Abfallinventar MIRA entwickelt (Modellhaftes Inventar der Radioaktiven Abfälle).

Das modellhafte Inventar beschreibt die konditionierten und verpackten Abfälle, die von den 5 schweizerischen Kernkraftwerken (3 GW(e)) während der 40-jährigen Betriebsdauer der KKWs produziert werden, wobei man davon ausgeht, dass alle Kernkraftwerke bis 2024 stillgelegt sein werden. Der Abfall wird entsprechend seiner Herkunft in 4 Kategorien eingeteilt: Betriebsabfälle, alle austauschbaren Reaktoreinbauten ausser Brennelemente, Stillegungsabfälle und Wiederaufarbeitungsabfälle (Wiederaufarbeitung aller abgebrannter Brennelemente). Das Inventar beinhaltet ferner Abfälle aus Medizin, Industrie und Forschung, die seit 1984 entstanden sind und bis 2053 anfallen werden (Abfälle vor 1984 wurden der Meeresversenkung zugeführt). Des weiteren sind die Stillegungsabfälle der Anlage des 600 MeV Protonen Beschleunigers (1.5 mA) vom PSI mitberücksichtigt, der voraussichtlich bis 2034 in Betrieb sein wird.

Das Abfallaufkommen wird in 80 unterschiedliche Abfallsorten eingeteilt, die jede für sich als ein durchschnittliches Abfallgebinde beschrieben ist und das zu erwartende Abfallaufkommen mit einbezieht. Band 2 enthält die reine Datenbasis über Aktivität der Radionuklide (insgesamt 117 Nuklide mit HWZ > 60 Tage), radiologische Eigenschaften, Materialinhalte, abfallspezifische Eigenschaften, Zeichnungen der Gebinde und Maximalwerte eines Abfallgebindes. Im Band 1 werden nach kurzer Beschreibung der Rohabfallherkunft alle Daten vorgestellt und diskutiert.

Im Band 1 werden die Abfallsorten vorläufig zwei Endlagertypen zugeordnet, dem Endlager "SMA" und "LMA/HAA" und weiter in folgende 3 Kategorien klassifiziert: kurzlebige schwach- und mittelaktive Abfälle (SMA), langlebige mittelaktive Abfälle (LMA) und hochaktive Abfälle (HAA). Unter Berücksichtigung aller Daten über Abfallvolumina, Aktivitäten, Radiotoxizitäten und Materialien kann das Inventar eines Endlagers und die damit verbundenen Unsicherheiten abgeschätzt werden. Die Ergebnisse erlauben Aussagen bezüglich einzuplanender Reserven für das zu erwartende Abfallaufkommen, sowie der Aktivitätsinhalte, die in die Sicherheitsanalyse einfliessen.

RÉSUMÉ

Un inventaire modélisé des déchets radioactifs (MIRA) a été fait pour être utilisé dans les projects suisses de stockage de ces déchets (i.e. construction des dépots, analyses de sûreté).

L'inventaire modélisé décrit les déchets conditionnés et containerisés qui, selon estimation, seront produits par les 5 centrales nucléaires suisses actuellement en opération (3 GW(e) sur 40 ans), la dernière étant arrêtée en 2024. Les catégories de déchets considérés sont: les déchets d'exploitation, les composantes échangeables du coeur du réacteur qui ne sont pas du combustible, les déchets de démantèlement et les déchets issus du retraitement du combustible usagé. (Pour ces derniers, l'hypothèse faite est celle du retraitement de tout le combustible déchargé). L'inventaire inclus aussi les déchets provenant de la médecine, de l'industrie et de la recherche entre 1984 et 2053 (ceux produits antérieurement ayant été immergés). Les déchets de démantèlement de l'accélérateur de particules du PSI (600 MeV p, 1.5 mA) et de ses annexes, supposés mis hors service en 2034, y sont aussi compris.

Les déchets sont décrits par 80 sortes de déchets, chacune définie par un colis de déchets de characteristiques moyennes et par le nombre de ces colis produits par année dans le temps. Les activités des radionucléides (en tout 117 isotopes de demi-période > 60 jours), les autres charactéristiques radiologiques, les matériaux du colis, d'autres propriétés, les dessins du colis et de sa configuration interne et les valeurs maximales du colis sont présentés dans une base de données des sortes de déchets constituant le Volume 2. Ces données sont introduites et discutées, après une brève description de l'origine des déchets, dans le Volume 1.

Une répartition provisoire des sortes de déchets entre les 2 types de dépots "SMA" et "LMA/HAA" a été faite (Volume 1). Elle définit ainsi 3 classes pour les sortes de déchets: déchets faiblement et moyennement radioactifs à vie courte (SMA), moyennement radioactifs à vie longue (LMA) et déchets hautement radioactifs (HAA). Les sommes par classe des volumes de déchets, de leur activités et radiotoxicités ainsi que de leur materiaux représentent le contenu des dépots. Les incertitudes sur l'inventaire des dépots sont analysées et indiquent les réserves en volume à considérer pour les dépots et la précision des activités pour les analyses de sûreté.

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1 INTRODUCTION

1.1 Objective

For the disposal of the Swiss radioactive waste two repositories are planned. One repository is called the SMA repository and will be for low and intermediate level waste that do not contain significant amounts of long-lived radionuclides ("SMA" waste: Schwach- und Mittelaktive Abfälle). The other repository will be for intermediate level waste that contains significant amounts of long-lived radionuclides ("LMA" waste: Langlebige Mittelaktive Abfälle) and high level waste ("HAA" waste: Hochaktive Abfälle). This repository is designated as the HAA/LMA repository.

This report presents a model of the Swiss radioactive waste inventory (MIRA: Modelhaftes Inventar der Radioaktiven Abfälle) and quantifies and characterises these wastes which are assumed to be packaged for disposal projects. Classification of these wastes into the 3 categories SMA, LMA and HAA and hence their allocation to the 2 repositories has been tentatively made. Due to the amount of information contained within the database, this report is made up of two volumes: the main report and the database.

The objective of the model inventory is to quantify and characterise all the Swiss waste packages that have been produced or are expected to be produced in the future. These data are for the design and safety assessment of the repositories including their operational phases. The database was provided from 1992 as an input for 3 specific Nagra projects "Projekt Endlager SMA-Anlagen und Betrieb" (SMA-repository facilities and operation), "Projekt Langzeit SA-WLB" (long-term safety analysis of the Wellenberg SMA-repository) and "Projekt Kristallin-I" (safety analysis of the high level waste repository in the crystalline basement in Northern Switzerland). Some values have been updated since 1992 leading to a few differences between the values provided in this report and those provided over the last 2 years.

1.2 Basic features of the model inventory

Within the model inventory the radioactive wastes are represented by Nagra as waste sorts. These waste sorts are defined by a typical waste package and a package production scenario. The SMA inventory is composed of 73 waste sorts, the LMA inventory of 6 waste sorts and the HAA inventory of 1 waste sort. The choice of 80 waste sorts is a result of a compromise between distinguishing the wastes in relation to their significant properties for disposal purposes (such as activities, materials and containers) and neglecting the deviations that occur in real conditioned wastes (such as variation of the 200 l drum type, of the cement matrix formula, of the raw waste, etc.). Waste that is expected to arise in the future and for which the conditioning and packaging has not yet been decided upon by the producer have been provisionally characterised in the model inventory so as to form conditioned waste sorts. The model inventory is a pro-

visional inventory, i.e. based on waste arising projections, as only about 5 % of the wastes considered have already arisen. Finally the inventory is based on two basic assumptions, one representing the present status of Swiss nuclear energy (without further nuclear power plants) and one for complete reprocessing of spent fuel.

As from December 1988 according to the revised HSK R-14 regulatory guide-line (HSK 1988) the producers are legally required to produce waste specifications for waste package types before starting new conditioning processes. Further to this, they are also required to specify the conditioned waste packages that have already been produced (including a data sheet for each individual package). This documentation (related to the real waste package types, as opposed to the waste sorts of the model inventory) will eventually form the basis for the real waste package inventory that will be used for the actual waste disposal. However, for the time being, the model inventory (MIRA) will be used as the basis for all planning activities and license applications. One modification will be made in the licence application for the construction of the SMA repository: the conceptual large concrete container for decommissioning waste used in this model inventory will be replaced by 4 new large containers of different sizes (see "Assumptions made for waste sorts that are no longer valid" in section 2.2.3).

1.3 Restrictions inherent in the model inventory

Restrictions inherent in the model inventory occur for the following reasons:

- the model inventory describes for some waste sorts the wastes as a whole, without detailing every specific real waste package type
- the model inventory, in some cases, has different designations than those commonly used or specified for real wastes

These restrictions are in some cases:

- difficulty in assigning a designated real waste to a model inventory waste sort using only their names
- a model inventory waste sort does not cover exactly the same type of real conditioned waste (differences in waste package components)
- a model inventory waste sort covers several real waste packages types each of which are conditioned (and specified) separately
- no model inventory waste sort exists for some specific real wastes. These real wastes are implicitly modeled within the waste sorts (e.g. waste from dismantled PSI accelerator components are included in the decommissioning waste of this facility)

In summary, not all wastes that are known (arisen or due to arise) are described explicitly in the model inventory. Such cases are mentioned in this report.

1.4 Relationship with the previous waste inventory

An initial Swiss waste inventory was produced for the Nagra "Projekt Gewähr 1985" (main report: NAGRA 1985, database: NAGRA 1984a), the intention being to demonstrate the feasibility of safe disposal of all expected arisings of Swiss radioactive waste.

The present model inventory is a development of this initial inventory. New waste sorts have been defined for raw waste that had previously not been completely characterised (e.g. accelerator waste and exchangeable non-fuel reactor components). The reprocessing wastes that were, in this initial inventory, only represented by the COGEMA waste sorts now also include those from BNFL. The waste sort characterisation has been extended and improved by using the results from various characterisation works. The developments in the waste production and conditioning as well as the requirements due to progress in repository planning and safety analyses since the time of "Project Gewähr 1985", have also been considered.

1.5 Inventory updating, QA/QC procedures

The inventory database is continuously updated on the basis of new information and complemented with specific data required by other projects. In particular, more information on wastes that have already been conditioned is expected in the next few years from formal waste specifications (see section 1.2).

The following procedures are used for the quality assurance of the database: 1) control of the characterisation work, 2) check of the data against any recently available independent characterisation data and 3) comparison with other radionuclide and material inventories. Controls, checks and comparisons are recorded. The basic nuclide and material data (half-lives, etc.) and computer programmes used within the MIRA data bank facility are documented and cross-checked.

1.6 Structure of the report

Chapter 2 presents the assumptions made for the model inventory, i.e. for the waste production scenario and waste descriptions.

Chapter 3 presents the type of data that is provided in the database of the waste sorts given in Volume 2.

Chapter 4 describes the raw wastes and the waste sorts and examines the methods used for the determination of the data.

Chapter 5 presents the SMA, LMA and HAA inventories for the 2 repositories and assesses these against possible uncertainties.

Volume 2 presents the database in which the primary data is used to describe the individual waste sorts.

2 ASSUMPTIONS MADE FOR THE MODEL INVENTORY

2.1 Waste production scenario

The waste production scenario for the model inventory only considers the installed 3 GW(e) of electricity generated by the 5 operational nuclear power plants. The 6 GW(e) scenario of "Projekt Gewähr 1985" has been reduced by half as a result of the moratorium introduced within the last decade. The five nuclear power plants in operation are:

KKB-I: Beznau I, PWR (Westinghouse), 350 MW(e),	operational since	1969
KKB-II: Beznau II, PWR (Westinghouse), 350 MW(e),	" "	1971
KKM: Mühleberg, BWR (General Electric), 320 MW(e),*	" "	1972
KKG: Gösgen, PWR (Kraftwerk Union), 920 MW(e),**	" "	1979
KKL: Leibstadt, BWR (General Electric), 990 MW(e),	" "	1984

* recently raised to 328 MW(e) and then to 355 MW(e)

** recently raised to 940 MW(e)

For the purposes of this study, the operational life of these power plants is assumed to be 40 years, at which time they will be dismantled to green field sites. Dismantling is defined as starting 2 years after shut-down and is expected to take 7 years. Waste production under this scenario will occur up to the year 2032.

It is assumed that all the spent fuel arising from the 3 GW(e) and 40 years power plant operation is reprocessed (i.e. total 120 GW(e)·years). Reprocessing waste production will occur up to the year 2033. It should however, be noted that the current reprocessing contracts cover only 30 % of the spent fuel expected to arise under this scenario and that reprocessing of the remaining 70 % is, as yet, undecided.

Waste from medicine, industry and research were sea-dumped until 1983. The production span of 1984 to 2053 assumed for the previous inventory, i.e. over 70 years, has been kept. Some decommissioning of research facilities or parts of them will certainly take place during this time. The decommissioning waste of the PSI 600 MeV proton accelerator facility, assumed to be shut-down at the earliest in 2034, has also been included in this inventory. Some minor technological type wastes will arise from the operation of the new central conditioning and storage facilities (zentrales Zwischenlager und Bundeszwischenlager, Würenlingen) and are mentioned here for the sake of completeness.

The time horizon for the model inventory is therefore taken to be 2053 (2033 for the waste from NPP's, 2040 for the PSI accelerator facility decommissioning waste and 2053 for the waste from medicine, industry and research).

2.2 Waste sort description

2.2.1 Waste sorts

The radioactive wastes of the model inventory are categorised into individual waste sorts. A waste sort is represented by a representative waste package consisting of average values and a production scenario for these waste packages. A waste sort is defined as containing similar raw wastes arising with similar conditioning and packaging characteristics, i.e. is defined by similar:

- type of raw waste (resins, activated material, etc.) including treatment if relevant
- activity spectrum and level (e.g. higher levels require additional internal shielding leading to differences in conditioning)
- type of conditioning material (cement, bitumen, etc.)
- type of container (200 l drum, large container, etc.)

Maximal values are also provided to cover the distribution of individual waste package values.

The waste sorts correspond as far as possible to expected arisings of real conditioned waste. However, the data reliability depends a great deal on the status of the real waste i.e. produced under specification, specified but not yet produced, produced but not yet specified, conditioning and packaging that is planned, envisaged or undefined. In these last cases the waste sorts have been tentatively defined by Nagra.

Distinction between waste sort and waste package type

According to Swiss regulatory guide-line HSK R-14 (HSK 1988), the real waste packages will be specified as "waste package types" (Abfallgebindetypen) which are described by using the same main characteristics as the MIRA waste sorts (Abfallsorten) and are characterised by using average values and guaranteed parameters (maximal allowable values). A data sheet is required by the guide-line for each real waste package produced. It is recognised that there will be many more waste package types for the real waste inventory than waste sorts for the model inventory. For the model waste inventory an attempt has been made to reduce the number of waste sorts for practical purposes by broadening the main characteristic ranges (for instance specificity of the raw waste, composition of the matrix cement and features of the drum). This reduction, however, must be compatible with the requirements for the design of the repositories (sufficient differentiation of waste containers, dose rates, etc.) and for the safety analysis (sufficient differentiation of nuclide activities, specific materials, etc.).

2.2.2 Designation of the waste sorts

A waste sort represents specific waste in the form of a typical waste package and a production scenario.

The system used for designating the waste sorts is as follows:

BA ("Betriebsabfälle") for routine operational wastes arising from NPPs, followed by the NPP abbreviation and a number defining the type of raw waste: for instance BA-KKB-2 (sludges and concentrates).

The power plants KKB-I and KKB-II are not any longer distinguished in this inventory. They are designated together by "KKB".

RA ("Reaktorabfälle") for exchangeable non-fuel reactor core components (such as control rods, fuel channels, etc.) followed by the NPP abbreviation and a number defining the type of raw waste: for instance RA-KKM-6 (fuel channels)

SA ("Stillegungsabfälle") for decommissioning waste, followed by the abbreviation of the facility and a number defining the type of raw waste: for instance SA-KKG-1 (NPP) or SA-PSIW-1 (PSI 600 MeV proton accelerator facility (Paul Scherrer Institute West Area))

WA ("Wiederaufarbeitungsabfälle) for waste arising from spent fuel reprocessing, followed by an abbreviation of the reprocessor, COG for COGEMA, BNF for BNFL and a number defining the type of raw waste: for instance WA-COG-4 (hulls and ends)

MIF ("Medizin, Industrie und Forschung") for waste arising from medicinal, industrial and research activities followed by a number defining the type of raw waste: for instance MIF-2 for tritium waste

Designation of waste sort categories and classes

In this report, categories and classes of waste sorts will be addressed.

A category (or a group) of waste sorts is designated by an appropriate reduction of the full waste sort designation:

WA: all reprocessing waste sorts (i.e. the reprocessing waste)
WA-COG: all reprocessing waste sorts from COGEMA
WA-4: hulls and ends from COGEMA and BNFL reprocessing

Waste sort classes SMA, LMA, HAA result from the allocation of the waste sorts to the repositories SMA and HAA/LMA.

Within the designation of the waste sorts an attempt has been made to cover through the numbering system the same raw waste. This system is however, in some cases approximative. BA-KKG-1 and BA-KKM-1 define the ion exchange resins arising from a PWR (KKG) and a BWR (KKM). SA-KKG-1 (core internals and reactor vessel part) is completely different from SA-PSIW-1 (activated

copper from a proton accelerator). Therefore, in the database, the waste sort designation is always accompanied by a very short main feature summary, i.e. BA-KKL-1, raw waste: resins, conditioning material: cement, container: 200 l drum.

To enable the reduction in the number of waste sorts (see the end of section 2.2.1), some waste sorts given in this model waste inventory result from the combination of waste sorts from the previous inventory (NAGRA 1984a) (for instance SA-KKG-3 to 6). In contrast, differentiations are made by using a letter; one example is SA-PSIW-1A and SA-PSIW-1B where the letter distinguishes the copper activity level and the requirement for different amounts of internal shielding within the same container type. Another example of differentiation is MIF-3A and MIF-3B where "3" stands for α-emitters, but A for actinide activities arising from fuel and B for Am-241 arising from smoke detectors.

The full list of the waste sorts included in this inventory is given in Appendix A of this main report.

Changes in waste sort designations from the previous inventory

In the previous waste inventory, issued for "Projekt Gewähr 1985" (NAGRA 1985, Nagra 1984a), some RA waste sorts were defined as BA waste sorts. The correspondence is:

BA-7 (NGB 85-02, NTB 84-47) → RA-6 (NTB 93-21)
BA-8 (NGB 85-02, NTB 84-47) → RA-7 (NTB 93-21)
BA-9 (NGB 85-02, NTB 84-47) → RA-4 (NTB 93-21)

2.2.3 Assumptions made for the waste sorts

In this section, it is intended to present the most important assumptions made for the waste sorts and their consequences for the waste characteristics given in the database. The more detailed assumptions will be examined in Chapter 4 which describes the determination of the waste sort characteristics.

Scope of assumptions

As already mentioned in section 1.2 and 2.2.1, the status of the real wastes can be classified as a result of the recent regulation for the waste specifications as:

1) waste package types specified and produced
2) waste package types specified but not yet produced
3) waste package types produced but not yet specified
4) waste package types planned
5) waste package types envisaged
6) waste package types undefined

The level of characterisation of the waste sorts of this inventory depends on these classes. For wastes of class 4, 5 and 6, the waste sorts have been tentatively defined by Nagra (activities of the raw waste, materials, internal configuration of the container). Therefore, the assumptions made by Nagra in the production of a waste sort database for use in the various disposal projects increase in scope from class 1 to 6. Some practical examples are: the reprocessing waste specifications (class 2) provide most of the required data (exceptions are some material components and some specific radionuclide activities) whilst for the reactor control rods (class 6) all data has had to be derived by Nagra except the control rod materials. One important consequence is that the reliability of the given maximal values that a waste package of a waste sort could show varies greatly. This reliability is high for the guaranteed values given for "specified" waste sorts and is low for waste sorts that have not, as yet, been formally specified.

Interdependence of assumptions

For the waste sorts whose packaging has been defined for this inventory, assumptions were made for 1) conditioning and packaging, 2) time of conditioning and packaging, 3) constraints on package dose rate and heat production and 4) simplification of the calculation. In some cases, the interaction between the assumptions leads to a few waste sort characteristics in the database which are not realistic. However, as they have no significant effect on the disposal projects, they have, as yet, not been modified. A typical example is given below.

The definition of NPP decommissioning waste sorts with conditioning and packaging 2 years after shut-down results in a few waste packages with very high levels of radioactivity and unacceptably high heat production values (e.g. SA-KKB-1: reactor internals and parts of the reactor vessel). In fact it is considered that it would not actually be feasible to condition these highly radioactive wastes 2 years after shut-down. These wastes would have to be cooled for a longer period of time in-situ. An additional benefit of a longer cooling time would be a reduction in the conditioned waste volumes due to a reduction in activity and hence shielding requirements (ALDER 1989).

Assumptions made for waste sorts that are no longer valid

Since 1980, Nagra has used the concept of a single large concrete container for decommissioning wastes and for some of the RA waste. This conceptual container has a external volume of 18.91 m^3 (2.05x2.05x4.5 m^3, 10 cm wall thickness) and a weight limit of 60 t. A longer version of this large container (4.8 m instead of 4.5 m, 20.17 m^3 instead of 18.91 m^3) was choosen for packaging the longer KKM control rods (RA-KKM-1A). This container has been found to be inadequate for many PSI accelerator facility decommissioning wastes. The weight limit leads to a restriction in the quantity of metallic components that can be placed into a container and as a result increases the overall total waste volume. For the model inventory the constraint of the weight limit was not followed in all cases as it was considered that, where this limit is exceeded, the package could

be viewed as 2 half containers. To solve this general weight problem (the weight limit already being critical for some NPP decommissioning waste) and to have a fixed design for the final planning of the repository, Nagra has recently defined 4 large concrete EC containers ("Endlagercontainer"):

EC-1: LxWxH 4.44x2.438x2.4 m^3; 25.98 m^3; max. 80 t; wall thickness 20 cm
EC-1S: LxWxH 4.44x2.438x2.4 m^3; 25.98 m^3; max. 80 t; wall thickness 20 cm
EC-2: LxWxH 2.438x2.438x2.4 m^3; 14.27 m^3; max. 56 t; wall thickness 20 cm
EC-2S: LxWxH 2.438x2.438x1.65 m^3; 9.81 m^3; max. 30 t; wall thickness 23 cm

S = heavy concrete container for high dose rate 200 l-drums

Therefore, the large container (18.91 m^3) used within the present model inventory has been superseded and has been replaced in the SMA-repository projects (facility and operation) by these EC containers. These EC containers will be considered in the 1996 licence application for construction of the SMA repository.

Note: A few large containers with dimensions other than given above for the SA-1,-2 and -3 containers have already been produced for components that have been replaced at the PSI accelerator facility. These containers have to be considered only in the repository planning study and are included in this model inventory as a dummy waste sort without characterisation of the waste (MIF-6).

With respect to the regulations, even the most active decommissioning waste could, previously, be packaged into such large containers (18.91 m^3). However, since 1985 a constraint of 1 rem/h (10 mSv/h) at 3 m distance for individual waste components without shielding was introduced into the IAEA transport regulations. A direct result of this constraint would be the use of type B containers for a small part of the waste if no other solution can be found.

A further point related to the regulations is the definition of when waste is classified as being radioactive. This can significantly affect the quantity of arisings of decommissioning raw waste. The Swiss regulation used in this work for the declassification of waste is summarised in Chapter 3 of reference (NAGRA 85). The activity concentration limits are in fact relatively high (20 nCi/g for β/γ emitters, 20 pCi/g for α emitters). In practice the actual waste activity concentrations are lower due to the effect of additional regulative constraints on the dose rate for specific waste categories (< 50 µrem/h at 10 cm distance for the NPP materials). A full revision of the radiological protection regulations are presently under way and could result in a slight increase in the decommissioning waste volumes.

Assumptions made for waste sort substitutions

The last point to be made, in concluding this section on assumptions, concerns the HLW vitrified residue arising from reprocessing by COGEMA and BNFL. COGEMA has specified this waste with nominal values corresponding to a refe-

rence PWR spent fuel. BNFL has done the same but with nominal values for 5 reference spent fuels (MAGNOX, AGR, PWR, BWR and Blend). As the waste package characteristics are very similar, the COGEMA residue has been selected so as to have only one waste sort (WA-COG-1) for the model inventory and characterisation. Note that the BNFL waste sorts (WA-BNF-MAG-1, WA-BNF-AGR-1, etc.) are not given in the database of this report.

2.3 Conclusions

A model inventory is essential for repository design studies, for safety analyses and for any further assessment which can make use of waste packages that more or less correspond to those expect to actually occur. The allocation of the waste to different repositories or repository caverns depending on radionuclide activities and materials is facilitated through the concept of waste sorts. However, the model features of the waste sorts constituting the inventory must always be remembered when using the database for such purposes.

In certain cases, the use of the current database in other projects has led to some modifications being made to the present model inventory. These modifications are indicated in the corresponding project reports.

3 DESCRIPTION OF THE DATA

3.1 Database and objectives of data

The principle of the MIRA database ("Modelhaftes Inventar der Radioaktiven Abfälle") given in Volume 2 is to present basic data describing the representative package of each waste sort and additional information needed by other projects. The objective in using this data is that all other subsidiary data that may be required can be derived from this basic data using associated computer programs (summations, activity decays, searches, etc.).

3.2 Introduction to the data reported

The MIRA database consists of seven forms for each waste sort and a material data file. An example of a set of forms is given in Appendix B:

Form 1: Representative waste package and production statistics
Form 2: Waste package maximal values
Form 3: Nuclide activities of the representative waste package (average and maximal values)
Form 4: Materials of the waste package components
Form 5: Chemical categories of materials
Form 6: Properties of the waste package
Form 7: Drawing of external container and of waste package internal configuration

The elemental composition of the materials quoted within the database are given at the end of Volume 2 of this report.

Note: To help the waste producer to determine and report the data to characterise their waste packages and to create their waste specifications a computer programme (ISRA) has been established. The model inventory (MIRA) uses the same basic programme, but to display the data, the 7 MIRA forms have been produced using a relational report writer.

These forms are described and explained in detail in the following sections. Where data are not available these are indicated by "k.A." (keine Angabe = no data).

3.3 Form 1: Representative waste package (Repräsentatives Abfallgebinde) Production statistics (Produktionsstatistik)

Note: Please see the forms given as example in Appendix B in conjunction with the following explanations.

This section summarises average values for the general waste package characteristics on two pages. The first page consists of:

Form designation and references ("Nagra-MIRA.....")

This section gives the MIRA form number (i.e. M-F1), the date of printing and internal database information. (This section occurs on all pages of the database).

"Abfallsorte"

This gives the waste sort designation ("WA-COG-5") with the type of the raw waste (Rohabfälle), the type of conditioning matrix (Zusatzstoff) and the type of container (Behälterhülle). (This information is repeated on the first page of each form).

"Representative waste package"

This gives summarised data for each waste package with average values for the variable parameters and includes: the geometry and external dimensions of the package, total mass, displacement volume, density, reference time for conditioning (valid for all time variables with respect to the waste discharge or spent fuel unloading), total activities (alpha, beta + gamma and sum), gamma dose rates (plus neutron dose rate for HAA/LMA wastes), heat output and surface contamination values.

The second page consists of:

"Production statistics"

This gives the period of arising, yearly arisings with cumulative total, total mass of arisings, total volume of arisings.

Caution: The annual arisings figure may entail a fraction of waste packages when the total number of waste packages (as a reference) has been divided by the total number of years of production for continuous production.

3.4 Form 2: Waste package maximal values
(Maximalwerte der Abfallgebinde)

This section gives maximal values calculated using engineering judgment if these values have not been specified elsewhere (very few specified values have been issued and received by Nagra from the waste producers). Maximal mass and dose rate data are required for the planning of the repository operation. Maximal total activities (alpha, beta + gamma and sum) and maximal radionuclide activities given separately in form 3 are used for the safety analyses of the repository operational phase.

The maximal mass has been estimated by assuming the most dense material given in the material composition of the waste matrix and filling material or from the transportation limit in the case of large containers. This should err on the pessimistic side but does not take into account instances where additional materials other than those specified are included in the packages (e.g. lead blocks, etc.).

The maximal activities, dose rate and heat output are simply derived from the average values by the use of an estimated factor. As no additional shielding has been considered, this can lead to inconsistencies when the dose rate is limited for a large container. In such cases, this factor is linearly derived from the maximal permitted dose rate of 0.1 mSv/h at 1 m for a large container.

Note: For the reprocessing waste sorts except WA-COG-5 and WA-BNFL-5 (both SMA waste), the maximal dose rates reported are the values specified by COGEMA and BNFL for the time of container dispatching. These are not therefore related to the reference time given for the waste sort characterisation.

The reliability of the values given depends on the status of the waste sorts (see section 2.2.3). Maximal values should be given by the waste package type specification or imposed by the waste acceptance criteria of the repository. Therefore, for most of the wastes the maximal values given are indicative.

3.5 Form 3: Activities of the representative waste package
(Aktivitäten des repräsentativen Abfallgebindes)

This section gives first the radionuclide activity spectrum in Bq for the average container of the waste sort. The values estimated for the maximum activities of a single container are also reported here to allow for a comparison with the average values. Activity values are reported only for nuclides with half-lives greater than 60 days. This applies also for members of long-lived decay chains such as U-237 ($t_{1/2}$ = 6.8 d). An activity "cutoff" nominally at 1 Bq per container is used to screen average activity levels when activities have been incorporated directly from computerised calculations (ORIGEN2 code, scaling factors, etc.).

Note 1: The activity "cutoff" leads to some nuclides which have been assessed as containing less than 1 Bq per container being deleted. Therefore, the absence of a nuclide from the list does not necessarily mean that its activity was not estimated.

Note 2: For most waste sorts the actinide and daughter activities are incomplete. When the activities have to be measured in a specific waste, only the more important actinide isotopes are usually measured and given for the waste characterisation.

To evaluate the effects of incomplete specification of actinide isotopes, a comparison was made between the 82 activities of the 4 decay chains calculated for PWR spent fuel (33 GWd/t) and a reduced set of 10 activities (U-235, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Am-243 and Cm-244) which give rise with time to the other decay chain members.

The results show that such a reduced set is sufficient for the characterisation. The α activity of the reduced set is 91 % at 4 y, ~ 100 % at 100 y, 98 % at 1,000 y, 93 % at 10,000 y, 77 % at 100,000 y and 88 % at 1,000,000 y. (The β,γ activity is ~ 100 % at 4 y, 92 % at 1,000,000 y with also a minimal value at 100,000 y of 88 %).

The maximum activity levels that a package could show are currently generated from the average values with an estimated "maximal" factor (see section 3.4) when a maximum value is not specified elsewhere. Some caution is required when using the individual (nuclide) maximum activity values, as individual uncertainties are not considered in these values.

3.6 Form 4: Materials of the waste package components
(Materialgemisch der Abfallgebindekomponenten)

Introduction

To describe and to characterise the material of the waste package components, a reference structure and designation of the waste package components has been defined by the authority (HSK 1988), see Figure 1. Typically a waste package is comprised of the following components:

- **raw waste** (Rohabfall)
- **additional material** (Zusatzstoff) which together constitute the
- **waste product** (Abfallprodukt), i.e. the waste matrix containing mixed raw waste and additional material, to which can be added
- **filling material** (Füllmaterial) to fill up the container in which some
- **inserts/fittings** (Einbauten), e.g. baskets, lost mixing paddles, etc. can be added and some

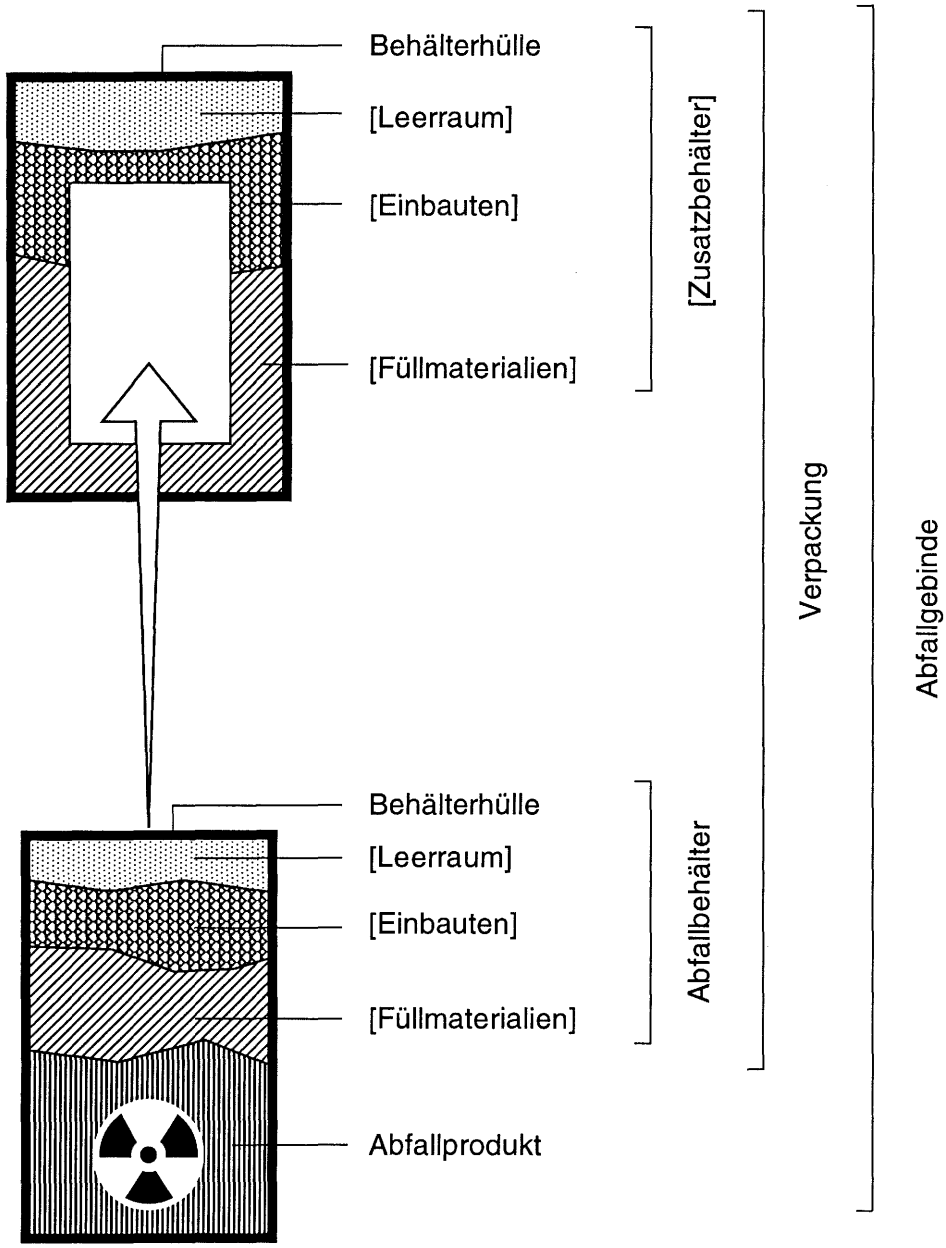


Fig. 1: Schematic drawing of a waste package as defined in reference (HSK 1988)

- **voidage/ullage** (Leerraum) can be left. The container envelope is named
- **casing** (Behälterhülle) and
  all these components constitute the
- **waste container** (Abfallbehälter)
  which can in some cases be put in a
- **cask/overpack** (Zusatzbehälter) having again
- **filling material, inserts, voidage and casing**
  The container, or the container with the cask/overpack, constitutes the
- **waste package** (Abfallgebinde)

The application of this system leaves some degrees of freedom especially for solid raw waste where the embedding materials can be viewed as either additional material or as a filling material. Another example is an internal cement liner which can be viewed as either filling material or as part of the casing.

Form description

Form 4 (in conjunction with the drawings of form 7) describes the components of the waste package, gives their mass, volume, density and their material composition (also with material, mass, volume and density data).

The introduction given above was made to explain the system used, the component designation and to indicate that, given the degrees of freedom mentioned, it could be difficult in some cases to understand the internal configuration of the waste package only from its components given in form 4. The waste package components with respect to the waste package configuration can in these cases only be understood by looking at the drawing of the waste package internal configuration given in form 7. In this form the components as used in form 4 are reported in a way so as to minimise any confusion.

Note: The materials are designated by explicit names (for instance zircaloy-4), ambiguous names (cement), chemical formula (SiO_2) or code name ("Gebinde X") ("drum X"). The clear understanding of the materials is obtainable from the material file of the database (Volume 2) where they are broken down into their elemental compositions.

Caution: The computer programme, when determining the masses, volumes and densities, starts by using the specified waste package component masses and fractional masses of their materials. However, the final constraint of the waste package volume must be satisfied. This is obtained by having, in certain instances, a minor inconsistency between the sum of the individual material volumes and the total volume given.

3.7 Form 5: Chemical categories of materials (Chemisches Inventar)

All materials specified in form 4 are categorised under the following chemical categories:

Organic materials (only shown if present)

This gives all identified organic materials in terms of their masses and mass fractions of total package mass. These materials can give rise to complexing agents which can result in increased nuclide mobility (e.g. cellulose, EDTA, etc.) or to gas production due to degradation.

Metallic materials (only shown if present)

This gives all identified "metallic" materials and its main purpose is to detail masses of all metals for corrosion studies leading to gas production.

Inorganic materials (only shown if present)

This gives all identified "inorganic materials", the information being used for various calculations (mass of cement to calculate the pH in the repository, etc.).

Unidentified materials (only shown if present)

In a very few cases, due to the nature of the waste, it has not been possible to complete the material composition and some unknown material is specified to indicate this.

Chemically toxic materials (only shown if present)

Some materials have been identified as being chemically toxic. If these materials are present their name is given here.

Hazardous materials (only shown if present)

Some materials have been identified as being potentially hazardous. If these materials are present their name is given here.

3.8 Form 6: Properties of the waste package (Eigenschaften der Abfallgebinde)

In this form some of the properties of the waste package which are used for repository planning and for safety analyses are given.

Dose rate distribution

A dose rate distribution, used for repository planning purposes, is given in this form.

The dose rate distributions attributed to each waste sort are based on engineering judgment (only a small amount of existing data is available) and must be considered as indicative. The package allocation to the dose rate log scale classes has been made to be approximately coherent with the average dose rate by considering the linear medium value of the classes and the number of packages in each class. As the log scale classes are in the form $2E^i$ to $2E^{i+1}$, their linear medium values are calculated in the following manner: (2E+1 + 2E+2)/2 = 1.1E+2.

Caution: The dose rate distribution considers all the waste packages at the time of production (reference time). To obtain values at any given time (up to 80 years), the subdistribution of each production year must be decreased using the normalised dose rate decrease given (see below).

An estimate of the maximal dose rate value for the waste package can be taken from the upper bound of the highest dose rate class given but the value that should be used is given in form 2. For containers having a capacity larger than ~ 200 l, the maximum dose rate at 1 m is assumed to be limited to 0.1 mSv/h (as specified in the transport regulations).

Waste sorts with packages having very low average dose rates cannot be characterised within the standard dose rate distribution classes. In these cases no dose rate distribution is given. This is the case for the following waste sorts:

RA:	KKB-1A, -1B; KKG-1A
SA:	KKB-8, KKM-3....6, KKG-3....6, KKL-8
	PSIW-1A,-1B, -2A, -2B, -3A, -4C, -6A
MIF:	MIF-3A, -3B, -4

Normalised decrease of the dose rate (gamma, 0 m)

To calculate the decrease of the waste package gamma dose rate at the surface as a function of time, normalised values (value = 1 at reference time or 0 year) are given for 5, 10, 20, 40 and 80 years.

Radiolytic gas production

Gas production rates (mainly H_2) as a function of time are given in litres per year for reference time (0 year), 5, 10, 20, 40, 80, 100 years. The cumulative production is also provided in m^3 for the same selected years.

The method to calculate the amount of gas produced is based on the determination of the energy deposited in each of the materials and the use of the material "G" values. The problem of defining, for all package configurations, the location and amount of energy deposited is solved for α and β particles by introducing into the computer code their deposited energy fractions for each of the materials involved. This parameter is estimated by expert judgment. For the γ particles these fractions are estimated through a simplified calculation of the γ

flux in (mixed) materials with consideration of the shielding and the escape probability.

Fissile materials

The fissile nuclides that have significant masses within MIRA are U-235, Pu-239 and Pu-241. For completeness U-233, Pu-238, Am-242m, Cm-243 and Cm-245 are also given. The masses of each nuclide are calculated for the average and maximal activity values at the reference time. The maximum fissile content can be used in conjunction with the transport limits to give an indication of any difficulties that may arise with transportation.

Note: As a result of a recent review, Th-232 activities have been added to MIF-3B (totaling 2.6 10^9 Bq) and to MIF-5B (totaling 2.0 10^7 Bq). The corresponding total masses (640 kg and 4.9 kg) have not, as yet, been attributed to the waste package materials of the 2 waste sorts.

Surface to mass ratio for metallic materials

For each metal occurring within the waste package the surface to mass ratio is given (in m^2/kg) in conjunction with the associated mass (in kg).

3.9 Form 7: Drawings of external container and of waste package internal configuration

(Abfallgebinde: Äussere Behälterhülle)
(Abfallgebinde: Innengeometrie)

The first page contains the drawing of the waste package external container.

Note: The container as displayed is the most representative container for the wastes described by the waste sort or a specific container to show the most important features of the waste. For example: 200 l drums with a zinc internal coating have been used for 2 different real waste streams and then the use of this zinc coating has been discontinued; within the model inventory, to represent the zinc (material of concern for gas production in the repository), a drum with a zinc coating has been ascribed to one of the waste sorts, and a drum without a zinc coating to the other.

The second page shows the drawing of the waste package internal configuration with the designation of the components according to their use in form 4.

3.10 Basic data not related directly to waste sorts

The data bank contains a central database containing reference data that is used in the generation of some form data. This information is also accessed during more complex calculations or summations.

The data contained within the central database can be divided into two parts:

1) Radiological reference data which gives data for each radionuclide recognised by MIRA: half-lives, decay energies, etc.

2) Material reference data which gives elemental breakdowns for each material recognised in the inventory. These material breakdowns are detailed at the end of Volume 2.

4 WASTE DESCRIPTION AND METHODS OF CHARACTERISATION

4.1 Objectives and work development

The three main objectives in developing a model inventory are: to improve the data that is most important in relation to disposal safety, to update the data according to the on-going developments in the production of the waste and to supply additional data for other projects.

The first inventory characterisation work was carried out for "Projekt Gewähr 1985" (see section 1.4). The scope of this first characterisation (data specificity and accuracy) was derived from an estimation of the significant data for the repository designs and safety analyses of this project. Most of the effort was devoted to the estimation of the waste sort radionuclide activities which were considered to be most important, especially the long-lived radionuclides for the long-term repository safety case.

Only after carrying out the repository safety analyses using this first inventory, was it possible to deduce from the results which data were most significant. To summarise, it was realised that some arisings of radionuclides had to be investigated more thoroughly because they led to predicted doses in the performance assessment that were relatively high, but well below the legal dose limit imposed. Further it was realised that the materials of the waste play a more important role in the radionuclide release and migration (and thereby in the resulting population dose) than assumed. This is due to their influence in the chemical processes involved (complexing agents, organic materials decomposing to complexing agents, specific metals such as Al, Zn responsible for high gas production, etc.).

The first objective of the current characterisation work was to improve the reliability or accuracy of the data (radionuclide activities and materials) that had been found to be significant. It has also to be mentioned here that knowledge about the repository near-field system has evolved in the past years, which has further modified the importance of some data.

To illustrate this with a practical example, the case of Cl-36 is given: Cl-36 ($t_{1/2}$ = 3.0 10^5 y) arising from Cl-35 neutron activation was estimated in the first inventory on the basis of the main Cl component of the activated materials. This was the general approach made everywhere. The improvements implemented for this inventory were 1) to check if all important amounts of activated Cl had been considered, 2) to take into account all the Cl impurities at ppm level in the relatively large amount of neutron activated materials and 3) to carry out a control measurement of Cl-36 (from Cl impurities in the reactor cooling water) in the ion exchange resins of some NPP's (XINQI et al. 1991). As an example of the situation for point 2, the COGEMA and BNFL specifications have not yet taken into account the 3.5 ppm Cl in activated zircaloy of the fuel rod hulls.

The second objective of the characterisation work was to update and improve the data in a more general way. In particular all new information available to Nagra (waste specifications, other papers) was considered. Not all the waste sorts are characterised to the same level of detail but an assessment of the level of characterisation is given later in this Chapter as well as indications of major changes that have occurred.

The third objective was to provide specific data required for the repository planning and the safety analyses as mentioned in the description of the data-base forms in section 3, especially 3.7 (chemical categories of materials) and 3.8 (properties of the waste package).

4.2 BA (NPP routine operational waste)

The majority of operational NPP wastes originate from the cleaning and purification of water (in various circuits and drains) and of the air. These consist of ion exchange resins, filter cartridges, air filters in which radioactive materials are retained as well as sludges/slurries and evaporator concentrates resulting from water purification. The radionuclides originate from fuel leakage (fission products and actinides), activated water impurities and corroded activated steel (activation products). The remaining operational wastes arise from repair, maintenance and inspection activities and include contaminated tools, protective materials, etc. as well as contaminated plant components. These solid wastes exhibit a similar spectrum of radioactivity and a part of them are combustible. The arisings, treatment and conditioning/packaging of the operational waste are detailed in (NAGRA 1984b).

For the present inventory the operational wastes are categorised into the following BA waste sorts:

BA-1: Ions exchange resins
BA-2: Sludges and concentrates
BA-3: Filter cartridges
BA-5: Solid waste
BA-6: Incinerated waste

The waste sort BA-KKB-4 formed for the air filters of KKB in the previous inventory (NAGRA 1985, NAGRA 1984b) has been incorporated in BA-KKB-5 (solid waste) and also in BA-KKB-6 (ashes) as is already the case for the filters of the other NPP's. BA-4 is therefore no longer existing in the present database. The waste sort BA-5 results from the combination of the previous waste sorts BA-5A (non compactable waste) and BA-5B (compactable waste). For the sake of simplification it has now the same characteristics for all NPP's. Finally, cement as conditioning material for the KKG resins (BA-KKG-1) has been replaced in the present inventory by bitumen following the final decision of the authority (HSK), at the end of 1993, to accept such a conditioning method for

the stored resins. (Note: this late modification could not be considered in all the projects using this inventory).

The method used for determining the average radionuclide activities is based on key nuclide measurements (Co-60 for activation products, Cs-137 for fission product and actinides as well as Cm-244 for actinides) and the use of correlation (also called scaling) factors to derive other radionuclide values (NAGRA 1985). The correlation factors used in 1984 were generic factors derived for German PWR and BWR reactors that were made confidentially available to Nagra. Since that time many other sets of correlation factors have been made confidentially available to Nagra. In comparison very few sets of scaling factors have been published in recent years (see (EPRI 1992) and references quoted therein). These factors have been reviewed taking into consideration additional measurements and theoretical estimations. From this review three revised sets of correlation factors have been determined, one for the PWR waste sorts excluding the filter cartridges (BA-3), one for the PWR filter cartridges (BA-3) and the third for the BWR waste sorts. It should be noted that these correlation factors are in fact NPP-specific and some care is required in their use. A study, including measurements, has recently been launched for the determination of the Swiss NPP values in connection with the waste specifications to be issued by the producers.

The radionuclide activities derived from these revised correlation factors have been compared with activity values obtained after "Projekt Gewähr 1985". As a result of this assessment of the radioactivities, the current activity values are considered to be satisfactory, with a few radionuclide activities still considered to be uncertain up to 2 orders of magnitude. Further improvements are expected after the receipt of the measured correlation factors and information from new waste specifications.

Remark on waste sort BA-KKW-A

A dummy waste sort has been created for BA wastes to describe those wastes that have been produced in type-B containers but not characterised ("KKW" means NPP). For the dummy waste sort only the number and general features of the containers are given for the repository planning. This waste sort is not included in any safety assessments as its radionuclide activities and materials can be considered to be included in the other BA waste sorts.

4.3 RA (Exchangeable non-fuel reactor core components)

4.3.1 General waste description

The RA wastes ("Reaktorabfälle") are components of the NPP reactor core, for instance control rod assemblies, fuel channels, etc. which are replaced after some years of operation when they become mechanically defective or when their operating life has ended. An overall definitive terminology for these wastes

does not, as yet, exist but as they do not contain fuel, they are commonly called in American reports "non fuel bearing components", "non fuel hardware", etc.. The RA wastes are materials which have been activated by the high neutron flux of the core and, in some cases, contaminated by crud deposits (components exposed to the reactor coolant water). The materials are stainless steels (structural materials), zircaloy (fuel channels), neutron absorbing materials (boron carbide (B_4C), Ag-In-Cd alloys, hafnium, borosilicate glass, borated steels) and finally aluminium oxide (Al_2O_3) (filling material). These RA waste components are specific to each reactor design. The operational life of these components varies, but the trend has been found to be an increase in operational life with experience, so that the estimates made for the numbers of components and their irradiation time for the 40 years of reactor operation is in most cases only approximate.

As these wastes take up very little space (except for BWR fuel channels) and as the metallic part of the waste are highly radioactive, due mainly to their Co-60 content, they are normally stored for cooling purposes in the NPP. Preliminary investigations into radionuclide characterisation, conditioning and packaging have recently started (mainly in the USA as there is concern there that some radionuclide concentrations could exceed the limits for US shallow land disposal).

As a result, the data, in terms of nuclide activities, materials, package types and package numbers, can only be taken as being a first approximation. The exception is for fuel channels for which waste producer specifications exist.

4.3.2 RA waste sorts

In the MIRA inventory the RA waste sort categories have been defined as follows:

RA-1A: Control assemblies ("Steuerelemente")
RA-1B: Part length control assemblies ("Trimsteuerelemente")
RA-4: n-absorbing elements ("n-Absorberelemente")
RA-5: Flow restriction rods ("Drosselkörper")
RA-6: Fuel channels ("Brennelementkasten")
RA-7: Incore instrumentation ("Mess-Lanzen")

The designation RA-2 and RA-3 is reserved for the primary and secondary neutron sources. An investigation has shown that it could be possible to package all the neutron sources into a large container of size 18.91 m^3. However, since this investigation, it has become apparent that the estimates of the radionuclide activities associated with an assumed time of packaging and hence the materials required for shielding and packaging are too uncertain for defining the type of container, i.e. the waste sort(s). However, to allow for consideration of these

neutron sources in the safety assessment of the repositories, conservative values of the radionuclide activities are given in Appendix C.

Examination of the waste on a reactor by reactor basis gives rise to the following waste sorts (an estimate of the total number of components that are expected to be discharged as waste are also given):

RA-KKB-1A: Rod cluster control assemblies (RCCA): 128
RA-KKB-1B: Part length control assemblies (PLCA): 12
RA-KKB-4: Burnable poison rod assemblies (BPRA): 91
(Vergiftungsstabelemente)
RA-KKB-5: Flow restriction rods: 240
RA-KKB-7: Incore instrumentation: 120

RA-KKM-1A: Control rod assemblies (CRA): 180
RA-KKM-4: Poison curtains : 48
(Vergiftungsbleche)
RA-KKM-6: Fuel channels: 1815
RA-KKM-7: Incore instrumentation: 160

RA-KKG-1A: Rod cluster control assemblies: 144
RA-KKG-4: Burnable poison rod assemblies: 75
(Vergiftungsstabelemente)
RA-KKG-5: Flow restriction rods: 150
RA-KKG-6: Incore instrumentation: 48

RA-KKL-1A: Control rod assemblies: 612
RA-KKL-6: Fuel channels: 6600
RA-KKL-7: Incore instrumentation: 400

Note: The RA-waste arising at the final shut-down of the NPP's, but preceding decommissioning, are classified as RA-waste and not as decommissioning waste.

4.3.3 Raw waste characterisation

For the characterisation of the raw waste quantities and their radionuclide content information is required with regard to the following parameters:

- material composition of the components (for instance of a rod cluster control assembly)
- number of components in the reactor core and their average life time, from which the number of components discharged as waste can be derived (these numbers are reported in the previous section)
- irradiation load factor: certain components like the control rod assemblies are not continually exposed to the full neutron flux
- neutron flux in the core

The average life and the irradiation load factor have been assumed by Nagra whereas the other parameters are defined.

The activation calculations have been made using the ORIGEN2 code. The application of this code for the RA components is deficient with respect to the following points:

- The neutron flux and cross sections used within the code do not take into account for certain elements the large depression in the thermal neutron flux caused by the neutron absorbing materials (Note: for the fuel channels the calculation is correct for the metal but not for the uranium impurities within the metal because uranium is considered by the code as being in the fuel where the epithermal neutron flux is more depressed).

- The code does not consider all the neutron reactions occurring from the isotopes of the elements constituting the n-absorbing materials and the metallic impurities. For instance the calculated tritium activity in the control rod boron carbide is much too low (B-10(n,t)Li-8 and secondary Li-7(n,nt)He-4 are not considered); the Be-10 ($t_{1/2}$ = 1.6 10^6 y) activity calculated for the reaction B-10(n,p)Be-10 could be 2 orders of magnitude too low because the cross section for fast neutrons is not considered. For these cases, separate "hand calculations" were carried out to supplement the code.

  It is expected that the development of applicable ORIGEN2 cross section libraries or other codes for radionuclide characterisation of some core materials will be undertaken in the future on an international level as and when the necessity arises to specify these wastes for disposal. Such a development has just started for the reactor decommissioning wastes (IAEA 1993) and could include the exchangeable non-fuel core components.

  The ORIGEN2 activity calculations have been carried out without the material compositions being specified down to the ppm impurity level (except for zircaloy) as mentioned for chlorine in section 4.1. However, it was possible to correct for the metal impurities in this inventory by using the results of work carried out elsewhere (EVANS et al. 1984). In this work the impurities in the steels of the reactors were measured and activation calculations carried out for decommissioning waste studies. As the same type of steels are used in the RA-components considered in this inventory, the results obtained for the activation of impurities from (EVANS et al. 1984) were added to the relevant RA waste sorts following normalisation to appropriate activities (Ni-59, etc.) arising from the main steel components and correction for different irradiation times.

  As stated above, results of ORIGEN2 calculations for other materials (B_4C, Al_2O_3, etc.) were modified by hand calculations, but only, as far as, for the production of disposal relevant radionuclides such as H-3, Be-10, Al-26, etc..

Finally the crud activities were estimated by choosing an estimate for the value of the Co-60 activity per m^2 (for instance 0.05 Ci/m^2 for the BWR control blades after 5 years cooling) and using correlation factors for the other nuclides. For this estimation, confidential data made available to Nagra as well as data contained in (ROBERTSON et al. 1984) were used.

4.3.4 Waste conditioning

Conditioning and packaging of RA-KKM-4 (poison curtains), RA-KKL and KKM-6 (fuel channels) and RA-7 (incore instrumentation) was already defined in the previous inventory (NAGRA 1985, NAGRA 1984a). The cooling time assumed before packaging is 5 years for all the RA waste sorts.

In 1989 a conditioning and packaging study was carried out by Nagra to define the control assemblies (RA-1A), the part length control assemblies (RA-1B), the absorber elements (RA-4) and the flow restriction rods (RA-5) as waste sorts and include them in the model inventory. The concept of a large container (18.91 m^3) was used as well as some smaller thick-walled metallic containers that were already available on the market. Very little work had been carried out worldwide with respect to the conditioning and packaging of these types of waste. The results of this study have been summarised elsewhere (IOLI & POPP 1990).

A general revision of these waste sorts should be made at a later stage when the incore management of these components, the calculation of the activities and the waste conditioning and packaging plans become more reliable. The comparison of the model inventory activities with available measured data (NUREG 1993, GIDARAKOS & SPALTHOFF 1983) confirms the estimates for 10 nuclides (H-3, C-14, Mn-54, Fe-55, Ni-59, Ni-63, Co-60, Nb-94, Ag-108m, Ag-110m) of the PWR rod cluster assemblies (RA-KKB and KKG-1A), the BWR control rod assemblies (RA-KKM and KKL-1A) and the burnable poison rod assemblies (RA-KKB-4). Finally, it should be noted that control rod assemblies of the hybrid type containing hafnium in addition to boron carbide rods have begun to be used in BWR reactors but have as yet not been considered in this inventory.

4.4 SA-KKW (NPP decommissioning waste)

Waste description and previous inventory

The decommissioning raw waste consists of: neutron activated steels arising from the reactor vessel and internals (these steels are also contaminated), neutron activated concrete arising from the biological shield, steels contaminated with crud deposits arising from the water circuits, various contaminated components (filters, insulation, etc.) and finally objects contaminated during dismantling (tools, plastic, filter).

Before the primary vessel and piping are dismantled, they may for radiological protection purposes be subject to decontamination. A decontamination factor of 10 has been assumed for these components. The resulting raw waste (as well as from other minor decontamination processes) are resins, concentrates and slurries.

In 1980, specific decommissioning studies were required by the Swiss nuclear regulatory authorities for each of the five NPP's to demonstrate technical feasibility, estimate doses and costs and assess the quantities of waste. The NPP's were assumed to have been shut-down after 40 years of operation with dismantling starting after either 2 or 30 years and taking 7 years to complete. The results of these studies were used by Nagra to determine the NPP decommissioning waste inventory by assuming dismantling 2 years after shut-down which is the more conservative option for the waste. The rough radiological characterisation of the material carried out for dismantling has been improved and the waste sorts were defined (based on raw waste exhibiting common characteristics, cement conditioning and a large container) and characterised. A detailed description of the NPP decommissioning waste and the characterisation carried out for the 1985 inventory can be found in the Projekt Gewähr 85 report (NAGRA 1985). A further analysis of packaging and disposal of the decommissioning waste has been made in (ALDER 1989).

Waste sorts in MIRA

As dismantled reactor internals and pieces of the reactor vessel are in general highly radioactive, have high surface dose rates and are found to exhibit significant variation from one component to another, each of the components has had to be examined individually for the determination of the shielding in the large containers. As a result, different masses of raw waste, cement and shielding materials have been considered for filling the large container, leading to different waste sorts. For instance in the 1985 inventory there are 9 waste sorts SA-KKL-1-1 to SA-KKL-1-9 for the reactor internals and for the most active part of the reactor vessel.

For the sake of simplification in the data presentation and their use in other disposal projects, the waste sorts SA-KKL-1-1 to SA-KKL-1-9 given as an example above, as well as the other original 1985 waste sorts (NAGRA 1985, Table 4.7 and 4.8; NAGRA 1984a) were combined in the current model inventory as follows:

SA-1-1 + SA-1-2 + ...SA-1-n	--> SA-1 (highly activated metals)
SA-2 + SA-3 + SA-4 + SA-5 + SA-6	--> SA-2...6 (no special features)
SA-7A + SA-7B	--> SA-7 (high organic content)
SA-8	--> SA-8 (actinides and organic content)

The SA-KKW waste sorts of the present inventory are thus:

SA-1:	core internals and reactor vessel parts
SA-2...6:	biological shield (2+4), reactor vessel parts (3), PWR primary circuits and BWR recirculation circuits (5), low contaminated components and piping (6)
SA-7:	other primary waste (other materials) plus secondary waste (such as tools, plastic, filters, etc.)
SA-8:	decontamination residues

The combination made is possible because the same container is used for all the decommissioning waste. The waste package materials and activities are averaged to form these new waste categories. In some cases the resulting additional material ("Zusatzstoff") given in form 4 of the database could be a mixture of different cement types. The information lost in combining these wastes can be the specific material configuration when different internal shielding arrangements exist in the original waste sorts. In such cases an indication is given on the drawing of form 7 that the configuration is representative and can vary inside the waste sort. If necessary the data of the original waste sorts can be retrieved from the data bank and other combinations can be made relatively easily.

Waste characterisation

The radionuclide characterisation is based on activation calculations carried out using ORIGEN2 and estimates made for the contamination of the water circuits. These estimates were made in 2 stages: first an estimate of the Co-60 activity per cm^2 was made and then radionuclide correlation factors to Co-60 were used (these values have been made confidentially available to Nagra or published).

The characterisation made for the previous inventory (NAGRA 1985, NAGRA 1984a) considered only the activation of the major elements occurring in the materials. In addition Co-60 contamination values and correlation factor values were at that time not widely available.

Comparisons of the Nagra data with other available decommissioning raw waste data (see (SKI 1990) for references to other data) has led to some improvements. As an example the determination of the biological shield radioactivities (SA-2 and 4) was improved using extended material composition data and new activation calculations carried out using ORIGEN2 (corrected for Ar-39 and H-3 production cross sections).

Quite recently the preliminary results of the safety analyses (NAGRA 1993) using the draft version of this inventory database have shown that Mo-93, Ag-108m and Ag-108 are significant as a result of the inclusion of the RA and the PSI accelerator facility decommissioning waste (SA-PSIW) in this inventory. The radionuclide activities of the highly activated steels and alloys arising in SA-

1 wastes were checked and completed for the material impurities using the impurity measurements and activity calculations of (EVANS et al. 1984), as carried out for RA wastes (see section 4.3.3). Less activated steels of SA-4 were not completed for impurity activities, as these activities were considered in comparison to be negligible.

The radionuclide activities of the decontamination residue (SA-8) were revised taking into account all the new information available to Nagra since 1984, extending from crud activity data in components that have been replaced in the Swiss NPP's to published data (ROBERTSON et al. 1984). The radionuclide activity completion is illustrated by the following figures: 15 nuclides were specified in the previous inventory whereas 37 are given in the present inventory (17 published by ROBERTSON et al. 1984). Other waste sorts were not revised for the contamination activities due to the relative low significance of these activities.

The waste sorts SA-7 were kept separate from SA-2...6 because of their organic contents. These wastes will be emplaced according to the present repository planning and safety analyses in a different repository compartment. As the significance of the radionuclide activities contained in these wastes has increased (as a result of their enhanced migration), a review, and hence an increase in the number of nuclides specified, was carried out on basis of the extended radionuclide spectrum given for the decontamination residue (SA-8). However, it should be noted with respect to disposal safety that the organic material of the waste sort SA-7 is solid and may be incinerated in the ZWILAG facility furnace before being conditioned.

Note: In most cases the raw waste activities were calculated at 2 years after plant shut-down. Conditioning and packaging were also defined at this reference time. Because of the early time assumed for conditioning (see section 2.2.3: Interdependence of assumptions), the resulting high heat production given for SA-1 packages, especially SA-KKB-1, is an artefact. In addition, these waste packages were assigned a production over 7 years leading to conservative values at time of production. In reality the internal container shielding and the heat production will be less due to the decay of Co-60.

4.5 SA-PSIW (PSI accelerator facility decommissioning waste)

4.5.1 Introduction

A significant part of the Swiss radioactive waste will include wastes from research particle accelerators. The two most important facilities are CERN at Geneva with many accelerators and PSI at Villigen with a very high 600 MeV proton beam intensity accelerator. Preliminary estimates of packaged waste volumes for the PSI accelerator facility of 15,000 m^3, for the CERN facility of 5,000 m^3 and for an additional reserve of 10,000 m^3 were made in 1985 (sec-

tion 6.3.5.1 in (NAGRA 1985)). For these wastes no activities were derived, but from general considerations, these were assigned to the SMA repository. As CERN is an international organisation, it is still not clear which decommissioning wastes will be disposed of in Switzerland. Currently it is assumed that the wastes arising on Swiss territory will be disposed of in Switzerland. The CERN policy is to keep the large components from accelerator decommissioning in place for in-situ radioactive decay and possible re-use. The operational research waste are disposed of in France and so far only the beam targets of a decommissioned experiment (ISOLDE) have been brought to PSI as MIF waste for future disposal.

As the PSI accelerator decommissioning waste was not described nor further characterised as waste sorts in the previous inventory (NAGRA 1985, NAGRA 1984a), and as there are no published papers on this type of waste, a more detailed characterisation than given for the other waste sort categories is contained in the following sections.

The question of the CERN decommissioning waste will be discussed in Chapter 5 in relation with the waste volume uncertainties.

4.5.2 Origin of waste and features of the waste radioactivity

Materials of an accelerator facility become radioactive when they are bombarded by the accelerated particles. These materials will become waste when the components these materials form are replaced or decommissioned.

The particles, forming a beam, are accelerated and this beam, once outside the accelerator, is directed through vacuum pipes by magnets onto beam dividers and targets which produce beams of other particles. The main beam is finally stopped in a beam dump. The various particle beams are used for different research experiments. A small fraction of the beam particles are lost along the beam trajectory and hit the surrounding materials. A larger fraction hits the targets and the rest the beam dump.

The high energy particles collide with the neutrons and protons in the nuclei of elements comprising the materials and not with the nucleus as a whole as is the case at low energies. These neutrons and protons with the high energy acquired in the collision hit further neutrons and protons of the nucleus and a cascading effect takes place in the nucleus with neutrons and protons being finally ejected from the nucleus. The residual nucleus can be almost any stable or radioactive isotope of lower mass. Protons and neutrons ejected from the nucleus will hit further nuclei and this external cascade stops when the ejected particles have no more energy. The neutrons below 30 MeV down to thermalisation induce nuclear reactions down to thermal (n,γ) captures. Therefore the main contributions to the radioactivity are from the spallation process and thermal neutron activation.

The range of protons in the materials is relatively small (23 cm of copper for 600 MeV) but for the high energy neutrons the range is much larger (about 23 cm is also the neutron mean free path describing the exponential decrease of the number of neutrons in the copper for energies down to 100 MeV). As a consequence a relatively large volume of the surrounding material becomes radioactive with the radioactivity decreasing with distance from the location of the initial collision. This last feature explains why the PSI accelerator facility decommissioning waste is characterised by an increasing volume of waste of decreasing activity. The most radioactive material will be copper, stainless steel, aluminium, etc. of the accelerator and beam system and the less radioactive material will be steel, iron shot concrete and concrete used as shielding.

4.5.3 Waste inventory and characterisation

The PSI 600 MeV proton accelerator was built at the beginning of the seventies with a high current capability (10-100 μA) for meson production and is now being upgraded to use a beam intensity of 1.5 mA.

The lay-out of the accelerator facility is given in Figure 2. This lay-out represents the facility configuration from 1995 on for an increased beam intensity of 1.5 mA. As compared to the previous lay-out used in this work to estimate the decommissioning waste, it shows a modified and increased shielding of the thin and thick targets producing the π beams and of the beam dump. The neutron source SINQ, which will be commissioned in 1995, was not a part of the previous lay-out and is discussed separately later.

The waste characterisation work was carried out on basis of the following assumptions:

- operation time 1995-2034, i.e. 40 years operation
- beam intensity: 1.5 mA with a load factor of 66.6 %, i.e. 40 mA·years
- cooling time: 1 year, i.e. waste activities referenced to 2035

To estimate the radionuclide activities of the different materials the following method was used:

PSI measured with a Ge-Li detector the γ-emissions from various material samples arising from different locations in the previous lay-out of the facility. From these measurements PSI deduced the γ-activity concentrations of the materials and then estimated the masses of the materials per decade of the measured γ-activity concentration (for instance 1 ton of steel between 1 and 10 Ci/m^3). PSI adjusted the results to 1985 including 1 year cooling, i.e. to 10 years operation and 0.36 mA·years.

Independently Nagra has conducted a full activity characterisation of the beam dump after it had been dismantled in 1989 (15 years operation, 0.73 mA·years). The Monte Carlo code HETC and the neutron transport code MCNP provided

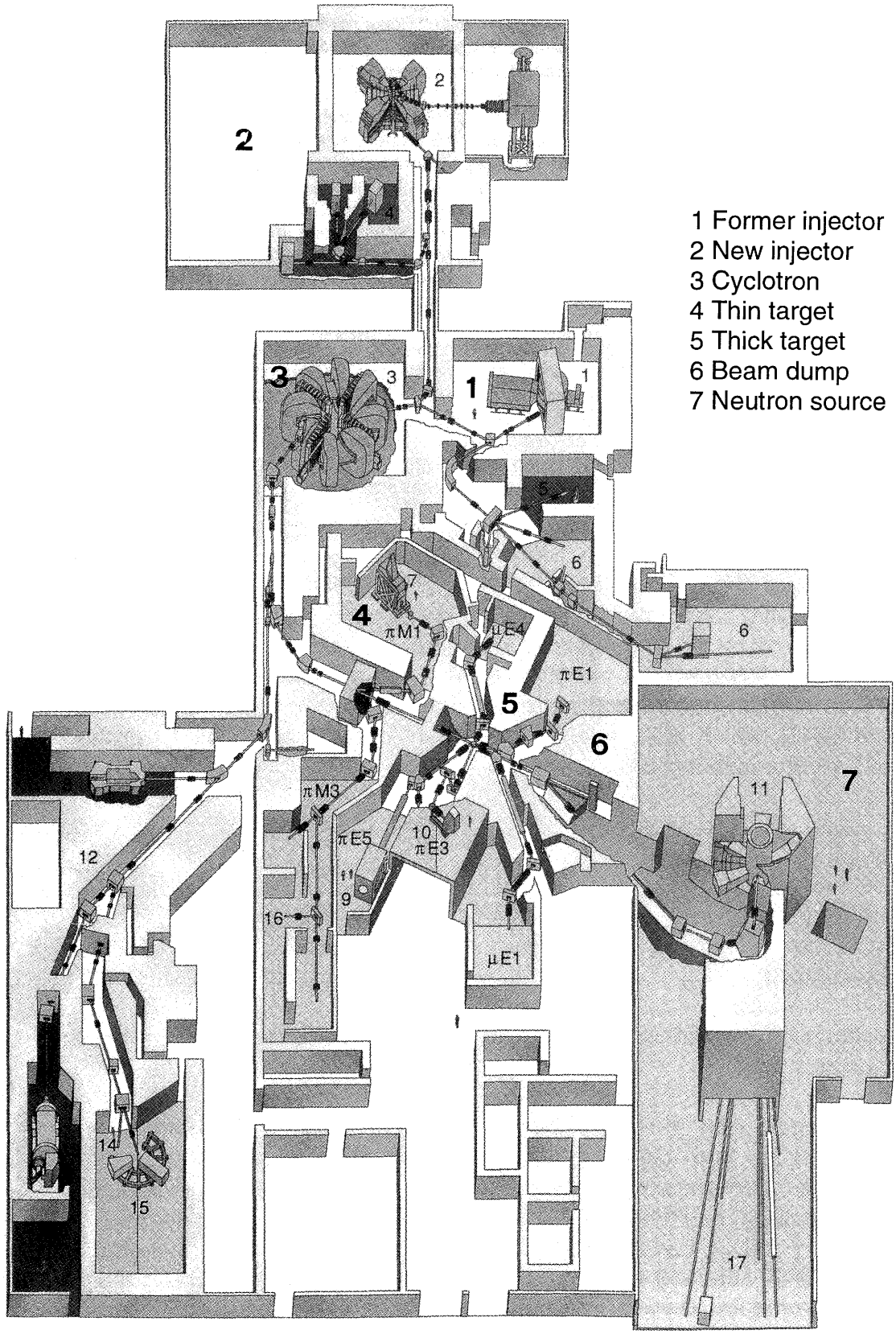


Fig. 2: PSI 600 MeV proton accelerator facility
(from PSI-Jahresbericht (yearly report) 1992)

the nuclide activities in the different materials of the beam dump. Radionuclide activity spectra of the materials were deduced after comparison with the partial results of a simpler code using only spallation cross sections and were calculated back to the conditions of the PSI 1985 waste inventory. In this inventory the PSI measured γ-activity spectra could thus be replaced by these calculated complete spectra after normalisation was made from available measured to calculated γ-activity ratios.

The completed PSI 1985 inventory (corresponding to 0.36 mA·years over 10 years) was then converted into the 2035 inventory (assuming 40 mA·years over 40 years). The additional materials arising as radioactive waste due to higher and longer irradiation were also taken into account but the activation produced pre-1995 (a few percent) was neglected.

Waste packaging was assumed with the activities defined after 1 year cooling and with cement conditioning. Filling factors for the different raw wastes were estimated to define the waste quantities arising in the large container ($2.05 \times 2.05 \times 4.5 = 18.91$ m^3). To cover the range of the waste activity concentrations in some materials three different internal shielding configurations were considered for such materials. To obtain the number of waste packages produced per year a decommissioning period of 5 years (2036-2040) was assumed with linear arisings of conditioned waste and without decaying the activities over this 5 year period.

4.5.4 Waste sorts and results

The activity concentration levels and internal package configurations are recognised by the letters A, B, C in the waste sort designation (A for the highest activity concentration and C for the lowest). The waste sorts are:

Designation	Raw waste	Packaged volume (m^3)	Activity (Bq)
SA-PSIW-1A SA-PSIW-1B SA-PSIW-1C	Copper Copper Copper	379	$4.6\ 10^{15}$
SA-PSIW-2A SA-PSIW-2B	Stainless steel Stainless steel	114	$1.7\ 10^{15}$
SA-PSIW-3A SA-PSIW-3B	Aluminium Aluminium	303	$8.3\ 10^{13}$
SA-PSIW-4A SA-PSIW-4B SA-PSIW-4C	Steel Steel Steel	5314	$9.7\ 10^{15}$
SA-PSIW-5	Concrete	5484	$3.2\ 10^{14}$
SA-PSIW-6A SA-PSIW-6B	Iron shot mortar Iron shot mortar	2553	$3.1\ 10^{15}$
Total		14147	$2.0\ 10^{16}$

These results show that the total decommissioning waste volume and activity of a nominal 1 MW beam power accelerator facility are comparable to those of a 1 GW(e) nuclear power plant under the same conditions of 40 years of operation and 1 year cooling. For instance, the values for NPP Gösgen, PWR, 940 MW(e) are ~ 7,000 m^3 and 1.3 10^{17} Bq (SA-PSIW: ~ 14,000 m^3 and 2.0 10^{16} Bq).

4.5.5 Analysis of the estimating method and of the results

Many assumptions and simplifications have been made in this characterisation work and these are discussed. Their effects on the results are difficult to evaluate but an overall confirmation is obtained from the comparison of the estimated total activity with the activity given by the following approximate formula (SULLIVAN 1992):

Activity (in GBq per W of beam power) $= 1.1E^{-0.8} \ln [(T + 1)/t]$

where E = proton energy in GeV = 0.6 GeV
T = irradiation time = 40 years
t = cooling time = 1 year

and effective beam power = $6\ 10^5$ W (1.5 mA x 2/3 load factor x $6\ 10^8$ V)

This formula gives $2.3\ 10^{15}$ Bq whereas the total activity derived in this work is found to be $2.0\ 10^{16}$ Bq.

The following major simplifications have been made here:

1) Evaluation of the radioactive material masses using only 23 values for the masses within 47 decades of γ-activity concentration (i.e. for instance 1 ton of steel between 1 and 10 Ci/m^3) by PSI.

2) The increase of the material activity concentrations of the previous lay-out neglects the change in the shielding configuration of the new lay-out and leads to a rough estimate of the quantities of materials previously considered as non radioactive (steel, concrete) that have become classified as radioactive.

3) Use of a single radionuclide vector for the materials based on the average nuclide vectors of the beam dump materials. In reality, the nuclide activity concentration varies in the materials with the neutron spectrum which is affected by the material type and configuration.

4) The material compositions used are those of the beam dump components. These were assumed for all other material types so that differences in alloys and impurities for steel and stainless steel resulting in differences in some nuclide activities were not considered.

It should be noted that reasonable agreement between the material γ-activities measured and those deduced from code calculations were obtained, which supports the activity completion method used.

As a result of the extrapolation from the previous lay-out to the new one based on the increase in beam intensity, the raw waste material quantities and activities (given in section 4.5.4) are to be considered only as a first approximation. As for the NPP decommissioning waste, the accelerator facility waste activities are dominated (in descending order) by the following nuclides Fe-55, Ni-63, Co-60, Mn-54, Co-57, He-3, etc. arising in steels and concrete. The following nuclide activities are specific to accelerator waste:

Na-22 (2.6 y), Al-26 ($7.3\ 10^5$ y), Si-32 (132 y), S-35 (87 d), Ar-42 (33 y), Ti-44 (47 y), Fe-60 ($1.5\ 10^6$ y), Co-56 (77 d), Ge-68 (271 d), As-73 (80 d), Zr-88 83 d), Nb-91 (680 y), Nb-91m (61 d), Nb-92 ($3.6\ 10^7$ y)

Other minor materials such as magnet araldite, lead, cables, targets, etc. have not been considered in the inventory. Of importance could be the radionuclides with long half-lives such as Cl-36 ($t_{1/2}$ = 3 10^5 y) arising in PVC cables, as they are relevant to the safety of disposal. A very conservative upper value in this case is 10^{11} Bq of Cl-36. The graphite thin and thick targets used up over the 40 years of operation (~ 100 kg) will contain at the time of the facility decommissioning a Be-10 ($t_{1/2}$ = 1.6 10^6 y) activity of the order of 10^8 Bq. The tritium produced will be of the order of 10^{13} Bq. These 3 activities are comparable with the total nuclide values already given for the SA-PSIW waste in Table 2 of section 5.3.

Finally, as already mentioned, the decommissioning waste of the neutron source using a Bi-Pb target are not included in this inventory. In reality it is expected that the beam will be dumped into the neutron source target 75 % of the time and 25 % into the copper of the new beam dump. This means that the activities in the shielding will be distributed into larger quantities of materials with reduced copper activities. In addition the Bi-Pb target activities will have to be considered (most safety relevant nuclides for disposal from the Bi-Pb target are Po-209 (102 y) and Hg-194 (520 y)).

In conclusion, the radionuclide characterisation of accelerator facility waste for disposal specifications will show the same type of calculational problems (i.e. reliability of code cross sections and of code applicability) as found for the decommissioned NPP reactor internals, vessel and biological shield (IAEA 1993).

4.6 WA (Reprocessing waste)

4.6.1 Waste description

The spent fuel of the NPP's is reprocessed by COGEMA (France) and BNFL (England) to recover the uranium and plutonium.

The fuel assemblies are chopped giving rise to sections of fuel rods and metallic end caps. The spent fuel is dissolved into an acid solution. The remaining zircaloy hulls are conditioned in cement together with the ends (waste sort WA-4). Uranium and plutonium are extracted from the solutions leaving a residual highly active "liquor" containing the fission products, other actinides formed in the spent fuel as well as some residual uranium and plutonium. This "liquor" is calcinated and conditioned with a glass frit and additives giving rise to the high level vitrified residues (WA-1). Low level technological waste from the operation of the plant are packaged in relatively thick wall containers (or metallic containers with an additional internal cement wall) (WA-5).

The other waste streams differ between COGEMA and BNFL due to the use of different reprocessing operations. For BNFL the high level fuel solution is centrifuged before the uranium and plutonium are separated. This gives rise to a centrifuge cake slurry containing undissolved fuel materials and metallic fines which

is then conditioned in cement (WA-BNF-7). The crud arising from the use of multiple element bottles for transport to and the storage of the fuel assemblies in the ponds (as well as the pond filter crud) is mixed by BNFL with barium carbonate (contains mainly radioactive carbon and iodine), arising from the scrubbing of the vitrification process off-gas, and conditioned in cement (WA-BNF-2). Other BNFL waste streams are not specified for return to the overseas customers and will be substituted by other specified waste (for instance vitrified residues). Liquid wastes arising at COGEMA are treated for water purification and the resulting precipitates and sludges are conditioned in bitumen (WA-COG-2). The COGEMA technological waste which cannot be declared as LLW due to its high α-emitter content is packaged into appropriate containers (WA-COG-6). Note: COGEMA recycle the deposited dissolution fines and the off-gas scrubbing products into the vitrified residues.

The WA waste sorts that are defined by the COGEMA and BNFL reprocessing waste specifications and included in this inventory are:

WA-COG-1:	HLW vitrified residues (also represents the BNFL vitrified residues WA-BNF-1)
WA-COG-2:	Precipitates and sludges in bitumen
WA-COG-4:	Hulls and ends in cement
WA-COG-5:	Technological waste (LLW)
WA-COG-6:	α-emitting technological waste
WA-BNF-2:	MEB (Multiple Element Bottle) crud and barium carbonate ($BaCO_3$) in cement
WA-BNF-4:	Hulls and ends in cement
WA-BNF-5:	Solid low level waste (technological waste)
WA-BNF-7:	Centrifuge cake slurry in cement

Note: WA-3 was originally allocated to COGEMA ion exchange resins whose draft specification has since been withdrawn.

With all Swiss spent fuel elements assumed to be reprocessed, a nominal total of 3,591 tU (ton Uranium) with a burn-up of 33,000 MWd/tU is used to derive the waste quantities. (The waste from an additional 135 tU reprocessed under previous contracts will not be returned to Switzerland). The value of 3,591 tU is an estimate corresponding to the reference fuel burn-up given in the waste specification which quotes the number of waste containers produced per tU. As a result of increasing fuel burn-up, it is currently estimated that 3,050 tU of fuel will arise for disposal, but it is not considered that this will modify the reprocessing waste quantities as these are defined by the radionuclide activities in the spent fuel which will be correspondingly higher.

The existing reprocessing contracts with the associated return of waste residues cover 942 tU and allocate 61 % of the spent fuel to COGEMA and 39 % to BNFL. For the nominal total 3,591 tU fuel considered for this inventory, an allocation of 1,790 tU for reprocessing at COGEMA and of 1,801 tU for repro-

cessing at BNFL (i.e. 50 %, 50 %) has been assumed based on information received from the NPP's in 1991.

4.6.2 COGEMA waste characterisation

In order to have all the data required for the planning of waste disposal and as a basis to assess the development of the specifications, Nagra has carried out an independent characterisation of the COGEMA waste for Projekt Gewähr 1985 on the basis of the draft specifications of that time (waste sort "WA-1" in (NAGRA 1985, NAGRA 1984a)). The vitrified residues were recharacterised when the final specification was issued (COGEMA 1986) and the waste sort is designated as WA-COG-1. The draft specifications for the other wastes have recently been modified and re-issued as final specifications (COGEMA 1990a, COGEMA 1990b, COGEMA 1990c, COGEMA 1991). The Nagra characterisation has been modified accordingly and completed where data is missing (e.g. specific activities, materials, neutron dose rates for LMA waste, etc.). Therefore, the data contained in this inventory are not completely those of COGEMA.

The Nagra characterisation of the COGEMA vitrified residue based on the COGEMA specification (COGEMA 1986), resulting in the data contained in the waste sort WA-COG-1, is described below.

The nuclide data for the COGEMA reference PWR spent fuel (33 GWd/tU, 3.5 % enrichment) and of its structural materials were calculated using the ORIGEN2 code. Carry-over fractions for the nuclides from the fuel into the glass were used. Not only all the radionuclides but also all the stable isotopes in the vitrified residues have thus been determined (neither of which were specified completely in the COGEMA specification). The glass chemical composition was modeled, taking into account the glass frit and the process additives, and is given in terms of the weight % of all the oxides (SiO_2, RuO_2, etc.) and elements (Ru, Pd, etc.). Radioactive and stable isotopes of elements as a function of time were requested for the repository near field safety analyses to investigate the elemental solubility for modeling the release of the radionuclides.

A summary of the glass chemical composition of the waste sort WA-COG-1 is given in Appendix D and compared with the COGEMA specification summary (COGEMA 1986). Approximately 1 weight % of the Nagra glass composition corresponds to material contributions not considered or given by COGEMA (activation products from fuel impurities (0.12 %), inconel fines from structural material (0.04 %)) and to an additional component added by Nagra (0.96 % of Gd and Gd activation products) to take into account the occurrence of Gd-poison in BWR fuel as BWR fuel residues will arise and could be mixed with the PWR fuel residues. Additional data to that reported in the database which have been used in the Kristallin-I project are also given in Appendix D.

4.6.3 BNFL waste characterisation

The characterisation of the various BNFL vitrified residues (Magnox, AGR, PWR, BWR and Blend types) has also been carried out independently by Nagra but with direct information from BNFL and was completed before BNFL issued the specification (BNFL 1990a). As explained in section 2.2.3 (Assumptions made for waste sort substitutions) the BNFL vitrified residues have not been included as separate waste sorts within this model inventory. In the HLW repository performance assessment studies, only the COGEMA vitrified residues given in this model inventory have been used. As a justification for this approach, Figure 3 shows the similarity between the vitrified residues which are all conditioned in identical stainless steel containers. The characterisation of the other BNFL waste sorts is based on the BNFL specifications (BNFL 1990b, BNFL 1990c) with completion of unspecified data in a similar way as for the COGEMA waste sorts.

4.6.4 Assessment of the waste sorts and expected modifications

The reliability of the data is, in general, considered to be good due to the data originating from specifications. However, for the technological wastes (WA-COG-5, WA-COG-6 and WA-BNFL-5), the reprocessors were not yet in a position to provide in the specifications detailed radionuclide activities and material compositions. These two basic sets of data have been estimated by Nagra. These data will be updated as soon as the results gained from experience in reprocessing operations become available. The neutron dose rates of the LMA waste sorts (all with exception of WA-1 and WA-5) have been calculated by Nagra because no values were given by COGEMA. In the case of BNFL the same nominal value given for WA-BNF-2, -4 and -7 was judged as being too high. The γ dose rates of the LMA waste sorts were also calculated to enable cross checking of the specification data and completion for hot spots.

Note: The cumulative waste package production of all the WA waste sorts has been modeled as being linear over 40 years between 1994 and 2033. According to the scenario for the operation times of the NPP's the spent fuel arises non-linearly over 49 years between 1976 (waste from the first reprocessing contracts will not be returned) and 2024. These arisings lead to a nominal package production over 56 years between 1979 and 2034 as the nominal package production times (reference time) after fuel discharge vary from 3 to 10 years for the different waste sorts.
The specified time between fuel discharge and reprocessing will be exceeded in the first years of reprocessing, as these operations only started in the early nineties: this will further modify the nominal actinide isotope distribution.
In conclusion, the modeled package production give rise to slightly conservative cumulated activities in the first decades as the longer cooling time for the initial arisings of the spent fuel is not considered.

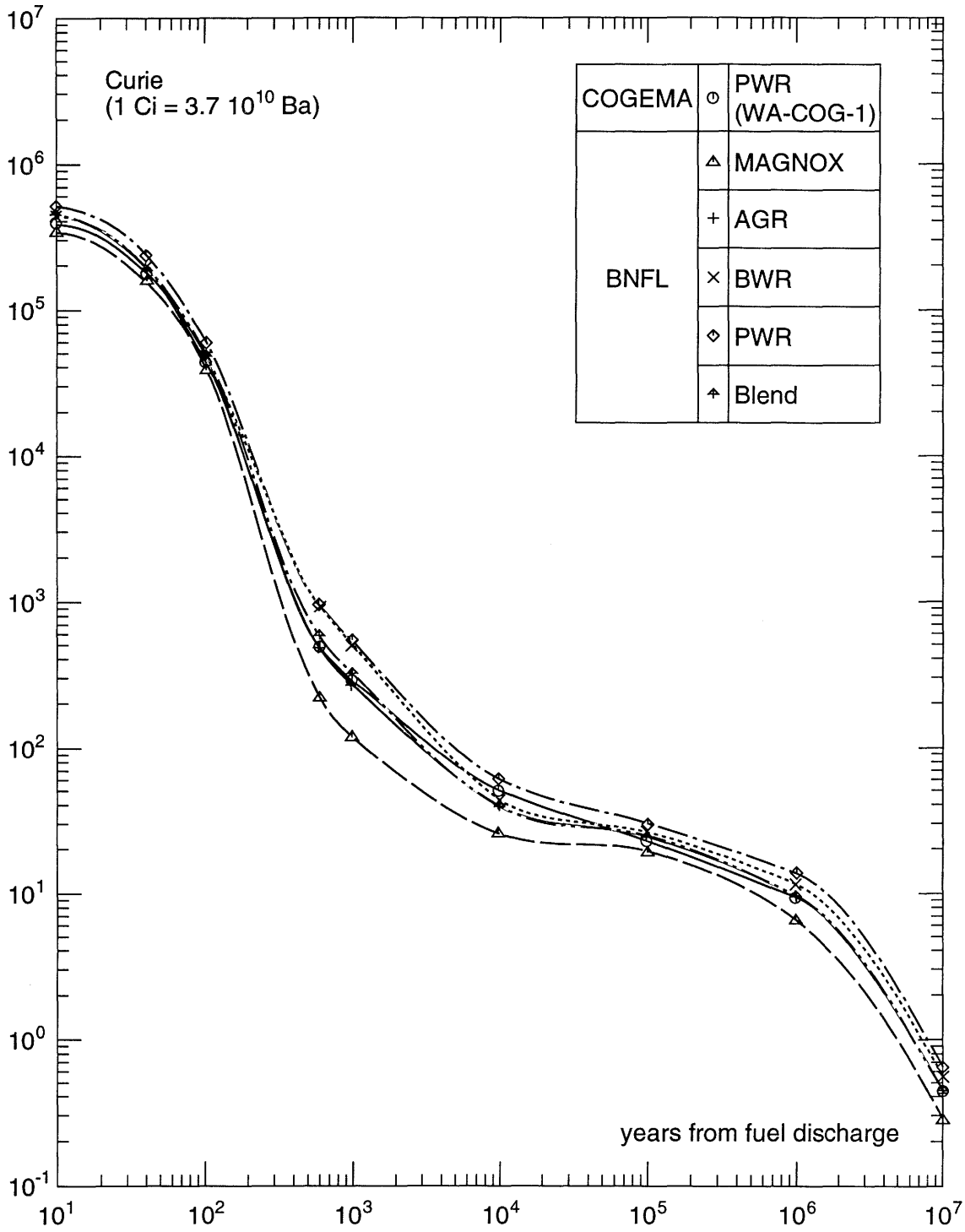


Fig. 3: Nominal activity in function of time of the different specified vitrified residue containers (Nagra data)

In spring 1993 COGEMA announced to the customers that changes in the waste conditioning will take place (with changes in the issued specifications): the conditioning of sludge and precipitates with bitumen (WA-2) will be discontinued and these waste added to the glass (WA-1), the number of WA-COG-6 containers will be reduced through improved waste sorting and a 200 l drum will be introduced in addition to the current WA-COG-5 container for this waste sort. Finally the WA-COG-6 asbestos cement container will be replaced with an iron fibre concrete container (COGEMA 1992).

4.7 MIF (Waste from medicine, industry and research)

Waste sort description

The wastes from medicine, industry and research are collected each year by the government health office (Bundesamt für Gesundheitswesen, BAG) under 5 separate categories. These wastes are then transferred to the "Paul Scherrer Institut" (PSI) for treatment (with sorting for incineration where possible), conditioning in cement and packaging. The research waste arising at PSI are added to the BAG waste during conditioning at PSI. Until 1983 the majority of the MIF waste were sea-dumped.

The following MIF waste sorts are based on these 5 collection categories:

MIF-1:	Beta,gamma emitting waste
MIF-2:	Tritium-bearing waste
MIF-3A:	"Alpha" waste from research (significant U, Pu)
MIF-3B:	"Alpha" waste from industry (significant Am)
MIF-4:	Radium-bearing waste
MIF-5A:	"Alpha,beta,gamma" waste from research
MIF-5B:	"Alpha,beta,gamma" waste from industry

The tritium-bearing waste (MIF-2) mainly arise from the production, use and return of luminescent paints and tritium light devices. MIF-3A wastes arise mainly as residues from reactor fuel (unirradiated) research and development at PSI, whereas MIF-3B waste are produced during the manufacture and return of smoke detectors containing Am-241. MIF-1 contains only β,γ-waste whereas MIF-5 also contains α-emitters (for instance, in MIF-5A from analysed irradiated fuel, and, in MIF-5B from miscellaneous industrial residues containing Am-241). The materials composing the MIF raw waste are diverse, e.g. solids, liquids, gases, slurries, resins, etc.. Very approximatively 15 % of the packaged waste volume arises from medicine, 25 % from industry and 60 % from research.

Note: Specific spallation wastes arising at the PSI accelerator facility (see section 4.5) from operations and modifications to the set-up are currently stored in-situ. These wastes are included in the model inventory within the decommissioning wastes SA-PSIW.

Note: The decommissioning waste of the research reactors are not reported explicitly in the current inventory but are considered with the NPP decommissioning waste (see point 6 of section 5.5).

Waste characterisation

The MIF waste inventory and characterisation was made in 1983-1984 for Projekt Gewähr 1985 (NAGRA 1985). The data from the producers and PSI covering a few years of waste collection, conditioning and packaging were analysed. As would be expected, the volumes, activities and materials of the 5 raw wastes categories have changed over the years and so average distributions were calculated. A yearly production was estimated from the raw waste arisings data by considering further information on total volume and activity data available for other years. The major difficulty found for this analysis was the limited information available for the waste items which, previous to 1984, had been ear-marked for sea-dumping.

The current MIF waste inventory covers, as the previous 1985 inventory, 70 years of production (1984-2053) with the same volumes for the different waste sorts and the same constant yearly production rate. The year 2053 was chosen to match the end of the NPP waste production within a 6 GW(e) scenario. However, under the present 3 GW(e) scenario used for this model inventory, this end of NPP waste production will be in 2033 and for the accelerator decommissioning waste 2040. Therefore, the end of the MIF production for this model inventory could have been considered as being earlier, as the current end date (2053) is an arbitrary value.

No global revision work has been done for the MIF waste by Nagra. Waste specifications are now required by the Swiss regulation (HSK 1988) to be delivered by the producers for each of their waste package types (see section 1.2) including individual waste package data sheets. The specifications for the MIF waste will be produced by PSI who is responsible for conditioning of the raw waste. The producer declaration for the raw waste is currently being revised. Nevertheless, the waste data, especially the radionuclide activities and the material compositions have been improved on a case-by-case basis using the information that has been obtained over recent years.

In a few years, the waste sorting, incineration (plasma furnace) and conditioning will be carried out in a new central facility (ZWILAG in Würenlingen). The MIF waste sorts categorisation and characterisation will at that time be modified.

Waste inventory

The MIF waste (i.e. volume, materials, radioactivities) arising over the next couple of decades cannot be predicted with the same degree of reliability as for the nuclear power plant waste. Industrial and research activities evolve rather rapidly whereas estimates can only be made by extrapolating past data. This extrapolation is even less certain when the significant wastes that have arisen

so far result only from a few specific activities (e.g. light generation with tritium and smoke detectors using Am-241). Comparison with MIF waste produced in other countries is of limited use. The assumption made of a constant yearly production rate cannot be contradicted from data available up to now but an undefined small increase can be expected in the next couple of decades.

In conclusion, the MIF waste inventory has to be considered as approximate and when used should be treated as such.

Remark on waste sort MIF-6

Wastes from the modification of the PSI accelerator facility have already been produced and it is expected that all of these wastes will be eventually conditioned and packaged into 16 m^3 containers built by PSI. For the repository safety analyses these wastes can be considered as being included within the PSI accelerator decommissioning waste (SA-PSIW) of this model inventory. However, for the repository planning knowledge of these 16 m^3 containers is required and therefore a dummy waste sort was created (MIF-6) with an undefined waste product as was the case for BA-KKW-A (section 4.2).

4.8 Concluding remarks

The model waste inventory has been introduced in Chapter 2 (nuclear scenario, waste sorts, etc.) and the forms of the database giving the characterisation data of the waste sorts (Volume II) explained in Chapter 3.

In this Chapter the raw wastes and the waste sorts as well as the general methods of characterisation and the assessments of the data have been reviewed. This Chapter is therefore the reference for the database whose content is given in Volume 2.

From this review it should be clear that the model database could contain different waste data than available from other characterisation papers (for instance the reprocessing waste specifications).

The list of the waste sorts is given in Appendix A with the main characteristics which define them (type of raw waste, material of conditioning and volume of containers). For providing an overview of their other properties (activities, dose rates, etc.) tables summarising these data are also presented in Appendix A.

Beside the database presented in Volume 2 and provided as input data for other projects (repository planning and safety analyses), further additional data have been provided during the course or the conclusion of these projects. The additional data for the waste sort WA-COG-1 are given in Appendix D.

5 WASTE INVENTORIES FOR THE TWO REPOSITORIES

5.1 Objectives

The Swiss waste model inventory presented in the previous Chapters and defined by the database in Volume 2 of this report is used in other Nagra projects. These projects include the repository planning and the safety analyses (operational phase and long-term safety) of the repository "SMA" and of the repository "LMA/HAA".

To enable these projects to be carried out, the first objective is to allocate the waste sorts to the 3 waste classes, SMA, LMA and HAA so that the waste inventories for the repositories are defined. This waste allocation allows the overall characteristics of the repository inventories (cumulative arising of the waste volume and activity, etc.) to be described. It is expected that the kinds of waste authorised to be disposed of in a repository will be defined in the operation licence.

The second objective is to estimate, from the knowledge gained to date, what could be the future deviations of the present model repository inventories. These could be regarded as the uncertainties associated with the present model SMA, LMA and HAA inventories which are the best current estimates and do not include any undue conservatism. The question of these uncertainties is not only related to the waste production but also to the future SMA, LMA and HAA waste acceptance criteria.

The repository design must take into account a waste volume reserve. The radiation dose rates predicted by the safety analyses of the operational phase and of the post-closure phase are related to the waste activities and are affected by the waste materials. Therefore the use of a model inventory for planning purposes requires an estimate of the uncertainties on the total volume of waste, the radionuclide activities and the waste materials.

It is expected that the repositories will be licensed for operation under the use of waste package acceptance criteria, registration and emplacement authorisation. Specified waste packages will reduce the uncertainties on the real repository inventory. A final safety assessment for all the waste disposed of will be carried out before sealing the repository, which will be based on a real inventory, as opposed to a model inventory. It is a possibility that conditions and limits will be set in advance for the total radionuclide activities and waste volume to be disposed of over specified time periods or up to the closure of the repository.

5.2 Waste allocation

The allocation of the waste sorts to the waste classes SMA, LMA and HAA and hence to the corresponding repositories SMA, LMA/HAA is carried out on the basis of the results obtained from the previous long-term safety analyses. The

difference between SMA and LMA waste sorts is a significantly higher α-toxicity resulting from long-lived radionuclides in comparison with the β,γ-toxicity. The allocation used in this report is provisional and intended for disposal projects. It bears no prejudice for the final allocation of the real wastes to the repositories licensed for operation. This provisional allocation is given in the list of the waste sorts contained in Appendix A. The HLW vitrified residues (WA-COG-1) have been designated as HAA. The reprocessing wastes with exception of the LLW technological wastes (WA-COG-5 and WA-BNF-5) have been designated as LMA. All the remaining waste sorts have been designated as SMA.

Due to the nature of a model inventory based on waste sorts describing sometimes differing wastes (for instance core internals and reactor vessel parts in SA-1), it is possible in a few cases that waste components of a designated SMA waste sort will in reality be packaged separately and some of them assigned to the LMA repository. This will depend on the final waste acceptance criteria and on the activity concentration and volume of these particular waste components. A further example of this is the arising of some residual solutions resulting from research fuel fabrication which contain uranium and plutonium. These have been included globally in MIF-3A (α-emitters containing Pu) and allocated to the SMA repository. A few waste packages containing these conditioned solutions may need to be disposed of in the LMA repository if their levels of uranium and plutonium are too high.

5.3 Waste contents of the repositories

According to the current model inventory and the waste allocation assumed, the SMA, LMA and HAA waste sort volumes are summarised below. The waste volume given is the displacement volume.

	Total (m^3)	SMA (m^3)	LMA (m^3)	HAA (m^3)
BA	9238	9238	--	--
RA	2371	2371	--	--
SA-KKW	42948	42948	--	--
SA-PSIW	14147	14147	--	--
WA	30835	23881	6469	485 (2693 containers)
MIF	8700	8700	--	--
Total	108239	101285	6469	485

Note 1: For calculational reasons in the model inventory, values are not rounded to the associated level of uncertainty.

Note 2: The nuclide activities and materials for the SMA waste sorts BA-KKW-A (201 m^3) and MIF-6 (1,120 m^3) have not been characterised (section 4.2 and 4.7). Therefore, the SMA waste sort volume considered in the safety assessment projects is reduced to 99,964 m^3.

A summary of the SMA and LMA waste materials is given in Table 1.

The waste arisings as a function of time are given in Figure 4 for the SMA waste and in Figure 5 for the LMA and HAA waste. According to the waste production scenarios assumed, the production of conditioned waste for each waste category will end in the following years: BA: 2024, RA: 2029, SA-KKW: 2032, WA: 2033, SA-PSIW: 2040, MIF: 2053.

The radionuclide activities of the SMA waste categories BA, RA, SA-KKW, SA-PSIW and sub-WA classified as SMA waste, sub-WA classified as LMA waste and sub-WA classified as HAA-waste, calculated at the end of their respective production period, are given in Table 2. This Table is intended for reporting the activities of these waste categories for comparison with other waste inventories.

The total radioactivity of the SMA, LMA and HAA wastes as a function of time allowing for decay and arising of the waste over the production period is given in Figures 6 and 7.

Note: The WA cumulated activities are slightly conservative over the first decades as a result of the way the waste package production has been modeled between 1994 and 2033 (see note in section 4.6.4).

The toxicity of the SMA, LMA and HAA wastes as a function of time has been calculated from the radionuclide activities. The radiotoxicity index (RTI) for ingestion is defined as:

$$RTI = \Sigma Aj \cdot DFj/DL$$

where Aj = activity of nuclide j in Bq
DFj = dose factor of ingestion for nuclide j in Sv/Bq
DL = dose limit from a repository for post-closure: 0.1 mSv/year

The dose factors, DFj, are taken from the Nagra "Activity to dose conversion factors" database (NAGRA 1991). The factors taken are the most pessimistic values without consideration of the different chemical forms of the radionuclides.

Figure 8 gives the radiotoxicity of the components of the SMA waste. It can be seen from this Figure that the dominant contribution to total toxicity originates from MIF wastes. However, with reference to Figure 6, the dominant contribution to the total radioactivity up to 10^5-10^6 years is provided by the decommis-

sioning SA-KKW wastes. This difference is due to MIF waste having a considerable Am-241 content (and hence α-toxicity) arising from smoke detectors (MIF-3B) and to a lesser extent (about an order of magnitude) from other industrial uses (MIF-5B).

A comparison between the total radioactivities and radiotoxicities of the SMA, LMA and HAA wastes given as a function of time is shown in Figures 9a and 9b.

The contribution to total toxicity from α-emitters and from β, γ-emitters are shown in Figure 10. It can be seen that the toxicity of α-emitters dominates for the majority of the time.

Finally it should be borne in mind that the activities and toxicities of the SMA, LMA and HAA waste sorts compared in Figures 9 and 10 are the result of a provisional allocation of the waste sorts currently made for the design and safety assessments of the projected repositories.

Table 1: Summary of SMA and LMA waste materials

Material	SMA wastes		LMA wastes	
	Mass (kg)	Weight-%	Mass (kg)	Weight-%
Metals	6.27E+7	25.47	3.77E+6	27.40
Steel	5.85E+7	23.78	2.76E+6	20.01
Al/Zn	3.85E+5	0.16	3.94E+3	0.03
Inconel	1.72E+6	0.70	5.11E+4	0.37
Zircaloy	3.24E+5	0.13	9.54E+5	6.94
Other metals	1.73E+6	0.70	1.32E+3	0.01
Inorganics	1.70E+8	69.30	9.10E+6	66.16
Salts	2.02E+5	0.08	1.57E+6	11.41
Ashes	3.93E+5	0.16	--	--
Glass	1.55E+6	0.63	3.86E+4	0.28
Cementitious materials	1.66E+8	67.58	7.29E+6	53.00
Other inorganics	2.08E+6	0.85	2.00E+5	1.45
Organics	1.28E+7	5.20	8.86E+5	6.44
High molecular weight (subtotal)	1.27E+7	5.16	8.73E+5	6.33
- Bitumen	4.60E+5	0.19	7.14E+5	5.17
- Ion exchange resins	1.97E+6	0.80	--	--
- Cellulose	5.46E+6	2.22	2.33E+3	0.02
- Others	4.79E+6	1.95	1.57E+5	1.14
Low molecular weight (subtotal)	8.23E+4	0.03	1.25E+4	0.09
- Detergents	4.38E+4	0.02	1.24E+4	0.09
- Flocculants	5.69E+2	2E-4	--	--
- Complexing agents	3.63E+4	0.01	1.25E+2	9E-4
- Others	1.55E+3	6E-4	--	--
Others unknown	8.53E+4	0.03	--	--
Total	2.46E+8	100.00	1.38E+7	100.00

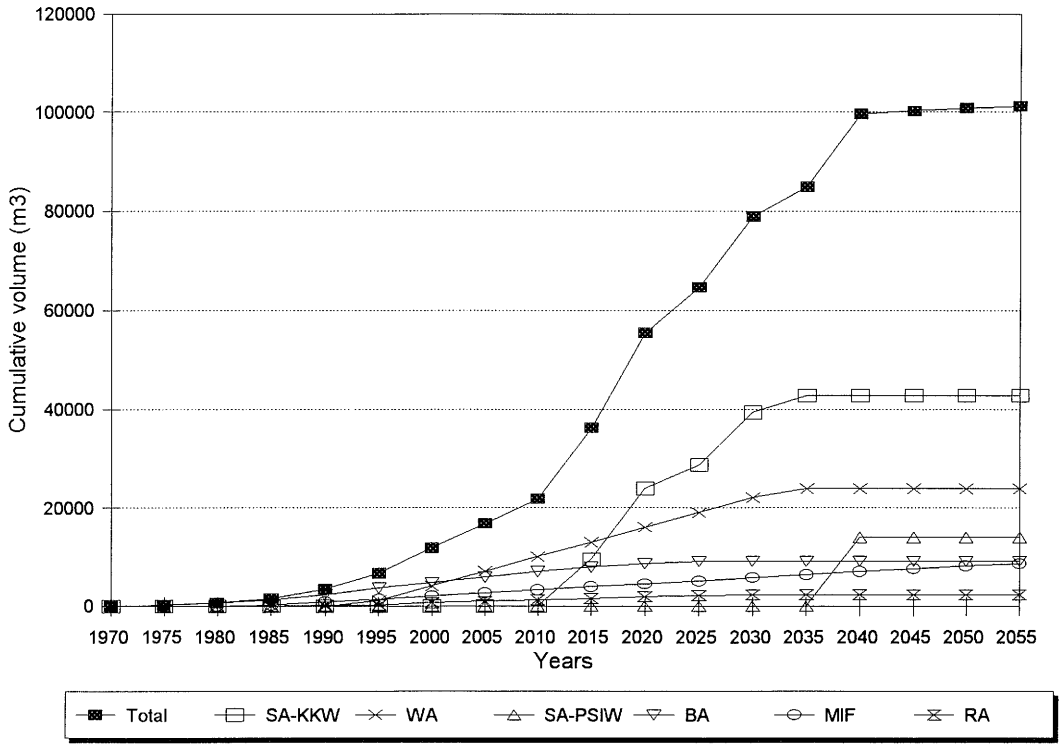


Fig. 4: Arising of SMA waste

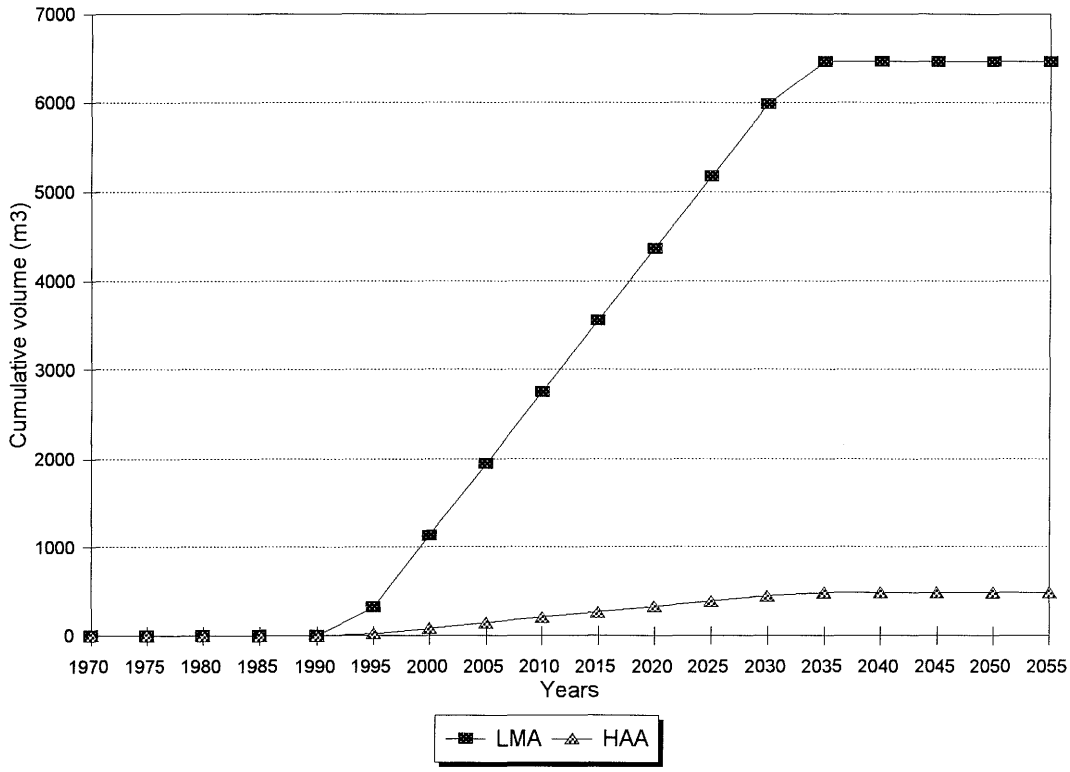


Fig. 5: Arising of LMA and HAA wastes

Table 2: Radionuclide activities (Bq) of the waste categories calculated at the end year of the waste package production period

Class →	SMA	SMA	SMA	SMA	SMA	SMA	LMA	HAA
Category →	BA	RA	SA-NPP	SA-PSIW	MIF	WA	WA	WA
Year →	2024	2029	2032	2040	2053	2033	2033	2033
Nuclide	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]
H-3	1.5E11	3.0E14	2.2E14	2.0E14	8.6E15	1.8E9	1.1E16	--
Be-10	--	2.1E10	3.1E7	1.4E9	--	2.8E0	1.6E7	5.9E8
C-14	2.0E12	3.1E13	1.5E14	3.3E11	7.6E13	1.2E5	5.1E13	1.2E11
Na-22	2.2E7	--	--	3.0E13	3.7E9	--	--	--
Al-26	--	--	--	9.7E9	--	--	--	--
Si-32	--	1.1E3	--	1.9E12	--	--	--	--
P-32	--	1.1E3	--	1.7E12	--	--	--	--
S-35	--	--	--	1.0E11	--	--	--	--
Cl-36	2.9E9	2.6E11	8.5E11	6.6E10	2.7E10	2.4E5	2.2E11	--
Ar-39	--	1.5E10	2.9E12	6.2E13	7.8E4	--	1.0E2	--
Ar-42	--	--	--	5.7E12	--	--	--	--
K-40	--	--	2.7E9	1.2E9	--	--	--	--
K-42	--	--	--	5.1E12	--	--	--	--
Ca-41	--	1.6E10	1.8E12	9.7E11	2.4E5	3.8E3	1.9E7	2.5E10
Ca-45	--	< 1	1.5E11	9.1E12	6.6E10	1.9E1	7.1E8	4.1E9
Sc-44	--	--	--	5.5E13	--	--	--	--
Sc-46	--	< 1	9.3E7	1.8E11	--	--	--	--
Ti-44	--	--	--	6.3E13	--	--	--	--
V-49	--	--	1.2E12	7.6E10	--	--	--	--
Mn-53	--	2.9E8	5.0E9	5.2E10	--	--	--	--
Mn-54	6.5E11	1.9E12	3.2E13	1.1E14	9.7E7	9.0E5	2.2E14	6.7E11
Fe-55	3.8E13	1.5E15	6.1E16	8.0E15	1.6E9	1.1E8	2.1E16	9.9E13
Fe-59	--	< 1	7.0E5	--	--	< 1	2.9E7	7.4E2
Fe-60	--	--	--	1.1E10	--	--	--	--
Co-56	--	--	--	3.2E11	--	--	--	--
Co-57	--	--	5.7E11	5.6E13	--	5.4E5	5.3E13	4.1E12
Co-58	6.0E2	3.5E5	2.2E11	7.1E11	9.6E8	8.5E0	1.3E11	1.1E8
Co-60	1.7E14	9.6E15	8.8E16	9.3E14	7.1E12	4.0E8	4.1E16	3.2E15
Co-60M	--	--	--	1.1E10	--	--	--	--
Ni-59	8.1E12	2.2E13	1.3E15	2.0E13	9.1E9	1.4E6	4.2E14	5.1E12
Ni-63	6.7E14	2.7E15	1.6E17	4.0E15	9.7E11	1.8E8	5.5E16	5.9E14
Zn-65	1.1E11	3.6E9	4.2E12	7.2E11	4.3E11	1.7E6	2.6E11	1.0E13
Ga-68	--	--	--	1.3E11	--	--	--	--
Ge-68	--	--	--	1.3E11	--	--	--	--
As-73	--	--	--	1.2E9	--	--	--	--
Se-75	--	--	--	1.3E10	1.2E8	--	--	--
Se-79	4.9E9	1.6E8	9.6E8	3.5E8	5.4E8	8.4E6	5.0E10	5.7E13
Kr-81	--	4.7E7	1.2E9	2.6E10	--	--	--	--
Kr-83M	--	--	--	6.0E9	--	--	--	--
Kr-85	--	2.2E11	5.1E11	--	5.8E12	--	5.3E14	--
Rb-83	--	--	--	7.8E9	--	--	--	--
Sr-85	--	--	--	5.8E8	--	--	--	--
Sr-90	1.3E12	1.0E13	2.3E12	--	5.0E13	9.1E11	7.5E15	6.1E18

Table 2: continued (1)

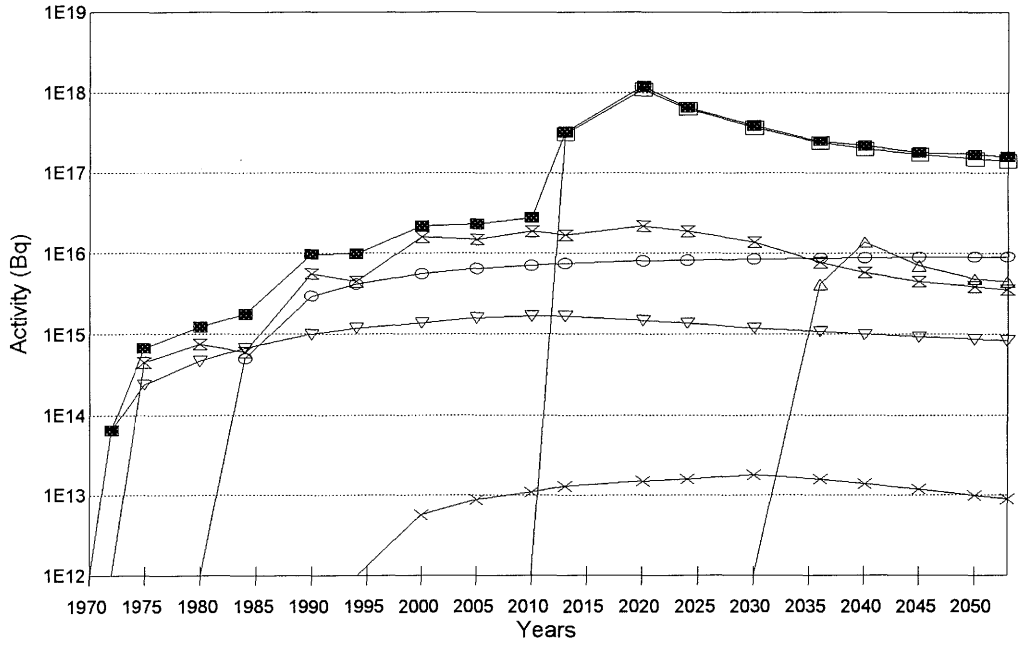
Class →	SMA	SMA	SMA	SMA	SMA	SMA	LMA	HAA
Category →	BA	RA	SA-NPP	SA-PSIW	MIF	WA	WA	WA
Year →	2024	2029	2032	2040	2053	2033	2033	2033
Nuclide	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]
Y-88	--	--	--	1.2E11	--	--	--	--
Y-90	1.3E12	1.0E13	2.3E12	--	5.0E13	9.1E11	7.5E15	6.1E18
Zr-93	3.6E8	9.7E12	1.8E8	1.5E9	2.4E9	1.2E7	2.0E13	2.5E14
Zr-88	--	--	--	7.6E9	--	--	--	--
Zr-95	--	2.0E5	2.0E4	6.6E7	--	--	--	--
Nb-91	--	--	--	1.0E11	--	--	--	--
Nb-91M	--	--	--	4.0E8	--	--	--	--
Nb-92	--	--	--	7.6E7	--	--	--	--
Nb-93M	2.5E8	6.7E12	4.4E12	5.0E13	9.9E7	7.5E6	1.3E13	1.6E14
Nb-94	1.4E9	3.8E12	5.3E12	2.2E11	3.2E6	8.9E2	1.5E14	2.7E11
Nb-95	--	4.5E5	4.7E4	1.4E8	--	8.0E4	1.1E12	1.7E12
Nb-95M	--	1.5E3	2.8E0	4.9E5	--	--	--	--
Mo-93	--	1.2E11	1.0E12	1.4E12	--	1.3E2	5.6E12	1.0E11
Tc-99	1.9E11	9.9E10	2.2E11	8.5E9	5.5E11	8.9E7	3.6E12	1.8E15
Ru-106	9.4E9	8.3E10	--	--	2.5E12	1.1E10	1.8E15	2.2E17
Pd-107	5.0E5	2.6E7	< 1	2.4E4	1.3E8	2.2E4	5.5E10	1.4E13
Ag-108	--	5.8E13	1.3E10	9.2E4	1.1E6	6.1E2	1.9E8	1.3E11
Ag-108M	--	6.6E14	1.5E11	1.0E6	1.3E7	7.1E3	2.1E9	1.5E12
Ag-110M	1.4E11	4.0E11	--	7.1E10	5.5E7	5.5E7	1.3E12	3.5E14
Cd-109	--	8.0E12	--	--	--	--	--	--
Cd-113	--	< 1	--	--	--	--	--	--
Cd-113M	--	3.6E9	--	--	--	--	--	--
In-113M	--	3.9E8	--	--	--	--	2.1E11	--
Sn-113	--	3.9E8	--	--	--	--	1.9E12	--
Sn-119M	--	1.3E13	2.4E6	7.6E7	--	3.2E6	1.2E15	2.4E13
Sn-121	--	2.1E13	4.8E9	1.1E7	--	2.3E6	3.1E15	1.5E13
Sn-121M	--	2.7E13	6.3E9	1.9E7	--	3.1E6	4.1E15	2.0E13
Sn-123	--	6.9E8	1.2E5	3.0E7	--	7.8E5	6.3E11	5.6E12
Sn-126	2.3E8	9.2E6	4.8E5	--	9.2E10	1.5E7	8.3E10	1.1E14
Sb-124	< 1	2.5E1	--	--	--	--	--	--
Sb-125	6.9E11	3.0E14	2.7E11	1.6E9	5.9E8	3.9E9	5.9E15	7.6E16
Sb-126	3.2E7	1.3E6	6.7E4	--	1.3E10	2.1E6	1.1E10	1.4E13
Sb-126M	2.3E8	9.2E6	4.8E5	--	9.1E10	1.4E7	8.2E10	1.0E14
Te-125M	1.7E11	7.4E13	5.6E10	2.7E8	1.3E8	9.5E8	1.3E15	1.9E16
I-129	1.4E8	3.2E9	5.4E7	--	4.1E7	6.5E5	1.3E11	4.3E9
Cs-134	9.9E12	3.7E11	4.1E11	4.3E11	5.0E12	7.5E10	9.7E14	4.5E17
Cs-135	1.4E9	1.3E8	7.4E7	--	7.4E8	8.4E6	2.9E11	5.1E13
Cs-137	2.7E14	2.0E13	4.3E12	--	9.8E13	1.3E12	2.3E16	8.6E18
Ba-133	--	9.4E10	1.8E12	5.9E9	2.3E10	--	--	--
Ba-137M	2.5E14	1.9E13	4.0E12	--	9.2E13	1.3E12	2.2E16	8.1E18
Ce-144	1.5E11	1.9E10	2.6E10	--	3.4E12	2.8E10	4.4E14	1.8E17
Pr-144	1.5E11	1.9E10	2.6E10	--	3.4E12	2.8E10	4.4E14	1.8E17
Pr-144M	1.1E9	3.4E8	2.0E8	--	2.5E10	2.1E8	3.3E12	1.3E15
Pm-145	--	4.9E7	1.4E9	4.7E8	--	--	--	--

Table 2: continued (2)

Class →	SMA	SMA	SMA	SMA	SMA	SMA	LMA	HAA
Category →	BA	RA	SA-NPP	SA-PSIW	MIF	WA	WA	WA
Year →	2024	2029	2032	2040	2053	2033	2033	2033
Nuclide	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]
Pm-147	1.3E11	4.9E11	1.6E9	--	1.3E13	1.0E11	7.7E14	7.2E17
Sm-146	--	--	1.5E2	--	--	--	--	--
Sm-147	< 1	3.6E0	< 1	--	7.2E1	3.6E0	8.1E5	6.6E8
Sm-151	--	3.6E9	7.9E10	3.7E10	3.9E11	6.1E9	1.3E14	4.2E16
Eu-152	7.5E7	6.0E7	9.1E12	8.3E12	9.7E8	4.7E7	3.0E11	2.9E14
Eu-154	7.7E10	1.1E11	1.4E12	1.0E12	2.3E12	4.3E0	9.5E14	3.2E17
Eu-155	1.9E10	2.0E10	1.8E11	2.0E10	9.2E11	1.3E0	6.3E13	9.2E16
Gd-152	< 1	< 1	1.1E0	< 1	< 1	< 1	< 1	4.2E2
Gd-153	--	--	--	--	8.0E8	--	--	--
Tb-158	--	1.8E8	2.9E9	--	--	--	--	--
Ho-166M	--	1.6E10	2.5E11	1.0E9	--	4.1E4	2.2E8	5.3E11
Tm-170	--	--	1.0E9	2.6E11	--	--	--	--
Hf-178M	--	6.4E9	5.4E10	--	--	--	--	--
Ta-182	--	1.1E9	1.3E9	6.4E5	--	--	2.9E13	1.1E11
W -185	--	< 1	--	--	--	--	--	--
Tl-204	--	--	--	--	1.0E11	--	--	--
Tl-206	--	--	2.3E1	--	--	--	--	--
Tl-207	3.3E2	< 1	< 1	--	3.0E1	2.2E2	2.2E6	1.2E9
Tl-208	4.9E6	< 1	< 1	--	1.7E7	1.4E5	2.7E8	1.8E10
Tl-209	2.6E0	1.2E5	1.1E3	--	< 1	< 1	2.6E2	4.1E5
Pb-205	--	1.7E5	2.9E6	--	--	--	--	--
Pb-209	1.2E2	5.5E6	4.9E4	--	< 1	3.4E1	1.2E4	1.9E7
Pb-210	3.6E0	< 1	< 1	--	7.0E10	3.3E4	8.6E3	2.8E7
Pb-211	3.3E2	< 1	< 1	--	3.0E1	2.2E2	2.2E6	1.2E9
Pb-212	1.3E7	< 1	< 1	--	4.6E7	3.9E5	7.6E8	5.0E10
Pb-214	1.2E2	1.1E1	< 1	--	2.3E12	4.7E1	2.9E5	8.5E7
Bi-210	3.5E0	< 1	< 1	--	6.8E10	3.1E4	8.4E3	2.8E7
Bi-211	3.3E2	< 1	< 1	--	3.0E1	2.2E2	2.2E6	1.2E9
Bi-212	1.3E7	< 1	< 1	--	4.6E7	3.9E5	7.6E8	5.0E10
Bi-213	1.2E2	5.5E6	4.9E4	--	< 1	3.4E1	1.2E4	1.9E7
Bi-214	1.2E2	1.1E1	< 1	--	2.3E12	4.7E1	2.9E5	8.5E7
Po-210	1.9E0	< 1	< 1	--	3.7E10	2.6E4	4.5E3	1.4E8
Po-218	1.2E2	1.1E1	< 1	--	2.3E12	4.7E1	2.9E5	8.5E7
Rn-222	1.2E2	1.1E1	< 1	--	2.3E12	4.7E1	2.9E5	8.5E7
Fr-221	1.2E2	5.5E6	4.9E4	--	< 1	3.4E1	1.2E4	1.9E7
Fr-223	5.2E0	< 1	< 1	--	< 1	3.3E0	3.1E4	1.6E7
Ra-223	3.3E2	< 1	< 1	--	3.0E1	2.2E2	2.2E6	1.2E9
Ra-224	1.3E7	< 1	< 1	--	4.6E7	3.9E5	7.6E8	5.0E10
Ra-225	1.2E2	5.5E6	5.1E4	--	< 1	3.4E1	1.2E4	1.9E7
Ra-226	1.2E2	1.1E1	< 1	--	2.4E12	4.8E1	2.9E5	8.5E7
Ra-228	< 1	< 1	< 1	--	2.9E8	< 1	3.8E0	7.3E3
Ac-225	1.2E2	5.5E6	4.9E4	--	< 1	3.4E1	1.2E4	1.9E7
Ac-227	3.7E2	< 1	< 1	--	3.9E1	2.4E2	2.2E6	1.2E9
Ac-228	< 1	< 1	< 1	--	2.9E8	< 1	3.8E0	7.3E3

Table 2: continued (3)

Class →	SMA	SMA	SMA	SMA	SMA	SMA	LMA	HAA
Category →	BA	RA	SA-NPP	SA-PSIW	MIF	WA	WA	WA
Year →	2024	2029	2032	2040	2053	2033	2033	2033
Nuclide	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]	[Bq]
Th-227	3.5E2	< 1	< 1	--	3.2E1	2.3E02	2.1E6	1.1E9
Th-228	1.3E7	< 1	< 1	--	4.8E7	4.0E5	7.9E8	6.9E10
Th-229	1.2E2	5.5E6	5.5E4	--	< 1	3.4E1	1.2E4	1.9E7
Th-230	1.7E4	1.3E1	< 1	--	1.5E4	4.7E3	4.7E7	8.4E9
Th-231	2.0E7	9.0E5	2.9E1	--	1.2E8	4.3E6	4.4E9	4.3E9
Th-232	< 1	< 1	< 1	--	2.6E9	< 1	3.4E1	8.7E3
Th-234	5.7E8	1.7E8	9.5E6	--	3.9E9	6.7E7	4.7E10	6.7E10
Pa-231	1.2E4	1.9E1	< 1	--	2.4E3	5.5E3	4.8E6	2.3E9
Pa-233	2.2E6	1.4E8	2.7E7	--	9.1E8	7.0E7	3.7E10	4.4E13
Pa-234	1.9E6	5.7E5	3.2E4	--	1.3E7	2.2E5	1.6E8	2.2E8
Pa-234M	5.7E8	1.7E8	9.5E6	--	3.9E9	6.7E7	4.7E10	6.7E10
U-232	1.4E7	--	--	--	--	1.2E6	8.5E8	7.5E9
U-233	4.6E4	2.7E9	5.8E8	--	3.5E3	9.5E3	6.4E6	3.9E9
U-234	7.1E7	3.0E6	1.9E5	--	1.7E9	3.0E8	2.3E11	3.9E11
U-235	2.0E7	9.0E5	2.9E1	--	1.2E8	4.5E6	4.5E9	4.3E9
U-235M	1.4E10	1.2E11	3.0E10	--	6.5E11	6.3E10	6.9E13	3.0E14
U-236	4.6E8	2.3E7	1.0E3	--	4.1E8	5.9E7	3.7E10	5.7E10
U-237	1.8E7	8.0E8	8.2E7	--	1.1E9	2.8E8	1.9E11	1.1E12
U-238	5.7E8	1.7E8	9.6E6	--	4.0E9	6.9E7	4.9E10	6.7E10
U-240	2.3E1	--	--	--	--	2.4E0	3.6E4	3.2E5
Np-237	2.2E6	1.4E8	2.8E7	--	9.1E8	7.1E7	3.7E10	4.4E13
Np-238	4.9E6	--	--	--	3.4E7	7.5E6	4.0E9	4.0E12
Np-239	7.0E10	4.4E10	5.2E8	--	1.9E10	3.8E9	1.9E12	2.0E15
Np-240M	2.3E1	--	--	--	--	2.4E0	3.6E4	3.2E5
Pu-236	--	--	--	--	--	--	--	1.5E10
Pu-238	1.1E11	1.0E12	6.8E10	--	2.2E12	4.3E11	3.9E14	2.3E15
Pu-239	1.4E10	1.2E11	3.0E10	--	6.5E11	6.3E10	6.9E13	3.0E14
Pu-240	2.3E10	2.4E11	3.5E10	--	7.4E11	7.7E10	9.4E13	7.8E14
Pu-241	7.2E11	3.3E13	3.4E12	--	4.7E13	1.2E13	7.9E15	4.6E16
Pu-242	1.2E8	6.4E6	1.3E8	--	2.1E9	4.1E8	3.2E11	1.6E12
Pu-244	2.3E1	--	--	--	--	2.5E0	3.8E04	3.2E5
Am-241	1.9E11	2.5E12	1.6E11	--	4.4E13	6.8E11	4.8E14	8.8E16
Am-242	1.0E9	--	--	--	7.0E9	1.5E9	8.3E11	8.5E14
Am-242M	1.0E9	--	--	--	7.2E9	1.7E9	8.6E11	8.6E14
Am-243	7.0E10	4.4E10	5.3E8	--	1.9E10	3.8E9	1.9E12	2.0E15
Cm-242	8.6E8	3.7E7	4.7E10	--	6.4E9	2.1E9	1.2E12	1.1E15
Cm-243	1.2E9	8.2E9	--	--	1.0E10	2.5E9	1.2E12	1.4E15
Cm-244	1.7E11	1.4E12	1.2E11	--	7.1E11	1.7E11	8.6E13	9.6E16
Cm-245	--	--	--	--	1.5E8	2.6E7	1.4E10	1.6E13
Cm-246	--	--	--	--	--	4.9E6	2.8E9	3.0E12
Total	1.4E15	1.5E16	3.1E17	1.4E16	9.1E15	1.8E13	2.2E17	3.1E19
Total β,γ	1.4E15	1.5E16	3.1E17	1.4E16	9.1E15	1.7E13	2.2E17	3.1E19
Total α	6.7E11	5.4E12	4.6E11	0.0E00	5.5E13	1.4E12	1.1E15	2.0E17



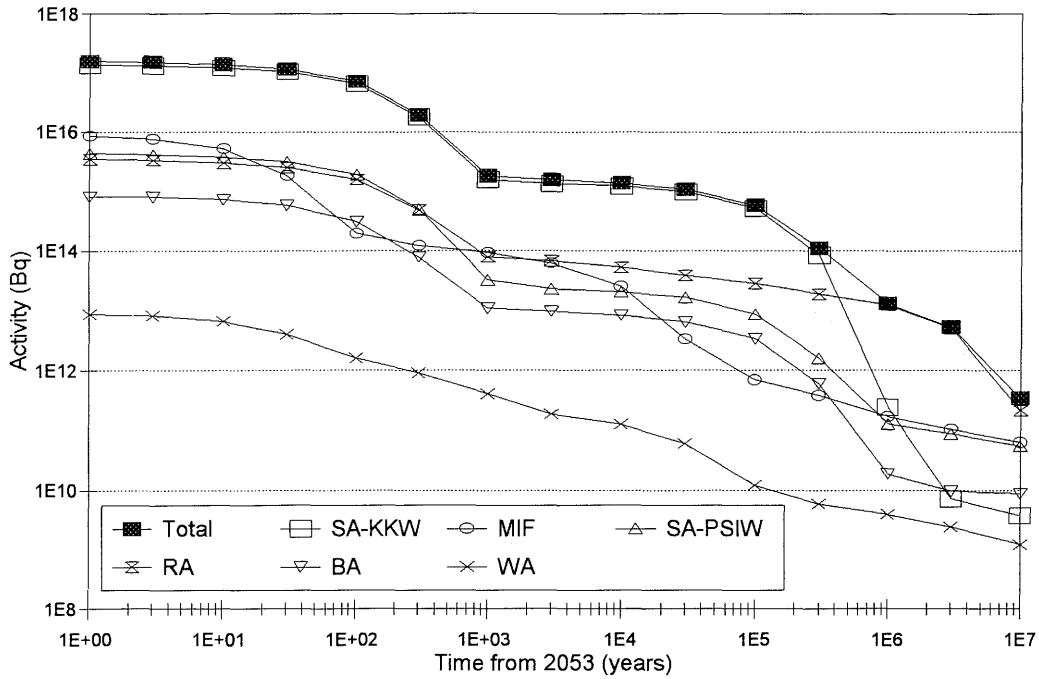
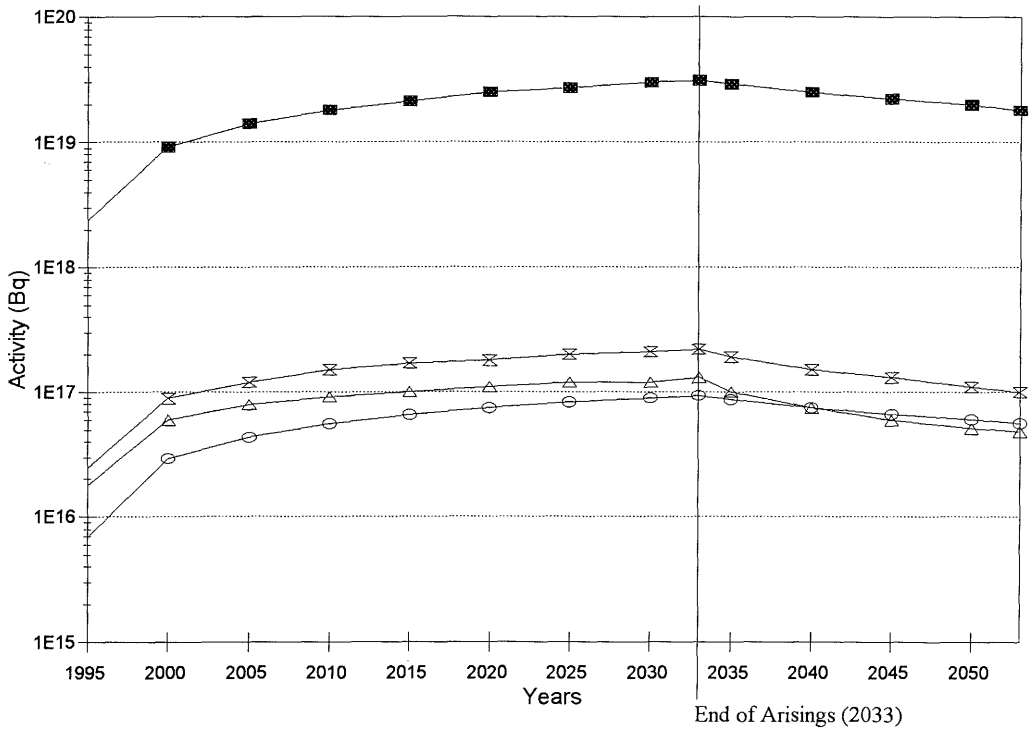


Fig. 6: Radioactivity of the SMA waste
(The total radioactivity is dominated by Ni-63 ($t_{1/2}$ = 100 y), then Ni-59 ($t_{1/2}$ = 7.6 10^4 y) and finally Zr-93 ($t_{1/2}$ = 1.5 10^6 y))



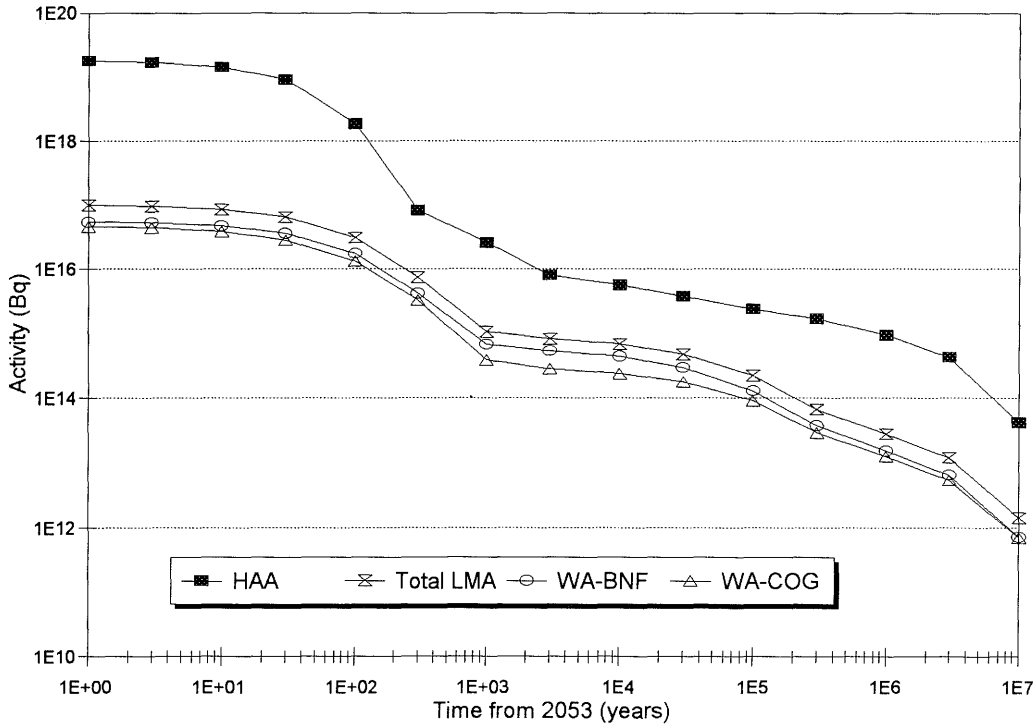


Fig. 7: Radioactivity of the LMA and HAA waste

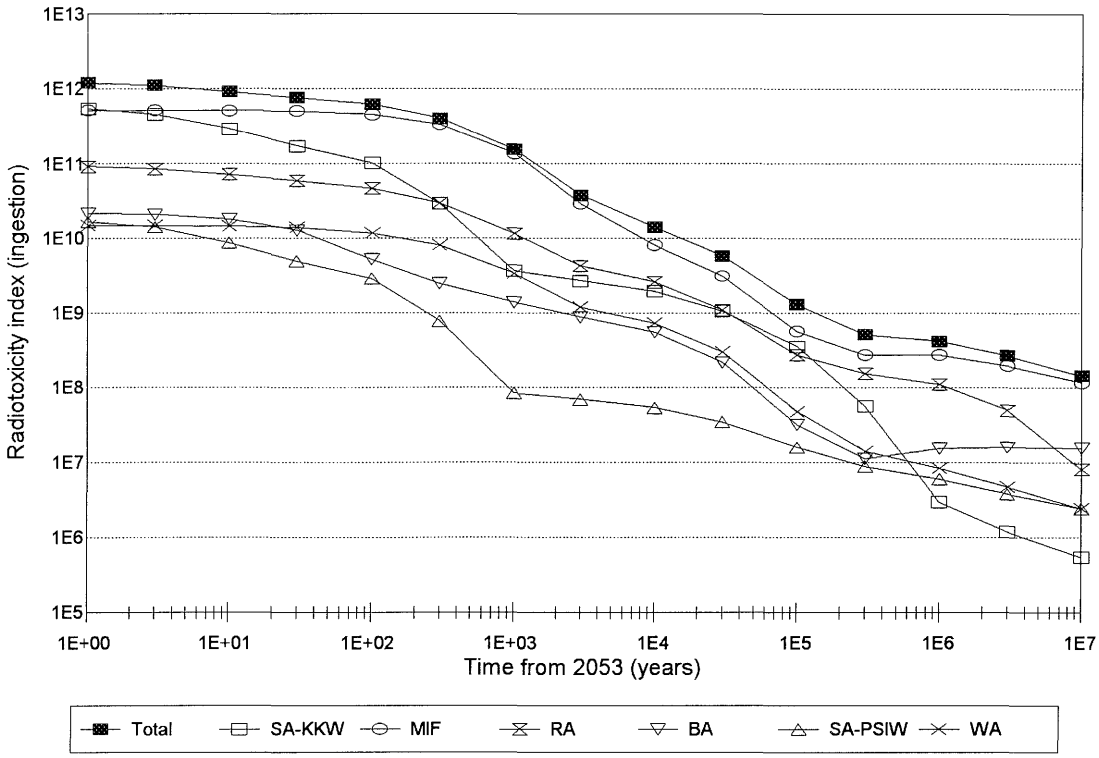


Fig. 8: Radiotoxicities (ingestion) of the SMA waste (to be compared with Fig. 6)
(The total radiotoxicity is dominated by Am-241 ($t_{1/2}$ = 432 y), then Pu-239 ($t_{1/2}$ = 24,100 y) and finally Np-237 (2.1 10^6 y))

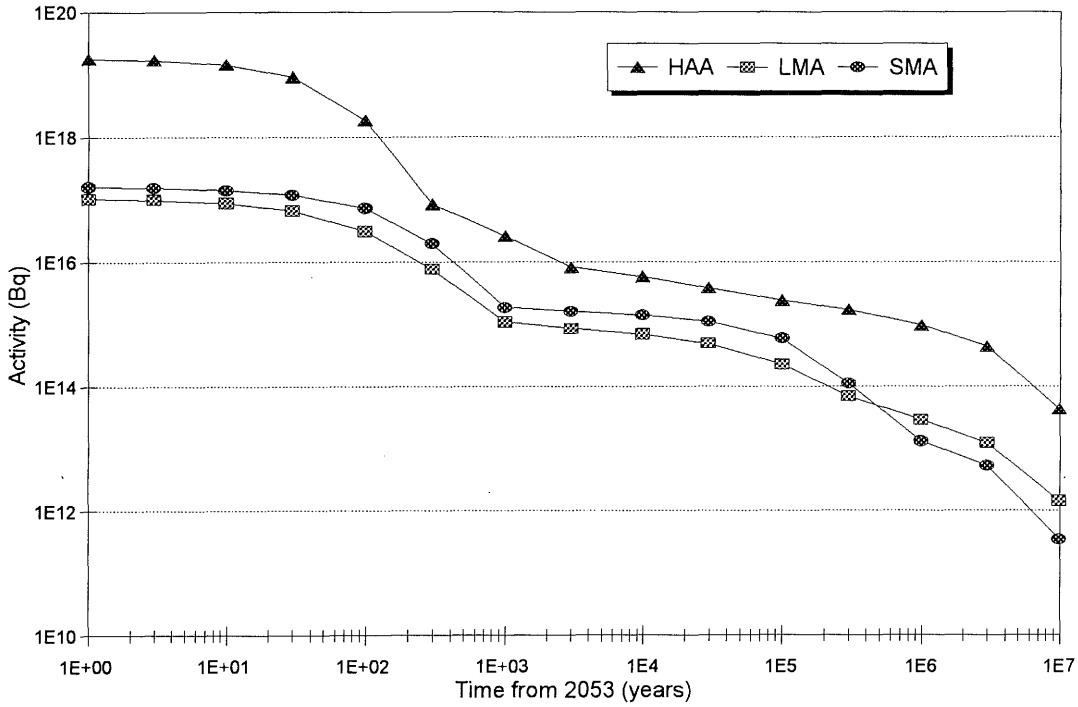


Fig. 9a: Comparison of the SMA, LMA and HAA waste activities

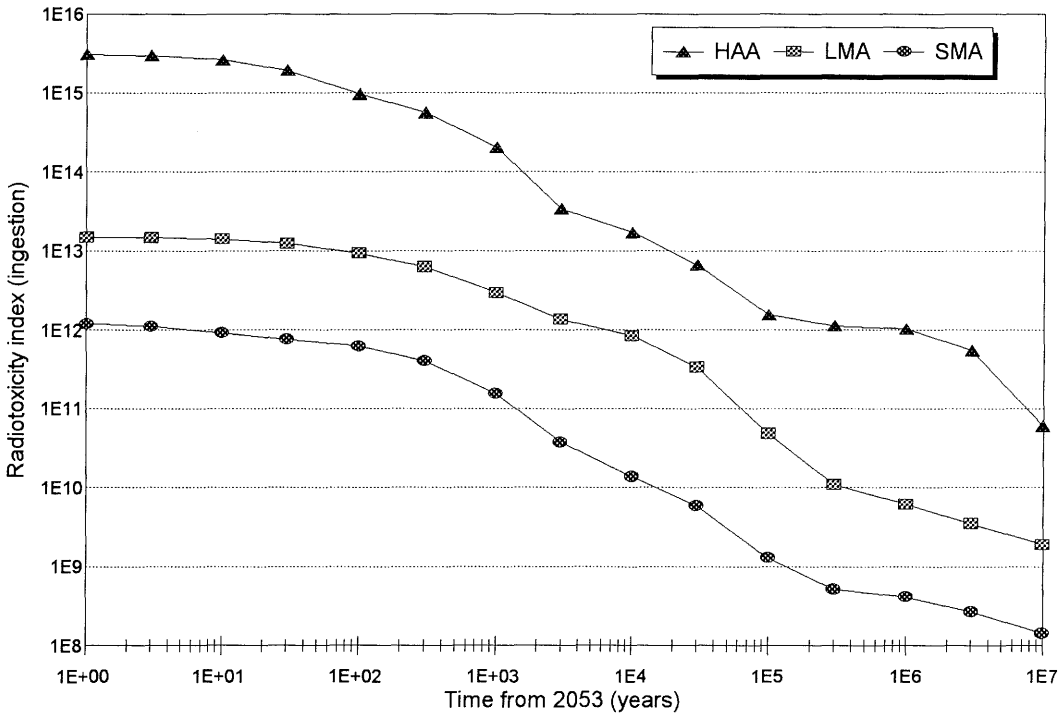


Fig. 9b: Comparison of the SMA, LMA and HAA waste radiotoxicities (ingestion)

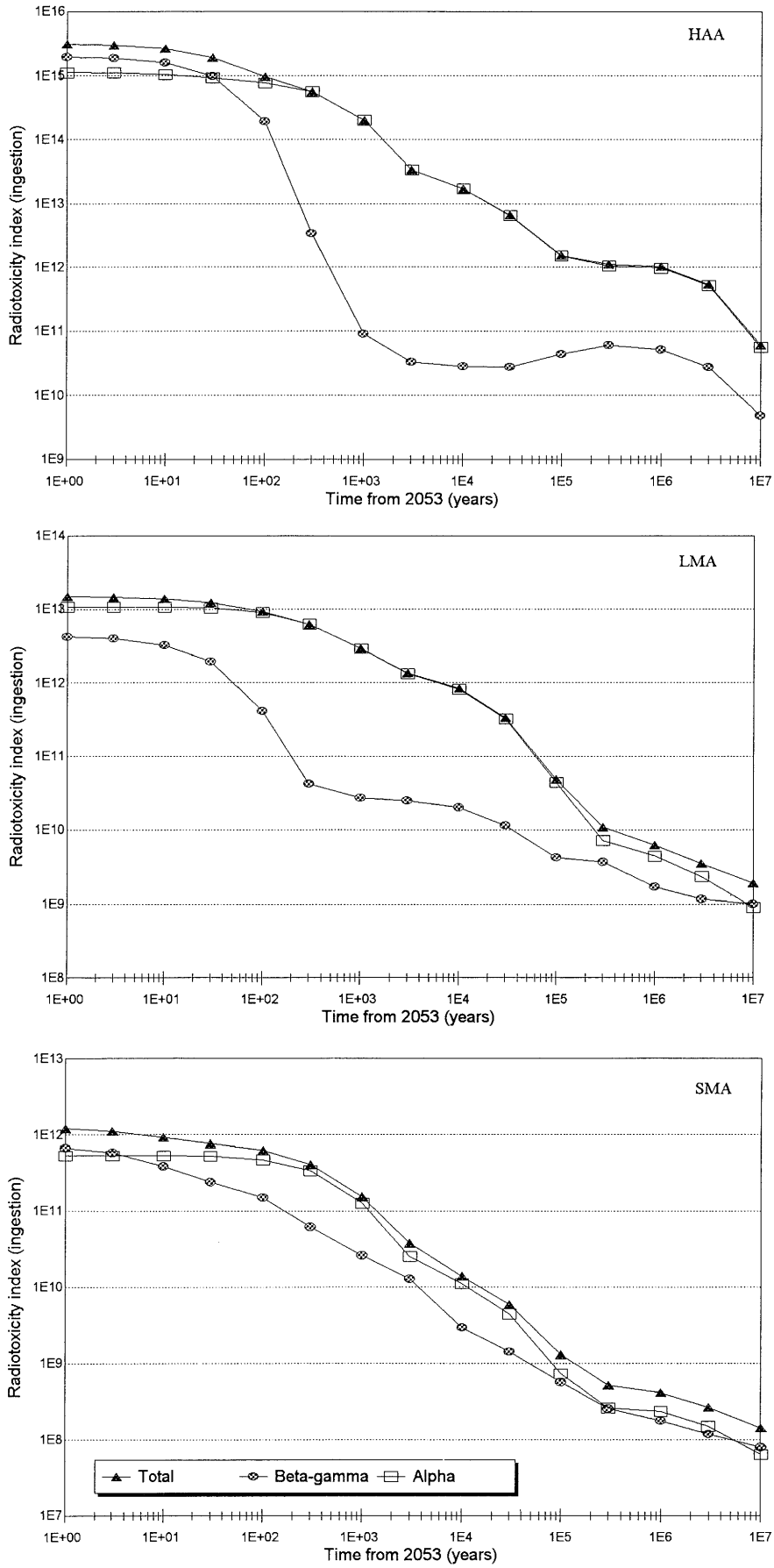


Fig. 10: Comparison of the β,γ- with the α-emitter component of the SMA, LMA and HAA waste radiotoxicities (ingestion)

5.4 Model waste inventory uncertainties

The model waste inventory is an estimate of waste arising in the future and the word "uncertainty" is intended to cover possible deviations from this estimate. These uncertainties can be more simply addressed if they are classified according to their origin. The following classes of uncertainties have been considered:

A) Waste production scenario assumed
B) Development of nuclear power technology
C) Changes occurring in medicine, industry and research activities
D) Improvement of waste arising, treatment and conditioning processes
E) Waste stored and declassified as not being radioactive or recycled
F) Waste which might be exchanged with other countries
G) Wastes that are known to occur, but as yet, are not included as waste sorts
H) Changes in regulations
I) Improved characterisation of the waste

All classes of uncertainties arise because the future is not "known", with exception of class I which is due to the current level of characterisation of the conditioned wastes.

It would be difficult to consider all the uncertainties on the same level because of the probability of the events the uncertainties are associated with. For the sake of simplification 2 levels of probability, low and high, will be assumed for judging the uncertainties considered in this report. The first objective of estimating these uncertainties is to quantify reserves in waste volumes and estimated activities for the repositories. A second objective is to examine the possibility of some particular SMA waste arising with a high specific safety relevant radionuclide content that may undermine the safety case (Note: such waste could always be reclassified as LMA waste).

5.5 Analysis of waste inventory uncertainties

A) Waste production scenario assumed

A-1) 120 GW(e)·years NPP programme

The present inventory considers only the existing NPP's and their projected 40 years of operation. Operation of a small advanced new NPP could be envisaged from 2010 on. Waste arising from the smaller NPP would occur during the period of waste production defined by the present NPP scenario. Decommissioning waste from the smaller NPP would arise much later.

- Assumption made: all wastes occurring from a 0.5 GW(e) NPP, i.e. 15 % more BA, RA, SA and WA, low probability.

Of a higher probability is the extension of the assumed 40 years of operation of the NPP's (of which the first case could be KKB in 2010).

- Assumption made: 25 % more BA, RA and WA (SA more activated), high probability.

A-2) Full spent fuel reprocessing

The possible deviation from the assumption of reprocessing all the spent fuel discharged from the NPP's is not included in the uncertainties considered in this section. The waste volume inventory for reprocessing only the spent fuel already under contract (~ 1/3) is given in Appendix E and compared with the current inventory.

A-3) MIF waste arising

The arisings of MIF waste have been considered until 2053, which is 20 years after the assumed end of the production of NPP wastes.

- Assumption made: no additional waste considered but the MIF activity distribution could be modified in the future.

B) Development of nuclear power technology

The power capacity of the NPP's will probably be raised (already carried out for KKM and KKG). The fuel burn-up will also be increased. MOX fuel can be loaded (already carried out in KKB). However, improvement in water chemistry and component replacement to reduce the worker doses will slightly decrease the waste activity and volume. A first estimate of the consequences will be an increase of the waste volume of BA, RA, WA wastes, proportional to the raised power and an increase in the SA waste activation. The relative concentration of the higher atomic number actinides will be increased because of the increased burn-up. The actinide α-activity in MOX fuel will be about 3 times higher than in UO_2 fuel.

- Assumption made: 10 % more BA, RA and WA wastes (SA more activated), a few % more α-emitters (non-linear with increase in burn-up), high probability.

C) Changes occurring in medical, industrial and research activities

Such changes will affect the MIF waste whose inventory was discussed in section 4.7 but further, will lead to the decommissioning of installations, parts of facilities, pilot plants, manufacturing systems, etc.. It is almost impossible to predict in the long-term the operational life of MIF installations, the technical changes in this field and the associated decommissioning of these installations.

One example that is known of is the addition of an intense neutron source to the PSI accelerator facility (see section 4.5.3) with waste arising from the associated modification of the last part of the proton beam set-up. As a further example, the PSI Hot Laboratory hot cells will sooner or later be decommissioned. It should also be noted that the waste producers have the possibility to store special waste items to allow for their cooling or an improved conditioning technique or the issue of the definitive repository waste acceptance criteria: such wastes which will only be declared in the future are hard to assess. Nevertheless MIF decommissioning wastes are to be expected and provision for their waste volume considered. Such wastes will be designated as SA-MIF and exclude the research reactor decommissioning wastes discussed later.

- Assumption made: 5,000 m^3 of additional MIF decommissioning waste (SA-MIF) excluding the research reactors, high probability.

D) Improvement of waste arising, treatment and conditioning processes

There is a constant worldwide effort to reduce the volumes of waste occurring in the nuclear fuel cycle. This trend is already observable in the NPP operational waste (BA) which were the first wastes produced. The estimate of the BA waste volume of the 5 existing NPP's made in 1984 was 14,500 m^3. This has been revised in the current inventory to 9,200 m^3.

The next improvement is expected for the reprocessing waste. COGEMA announced in 1993 that they would modify the treatment and conditioning of their waste such that some waste streams will be reduced. The bituminisation of the precipitate and sludge waste (WA-COG-2) will be discontinued as a result of their incorporation into the vitrified HLW (WA-COG-1). Under the assumptions made of full reprocessing and a COGEMA share of 50 %, the 5,370 containers (1,142 m^3) of the bituminised waste inventory will be reduced to about 406 containers (87 m^3). Furthermore, through improved raw waste sorting and conditioning processes, the 2,506 containers (2,990 m^3) of WA-COG-6 (α-bearing technological waste) will be reduced to 1,002 containers (1,196 m^3) and for WA-COG-5 (technological waste) this improvement will entail 200 l drums replacing some of the specified 1.67 m^3 containers. Further for hulls and end-fittings (WA-COG-4), COGEMA is currently assessing a hull compaction technology to replace the cementation process. The pellets produced by compaction will be loaded in a canister identical in shape to a glass canister. This new conditioning technique is intended to be implemented before the end of the decade, producing a final volume of 0.15 m^3/t instead of 0.6 m^3/t. The estimated volume reduction for the LMA WA-COG-4 is 600 m^3. Beside these announced improvements for the LMA reprocessing waste, a reduction of 10 % of the SMA reprocessing waste volume can be estimated with respect to the present specification values.

A reduction of the waste volume can be expected from the treatment of some BA and MIF wastes at the new central conditioning facility (ZWILAG in Würenlingen) which is expected to start operation in 1998. An estimate of the

reduction is not yet feasible. For the MIF waste this reduction could compensate for a small undefined production increase with time mentioned in section 4.7. Both effects are therefore neglected.

For the future wastes RA, SA and PSIW, a volume reduction on the assumed conditioning and packaging is not relevant. The volumes given are the current best estimates whose deviations could be positive or negative.

- Assumptions made: reduction of waste volume for LMA-WA: 3,450 m^3 (with volume and activity reduction for discontinued WA-COG-2) and reduction for SMA-WA: 10 % globally, both reductions with high probability.

E) Waste stored and declassified or recycled

This strategy affects the overall waste volumes but does not significantly affect the total waste activity, as it only concerns declassified or recycled waste with low activity concentrations. The majority of these types of waste can be assumed to be NPP SA waste.

Also to be mentioned, is the use of waste packages as shielding blocks at the PSI accelerator facility. These packages result from decommissioning waste arising from modifications to the set-up. These irradiated waste packages can be considered as part of the global accelerator decommissioning waste.

- Assumptions made: SMA wastes: 5 % reduction of the NPP SA waste (2,500 m^3) with high probability.

F) Waste which might be exchanged with other countries

This concerns the reprocessing waste. The customers and the reprocessors are investigating this possibility on the basis of a waste equivalence principle. One consequence of this investigation may be the receipt of more HLW containers as replacement for the reprocessing residues which are, as yet, not covered by the specifications. A second consequence is that HLW may be exchanged against SMA waste and LMA waste against SMA, as SMA waste are readily disposable in the reprocessor's countries. Such an exchange could reduce significantly the SMA waste volume through reduction of WA-BNF-5 (17,456 m^3) and WA-COG-5 (6,425 m^3). The HLW volume increase (represented in the current inventory by WA-COG-1) could reach 10 %. Waste exchange between the customers themselves can not be estimated at present.

- Assumptions made: 10 % more HAA (WA-COG-1) with high probability; no return of the 23,881 m^3 of the SMA reprocessing waste (WA-COG-5, WA-BNF-5) with low probability.

G) Wastes that are known to occur but as yet are not included in the inventory

The comparison of this inventory with the status of the waste known to have already arisen or due to arise in Switzerland indicates that some wastes have not been specified in this inventory. These wastes may be classified into 4 groups for discussion.

1) Small reactor decommissioning waste

This group includes the decommissioning wastes from the following facilities given with an estimate of their total conditioned waste volume:

Pilot power reactor LUCENS (shut-down 1969): 200 m^3 (the 1984 estimate of 1,200 m^3 has been reduced to only the waste volume that requires removal and disposal after the decision was made not to re-use the reactor caverns).
Research reactor DIORIT (shut-down 1983): 500 m^3 (the 1984 estimate of 1,500 m^3 has been revised to 500 m^3 in 1994 by PSI following conditioning studies).
Research reactor SAPHIR (in operation): 800 m^3

Further, minor research facilities, i.e. PROTEUS and CROCUS (critical reactors), LOTUS (under critical reactor) and the swimming pool reactor Type AGN-211-P, are estimated to give rise to 500 m^3 of conditioned waste.

About 2,000 m^3 of reactor decommissioning waste is therefore to be added to the NPP decommissioning waste (~ 43,000 m^3), i.e. ~ 5 % (earlier estimate of 1984: ~ 10 %). As a first estimate the materials and the activities could be assumed to be the same as the NPP decommissioning waste (conservative for the activities). The main exception is graphite which was used as reflector in DIORIT (38 t, 4 ppm Cl impurities). Estimates of the activities of the two important long-lived safety relevant radionuclides C-14 and Cl-36 in this graphite give ~ 4 10^{10} Bq of C-14, i.e. ~ 2 10^{-4} of the C-14 in the SMA inventory, and ~ 4 10^{9} Bq of Cl-36, i.e. ~ 1 % of the Cl-36 in the SMA repository.

- Assumption made: 10 % more NPP SA (corresponding to the earlier estimate), high probability.

2) CERN accelerator decommissioning waste

As mentioned in section 4.5.1 some CERN decommissioning wastes could be expected to be disposed of in Switzerland. There are too many unknowns concerning the decommissioning schedule and the waste management to provide a well founded estimate of the waste quantities. However, the material and activity features from the CERN accelerator facilities should be approximately the same as those of the PSI accelerator waste.

- If an assumption has to be made for the CERN decommissioning waste which may be expected, it will be modeled as 100 % of SA-PSIW with a low probability.

3) Specific existing waste

Some particular raw wastes have already arisen and in some cases have already been conditioned. These are included implicitly within the existing waste sorts defined within the model inventory, but as they could be queried by people managing these wastes, they are mentioned here explicitly but not exhaustively: "KKB-Kies" (gravel waste), "FIX-Box" (research fuel fabrication solution), small metallic pieces from the KKM reactor core, KKB replaced steam generators, etc..

The NPP reactor neutron sources are not included in this inventory (see section 4.3.2). These wastes cannot be considered as being included implicitly in the model inventory because of their particular characteristics which are reported and discussed in Appendix C.

4) Technological waste from central conditioning and storage facilities

A central storage facility for MIF waste (Bundeszwischenlager, Würenlingen) has been in operation for a few years. A central conditioning and storage facility (zentrales Zwischenlager, Würenlingen) is expected to come into operation in 1998. These facilities will produce some operational technological wastes. These wastes are not included specifically in the model inventory. Activities, materials and volume can be neglected from a quantitative point of view with respect to other uncertainties.

H) Changes in regulations

The trend of the applicable regulations is to become more restrictive with regard to radiological protection. The Swiss radiological protection regulations are now being fully revised and will be implemented in the near future. The criteria defining what materials under what circumstances are classified as radioactive waste, and hence subject to disposal in a repository, will be modified to meet two objectives: decreasing the general release of radioactive materials, and hence increasing the volume of waste for disposal, whilst allowing the re-use or recycling of radioactive materials under well defined safety conditions. Without definitive regulations or specific investigations into the declassification and the re-use of materials, it is not currently possible to make any quantitative estimate of the resulting increase in waste volumes. This increase mainly concerns the decommissioning materials for which the activity concentrations may be as low as background levels.

For the declassification of γ-emitting materials as wastes, the current, relatively high, nuclide concentration limits are in practice reduced by additional limits on the dose rates. For contaminated materials, it would be advantageous to decon-

taminate high volume materials if the forthcoming limits are significantly lower. The decommissioning waste volume of 57,000 m^3 (SA-KKW, SA-PSIW) is not expected to increase significantly.

- Assumption made: no quantification of the slight increase in waste volume that could be expected; however, it is estimated that this uncertainty is covered by the overall waste volume reserve given for the SMA waste in section 5.6.1.

I) Improved characterisation of the waste

The underestimation of some safety relevant radionuclides and of some materials such as organics, complexing agents, etc. is of major concern. However, for effective disposal, an appropriate characterisation of the waste must be carried out to allow for the legally required specification and acceptance into the repository.

A further source of uncertainty, inherent in a model inventory, is the possibility of overlooking some small volume waste components having a high concentration of safety relevant or unusual radionuclides. These wastes could include any unusual materials that have been highly activated by neutrons in reactors or by protons and neutrons in accelerators. Such examples of wastes have already been reported for the target graphite, the target Bi-Pb, and the PVC cables of the PSI accelerator (section 4.5.5) as well for the graphite reflector of the DIORIT reactor (point G of this section). A further case is the planned preliminary target (lead in zircaloy cladding) for the testing of the PSI SINQ neutron source.

As far as the waste volume is concerned it can be stated, from experience gained, that the waste projections given in this type of inventory are generally slightly underestimated as compared to reality, due to the inherent difficulty of identifying every occurrence of every waste. Therefore, an overall uncertainty of 10 % is assumed on the waste volume of this inventory, except for the HAA waste.

- Assumption made: + 10 % waste volume on all waste sorts except WA-COG-1, with high probability.

5.6 Repository inventory uncertainties

Even though only about 5 % of the total volume of the waste described by this model inventory has arisen so far, a tentative estimate of the repository inventory uncertainties has to be made. These uncertainties indicate the reserves modeled within this inventory and should be borne in mind when using this inventory for other projects.

The uncertainties assumed in section 5.5 have been considered as either high or low probability. The propagation of uncertainties is not considered (for instance the effect of raising the NPP power on the possible extension of their operation over 40 years). The resulting waste volume uncertainties are given for the different waste categories BA, RA, etc. to provide a further indication of the waste sorts involved.

5.6.1 SMA waste

The estimated SMA waste volumes with high and low probability uncertainties are given without rounding below:

	SMA inventory (m^3)	High probability uncertainties (m^3)	Low probability uncertainties (m^3)
BA	9238	4157	1386
RA	2371	1067	356
SA-KKW	42948	6090[2)]	6422
SA-PSIW	14147	1415	14147[3)]
WA	23881	8360	3583 + (-23881)[4)]
MIF	8700	872	
SA-MIF[1)]		5000	
Total	101285	26959	25894 + (-23881)[4)]

1) MIF decommissioning waste (section 5.5 (C) excluding the research reactors (section 5.5 (G))
2) including the research reactors
3) CERN decommissioning waste (section 5.5 (G))
4) SMA reprocessing waste exchanged against HAA waste (section 5.5 (F) and text)

The subtraction of a very large uncertainty with a low probability (i.e. -23,881 m^3 WA) in the uncertainty summation does not make sense for defining a waste volume reserve for the repository and is therefore given separately.

These results can be expressed as follows: a waste volume reserve of 27,000 m^3 (rounded) required for high probability uncertainties and another of 26,000 m^3 (rounded) required for low probability uncertainties are to be taken into account for the SMA waste in addition to the volume of 101,000 m^3 (rounded) given by the model inventory. In summary: **the waste volume including reserve estimated for the SMA repository is 150,000 m^3** (see also (NAGRA 1992)).

The safety relevant radionuclides are these that in the safety analyses turn out to be the closest to the regulatory limits. Uncertainties on the safety relevant nuclide activities are therefore important. Depending on the results of the safety analyses based on this inventory, further specific investigations on nuclide acti-

vities could be made and/or specific activity determinations required for the specified waste before acceptance into the repository. The uncertainties on the activities of the non-safety relevant nuclides are negligible in comparison with the safety margins shown in the safety analyses which are of many orders of magnitude. However, in order to provide an indication of the uncertainties on the activities, the following statement can be made:

> A reserve of one order of magnitude is estimated for the radionuclide activities of the SMA waste. Additional reserves of 2 to 3 orders of magnitude are to be considered for a few nuclides arising in the individual waste sorts due to difficulties in the characterisation (correlation factors for BA waste and crud in SA-KKW and RA, applicability of computer codes for unusual waste, etc.).

A comparison of the SMA total radionuclide activities has been made with the 50 activities of the comparable Swedish waste inventories (SKBF/KBS 1981, SKB 1993, SKB 1994). The agreement lies within about plus or minus one order of magnitude for most nuclides. A notable exception to this is the SMA Cl-36 activity which is about 4 orders of magnitude higher in the Swiss inventory. From Table 2 of section 5.3, it can be seen that Cl-36 ($t_{1/2}$ = 3 10^5 y) arises in all SMA waste categories with SA-KKW (8.5 10^{11} Bq) and RA (2.6 10^{11} Bq) dominating. However, the SMA SA-KKW + RA value (1.1 10^{12} Bq) is confirmed (within a factor of 5) by a Finnish estimate (YJT 1993) for the equivalent waste (taking into account the difference in the NPP power ratings). The comparison with the other 12 Finnish activities gives the same agreement as for the Swedish activities. A third comparison with the 70 activities of the United Kingdom (UK) inventory for BA, RA and MIF (VAN DE PUTTE 1994) shows also the same relative agreement including Cl-36.

Finally the first 2 comparisons have shown that activities specific to concrete activation (Ar-39, Ca-41, etc.) are relatively high in the Swiss SMA inventory due to a higher activation of the 2 KKB biological shields (SA-KKW) and the accelerator facility shielding concrete (SA-PSIW).

It should be noted that reserves for the SMA radionuclide activities cannot be estimated with confidence as it cannot be excluded that future arisings of unknown wastes could increase some activities by many orders of magnitude. But in such cases, these wastes could always be classified as LMA or HAA waste.

The SMA waste material distribution can be considered as representative for wastes to be disposed of in the future. Some specific materials that can affect the repository safety (cellulose, Zn, etc.) are expected to be partially reduced as a result of counter measures that will be taken during the waste arising and conditioning processes. Relatively low limits are set for their declaration in the specifications.

5.6.2 LMA waste

The combination of the high probability uncertainties gives a waste volume reduction of about 500 m^3 but the extent of this reduction is very uncertain. As a confirmation COGEMA has just announced the possibility for substitution of the SMA WA-COG-5 waste against LMA WA-COG-6 waste (both technological waste). This may reverse the reduction of about 500 m^3 to an increase of about this value. The reserves with a low probability amount to about 1,000 m^3. The reprocessing wastes, constituting the LMA inventory, have recently started to be conditioned by COGEMA with modifications to the specifications issued a few years ago (see section 5.5 (D)). Announcements of the reduction in waste volumes will probably occur after further operational experience.

The main waste volume reserve to be added is related to the waste allocation uncertainties. The present allocation made for the waste sorts can be considered to be adequate according to our current understanding. However, the actual future activity limits of the SMA waste derived from the safety analyses of the repository can be implemented in different ways and could affect the waste allocation to the LMA repository. There will be many more waste package types than currently specified waste sorts and further to this there will be many individually specified waste packages. Waste packages exceeding defined limits can be excluded, or by compensation with other less active packages accepted; on the other hand, waste packages of one type can be excluded globally to avoid any problems arising from a few individual waste packages (waste packages that deviate too far from average values).

In addition it may be possible that some highly radioactive or radiotoxic, but of small volume, waste components covered by the SMA waste sorts will be allocated to the LMA repository.

The LMA repository, for which the current estimate waste volume is 6,500 m^3 (section 5.3), will not be constructed in the near future. For preliminary studies a tentative estimate including reserves of a total of 15,000 m^3 (see also NAGRA 1992) should be considered. The LMA waste activities based on reprocessing specifications are relatively accurate. Other waste excluded from the SMA repository should not in principle modify to any great extent the radionuclide inventories.

5.6.3 HAA waste

With the assumption of all spent fuel being reprocessed a reserve of 1,211 vitrified residue containers should be taken into account in addition to the present inventory of 2,693 containers which is equivalent to 485 m^3 (section 5.3). This reserve takes account of the possible extension of NPP operational life and increase in power as well as the possible exchange of HAA waste against other wastes (see section 5.5 A, B and F). A low probability additional reserve of 464 containers should also be further considered (additional 0.5 MW(e) NPP, see

section 5.5 A). Note: containers with very low waste loading arising from maintenance operations could be returned by the reprocessors and would represent a few percent of the total number of containers.

The specified high level wastes are very well characterised with respect to activities, materials etc. in comparison to other wastes.

The specified COGEMA vitrified residues based on PWR fuel have been selected for this inventory (WA-COG-1) as a representative HLW waste and this selection is in agreement with the objectives of the other projects using this inventory. PWR fuel shows slightly more conservative characteristics than BWR fuel.

The burn-up of the spent fuel before discharge is increasing. The resulting increase of the long-lived actinide activities is non-proportionally larger than for the fission products. This effect can be seen in Figure 3 (see section 4.6.3) where the difference between the activity from PWR fuels and the activity from the comparatively low burn-up Magnox fuel is seen to be increasing with time due to a higher actinide content of the PWR fuels. The average burn-up of all the spent fuel is expected to be higher than the reference burn-up used in the WA-COG-1 characterisation. Therefore, it should be noted, for sake of completeness, that the HAA waste inventory underestimate slightly the activities of the actinides and their daughters. These activities could further be increased with the use of MOX fuel.

6 CONCLUSIONS

Summary of the work

A model radioactive waste inventory (MIRA) has been developed for Swiss waste disposal projects (repository planning, safety assessment).

The inventory describes the conditioned and packaged wastes that are expected to be produced by the 5 operational Swiss NPP's (3 GW(e) over 40 years), with the last NPP being shut-down in 2024. The waste categories assumed are: operational waste, exchangeable non-fuel reactor core components, decommissioning waste and reprocessing waste (with assumed complete spent fuel reprocessing through COGEMA and BNFL; however, the data for partial reprocessing are presented in Appendix E). These wastes will arise up to 2032. The inventory also details the wastes arising from medicine, industry and research between 1984 (previous waste arisings were sea-dumped) and 2053. The decommissioning wastes of the PSI 600 MeV proton (1.5 mA) accelerator facility, assumed to be shut-down in 2034 with waste arising up to 2040, are also included.

The waste arisings are described by 80 waste sorts each defined using an average waste package description linked to the number of packages arising with time. Radionuclide activities (overall 117 nuclides with $t_{1/2}$ > 60 days), other radiological characteristics, material content, specific properties, package internal drawing and possible maximal values of the package are provided in a database of the waste sorts given in Volume 2. This data has been presented and discussed in this Volume, along with a short description of the raw waste origin.

Provisional allocation of the waste sorts to the 2 repository types SMA and LMA/HAA has been made, thus defining 3 waste sort classifications: SMA (short-lived LLW and ILW), LMA (long-lived ILW) and HAA (HLW). The summation of the waste volumes, activities, toxicities and materials gives an overview of the repository waste contents. Uncertainties associated with the repository inventories have been analysed. The results provide an indication of the waste volume reserves to be considered for the repositories and of the accuracy of the activities for the safety analyses.

Model nature of the waste inventory

The reported waste inventory is a model inventory.

The model nature of the inventory is due to:

- projections having to be made for waste arising in the future (only about 5 % of the total waste inventory has already arisen).
- assumptions having to be made for the conditioning and packaging of most wastes that are expected to arise in the future.

- reduction of the radioactive waste diversity by the use of a limited number of waste sorts.

These features do not apply to the real waste inventory which was started a few years ago. All real waste packages that are produced are subject to specifications from the producers. These specifications will be used for the acceptance of the waste into the repository and will define the real waste inventory.

Assessment of the model waste inventory

The characterisation of the waste reflects the current world status in this field and the common effort undertaken. Much progress has been made in the last decade, giving confidence in the results of the characterisation, especially through international comparison. The general trend is to go from preliminary characterisations made for waste management strategies to more detailed characterisations that are sufficient for disposal projects and finally to more precise characterisation of individual wastes made for the specification and acceptance into the repositories. This development is not uniform and depends on which waste types arise first or for which waste types a repository is currently or soon to be available. The last step may require some improvement in the characterisation tools (correlation factors, computer codes, measurement techniques, etc.).

The projection and estimation of waste arisings is different for each country. This estimate should reflect the uncertainties of the future, i.e. atomic and nuclear developments, and the decisions of the producers affecting the waste production (reprocessing, conditioning, substitutions, etc.).

Use of the model waste inventory

The data of this model inventory have been used in the Swiss disposal projects. However, some data could differ between these projects and this current inventory as these data were provided over the last 2 years and in some cases have been updated or improved. Some other data have been modified within these projects as constraints may be put on waste packages that have not yet arisen to improve disposal operations. Specific inventories have also been produced from the inventory database for practical use within these projects (grouping of 200 l drums, SMA sub-inventories for different repository caverns, etc.). Modified data and inventories specific to projects are documented in the reports of these projects.

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Appendix A: 1) List of waste sorts with main features
2) Summary of waste sort properties

1) List of waste sorts with main features

Waste sort	Type of raw waste	Material of conditioning	Volume of container [m^3]	Containers	Total volume of waste [m^3]	Class of waste
BA (NPP routine operational waste)						
BA-KKB-1	Resins	Polystyr.-DVB	0.1	3732	378	SMA
BA-KKB-2	Sludges/concentrates	Cement	0.2	970	214	SMA
BA-KKB-3	Filtercartridges	Cement	0.2	324	71	SMA
BA-KKB-5	Solid waste	Cement	0.2	3000	650	SMA
BA--KKB-6	Incinerated waste	Cement	0.2	440	96	SMA
BA-KKM-1	Resins	Cement	0.2	5300	1148	SMA
BA-KKM-2	Sludges/concentrates	Cement	0.2	300	66	SMA
BA-KKM-5	Solid waste	Cement	0.2	1200	260	SMA
BA-KKM-6	Incinerated waste	Cement	0.2	240	52	SMA
BA-KKG-1	Resins	Cement	0.2	800	171	SMA
BA-KKG-2	Sludges/concentrates	Bitumen	0.2	2400	513	SMA
BA-KKG-3	Filtercartridges	Cement	0.2	160	34	SMA
BA-KKG-5	Solid waste	Cement	0.2	1280	277	SMA
BA-KKG-6	Incinerated waste	Cement	0.2	300	66	SMA
BA-KKL-1	Resins (+sludges/concentrates)	Cement	0.2	18800	4061	SMA
BA-KKL-5	Solid waste	Cement	0.2	4200	910	SMA
BA-KKL-6	Incinerated waste	Cement	0.2	320	70	SMA
BA-KKW-A	Undefined in B-cont.	--	0.98	205	201	SMA
RA (NPP exchangeable non-fuel reactor core components)						
RA-KKB-1A	Control rods	Cement	18.9	2	38	SMA
RA-KKB-1B	Trim rods	--	0.7	2	1	SMA
RA-KKB-4	Absorber elements	--	0.7	13	9	SMA
RA-KKB-5	Flow restriction rods	--	0.7	8	6	SMA
RA-KKB-7	Incore instrumentation	Cement	0.2	120	26	SMA
RA-KKM-1A	Control rods	Cement	20.2	20	403	SMA
RA-KKM-4	Absorber elements	Cement	0.98	48	47	SMA
RA-KKM-6	Fuel channels	Cement	0.2	330	71	SMA
RA-KKM-7	Incore instrumentation	Cement	0.2	160	35	SMA
RA-KKG-1A	Control rods	Cement	18.9	4	76	SMA
RA-KKG-4	Absorber elements	--	1.32	5	7	SMA
RA-KKG-5	Flow restriction rods	--	0.7	15	11	SMA
RA-KKG-7	Incore instrumentation	Cement	0.2	48	10	SMA
RA-KKL-1A	Control rods	Cement	18.9	68	1286	SMA
RA-KKL-6	Fuel channels	Cement	0.2	1200	259	SMA
RA-KKL-7	Incore instrumentation	Cement	0.2	400	86	SMA
SA-KKW (NPP decomissioning waste)						
SA-KKB-1	Middle active waste	Shot mortar	18.9	124	2346	SMA
SA-KKB-2..6	Low active waste	Cement	18.9	812	15355	SMA
SA-KKB-7	Low active waste with organics	Cement	18.9	66	1248	SMA
SA-KKB-8	Decontamination waste	Cement	18.9	22	416	SMA
SA-KKM-1	Middle active waste	Shot mortar	18.9	47	888	SMA
SA-KKM-3..6	Low active waste	Cement	18.9	123	2326	SMA
SA-KKM-7	Low active waste with organics	Cement	18.9	20	379	SMA
SA-KKM-8	Decontamination waste	Shot mortar	18.9	55	1040	SMA

1) List of waste sorts with main features: continued

Waste sort	Type of raw waste	Material of conditioning	Volume of container [m^3]	Containers	Total volume of waste [m^3]	Class of waste
SA-KKG-1	Middle active waste	Shot mortar	18.9	30	568	SMA
SA-KKG-3..6	Low active waste	Cement	18.9	232	4387	SMA
SA-KKG-7	Low active waste with organics	Cement	18.9	38	719	SMA
SA-KKG-8	Decontamination waste	Shot mortar	18.9	56	1059	SMA
SA-KKL-1	Middle active waste	Shot mortar	18.9	52	984	SMA
SA-KKL-3..6	Low active waste	Cement	18.9	441	8339	SMA
SA-KKL-7	Low active waste with organics	Cement	18.9	51	965	SMA
SA-KKL-8	Decontamination waste	Cement	18.9	102	1929	SMA
SA-PSIW (PSI accelerator facility decommissioning waste)						
SA-PSIW-1A	Copper (Grade A activity)	Cement	18.9	4	76	SMA
SA-PSIW-1B	Copper (Grade B activity)	Cement	18.9	12	227	SMA
SA-PSIW-1C	Copper (Grade C activity)	Cement	18.9	4	76	SMA
SA-PSIW-2A	St. Steel (Grade A activity)	Cement	18.9	4	76	SMA
SA-PSIW-2B	St. Steel (Grade B activity)	Cement	18.9	2	38	SMA
SA-PSIW-3A	Aluminium (Grade A activity)	Cement	18.9	3	57	SMA
SA-PSIW-3B	Aluminium (Grade B activity)	Cement	18.9	13	246	SMA
SA-PSIW-4A	Steel (Grade A activity)	Cement	18.9	11	208	SMA
SA-PSIW-4B	Steel (Grade B activity)	Cement	18.9	90	1702	SMA
SA-PSIW-4C	Steel (Grade C activity)	Cement	18.9	180	3404	SMA
SA-PSIW-5	Concrete	Cement	18.9	290	5484	SMA
SA-PSIW-6A	Iron shot mortar	Cement	18.9	5	95	SMA
SA-PSIW-6B	Iron shot mortar	Cement	18.9	130	2458	SMA
WA (Reprocessing waste)						
WA-COG-1	Vitrified residues (HLW)	Glass	0.18	2693	485	**HAA**
WA-COG-2	Sludges/concentrates	Bitumen	0.2	5370	1142	**LMA**
WA-COG-4	Hulls and ends	Cement	1.5	689	1030	**LMA**
WA-COG-5	Low level tech. waste	--	0.66	9666	6425	SMA
WA-COG-6	Alpha tech. waste	--	1.2	2506	2990	**LMA**
WA-BNF-2	MEB crud/$BaCO_3$ slurry	Cement	0.5	158	90	**LMA**
WA-BNF-4	Hulls and ends	Cement	0.5	1324	758	**LMA**
WA-BNF-5	Solid low level waste	--	1.2	14768	17456	SMA
WA-BNF-7	Centrifuge slurry	Cement	0.5	802	459	**LMA**
MIF (Waste from medicine, industry and research)						
MIF-1	Beta,gamma-waste	Cement	0.2	14000	3032	SMA
MIF-2	Tritium-bearing waste	Cement	0.2	2800	606	SMA
MIF-3A	Alpha-waste (Research)	Cement	0.2	5600	1213	SMA
MIF-3B	Alpha-waste (Industry)	Cement	0.2	3150	682	SMA
MIF-4	Radium-bearing waste	Cement	0.2	1050	227	SMA
MIF-5A	Alpha,beta-gamma-waste (Research)	Cement	0.2	7700	1668	SMA
MIF-5B	Alpha,beta,gamma-waste (Industry)	Cement	0.2	700	152	SMA
MIF-6	Undefined in 16 m^3-cont.	--	16.0	70	1120	SMA

2) Summary of waste sort main properties

- The "same" waste sorts from the NPP's are considered together (for instance: BA-1 = BA-KKB-1 + BA-KKM-1 + BA-KKG-1 + BA-KKL-1). The "same" waste sorts from the reprocessors are also considered together.
- The range of the average values of the waste packages are given with, in brackets, for indication, the maximal values.

Waste sort	Raw waste	Material of conditioning	Alpha activity (Bq/container)	Beta-gamma activity (Bq/container)	Dose rate at 0 m (Sv/h)	Heat output (W)
BA (NPP routine operational waste)						
BA-1	Ion exchange resins	Bitumen Cement Polymer	1.0E5-1.5E6 (7.2E6)	5.8E10-1.4E12 (7.2E12)	2.9E-2-9.3E-1 (4.7E0)	1.0E-2-2.7E-1 (1.4E0)
BA-2	Sludges and concentrates	Cement Bitumen	9.7E3-1.2E4 (9.7E5)	1.7E10 (1.7E12)	9.4E-3-1.3E-2 (9.8E-1)	3.2E-3 (3.2E-1)
BA-3	Filter cartridges	Cement	1.7E9-3.2E9 (3.2E10)	2.0E11-3.6E11 (3.6E12)	9.6E-2-2.2E-1 (2.0E0)	4.2E-2-7.7E-2 (7.7E-1)
BA-5	Solid waste	Cement	7.0E2 (7.0E4)	8.6E8 (8.6E+10)	3.4.E-4 (3.4E-2)	1.7E-4 (1.7E-2)
BA-6	Incinerated waste	Cement	6.7E4-9.1E4 (3.0E+6)	2.4E10 (8.2E11)	6E-3 (2.0E-1)	3.9E-3 (1.3E-1)
RA (NPP exchangeable non-fuel reactor core components)						
RA-1A	Control assemblies	Cement	5.7E7-2.5E8 (4.4E8)	4.1E14-1.1E15 (1.5E15)	1.3E-4-2.5E-4 (3.1E-4)	7.4E1-3.8E2 (5.0E2)
RA-1B	Part length control assemblies	None	5.0E7 (5.0E7)	2.5E14 (2.5E14)	5.6E-4 (5.6E-4)	4.9E1 (4.9E1)
RA 4	n-absorbing elements	None/Cement	5.5E6-3.4E7 (4.3E7)	6.0E12-1.0E14 (1.3E14)	2.7E-4-3.2E-4 (5.9E--4)	8.8E-1-1.6E1 (2.0E1)
RA-5	Flow restriction rods	None	1.1E8-1.4E8 (2.0E8)	2.4E14-2.6E14 (4.1E14)	3.2E-4-6.7E-4 (6.7E-4)	4.2E1 (7.1E1)
RA-6	Fuel channels (BWR)	Cement	2.9E9-3.8E9 (2.3E10)	8.9E12-1.2E13 (7.0E13)	1.2E0-1.6E0 (9.4E0)	6.4E-1-8.3E-1 (5.0E0)
RA-7	Incore instrumentation	Cement	2.7E6-4.1E6 (4.1E7)	3.9E12-1.3E13 (1.3E14)	6.2E-1-6.9E-1 (6.9E0)	6.3E-1-2.9E0 (2.9E1)
SA-KKW (NPP decommissioning waste)						
SA-1	Core internals and reactor vessel parts	Cement	3.4E7-1.4E9 (2.5E9)	8.8E14-1.8E16 (3.2E16)	3.5E-5-1.1E-4 (1.9E-4)	7.7E1-1.8E3 (3.3E3)
SA-2...6	Biological shield (2+4), reactor vessel parts (3), prim. and recirc. circuits (5), piping (6)	Cement	4.4E6-3.4E7 (1.7E9)	2.5E10-2.2E12 (2.0E13)	3.7E-7-1.5E-5 (1.4E-4)	2.4E-3-1.2E-1 (1.1E0)
SA-7	Other primary (materials) and secondary (tools, etc.) wastes	Cement	7.0E7-8.7E7 (4.5E8)	3.5E10-4.2E10 (2.1E11)	2.2E-5-3.0E-5 (1.4E-4)	2.9E-3-3.6E-3 (1.9E-2)
SA-8	Decontamination residues	Cement	8.8E8-6.9E10 (4.4E12)	4.4E11-3.4E13 (2.1E15)	2.1E-6-9.7E-5 (1.3E-4)	3.7E-2-2.8E0 (1.8E2)

2) Summary of waste sort main properties: continued

Waste sort	Raw waste	Material of conditioning	Alpha activity (Bq/container)	Beta-gamma activity (Bq/container)	Dose rate at 0 m (Sv/h)	Heat output (W)
SA-PSIW (PSI accelerator facility decommissioning waste)						
SA-1	Copper	Cement	0.0	1.3E11-9.9E14 (4.5E15)	1.3E-5-8.6E-5 (1.4E-4)	1.6E-2-1.2E2 (5.6E2)
SA-2	Stainless steel	Cement	0.0	1.9E12-4.2E14 (6.1E14)	6.8E-5-8.2E-5 (1.4E-4)	2.0E-2-4.2E0 (6.2E0)
SA-3	Aluminium	Cement	0.0	5.9E10-2.7E13 (3.0E14)	1.1E-5-4.8E-5 (1.4E-4)	6.1E-3-2.9E0 (3.1E1)
SA-4	Steel	Cement	0.0	1.2E11-7.8E14 (1.2E15)	4.1E-6-4.1E-4 (4.1E-4)	1.1E-3-5.7E0 (8.5E0)
SA-5	Concrete	Cement	0.0	1.1E12 (1.1E12)	3.4E-4 (3.4E-4)	4.5E-2 (4.5E-2)
SA-6	Iron shot mortar	Cement	0.0	1.3E13-2.7E14 (2.7E14)	1.3E-4-4.1E-4 (4.1E-4)	1.2E-1-2.3E0 (2.3E0)
WA (Reprocessing waste)						
WA-1	High level vitrified residues	Glass	1.1E14 (2.5E14)	2.8E16 (3.8E16)	3.4E3 (5.1E3)	2.8E3 (4.1E3)
WA-2	Precipitates and sludges (COGE-MA), $BaCO_3$ and MEB crud (BNFL)	Bitumen Cement	3.6E8-1.8E10 (3.2E10)	3.5E11-4.3E12 (1.2E13)	1.2E-1-5.2E-1 (4.0E0)	8.2E-2-4.5E-1 (2.8E0)
WA-4	Hulls and ends	Cement	1.4E11-2.8E11 (2.7E12)	1.4E14-8.2E14 (4.7E15)	2.4E1-3.9E1 (7.5E1)	2.0E1-1.2E2 (1.0E3)
WA-5	Technological lowlevel waste	Compacted	2.6E6-1.2E8 (1.2E10)	8.0E7-3.8E9 (1.3E12)	2.1E-5-1.0E-4 (2.0E-3)	4.5E-6-2.1E-4 (1.2E-1)
WA-6	α-emitting technological waste(COGEMA)	Cement	7.4E10 (6.4E11)	7.3E11 (3.1E12)	7.0E-3 (4.0E-2)	8.0E-2 (6.4E-1)
WA-7	Centrifuge cake slurry (BNFL)	Cement	3.8E11 (6.5E11)	4.0E13 (4.4E14)	2.0E0 (1.0E1)	3.5E0 (5.5E1)
MIF (Waste from medicine, industry and research)						
MIF-1	Beta, gamma emitting waste	Cement	0.0 (-)	3.7E10 (3.7E12)	9.5E-3 (9.5E-1)	3.1E-3 (3.1E-1)
MIF-2	Tritium-bearing waste	Cement	0.0 (-)	1.2E13 (2.4E14)	3.0E-5 (6.0E-4)	1.1E-2 (2.2E-1)
MIF-3	"Alpha" waste	Cement	1.2E8-1.3E10 (1.3E11)	0.0-1.1E9 (1.1E10)	3.0E-9-2.5E-6 (2.5E-5)	1.0E-4-1.2E-2 (1.2E-1)
MIF-4	Radium-bearing waste	Cement	2.1E9 (2.1E10)	0.0 (-)	5.4E-6 (5.4E-5)	1.6E-3 (1.6E-2)
MIF-5	"Alpha, beta, gamma" waste	Cement	7.6E8-1.9E9 (1.7E10)	5.9E10-1.7E11 (1.7E12)	1.7E-2-3.2E-2 (3.2E-1)	7.7E-3-1.5E-2 (1.5E-1)

APPENDIX B: Example of a waste sort database formular set

Nagra-MIRA : M-F1	Datum: 01.09.1994
Formular Nr. : J_R_000259	Seite: 1
	3.62A

Abfallsorte : WA-COG-5
Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE Zusatzstoff : - Behälterhülle : 0.67m3-FASS

Repräsentatives Abfallgebinde

Geometrieart	Länge [m]	Breite [m]	Höhe [m]	Durchm. [m]
Zylinder	——	——	1.2000	0.8400

Masse[kg]	Volumen[m3]	Dichte[kg/m3]
1.199E+03	6.647E-01	1.80E+03

Referenz-Zeit [a] zw.Entlad. und Kondit.
3.0

Aktivitäten [Bq]	Alpha	Beta/Gamma	Total
	1.2E+08	3.8E+09	3.9E+09

Dosisleistung [Sv/h]	Gamma
Abstand 0m	1.000E-04
Abstand 1m	1.000E-05
Abstand 2m	k.A.
Abstand 3m	k.A.

Wärmeleistung [W] : 2.125E-04

Oberflächenkontamination [Bq/cm²]	Alpha	Beta/Gamma
	3.700E-01	3.700E+00

Nagra-MIRA : M-F1 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 2
3.62A

Produktions-Statistik

Szenario:

Intervall [a] von	bis	Produktion [Stk] pro Jahr	Total	kumm.
1994	2033	241.65	9666	9666

Tot. Anzahl	Tot.Masse[kg]	Tot.Volumen[m3]
9666	1.16E+07	6425

Nagra-MIRA : M-F2	Datum: 01.09.1994
Formular Nr. : J_R_000259	Seite: 1
	3.62A

Abfallsorte : WA-COG-5
Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE Zusatzstoff : - Behälterhülle : 0.67m3-FASS

Maximalwerte der Abfallgebinde

Masse [kg]	1.800E+03

Aktivitäten [Bq]	Alpha	Beta/Gamma	Total
	5.7E+09	1.3E+12	1.3E+12

Dosisleistung [Sv/h] geschätzt	Gamma
Abstand 0m [Sv/h]	k.A.
Abstand 1m	k.A.
Abstand 2m	k.A.
Abstand 3m	k.A.

Dosisleistung [Sv/h] spezifiziert	Gamma
Abstand 0m	2.000E-03
Abstand 1m	1.000E-04
Abstand 2m	k.A.
Abstand 3m	k.A.

Wärmeleistung [W]	1.182E-01

Oberflächenkontamination [Bq/cm²]	Alpha	Beta/Gamma
	3.700E-01	3.700E+00

Nagra-MIRA : M-F3 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 1
3.62A

Abfallsorte : WA-COG-5

Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff : -
Behälterhülle : 0.67m3-FASS

Aktivitäten des repräsentativen Abfallgebindes [Bq]

Referenz-Zeit zw. Entlad. und Kondit. [a]	Cutoff [Bq]
3.0	1.0000E+00

Nuklid	Mittel [Bq]	Maximal [Bq]
H -3	4.4E+05	5.0E+08
C -14	1.2E+01	1.3E+04
Cl-36	2.4E+01	2.7E+04
Mn-54	2.0E+03	2.0E+06
Fe-55	9.8E+04	1.1E+08
Co-57	1.3E+03	1.4E+06
Co-60	2.0E+05	2.3E+08
Ni-59	1.5E+02	1.7E+05
Ni-63	2.0E+04	2.3E+07
Zn-65	4.2E+03	4.7E+06
Se-79	8.4E+02	9.5E+05
Sr-90	1.4E+08	1.5E+11
Y -90	1.4E+08	1.5E+11
Zr-93	1.2E+03	1.4E+06
Nb-93M	2.9E+02	3.3E+05
Nb-95	3.2E+02	3.5E+05
Tc-99	8.9E+03	1.0E+07
Ru-106	2.3E+07	2.6E+10
Rh-106	2.3E+07	2.6E+10
Ag-110M	1.4E+05	1.5E+08
Sn-119M	7.4E+03	8.3E+06
Sn-121M	3.9E+02	4.2E+05
Sn-123	2.7E+03	3.2E+06
Sn-126	1.5E+03	1.8E+06
Sb-125	3.5E+06	3.9E+09
Te-125M	8.4E+05	9.5E+08
I -129	6.5E+01	7.2E+04
Cs-134	8.6E+07	9.5E+10
Cs-135	8.4E+02	9.5E+05
Cs-137	2.1E+08	2.4E+11
Ba-137M	2.0E+08	2.3E+11
Ce-144	6.5E+07	7.2E+10
Pr-144	6.5E+07	7.2E+10
Pm-147	9.8E+07	1.1E+11
Sm-151	7.1E+05	8.0E+08
Eu-152	1.1E+04	1.2E+07
Eu-154	1.4E+07	1.7E+10
Eu-155	7.1E+06	7.8E+09

Nagra-MIRA : M-F3 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 2
3.62A

Abfallsorte : WA-COG-5

Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff : -
Behälterhülle : 0.67m3-FASS

Aktivitäten des repräsentativen Abfallgebindes [Bq]

Nuklid	Mittel [Bq]	Maximal [Bq]
Ho-166M	4.2E+00	4.8E+03
U -232	1.5E+02	6.9E+03
U -234	2.7E+04	1.2E+06
U -235	4.5E+02	2.1E+04
U -236	5.9E+03	2.7E+05
U -238	6.9E+03	3.3E+05
Np-237	6.8E+03	3.2E+05
Np-239	3.8E+05	1.8E+07
Pu-238	5.0E+07	2.4E+09
Pu-239	6.3E+06	3.0E+08
Pu-240	7.7E+06	3.6E+08
Pu-241	2.7E+09	1.2E+11
Pu-242	4.1E+04	2.0E+06
Am-241	2.1E+07	1.0E+09
Am-242M	1.8E+05	8.4E+06
Am-243	3.8E+05	1.8E+07
Cm-242	2.4E+06	1.1E+08
Cm-243	3.8E+05	1.8E+07
Cm-244	3.2E+07	1.5E+09
Cm-245	2.6E+03	1.2E+05
Cm-246	5.0E+02	2.3E+04
Σ Alpha	1.2E+08	5.7E+09
Σ Beta/Gamma	3.8E+09	1.3E+12
Σ Total	3.9E+09	1.3E+12

Nagra-MIRA : M-F4 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 1
3.62A

Abfallsorte : WA-COG-5

Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff : -
Behälterhülle : 0.67m3-FASS

Materialgemisch der Abfallgebindekomponenten

	Material	Mas.[kg]	Vol.[m3]	[kg/m3]
Abfallprodukt		1.75E+02	2.45E-01	7.15E+02
Rohabfall	TECH.SCHW.AKT.ABF.	1.50E+02	2.39E-01	6.29E+02
└──>	ALUMINIUM	1.64E-01	6.06E-05	2.70E+03
└──>	DIVERSE ORGANISCH	1.25E-02	1.25E-05	1.00E+03
└──>	GLAS	3.21E+00	1.34E-03	2.40E+03
└──>	GUMMI	2.27E+01	2.27E-02	1.00E+03
└──>	KUPFER	5.04E-02	5.65E-06	8.93E+03
└──>	LUFT	1.64E-01	1.64E-01	1.00E+00
└──>	POLYETHYLEN	7.60E+00	8.17E-03	9.30E+02
└──>	PVC	3.02E+01	2.51E-02	1.20E+03
└──>	ST STAHL	3.84E-01	4.89E-05	7.85E+03
└──>	STAHL	7.32E+01	9.33E-03	7.85E+03
└──>	TRUEMMER	4.83E+00	1.61E-03	3.00E+03
└──>	ZELLULOSE	7.60E+00	7.80E-03	9.74E+02
└──>	ZINK	5.04E-02	7.07E-06	7.13E+03
Zusatzstoff	STEEL DRUMS	2.50E+01	5.44E-03	4.61E+03
└──>	POLYETHYLEN	2.38E+00	2.56E-03	9.30E+02
└──>	STAHL	2.27E+01	2.89E-03	7.85E+03
Füllmaterial	IRON FIBRE CONCRETE	3.44E+02	1.41E-01	2.44E+03
└──>	EISEN	4.23E+00	5.38E-04	7.86E+03
└──>	KIESSAND COG-5A	1.69E+02	6.74E-02	2.50E+03
└──>	SAND COG-5A	9.15E+01	3.66E-02	2.50E+03
└──>	SIO2 IN HAW-COG	2.82E+00	1.02E-03	2.75E+03
└──>	VERFLUESSIG.POZN400	6.78E-01	2.82E-04	2.40E+03
└──>	WASSER	1.90E+01	1.90E-02	1.00E+03
└──>	ZEMENT CLC 45	5.64E+01	1.79E-02	3.15E+03

Nagra-MIRA : M-F4 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 2
3.62A

Materialgemisch der Abfallgebindekomponenten

	Material	Mas.[kg]	Vol.[m3]	[kg/m3]
Behälterhülle	COG-5	6.80E+02	2.79E-01	2.44E+03
└──>	EISEN	8.36E+00	1.06E-03	7.86E+03
└──>	KIESSAND COG-5A	3.35E+02	1.34E-01	2.50E+03
└──>	SAND COG-5A	1.81E+02	7.24E-02	2.50E+03
└──>	SIO2 IN HAW-COG	5.58E+00	2.03E-03	2.75E+03
└──>	VERFLUESSIG.POZN400	1.34E+00	5.58E-04	2.40E+03
└──>	WASSER	3.76E+01	3.76E-02	1.00E+03
└──>	ZEMENT CLC 45	1.12E+02	3.54E-02	3.15E+03
Abfallgebinde	WA-COG-5	1.20E+03	6.65E-01	1.80E+03

Nagra-MIRA : M-F5 Datum: 01.09.1994
Formular Nr. : J_R_000259 Seite: 1
3.62A

Abfallsorte : WA-COG-5

Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff : -
Behälterhülle : 0.67m3-FASS

Chemisches Inventar

Organisches Material	Masse [kg]	Anteil [-]
Zellulose	7.60E+00	6.34E-03
Kunststoffe	4.02E+01	3.35E-02
Gummi	2.27E+01	1.89E-02
Harze		
Bitumen		
Komplexbildend		
Andere	2.03E+00	1.69E-03
Total	7.25E+01	6.05E-02

Metallisches Material	Masse [kg]	Anteil [-]
Stahl	1.09E+02	9.09E-02
Aluminium	1.64E-01	1.37E-04
Zink	5.04E-02	4.20E-05
Legierungen		
Andere	5.04E-02	4.20E-05
Total	1.09E+02	9.11E-02

Inorganisches Material	Masse [kg]	Anteil [-]
Zement	1.00E+03	8.34E-01
Salz	8.39E+00	7.00E-03
Asche		
Glas	3.21E+00	2.68E-03
Komplexbildend		
Andere	1.64E-01	1.37E-04
Total	1.01E+03	8.44E-01

Nagra-MIRA : M-F6 Datum: 01.09.1994
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Abfallsorte : WA-COG-5

Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff : -
Behälterhülle : 0.67m3-FASS

Eigenschaften der Abfallgebinde

Dosisleistungs-Verteilung Abstand: 0m

[Sv/h]	Anz.Geb.	kumm.
2E-6 bis 2E-5	966.00	966.00
2E-5 bis 2E-4	8603.00	9569.00
2E-4 bis 2E-3	97.00	9666.00

Mittel: 1.000E-04

Abnahme der normalisierten Dosisleistung (Gamma / Abstand 0m)

Ref.Zeit	Anteil [-]	0.0 ... 1.0
0	1.0000E+00	
5	6.5448E-01	
10	4.9682E-01	
20	3.1490E-01	
40	1.3569E-01	
80	3.0354E-02	

Radiolytische Gasproduktion

Ref.Zeit	Prod. [l/a]	Integral [m3]
0	2.7E-02	0.0E+00
5	2.9E-02	1.4E-04
10	3.0E-02	2.8E-04
20	3.2E-02	5.9E-04
40	3.2E-02	1.2E-03
80	3.0E-02	2.5E-03
100	2.8E-02	3.1E-03

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Abfallsorte : WA-COG-5
Rohabfall : TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE Zusatzstoff : - Behälterhülle : 0.67m3-FASS

Eigenschaften der Abfallgebinde

Spaltbare Materialien

Referenzzeit zw.Entlad und Kond. [a]: 3.0

Nuklid	Aktivität mittel [Bq]	[g]	Aktivität maximal [Bq]	[g]
U -233	4.2E-01	1.2E-09	2.0E+01	5.6E-08
U -235	4.5E+02	5.6E-03	2.1E+04	2.6E-01
Pu-238	5.0E+07	7.9E-05	2.4E+09	3.8E-03
Pu-239	6.3E+06	2.7E-03	3.0E+08	1.3E-01
Pu-241	2.7E+09	7.1E-04	1.2E+11	3.1E-02
Am-242M	1.8E+05	4.6E-07	8.4E+06	2.2E-05
Cm-243	3.8E+05	2.0E-07	1.8E+07	9.4E-06
Cm-245	2.6E+03	4.1E-07	1.2E+05	1.9E-05
Total	2.8E+09	9.2E-03	1.2E+11	4.3E-01

Oberflächen/Massen Verhältnis der metallischen Materialien

Material	[m2/kg]	[kg]
R_ALUMINIUM	5.00E-02	1.64E-01
R_EISEN	5.00E-01	1.26E+01
R_KUPFER	1.50E-01	5.04E-02
R_ST STAHL	1.00E-02	3.84E-01
R_STAHL	6.55E-02	9.59E+01
R_ZINK	1.00E-01	5.04E-02

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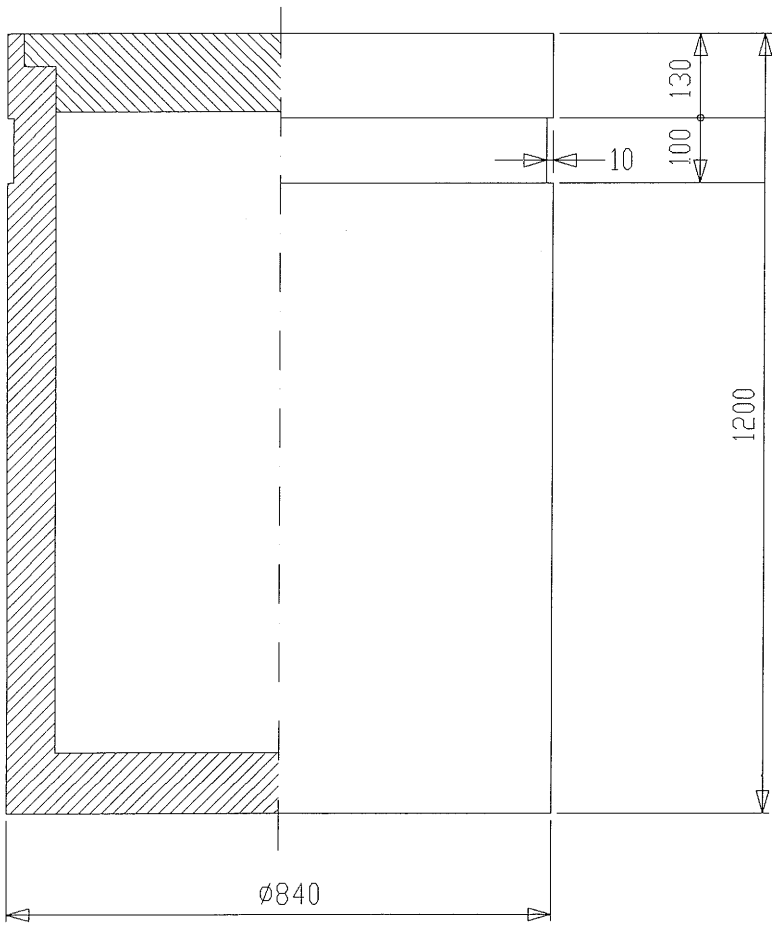
		3.62A

Abfallsorte	: WA-COG-5
Rohabfall	: TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff	: -
Behälterhülle	: 0.67m3-FASS

Abfallgebinde : Äussere Behälterhülle (alle Masse in mm)

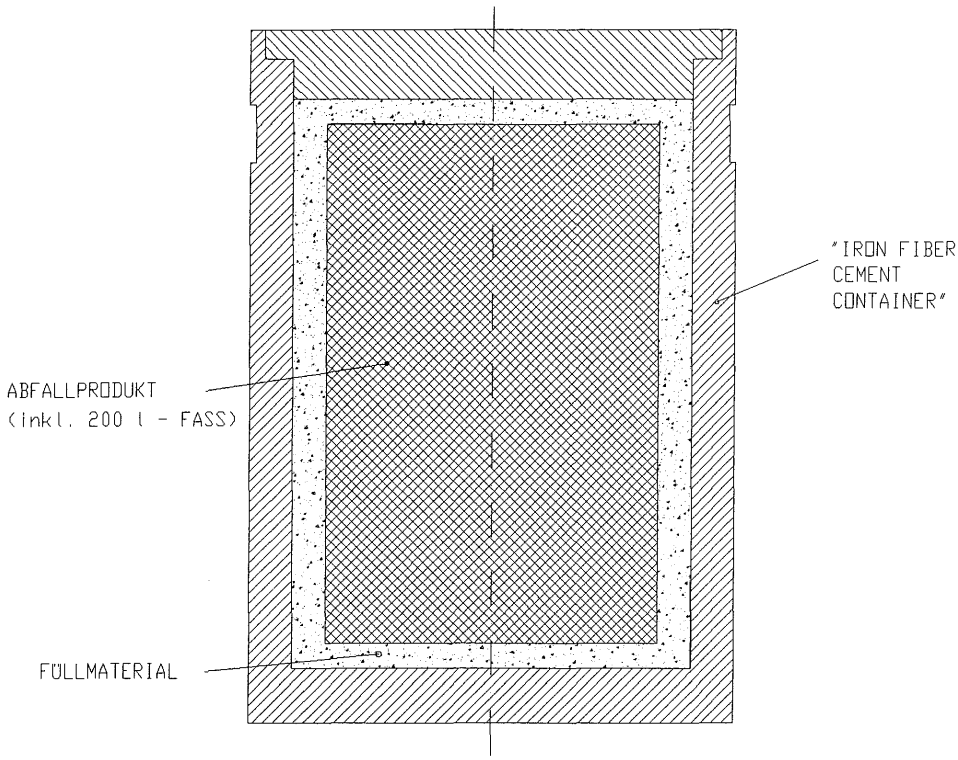


Nagra-MIRA	: M-F7	Datum: 01.09.1994
Formular Nr.	: J_R_000259	Seite: 2

3.62A

Abfallsorte	: WA-COG-5
Rohabfall	: TECHNOLOGISCHE SCHWACHAKTIVE ABFÄLLE
Zusatzstoff	: -
Behälterhülle	: 0.67m3-FASS

Abfallgebinde : Innengeometrie (alle Masse in mm)



APPENDIX C: Reactor neutron source inventory

1) Introduction

The reactor neutron sources have not been included in the waste model inventory. Too many estimates for the activities, γ- and n-dose rates and the times of arising for the various types of sources are required. Further to this, too many assumptions for the cooling times and appropriate packaging would be required to allow for realistic waste sorts. In this appendix, overall characterisation values are given for the definition of these wastes and they are compared with the values of the other RA wastes (non-fuel exchangeable reactor wastes) and SMA wastes to allow for an assessment of their importance for the long-term repository safety assessment.

2) Description

Reactor neutron sources are used for starting the reactor operation. There are many types and they consist of an assembly of one or more long tubes (mainly stainless steel) containing the neutron sources and sometimes boron glass (neutron absorber) and Al_2O_3 filling material. The total weight lies between about 10 and 40 kg. The neutron sources of the Swiss reactors are made of Cf-252 ($t_{½}$ = 2.64 y) (primary sources) and of Sb-Be (secondary sources emitting neutrons after incore activation of Sb-123 to γ-emitting Sb-124 ($t_{½}$ = 60.2 days)). A reactor neutron source may contain primary and secondary neutron sources together.

The reactor neutron sources remain in most cases for 1 year in the reactor where their materials are activated by the core neutron flux. Some reactor sources may be used for different cycles for up to 10 years in the reactor.

3) Characteristics of the reactor neutron source waste

Materials and activities of the waste arising from the NPP neutron sources have been estimated by Nagra using information on the foreseen operation of these sources. The overall estimates given below are conservative in some cases as the numbers of reactor neutron sources and their operational cycles are less than was foreseen.

Materials

Steel (mainly 1.4301, 1.4541, 1.4550):	~ 140 kg
Al_2O_3 :	~ 190 kg
Boron glass:	~ 30 kg
Sb-Be (~78% Sb, ~ 22% Be):	~ 90 kg
Cf-252 (~ 2 mg):	negligible
Total	~ 450 kg

Activities

Activity values at 5 years after discharge are given for the main radionuclides. The activities of the reactor neutron sources are summed without consideration of the different discharge times.

Nuclide	Half-life	Reactor neutron sources (Bq)	RA (reactor exchangeable components) in 2029 (Bq)
H-3	12.3 y	$1\ 10^{15}$	$3.0\ 10^{14}$
Be-10	$1.6\ 10^{6}$ y	$3\ 10^{8}$	$2.1\ 10^{10}$
C-14	$5.7\ 10^{3}$ y	$3\ 10^{9}$	$3.1\ 10^{13}$
Mn-54	312 d	$2\ 10^{12}$	$1.9\ 10^{12}$
Fe-55	2.7 y	$2\ 10^{14}$	$1.5\ 10^{15}$
Co-60	5.3 y	$1\ 10^{14}$	$9.6\ 10^{15}$
Ni-59	$7.5\ 10^{4}$ y	$2\ 10^{11}$	$2.2\ 10^{13}$
Ni-63	$1.0\ 10^{2}$ y	$3\ 10^{13}$	$2.7\ 10^{15}$
Nb-94	$2.0\ 10^{4}$ y	$4\ 10^{10}$	$3.8\ 10^{12}$
Sb-124	60 d	$4\ 10^{7}$	$2.5\ 10^{1}$
Sb-125	2.8 y	$8\ 10^{12}$	$3.0\ 10^{14}$
Cs-134	2.1 y	$5\ 10^{7}$	$3.7\ 10^{11}$
Ba-133	10.5 y	$5\ 10^{9}$	$9.4\ 10^{10}$
Cf-252	2.6 y	$4\ 10^{10}$	--

The comparison with the activities of the RA waste given at the end of their arising in 2029 according to the nuclear scenario considered exhibits the following:

1) Long-lived activities from the steel (Ni-59, Ni-63, Nb-94) are 100 time lower. Fe-55 and Mn-54 activities are not so much lower, as they have not been decayed to take account of differences in arising times.

2) Cs-134 and Ba-133 arise from boron glass but additional contributions from metal impurities and crud are considered for the RA waste.

3) H-3 and Be-10 arise from Be in the Sb-Be neutron sources. They also arise with C-14 in the boron carbide of the control rod assemblies. The significant amount of H-3 in the neutron sources (3 time higher than in the RA) comes from the reactions Be-9 (n,α) He-6, He-6 $\rightarrow$ Li-6, Li-6 (n,α) H-3.

4) As Cf-252 is not present in other SMA waste and as H-3 is higher than in the other RA wastes, these nuclides are assessed in relation to the overall SMA disposal safety.

   The H-3 contribution from the neutron sources of $1\ 10^{15}$ Bq is less than that arising in the other (combined) SMA wastes (RA: $3\ 10^{14}$ Bq, SA-KKW: $2.2\ 10^{14}$ Bq, SA-PSIW, $2.0\ 10^{14}$ Bq and MIF $8.6\ 10^{15}$ Bq (see Table 2 in section 5.3)).

The remaining Cf-252 (4 10^{10} Bq) decays after 100 year into 3 10^{5} Bq of Cm-248 ($t_{½}$ = 3.4 10^{5} y, → Pu-244 → Pu-240 → .....). These activities are negligible with respect to the other SMA actinide activities (see Table 2 in section 5.3). The spontaneous fission of Cf-252, Cm-248 and Pu-244 give rise to similarily negligible amounts of fission products.

4) Conclusions

The reactor neutron sources can be classified as SMA wastes. Their tritium contributions to the SMA inventory activities are of the same order of magnitude as the other SMA wastes. Cf-252 and Cm-248 activities, which are not estimated to be present in the other SMA wastes, are not considered to be significant as regards to disposal safety.

APPENDIX D: Additional characterisation data for the waste sort WA-COG-1 (COGEMA vitrified residues)

More characterisation data than given in the standardised database of Volume 2 for the waste sort WA-COG-1 were required for the Kristallin-I Project on the disposal of the high level vitrified residues from reprocessing. These data are reported in the following Tables. (For description of the Nagra characterisation, see report section 4.6.2).

Table D-1: Characteristics of a flask of vitrified HLW-waste sort WA-COG-1

Overall volume of flask	0.18 m^3	
Glass volume	0.15 m^3	
Weight of glass	412 kg	
Initial heavy metal equivalent	1.37 tU [1)]	
Radioactivity content:	beta/gamma	alpha
4 years after fuel unloading	2.8 10^{16} Bq	1.1 10^{14} Bq
40 years after fuel unloading	7.1 10^{15} Bq	5.2 10^{13} Bq
Radiogenic heat output		
Time after fuel unloading (years)	Heat output (watts) [2)]	
4	2810	
40	589	
50	469	
100	161	
300	21.6	
600	12.9	
1,000	7.3	
10,000	0.6	

1) 1.37 tU of the reference PWR fuel is used for the flask characterisation as in the COGEMA specification. However, to deduce the number of flasks for the model inventory the conservative value given in the COGEMA specification of 0.75 container/tU (1.33 tU/cont.) rounded from 0.73 cont./tU (1.37 tU/cont.) has been used.

2) Preliminary set of values calculated by the MIRA data bank facilities. The database value (Volume 2) is 2772 W at 4 years.

Table D-2: Comparison of glass composition of the waste sort WA-COG-1 and the COGEMA specification

Component	Nagra model (WA-COG-1)	COGEMA specification
SiO_2	45.36	45.1
B_2O_3	14.04	13.9
Al_2O_3	4.50	4.9
Na_2O	9.19	9.8
CaO	4.01	4.0
Fe_2O_3	2.75	2.9
NiO	0.39	0.4
Cr_2O_3	0.49	0.5
P_2O_5	0.29	0.3
Li_2O	2.01	2.0
ZnO	2.47	2.5
Fission product oxides	10.87	11.1
Zr oxides [1]	0.98	1.0
Metallic particles	0.67	0.7
Actinide oxides	0.86	0.9
Subtotal	98.88	
Others [2]	1.12	
Total	100.0	100.0

1) Excluding Zr from fission products

2) Fuel activation products: 0.12 %, Gd and Gd activation products (taken from BWR spent fuel): 0.96 %, inconel fines: 0.04 %

Table D-3: Masses of 25 selected elements in one nominal vitrified residue container of the waste sort WA-COG-1 after 10^3 and 10^6 years

Time (years)	A 1'000 (g)	B 1'000'000 (g)	Ratio B/A
C	1.98 E -01	1.98 E-01	1.00
Ca	1.18 E+04	1.18 E+04	1.00
Ni	1.36 E+03	1.36 E+03	1.00
Se	7.80 E+01	7.00 E+01	0.90
Zr	8.63 E+03	8.26 E+03	0.96
Nb	1.96 E+00	3.67 E+02	187.09
Mo	4.60 E+03	4.60 E+03	1.00
Tc	1.06 E+03	4.10 E+01	0.04
Pd	1.76 E+03	1.74 E+03	0.98
Ag	9.78 E+01	1.26 E+02	1.29
Sn	1.46 E+02	1.11 E+02	0.76
I	3.15 E-01	3.05 E-01	0.97
Cs	1.99 E+03	1.88 E+03	0.94
Sm	1.18 E+03	1.18 E+03	1.00
Ho	4.10 E-01	4.08 E-01	1.00
Pb	1.37 E+00	3.85 E+00	2.81
Rn	3.27 E-10	1.34 E-08	40.97
Ra	5.08 E-05	2.10 E-03	41.23
Th	1.19 E-02	3.84 E+00	323.63
Pa	5.18 E-04	8.46 E-03	16.32
U	1.99 E+03	2.24 E+03	1.13
Np	8.15 E+02	6.29 E+02	0.77
Pu	1.04 E+02	7.52 E-01	7.2 E-03
Am	1.46 E+02	3.66 E-07	2.5 E-09
Cm	9.11 E-01	7.91 E-04	8.7 E-04

APPENDIX E: Inventory waste volumes for partial reprocessing of spent fuel

One of the major assumption made for the model waste inventory is the full reprocessing of the spent fuel arising from the 3 GW(e) scenario over 40 years (i.e. 120 MW(e) · years). This assumption maximises the SMA and LMA waste volume. However, it is not certain that the NPP's will extend their reprocessing contracts which currently cover 942 t uranium (~ 1/3 of total). Therefore, the waste volumes for partial reprocessing of 942 tU (COGEMA 61 %, BNFL 39 %) is given below, along with the data for full reprocessing (nominal 3,591 tU, COGEMA 50 %, BNFL 50 %) for comparison. (For explanation of these 6 values see section 4.6.1)

Waste sort [1] Category of waste sorts Class of waste sorts [1]	Partial reprocessing of spent fuel (contracted) Waste volume (m^3)	Fuel reprocessing of spent fuel Waste volume (m^3)
WA-COG-1 (HAA)	127 (707 cont.)	485 (2,693 cont.)
WA-COG-2 (LMA)	366	1,142
WA-COG-4 (LMA)	329	1,030
WA-COG-5 (SMA)	2,053	6,425
WA-COG-6 (LMA)	957	2,990
WA-BNF-2 (LMA)	20	90
WA-BNF-4 (LMA)	156	758
WA-BNF-5 (SMA)	3,586	17,456
WA-BNF-7 (LMA)	95	459
HAA-WA	127 (707 cont.)	485 (2,693 cont.)
LMA-WA	1,923	6,469
SMA-WA	5,639	23,881
Spent fuel	4,036 [2] (1,621 cont.) [2]	--
HAA	4,163 (2,328 cont.)	485 (2,693 cont.)
LMA	1,923	6,469
SMA	83,043	101,285

1) For main features see Appendix 1
2) Indicative values for direct disposal of spent fuel

Without further reprocessing contracts the model waste volume of the SMA repository of 101,285 m^3 would be reduced by about 18,000 m^3 to 83,043 m^3. The waste volume including reserve of 150,000 m^3 (section 5.6.1) would be reduced to 120,000 m^3.