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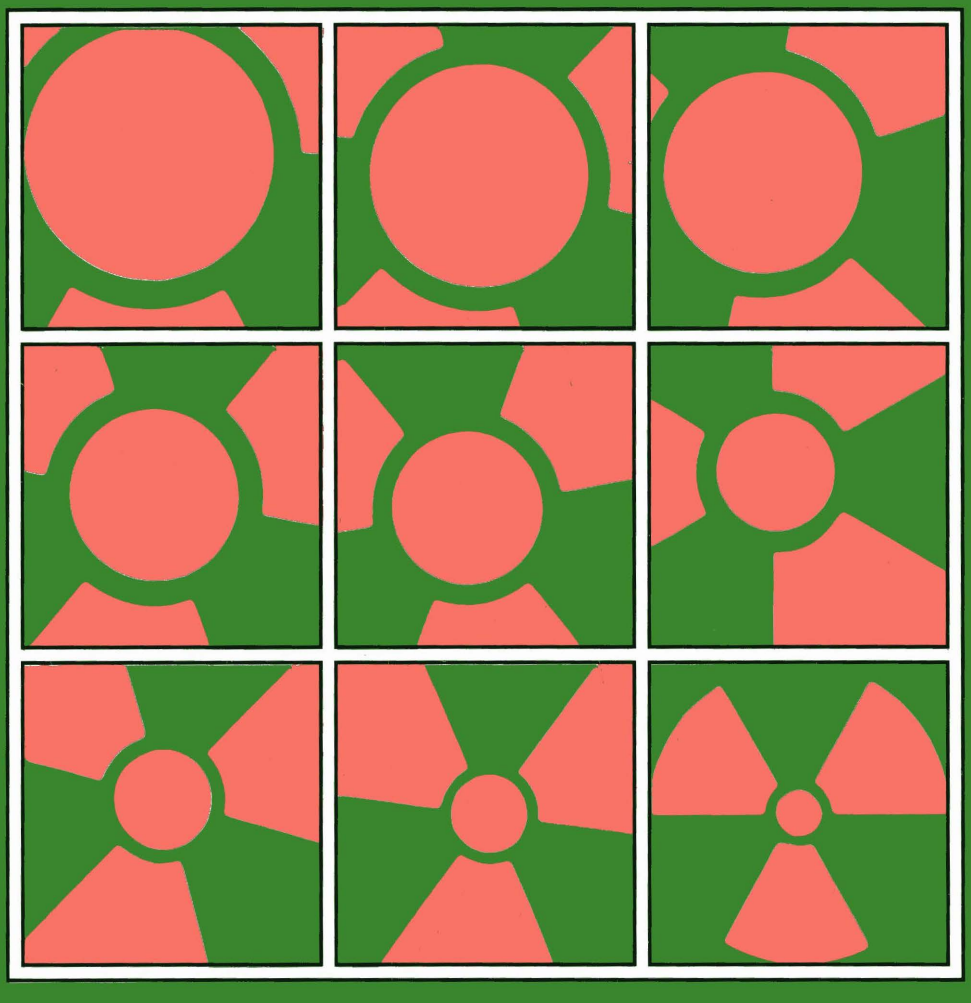
(Volume 3)

Description of German scenarios for PWR and BWR wastes



Report

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Description of German scenarios for PWR and BWR wastes

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Final report

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FOREWORD

This report deals with the description of management routes for PWR and BWR wastes relying to a certain extent on German practices in this particular area. This description is part of an overall assessment study aiming at evaluating a selection of management routes for LWR waste based on economical and radiological criteria.

Actually the assessment study was implemented through complementary contributions provided by nine organisations and companies, i.e.
CEN - Fontenay-aux-Roses, INITEC - Madrid, KAH - Heidelberg, BELGATOM - Brussels, TASK R&S - Ispra, SGN - St. Quentin-en-Yvelines, EDF/SEPTEN - Villeurbanne, FRAMATOME - Paris-la-Défense, GNS - Essen, co-ordinated by the Commission of the European Communities (Brussels).

The main achievements of the assessment study have been summarised by BELGATOM-Brussels.
These different contributions are published as EUR Reports in 1992 (listed as below):

VOLUME N°	MAIN AUTHORS	ORGANISATION	TITLE	EUR REPORT N°
1	R. Glibert	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Main achievements of the joint study	14043 EN/Vol 1
2	E. de Saulieu C. Chary	SGN EDF	Assessment of Management Alternatives for LWR Wastes ; Description of a French scenario for PWR waste	14043 EN/Vol 2
3	S. Santraille K. Janberg H. Geiser	FRAMATOME - GNS	Assessment of Management Alternatives for LWR Wastes : Description of German scenarios for PWR and BWR wastes	14043 EN/Vol 3
4	J. Crustin R. Glibert	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Description of a Belgian scenario for PWR waste	14043 EN/Vol 4
5	B. Centner	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Assessment of the radiological impact to the public resulting from discharges of radioactive effluents	14043 EN/Vol 5
6	G.M. Thiels S. Kowa	TASK R & S KAH	Assessment of Management Alternatives for LWR Wastes : Cost determination of the LWR waste management routes (Treatment/Conditioning/Packaging/ Transport Operations)	14043 EN/Vol 6
7	J. Malherbe	CEA	Assessment of Management Alternatives for LWR Wastes : Cost and radiological impact associated to near surface disposal of reactor waste (French concept)	14043 EN/Vol 7
8	N. Sanchez-Delgado	INITEC	Assessment of Management Alternatives for LWR Wastes : Cost and radiological impact associated to near surface disposal of reactor waste (Spanish concept)	14043 EN/Vol 8

SUMMARY

During the last few years, reactor waste management practices have taken advantage of many improvements as far as processes, organization and safety are concerned. Within the framework of the 3rd EC Waste Management programme (item 1 : system studies), GNS and FRAMATOME are taking part in the present joint study, the purpose of which is first to characterize various overall management schemes resulting from these new developments. After a complete description of each scheme, an analysis is carried out which consists in evaluating industrial feasibility, relevant costs and radiological impact on workers and public.

The assessment of the different European management alternatives will make up basic data for the study. Each national scheme will be applied to a hypothetical site with 20GWe capacity consisting of light-water cooled reactor (PWR's and BWR's) and to low and intermediate radioactive wastes generated during normal operation. After having analysed each basic case, sensitivity studies will be carried out, varying the most important parameters such as waste characteristics, treatment methods, discharge limits and disposal criteria having both cost and radiological impact.

GNS and FRAMATOME are in charge of providing the assessment of German and Italian approaches. The evaluation of these management schemes will comprise waste inventories, treatment, conditioning, packaging, interim storage, transport and final disposal operations, cost and radiological impact on workers and public.

CONTENTS

1. INTRODUCTION

1.1. OBJECTIVES AND SCOPE

During the last few years, reactor waste management practices have taken advantage of many improvements as far as processes, organization and safety are concerned. Within the framework of the 3rd EC Waste Management programme (item 1 : system studies), GNS and FRAMATOME are taking part in the present joint study, the purpose of which is first to characterize various overall management schemes resulting from these new developments. After a complete description of each scheme, an analysis is carried out which consists in evaluating industrial feasibility, relevant costs and radiological impact on workers and public.

The assessment of the different European management alternatives will make up basic data for the study. Each national scheme will be applied to a hypothetical site with 2ØGWe capacity consisting of light-water cooled reactor (PWR's and BWR's) and to low and intermediate radioactive wastes generated during normal operation. After having analysed each basic case, sensitivity studies will be carried out, varying the most important parameters such as waste characteristics, treatment methods, discharge limits and disposal criteria having both cost and radiological impact.

GNS and FRAMATOME are in charge of providing the assessment of German and Italian approaches. The evaluation of these management schemes will comprise waste inventories, treatment, conditioning, packaging, interim storage, transport and final disposal operations, cost and radiological impact on workers and public.

1.2. WORK PROGRAMME

In order to evaluate the management schemes, a methodology has to be applied which consists of performing the following sequence of tasks :

1. Definition of typical primary waste inventories resulting from normal operation of PWR's and BWR's.

2. Definition of discharge limits, waste acceptance criteria and transport regulations.

3. Drawing up basic management routes, taking into account treatment and conditioning variants.

4. Sensitivity studies for each basic management route.

5. Costing of the basic management routes as well as the variants arised from sensitivity studies.

6. Assessment of the related occupational exposure.

7. Assessment of the related radiological impact on the public (short and long term).

In this regard, FRAMATOME/GNS will be partly responsible for tasks 1, 2, 5 and 7 and wholly responsible for tasks 3, 4 and 6 for two basic management schemes relying on the German and Italian philosophy and practice.

1.3. REPORT CONTENTS

In this report :

- Chapter 2 describes the basic data for the study : scenario version, waste inventories, discharge limits, transport regulations and waste acceptance criteria.

- Chapter 3 deals in detail with the German waste management scheme for PWR'S. The operation of gaseous, liquid and solid waste treatment systems is described. Quantities, activities and radionuclide composition of released gaseous and liquid effluents are calculated and compared with discharge limits. These data are used for radiological impact assessment. The evaluation of conditioned solid waste characteristics and relevant packagings allows a judgement upon the management scheme efficiency and volume of packagings is used in the definition of final disposal site. The costing of the overall scheme is carried out in order to do a further comparison between the national practices.

- Chapter 4 deals with the German waste management scheme for BWR's in the same way as chapter 3.

- Chapter 5 is concerned with sensitivity studies about waste conditioning : incineration and compaction for dry wastes, drying methods and cementation for wet wastes.

- Chapter 6 gives assessment of occupational exposure.

2. BASIC DATA

2.1. SCENARIO VERSION

2.1.1. General assumptions for the European study

The reference scenario is the radioactive waste management scheme of a 20 GWe nuclear park of Pressurized Water Reactors (respectively Boiling Water Reactors).

Low and intermediate radioactive wastes are issued from plant normal operation. Gaseous and liquid streams are purified into treatment systems and recycled or released after activity control. Wet active solid wastes (concentrate - sludge - resins) are generated during these effluent processing (they are called secondary solid waste) and are collected together with primary solid wastes issued from purification systems which are part of the nuclear processing and not of the waste treatment. Dry active solid wastes (cloth - paper - tools - pipes ...) are produced inside the plant and are to be removed so as wet active ones. Before being conveyed to a final disposal, these wastes must be conditioned by means of solid waste treatment units (on-site or mobile stations).

The intermediate storage capacity is adapted to the volume of packages produced during one year of operation.

The distance between the plant and the final disposal center is 500 km.

2.1.2. The German Nuclear Power Stations

The German nuclear park is rather diversified. It starts with the now under decommissioning boiling water reactor of Kahl (VAK) with only 15 MWe, and its latest commissioned PWR is Brokdorf with a net electrical output of 1365 MWe.

The trend to bigger units started in 1969 with the order for Biblis (1200 MWe). It was the world's first nuclear steam supply system (NSSS) which was linked with a single-case turbine.

On the PWR side, the intermediate level of 900 MWe was skipped. The biggest reactor with respect to electrical output prior to the construction of Biblis was Stade, with 662 MWe. Within the framework of this study, it would be tempting to use the most recent power stations as reference, because their waste-treatment systems are the most up-to-date. However, it seems suitable to use the Biblis operating data with respect to the production of nuclear wastes. The reason is rather simple, only a prolonged operation shows the tendency of the waste produced during long-term normal operation. As Brokdorf has a little more than 1 year operating history, it can only serve as reference for the state-of-the-art of waste management, but not for the waste operating valves.

On the BWR side, there is Philippsburg I with 900 MWe, but again it is not the most recent. KRB B/C (1244 MWe) may be more representative for the equipment, but not yet for the waste production under long-term normal operation.

2.1.3. The Italian Nuclear Power Stations

Italy was in the fore-field of the nuclear industry in the 60's with a gas-cooled reactor, LATINA, a BWR, GARIGLIANO (now under decommissioning), and a small PWR, TRINO, similar to CHOOZ in Belgium.

In the 70's only one more reactor followed, a BWR of 875 MWe, CAORSO, which can now serve as a basis for this study.

Of this plant the basic waste system information is made available by ENEL and the operational characteristics with respect to the radioactive waste are known over several years of operation.

Two new nuclear power stations are under construction at Montalto di Castro, also BWR's, and at the Vercellese site TRINO II, a PWR, is planned.

The general future of the nuclear industry in Italy is similar to the German one, as it is facing stiff opposition from a public traumatized by the Chernobyl accident ; there is only a minor prospect for further nuclear power stations to be put on the grid in the 90's beyond those mentioned .

2.1.4. Waste management schemes applied to the European study

For further discussion in this study, the basic data on waste management are derived from German nuclear power plants only because currently all Italian plants are not in operation and operational data are therefore not available.

2.1.4.1. Waste management scheme for PWR's

For the assessment of German practice in waste management for PWR's, the following assumptions are made :

- The operating values are taken from plants that have been in operation for several years (Biblis for instance) while the waste treatment system design corresponds to more recent plants (1300 MWe Convoy plants).

- The capacity of German PWR'S is approximately 1300 MWe. The inventories of primary wastes assumed for the European study are those usually found in 900 MWe French and Belgian plants as far as quantities are concerned. It can be noted that the operating values taken for German plants in operation for several years (Biblis for example) are close to those of the study inventories.

To take into account this difference, the study is carried out as follows :

. flow sheet elaboration with European inventories in order to compare the German scheme efficiency with French and Belgian ones,

. equipment cost evaluation of one 1300 MWe plant,

. extension to the 20 GWe park by multiplying with the rate 20/1.3 (amounts of produced waste, total costs of management scheme and transport).

2.1.4.2. Waste management scheme for BWR's

For the assessment of German practice in waste management for BWR's, the following assumptions are made :

- The operating values are taken from plants that have been in operation for several years (ISAR for instance) while the waste treatment system design corresponds to more recent plants (KRB II for instance).

- The capacity of German BWR's is approximately 1300 MWe. The inventories of primary wastes assumed for the European study are those usually found in German BWR's.

The study is carried out as follows :

- flow sheet elaboration with European inventories in order to compare the German scheme efficiency with Spanish one,

- equipment cost evaluation of one 1300 MWe plant,

- extension to the 20 GWe park by multiplying with the rate 20/1.3 (amounts of produced waste, total costs of management scheme and transport).

2.2. WASTE INVENTORIES

2.2.1. Assumptions

The working group has decided to use the same primary waste inventories to draw up national management schemes in order to make the next part of the study, more consistent.

For this purpose, the wastes have been divided into :

- gaseous effluents,

- liquid effluents,

- solid wastes.

The characteristics of these inventories (origin, volume, activity and radionuclide composition) are shown in tabular form with the waste broken down into streams and quantities corresponding roughly to those existing in France for PWR's and in the F.R.G. for BWR's, because of greater national experience in these areas. The real values are operating values and the design values will be used to size treament units. As far as the radionuclide composition of effluents is concerned, the given values are theoretical because they are likely to vary greatly for the plant according to type, age, operation point in operating cycle and time of measurement (the composition depends on the half life of isotopes). For the study, it seems more appropriate to deal with the longer half-life isotopes and in the case of liquid waste to limit oneself to those isotopes which are the most radioactive.

2.2.2. PWR's

2.2.2.1. Gaseous effluents

European study :

The gas characteristics (origin, volume, activity, nuclide composition) have been defined as follows :

TABLE 2.1 : GASEOUS EFFLUENT ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE	REAL VALUE
Primary coolant degassing, evaporation and storage	10 000 Nm^3/a 2 000 Ci/Nm^3	6 000 Nm^3/a 20 Ci/Nm^3
Ventilation	150 000 Nm^3/h 10^{-4} Ci/Nm^3	150 000 Nm^3/h 5 x 10^{-7} Ci/Nm^3

TABLE 2.2 : GASEOUS EFFLUENT RADIONUCLIDE COMPOSITION

RADIONUCLIDE	C-14	Kr-85	Kr-85 m	Kr-87	Kr-88	Xe-133	Xe-133m
%	0.00001	0.03	1.83	1.25	3.32	80.41	1.75
RADIONUCLIDE	Xe-135	I-131	I-132	I-133	I-134	I-135	Aerosols
%	11.31	0.01	0.02	0.03	0.01	0.02	0.00001

The radionuclide composition corresponds to the one to be expected at the entrance of the off gas system. For sake of easiness, the same composition is applied to ventilation gases.

German management scheme :

♦ VENTILATION

The flowrate released through the stack (150000 Nm^3/h) corresponds to the mean value of German practice.

♦ OFF-GAS SYSTEM

In the F.R.G. (Federal Republic of Germany), the activated gases are continuously fed through the decay unit of the system. The throughput is 10 Nm^3/h (90000 Nm^3/a*) and half of it, 45000 Nm^3, are released each year through the stack (6000 Nm^3/a habe been defined for the study).

As part of our study, we assume that the total activity treated and released by the off-gas system is 120000 Ci per year in order to be consistent with the European gaseous inventory.

Nevertheless, the activity treated in German off-gas system is nowadays about 10000 Ci/a because of system optimization. The specific activity at the system inlet is then about 10^{-1} Ci/Nm^3 instead of 20 Ci/Nm^3 which is the value used for the study.

* Nm^3/a : Nm^3 per annum

2.2.2.2. Liquid effluents

European study :

The liquid characteristics (origin, volume, activity, nuclide composition) have been defined as follows :

TABLE 2.3. : LIQUID EFFLUENT ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE	REAL VALUE
Primary system effluents (hydrogenated)	24 000 m^3/a 10 Ci/m^3 (without gas) 100 Ci/m^3 (with gas)	10 000 m^3/a 0.1 Ci/m^3 (out of gas)
Secondary drain wastes	4 000 m^3/a 10^{-1} Ci/m^3 (on average) 1 Ci/m^3 (peak value)*	2 500 m^3/a 10^{-2} Ci/m^3
Laundry wastes	4 000 m^3/a 10^{-4} Ci/m^3	4 000 m^3/a 10^{-5} Ci/m^3 (on average 10^{-4} Ci/m^3 (peak value)*
Decontamination operations	500 m^3/a 10^{-1} Ci/m^3	10 m^3/a 10^{-2} Ci/m^3
Chemicals	1 500 m^3/a 10^{-2} Ci/m^3	1 500 m^3/a 10^{-3} Ci/m^3
Building wastes	6 000 m^3/a 10^{-3} Ci/m^3	3 000 m^3/a 10^{-3} Ci/m^3

* The peak value corresponds to max 10 % of the operation time.

TABLE 2.4 : PRIMARY COOLANT RADIONUCLIDE COMPOSITION

RADIONUCLIDE		Mn-54	Co-58	Co-60	Sr-90	Nb-95	Mo-99	Ag-110 m
%		0.44	3.0	0.6	0.018	0.001	0.44	0.44
RADIONUCLIDE	Sb-124	I-131	I-132	I-133	I-134	I-135	Cs-134	Cs-137
%	0.44	10.4	18.2	31.2	10.4	20.8	1.79	1.79

The specific activity of H_3 is 0.60 Ci/m^3.

TABLE 2.5. : LIQUID WASTE RADIONUCLIDE COMPOSITION

RADIONUCLIDE	H-3	Mn-54	Co-58	Co-60	Sr-90	Nb-95	Mo-99	Ag-110 m
%	1	4.75	31.66	6.33	0.19	0.013	4.75	4.75
RADIONUCLIDE	Sb-124	I-131	I-132	I-133	I-134	I-135	Cs-134	Cs-137
%	4.75	0.46	0.79	1.38	0.46	0.92	19.00	19.00

Because of large discrepancies between the data supplied by the participants, it was decided to use the same Cesium/Colbalt ratio throughout the drawing-up of basic management routes and to investigate the impact of various ratios only during sensitivity studies.

For calculating the equivalent activity in iodine-131, the following formula is used :

$$I - 131\ eq = I\text{-}131 + \frac{I\text{-}132}{30} + \frac{I\text{-}133}{4} + \frac{I\text{-}134}{50} + \frac{I\text{-}135}{10}$$

German management scheme :

♦ PRIMARY COOLANT TREATMENT SYSTEM

The volume of hydrogenated effluents greatly depends on the plant operation ; 10000 m^3/a is a mean value.

♦ LIQUID WASTE TREATMENT SYSTEM

In German plants, the segregation of effluents is not similar to the one used for our study. Secondary and building drains are collected together either from equipment rooms or from operating rooms ; active effluents are therefore seperated from inactive ones.

Until now, liquid treatment has depended on their origin but now, effluents are segregated by the operator with new criteria : clear ones are separated from dirty ones. The origin of the effluents is no longer taken into account. The effluents are routed to their treatment unit after chemical and radiochemical analysis.

The inventories defined for the European study are suited to the German new practice.

2.2.2.3. Solid wastes

During normal plant operation, different types of solid wastes are generated.

They mainly consist of :

- wet solid wastes (issued from liquid processing) :

  . resins used for coolant purification,

  . mechanical or precoat filters ;

- dry solid wastes :

  Technological wastes which are sorted (combustible and non combustible, compactable and non compactable).

The following tables show the primary solid waste inventories defined for the study : origin, volume and activity (TABLE 2.6) and the relevant radionuclide composition (TABLE 2.7).

Secondary solid wastes are produced during liquid waste processing. Their characteristics depend on national management scheme.

The inventories defined for the European study are suited to the German practice.

TABLE 2.6 : PRIMARY SOLID WASTE ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE		REAL VALUE	
Primary resins :				
- highly active	1.3 m^3/a	700 Ci/m^3	1.3 m^3/a	500 Ci/m^3
- low active	2.6 m^3/a	100 Ci/m^3	2.6 m^3/a	50 Ci/m^3
Primary filters :				
- Primary coolant purification	15 filt/a	100 Ci/filt.	10 filt/a	50 Ci/filt.
- Spent fuel pit purification	25 filt/a	2 Ci/filt.	20 filt/a	1 Ci/filt.
Core components	1 m^3/a	1 000 Ci/m^3	1 m^3/a	1 000 Ci/m^3
Normal equipment combustible + compactable	260 m^3/a	0.01 Ci/m^3	260 m^3/a	0.01 Ci/m^3
Normal equipment non combustible + compactable	100 m^3/a	0.01 Ci/m^3	100 m^3/a	0.01 Ci/m^3
Normal equipment combustible + non compactable	20 m^3/a	0.01 Ci/m^3	20 m^3/a	0.01 Ci/m^3
Normal equipment non combustible + non compactable	20 m^3/a	0.2 Ci/m^3	20 m^3/a	0.2 Ci/m^3

A distinction is made between the most and least active fractions of primary resins as well as for the primary filters. Core components will be included in the overall inventory although the management of these waste categories can be considered as common for all the reference routes. Therefore, it will not be taken into account in the subsequent sensitivity studies. The low level solid waste category (normal equipment)is subdivised in four sub-categories to enable its further processing by means of a large number of variants. The radionuclide composition for solids is largely dominated by caesium and cobalt and must be considered identical for all the solid waste streams.

TABLE 2.7. : PRIMARY SOLID WASTE RADIONUCLIDE COMPOSITION

RADIONUCLIDE	Mn-54	Co-58	Co-60	Mb-90	Ag-100m	Sb-124	Cs-134	Cs-137
%	5	33.32	6.68	5	5	5	20	20

German management scheme :

The inventories defined for the study are representative of German practice.

In the FRG, Cobalt/Cesium proportions are quite different.

Co58-60 : 60 % - 70 %

Cs134-137 : 30 % - 40 %

2.2.3. BWR's

2.2.3.1. Gaseous effluents

The inventories defined for the European study are suited to the German practice.

TABLE 2.8. : GASEOUS EFFLUENT ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE	REAL VALUE
Entrance of the off-gas system	200000 Nm^3/a in : 5.10^4 Ci/h out : 0.1 Ci/h	200000 Nm^3/a in * : 50 Ci/h out* : 0.01 Ci/h
Ventilation	200000 Nm^3/h 10^{-5} Ci/Nm^3	200000 Nm^3/h 10^{-7} Ci/Nm^3

* Difference due to isotopes with longer period.

TABLE 2.9 : GASEOUS EFFLUENT RADIONUCLIDE COMPOSITION

RADIONUCLIDE	C-14	Kr-85	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-133m
%	0.00001	0.03	1.83	1.25	3.32	80.41	1.75
RADIONUCLIDE	Xe-135	I-131	I-132	I-133	I-134	I-135	Aerosols
%	11.31	0.01	0.02	0.03	0.01	0.02	0.00001

2.2.3.2. Liquid effluents

The inventories defined for the European study are suited to the German practice.

TABLE 2.10 : LIQUID EFFLUENT ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE	REAL VALUE
Feed water, primary system leaks, samples, back-flushing waters, system discharges	25000 m^3/a 10^{-3}-10^{-1} Ci/m^3	15000 m^3/a $0.2(10^{-3}$-$10^{-1})$ Ci/m^3
Drain waters from buildings	5000 m^3/a 10^{-5}-10^{-3} Ci/m^3	5000 m^3/a 10^{-6}-10^{-4} Ci/m^3
Laundry waste, showers, washing rooms	5000 m^3/a (peak : 25 m^3/d) 10^{-6}-10^{-3} Ci/m^3	5000 m^3/a 10^{-6}-$5x10^{-4}$ Ci/m^3
Laboratory, decontamination	1500 m^3/a 10^{-5}-10^{-2} $Ci/.m^3$	500 m^3/a 10^{-5}-10^{-2} Ci/m^3
Regenerated water condensate cleaning (mixed bed resins)	800 m^3/a 10^{-5}-10^{-2} Ci/m^3	400 m^3/a 10^{-5}-10^{-2} Ci/m^3
Waste from decantation, filtration	3000 m^3/a 10^{-5}-10^{-3} Ci/m^3	2000 m^3/a 10^{-5}-10^{-3} Ci/m^3

* The peak value corresponds to max. 10 % of the operational times.

TABLE 2.11 : LIQUID EFFLUENT RADIONUCLIDE COMPOSITION

RADIONUCLIDE	H - 3	Mn-54	Co-58	Co-60	Sr-90	Nb-95	Mo-99	Ag-110m
%	1	4.75	44.33	8.87	0.003	0.013	4.75	4.75
RADIONUCLIDE	Sb-124	I-131	I-132	I-133	I-134	Cs-134	I-135	Cs-137
%	4.75	0.46	0.79	1.38	0.46	11.4	0.92	11.4

2.2.3.3. Solid wastes

The inventories defined for the study are not suited to the German practice as far as volumes are conerned.
The inventories used for German BWR study are defined in TABLE 4.8.

TABLE 2.12 : PRIMARY SOLID WASTE ORIGIN, VOLUME AND ACTIVITY

WASTE ORIGIN	DESIGN VALUE	REAL VALUE
Resins (bead)	15 m^3/a 50 Ci/m^3	+/- 5 m^3/a 10 - 20 Ci/m^3
Filters (no cartridge but powder for primary cleanup)	50 m^3 100 Ci/m^3 **	< 20 m^3 10 - 20 Ci/m^3
Core components	5 m^3/a 1000 Ci/m^3	5 m^3/a 1000 Ci/m^3
Normal equipment non combustible + compactable*	260 m^3/a 0.01 Ci/m^3	260 m^3/a 0.01 Ci/m^3
Normal equipment non combustible + compactable	100 m^3/a 0.01 Ci/m^3	100 m^3/a 0.01 Ci/m^3
Normal equipment combustible + non compactable	20 m^3/a 0.01 Ci/m^3	20 m^3/a 0.01 Ci/m^3
Normal equipment non combustible + non compactable	20 m^3/a 0.2 Ci/m^3	20 m^3/a 0.2 Ci/m^3

* Density before compaction : 0.15
** Due to corrosion products

TABLE 2.13 : PRIMARY SOLID WASTE RADIONUCLIDE COMPOSITION

RADIONUCLIDE	Mn-54	Co-58	Co-60	Mo-99	Ag-110m	Sb-124	Cs-134	Cs-137
%	5	46.65	9.35	5	5	5	12	12

2.3. DISCHARGE LIMITS

2.3.1. Assumptions

The data sets proposed by the participants were quite different because they adapted to national practices.

In order to make the study more consistent, it has been decided to agree about common annual averaged discharge limits.

Two kinds of values have been defined for each radionuclide category :

- Design values : they will be used to design waste treatment installations and correspond to large values allowing a more genuine assessment of each national management route.

- Objective values : they are usually encountered values or targets to be reached during normal operation but without any binding constraint.

The possible effects is imposing additional discharge limits for alpha emitters on mangement routes for LWR waste will be investigated later on in the frame of sensitivity studies.

2.3.2. PWR's

2.3.2.1. FRG

2.3.2.1.1. Gaseous effluents

Origins

The reactor-building is ventilated. This is also true for those buildings in which the secondary and auxiliary systems are located. The ventilation and the emission of air is limited to the minimum necessary to maintain the different pressure steps in between the respective buildings. The pressure is lower in those buildings where the air could be theoretically contaminated by leaks from the systems.

Leaks can appear from all systems in which liquids are under high pressure or where the temperature of the liquids is fairly high. For this reason, the air is continuously filtered.

Prior to the emission of air through the stack, aerosols and iodine filters, with an efficiency higher than 99 %, purify the air.

An additional input to the venting system comes from the off-gas system. During the fission of the Uranium, radioactive fission gases are produced within the primary coolant. These have to be ventilated via the off-gas treatment system. For this purpose, the off-gases are led via a

delay-system and in this system, they decay, to non-dangerous concentrations. Then, they are evacuated via the stack. The delay times (min. 60 days for Xenon, greater than 60 hours for Krypton) are sufficient to assure a complete decay of the short-lived Xenon and Krypton isotopes. The longer half-life of Xenon-133 is significantly reduced while Krypton-85 is totally released resulting from its long half-life time.

Release Requirements

During normal operation, the activity released from the off-gas system is reduced, as the primary coolant is not continuously degassed. The release of gaseous activities is subject to greater variations. It depends on the number of defective fuel elements in the storage pool, the operational characteristics of the power station (for example load following) and the possible leaks of components.

A good handling and maintenance can significantly reduce the amount of gaseous radioactivity to be released.

In order to give some indications for 1300 MW PWR (Brokdorf), the following yearly gaseous releases have been required for the license application :

TABLE 2.14 : DISCHARGE LIMITS FOR GASEOUS EFFLUENTS (GERMAN PWR)

GASEOUS EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Noble gases	30 000	110 000 10^{10}
Iodine 131	0.15	0.56 10^{10}
Long living aerosols	1	3.7 10^{10}
Tritium	4 000	14 800 10^{10}

These values give a significant operational margin, as experience shows. Even if these values are actually used in normal operation, the personal dose of the people living close to or in greater proximity of the power station is significantly under those limits given by the radio-protection law.

2.3.2.1.2.Liquid effluents

Origins

Water of the nuclear auxiliary systems can theoretically have the same specific activity on the form of dissolved or suspended substances as the primary coolant.

The water leaking from the primary and auxiliary systems is collected together with those liquids coming from the operation of the laboratory, decon-systems, showers, etc. Their specific activity is however, significantly lower. All radioactive liquids are treated in the liquid waste treatment system. Depending on the origin and activity, these waters are cleaned within a centrifuge and/or an evaporating system.

Release requirements

After purification, the liquids are stored in control tanks and the activity concentration is measured. If the concentration is lower than 1.85×10^{7} Bq/m^3 (5×10^{-4} Ci/m^3), then the liquids may be released after dilution with the cooling water.

In the nuclear power station of Brokdorf (1300 MW), the release of radioactive-liquid effluents is authorized up to an annual limit.

TABLE 2.15 : DISCHARGE LIMITS FOR LIQUID EFFLUENTS (GERMAN PWR)

LIQUID EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Total (w/o Tritium)	1.5	$5.6\ 10^{10}$
Tritium	950	$3500\ 10^{10}$

Tritium is produced because of the capture of neutrons during the fission process and transformation of the boron used for the regulation of the power station. It has a decay-time of 12.3 years and cannot be extracted

with technically acceptable performance from the liquid effluents. Therefore the tritium is released with the operational liquid effluents in the cooling-water.

2.3.2.2. European study

TABLE 2.16 : DISCHARGE LIMITS RELATED TO ONE PWR UNIT

VALUES EFFLUENTS	DESIGN VALUES (Ci/a)	OBJECTIVE VALUES (Ci/a)
Liquid effluents		
. Total (Tritium excluded)	9	2
. Tritium	950	750
Gaseous effluents		
. Noble gases	20000	2000
. Halogens	0.3	0.02
. Aerosols	0.5	0.02
. Tritium	200	100

2.3.3. BWR's

2.3.3.1. FRG

2.3.3.1.1. Gaseous effluents

TABLE 2.17 : DISCHARGE LIMITS FOR GASEOUS EFFLUENTS (GERMAN BWR)

GASEOUS EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Noble gases	90 000	333 10^{14}
Iodine 131	0.5	1.85 10^{10}
Long living aerosols	1.5	5.55 10^{10}
Tritium	500	1 850 10^{10}

2.3.3.1.2. Liquid effluents

Water coming from primary and auxiliary systems (leaks, samples, back-flushing...) is collected with drain waters from building, laundry waste, laboratory and decontamination effluents.

The authorized releases are the following :

TABLE 2.18 : DISCHARGE LIMITS FOR LIQUID EFFLUENTS (GERMAN BWR)

LIQUID EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Total (w/o Tritium)	8	29.6 10^{10}
Tritium	500	1850 10^{10}

2.3.3.2. Italy

2.3.3.2.1. Gaseous effluents

Licensing limits

Like in the German power stations, the license of CAORSO reflects the different radiotoxicities of the constituents of the gaseous effluent stream, thus allowing for :

TABLE 2.19 : DISCHARGE LIMITS FOR GASEOUS EFFLUENTS (ITALIAN BWR)

GASEOUS EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Noble gases	20 000	74 000 10^{10}
Halogens	0.2	0.74 10^{10}
Aerosols	0.007	0.026 10^{10}

A particularity of CAORSO is the fact that limits are specified per stack ; this is due to the fact that CAORSO has three different stacks of various heights resulting in different emissions, and thus dose-rates to the public.

Actual releases

In the years 1981-1986, the following values for the major nuclides have been measured :

TABLE 2.20 : RELEASED GASEOUS EFFLUENT ACTIVITY (ITALIAN BWR)

	ACTIVITY (Ci/a)
H-3	3 : 30
Kr-85/88	30 : 150
Xe-133/135	100 : 700
I-131	001 : 0.013
Sr-90	< 0.001
Cs-134	< 0.001

The comparison with the license limits shows that the annual releases have always only been fractions of the authorized ones.

This also reflects the fact, that the gaseous effluent treatment systems are adequate.

However,it has not been examined in detail,whether weekly or daily releases are equally low or whether some have been closer to the license limits.

2.3.3.2.2.Liquid effluents

License limits

Like in the case of gaseous effluents the licensed liquid effluent radioactive releases are linked via a formula which reflects the different radiotoxicities of the various isotopes. This means, that in case of a 100 % release of one category, the other categories have to be reduced to zero.

In the framework of this study, we shall, however, only refer to the maximum values per category, which are :

TABLE 2.21 : DISCHARGE LIMITS FOR LIQUID EFFLUENTS (ITALIAN BWR)

LIQUID EFFLUENTS	DISCHARGE LIMITS	
	Ci/a	Bq/a
Total (w/o Tritium) :		
- β nuclides	6	$22.2\ 10^{10}$
- α nuclides	0.6	$2.2\ 10^{10}$
Tritium	6 000	$22\ 200.10^{10}$

Actual liquid leleases under normal operation

Here again we have extracted the minimum and maximum numbers for the different nuclides over the years 1981 to 1986 :

TABLE 2.22 : RELEASED LIQUID EFFLUENT ACTIVITY (ITALIAN BWR)

	ACTIVITY (Ci/a)
H-3	11.6 : 32.4
Co-60	0.062 : 0.264
Cs-137	0.003 : 0.015
I-131	0.002 : 0.012
Sr-90	< 0.001 : 0.006
Pu-239	<< 0.001

From these figures it can be concluded again that the waste management systems apparently work well, as the actual releases are only fractions of the authorized ones.

This statement, however, does not yet conclude that the system is optimized. This will be examined in more detail later.

2.3.3.3. European study

TABLE 2.23 : DISCHARGE LIMITS RELATED TO ONE BWR UNIT

VALUES / EFFLUENTS	DESIGN VALUES (Ci/a)	OBJECTIVE VALUES (Ci/a)
Liquid effluents		
. Total (β,) (w/o Tritium)	8	1
. Tritium	500	100
Airborne effluents		
. Noble gases	20000	3000
. Halogens	0.05	0.01
. Aerosols	1.5	0.03
. Tritium	500	150

2.4. TRANSPORT REGULATIONS

2.4.1. FRG

In practically all cases within the European Community, nuclear power stations are built at a distance from any site for final disposal of radioactive wastes. Thus wastes have to be transported to the final disposal site on public roads. As a consequence, transport regulations apply.In the FRG, until the end of 1987, the basis were the IAEA "Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition, Safety Series N°6".

From 1988 onwards, the standards given in the 1985 edition to the IAEA-Regulations will apply.

These regulations deal primarily with requirements related to package design and test requirements. In the new edition, emphasis has also been given to the principle of keeping radiation exposures as low as practicable.

For example it is specified in II.202 : "Radiation exposures from the handling, storage and transport of radioactive material shall be kept as low as reasonably acheviable, economic and social factors being taken into account ".

It is further said in II.205 b :

"For members of the public, in the determination of segregation distances or dose rates in regularly occupied public areas or in areas where the public has regular access, a dose level of not more than 1 mSv (100 mrem) per year to the critical group shall be used as the limiting value. This value shall be used together with hypothetical but realistic models and parameters to determine segregation distances or dose rates for members of the public, with the objective of providing reasonable assurance that actual doses from transport of radioactive material will not exceed small fractions of the appropriate dose limits".

Such relatively vague advises are difficult to translate into precise numbers and specific design choices. If overemphasized, these new requirements can cause significant cost increases in waste transportation. We shall therefore later examine the consequences in more detail, when the waste management systems are defined and as well as the packages for the waste types.

Package means in this context the packaging with its radioactive contents as presented for transport. A material is considered radioactive, if its specific activity exceeds 74 kBq/kg (2 n Ci/g). The packaging performance standards, in terms of retention of integrity of containment and shielding, depend upon the quantity and nature of the radioactive material (RAM) transported.

In general within the framework of this study, we shall refer to the following package classifications of rising degree of accident resistance :

1 - Industrial packages Type 1, 2, 3 for LSA (Low Specific Activity) or SCO (Surface Contaminated Object) RAM.

2 - Type A package, containing an activity of up to A_{11} if special form RAM, or up to A_{12} if not special form RAM and designed to meet certain General Requirements to assure the integrity during transport accidents.

For some typical waste isotopes, the A_{11}, A_{12} values are given as follows :

TABLE 2.24 : ACTIVITY LIMITS FOR PACKAGES

	A_1 (TBq ; Ci)	A_2 (TBq ; Ci)
Co_{60}	0.4 ; 10	0.4 ; 10
Cs_{134}	0.6 ; 10	0.5 ; 10
I_{131}	3 ; 80	0.5 ; 10
Sr_{90}	0.2 ; 5	0.1 ; 2

3 - Type B package, defined by an activity in excess of A1 if special form RAM or in excess of A2 if not special form RAM.

In difference to the 1973 issue of the IAEA Regulations, the 1985 edition specifies in para 422 :

"The quantity of LSA material or SCO in a single package or object or collection of objects, if appropriate, shall be so restricted that the external radiation level at 3 m from the unshielded material or object or collection of objects does not exceed 10 mSv/h (1 rem/h)."

From this follows also independently from the A_{11}/A_{12} - limits, that beyond a certain radiation dose without shielding, the packaging must meet the very stringent type B - requirements, which are :

1 - Withstand a 9 m drop on an unyielding surface in the most damaging drop position at - 40°C.

2 - Withstand a 1 m drop onto a specified steel pinion.

3 - Withstand without substantial loss of shielding capacity a 1/2 h, 800 °C fire, etc...

It is evident, that the different packaging categories have widely differing prices due to the greater demands on material, quality assurance and quality control during fabrication.

These costs will be one of the items which will influence the optimum in waste management as it is theoretically possible to " dilute " all reactor wastes in such a way that they are transportable in IPs ; the other extreme is to concentrate i.e. all liquid wastes by drying, which then leads to high specific activities requiring exclusively the use of Type B-packagings.

2.4.2. European study

Since the national waste transport regulations are quite different, the basic management schemes will first be drawn up with the current national practice. The impact of adopting the same IAEA recommendations will be examined as part of the parametric study. For the F.R.G, these recommendations are already in application. Nevertheless, they are going to be amended because of recent changes.

2.5. WASTE ACCEPTANCE CRITERIA (FRG)

2.5.1. Storage and final disposal background

The final destination of the radioactive wastes resulting from the nuclear fuel cycle is the final depository.

Thus the acceptance criteria for this depository will have an up-front influence not only on the storage conditions but also on all conditioning techniques and thus on waste management optimization.

It is therefore necessary to carefully analyze the boundary conditions for final disposal in the F.R.G. in order to assess the possibilities for modifications within the existing waste management scheme. Otherwise any proposal for theoretical changes with the goal of optimization may be unrealistic.

The actual German situation is determined by the acceptance criteria of :

a) the KONRAD iron-mine for non-heat generating wastes and,

b) the GORLEBEN salt-mine for heat generating wastes.

As this study is only concerned with wastes resulting directly from nuclear power stations operation and not from reprocessing, we shall only deal with non-heat generating wastes, thus only the KONRAD acceptance criteria are of relevance within this study framework.

In the F.R.G., shallow-land burial or the erection of tumuli like in France is not considered. Thus a differentiation of wastes with shorter and longer decay times is not possible.

2.5.2. The waste acceptance criteria of the Konrad iron-mine

The relevant criteria were developed by PTB (Physikalische Technische Bundesanstalt, a federal institution responsible for long term storage and disposal of radioactive waste in FRG) on the basis of a site-specific safety analysis, which includes the following 14 repository-relevant features :

1. Total activity of the waste package.

2. Activity of relevant individual nuclides.

3. Dose rate at the surface of the package at distances of 1 and 2 m.

4. Surface contamination of the waste package.

5. Chemical composition of the crude waste.

6. Quality of the fixing agent.

7. Quality of the waste canister.

8. Quantity ratio : waste/fixing agent/water/aggregates.

9. Mixture (consistency).

10. Mass.

11. Setting conditions of the matrix.

12. Water content or residual moisture.

13. Thermal behaviour.

14. Stackability.

Only such waste may be disposed of, whose properties, on the basis of the documents from the party obliged to surrender the waste and on the basis of product controls, correspond to the acceptance conditions laid down in the license application to the mining authority by PTB. As the license is not yet issued but probably only in 1992, the acceptance criteria are still preliminary, but for the operators of nuclear power stations they constitute the practical basis for their waste management.

2.5.3. Preliminary waste acceptance requirements

Waste acceptance requirements were prepared for the delivery of waste packages to the repository. They cover all the technical requirements and are formulated in such a way that they first describe the general aspects and then develop into more specific requirements.

2.5.3.1. General requirements

Two general and basic requirements are to be fulfilled by all wastes :

- compliance with the safety requirements,

- prohibition of mixing non-radioactive waste with radioactive ones.

2.5.3.2. Requirements for waste forms

The various radioactive waste forms to be disposed of in the Konrad mine must fulfill the previous basic requirements and the additional basic requirements if an immobilization material is used, as well as the special requirements formulated for the waste form groups they belong to (the different groups are defined thereafter).

2.5.3.2.1. Basic requirements

All waste forms must be solid or solidified and they must neither rot nor ferment. The content of liquids and gases in bottles, lamps etc... and the content or release of free-moving liquids are regulated and is to be limited. Furthermore, no self-igniting or explosive materials are permitted and the concentration by mass of fissile materials is restricted to 50g per 100 l of waste form. The processing of radioactive wastes in a packaging is permitted if the safety-related barrier functions of the packaging are not adversely affected.

Immobilized wastes must have completely set or must be completely solidified. If, for example, solid radioactive wastes are cast or if the void spaces between inner packagings are filled, suitable free-flowing immobilization materials must be used which can be consolidated, for example, by vibration. Contaminated liquids may be used for the mixing of the immobilization material if the quality requirements of the respective waste form group are fulfilled and if compatibility with those materials to be cast is guaranteed.

If radioactive wastes can release Rn-220 from packagings without specified tightness, the waste form must be enclosed by at least 40 mm of inactive concrete.

2.5.3.2.2.Waste Form Groups

All the radioactive waste forms can be assigned to one of the six waste form groups which were established as a result of the incident analysis for the operational phase of the Konrad repository based on radionuclide release behaviour. These waste form groups are :

* group Ø1 : bitumen and plastic products,

* group Ø2 : solid matter,

* group Ø3 : metallic solid matter,

* group Ø4 : compacted waste,

* group Ø5 : cemented/concreted waste,

* group Ø6 : concentrates.

There are basic requirements which are to be fulfilled by all waste form groups. If a waste form is e.g., assigned to group Ø1, it must be guaranteed that the basic requirements are met. All waste forms can therefore be assigned to group Ø1. If an assignment to other groups is made, specific requirements must be fulfilled in addition to the basic requirements. For example, compacted wastes must be compacted with a pressure of 3Ø MPa or more into an inherently stable form, and concentrates must consist of a non-powdery solid residue of evaporated liquids or dried sludges which is not combustible.

Due to the improved release behaviour of the waste forms, the maximum permissible radionuclide inventory increases from group Ø1 to group Ø6.

2.5.3.3. Requirements for Packagings

Radioactive waste forms must be packed into packagings for transport, handling and stacking. The packagings must be filled without undue damage. The requirements for packagings are divided into basic requirements and special requirements for the waste classes. The use of inner packagings is also regulated.

2.5.3.3.1.Filling of Packagings

The filling of packagings must be performed in such a way that the limits of the local dose rates which comply with the transport regulations are not exceeded and that the packagings are not unduly damaged by the waste form. The packagings should be filled as completely as possible and a uniform mass distribution should be ensured during transport, handling and stacking.

2.5.3.3.2.Basic Requirements

Standardized packagings are to be used for the diposal of radioactive wastes in the Konrad repository. 3 types of cylindrical concrete packagings and 3 of cylindrical cast-iron packagings may be used as well as 6 different types of containers (rectangular boxes) which might be manufactured from sheet steel, concrete or cast iron. Drums are not accepted and must be packed, for example, into containers. Packagings which are used for the conditioning of radioactive wastes arising from the reprocessing of spent fuel from German nuclear power plants in other European countries may differ from the standardized packagings. The packagings must be type-tested.

Packagings must be designed in such a way that their stackability in the repository is ensured for a height of at least 6 m without adverse effects on their thightness and integrity.

On delivery, the packagings must be free of apparent corrosive and mechanical damages which adversely affect their tightness and integrity.

If necessary, the packagings or the inner packagings must be designed to be corrosion resistant.

2.5.3.3.3.Waste Classes

Two waste classes were defined within the safety assessments for the Konrad repository which refer to the quality of the packaging. Waste class I imposes minor requirements whereas waste class II necessitates improved requirements. In all cases the basic requirements must be fulfilled.

Packagings may be assigned to waste class I if their design is such that their integrity after a mechanical impact with 4 m/s or less is preserved to such an extent that in the case of a subsequent fire flammable waste forms with a melting point above 300 °C do not burn but pyrolyse.

Packagings assigned to waste class II must withstand a drop from a height of 5 m on an unyielding target in such a way that the leak rate after the drop does not exceed 10^{-4} Pa m^3/s (with the exception of waste homogeneously immobilized in bitumen) and must have either a thermal conductivity resistance of at least 0.1 m^2K/W or a leak rate of 10^{-5} Pa m^3/s or less (standard conditions) ; it must also be ensured that the integral leakage in the case of a fire (800°C, 1 hour) does not exceed 1 mole gas within 24 hours.

2.5.4. Consequences of the general and basic requirements

The above-mentioned general requirements lead to quantitative limitations with respect to the dose-rates of acceptable waste packages :

TABLE 2.25 : GENERAL REQUIREMENTS FOR WASTE PACKAGES

LIMITATIONS OF THE LOCAL DOSE RATE

WASTE PACKAGE	MAX. LOCAL DOSE RATE Sv/h	
Cylindrical waste packages	on average at surface	2×10^{-3}
	locally	1×10^{-2}
	at 1 m distance	1×10^{-4}
Containers	on average at surface	2×10^{-3}
	locally	1×10^{-2}
	at 2m distance	1×10^{-4}

The safety analysis covering all handling and disposal incidents leads to a further quantification resulting in annually disposable activity limits presented in the following tables.

Activity values per waste package are derived for four radionuclides and two radionuclide groups (non-specified further alpha and beta/gamma emitters) which are listed in TABLE 2.26. With respect to the requirements inferred from normal operation, a distinction is made between packagings without and with specified tightness (annual portion of release less than 0.01 to less than 0.001). The portion of release is defined as that part of the activity which is released from the waste form into the packaging atmosphere (residual voids) and reaches the ambient atmosphere by leakages.

The activity limiting values derived from the incident analysis for the radionuclides being radiologically most important (key nuclides), for non-specified further alpha and beta/gamma emitters are compiled in TABLE 2.27. For other radionuclides (individual radionuclides), activity limiting values are defined but not reported here.

TABLE 2.26 : ACTIVITY LIMITING VALUES FOR RADIONUCLIDES AND RADIONUCLIDE GROUPS DISPOSABLE PER YEAR IN THE KONRAD REPOSITORY RESULTING FROM THE SAFETY ASSESSMENT OF NORMAL OPERATION

Data in Bq/a

Radionuclide/ Group of Radionuclides	Packaging without Specified Tightness		Packaging with Specified Tightness					
			Annual Portion of Release					
			≤0,01		≤0,001		≤0,0001	
	metallic solid matter	other waste form groups	metallic solid matter	other waste form groups	metallic solid matter	other waste form groups	metallic solid matter	other waste form groups
Tritium								
- unspecified		$2,3\ 10^{13}$		$7,1\ 10^{13}$		$7,1\ 10^{13}$		$7,1\ 10^{13}$
- as HTO*		a) $6,9\ 10^{13}$ b) $6,7\ 10^{13}$ c) $2,3\ 10^{13}$		a) $6,8\ 10^{15}$ b) $2,4\ 10^{15}$ c) $7,1\ 10^{13}$		a) $5,8\ 10^{16}$ b) $3,4\ 10^{15}$ c) $7,2\ 10^{13}$		a) $2,4\ 10^{17}$ b) $3,6\ 10^{15}$ c) $7,2\ 10^{13}$
- as HT	$4,7\ 10^{15}$		$9,6\ 10^{15}$		$9,6\ 10^{15}$		$9,6\ 10^{15}$	
C 14								
- unspecified or in volatile form	$5,3\ 10^{16}$	$1,0\ 10^{12}$	$9,6\ 10^{16}$	$1,9\ 10^{12}$	$9,6\ 10^{16}$	$1,9\ 10^{12}$	$9,6\ 10^{16}$	$1,9\ 10^{12}$
- volatile portion ≤ 10 %		$1,0\ 10^{13}$		$1,9\ 10^{13}$		$1,9\ 10^{13}$		$1,9\ 10^{13}$
- volatile portion ≤ 1 %		$1,0\ 10^{14}$		$1,9\ 10^{14}$		$1,9\ 10^{14}$		$1,9\ 10^{14}$
J 129								
- unspecified		$2,1\ 10^{10}$		$2,1\ 10^{12}$		$2,1\ 10^{13}$		$2,1\ 10^{14}$
- on silver containing filters from off gas purification in reprocessing plants		$2,1\ 10^{12}$						
Ra 226								
- non-immobilized		$1,0\ 10^{10}$		$5,2\ 10^{13}$		$5,2\ 10^{14}$		$5,2\ 10^{15}$
- immobilized		$1,6\ 10^{11}$						
β/γ-emitters		d) $4,2\ 10^{18}$ e) $4,2\ 10^{16}$		d) $4,2\ 10^{20}$ e) $4,2\ 10^{18}$		d) $4,2\ 10^{20}$ e) $4,2\ 10^{18}$		d) $4,2\ 10^{20}$ e) $4,2\ 10^{18}$
α-emitters		$2,1\ 10^{17}$		$2,1\ 10^{19}$		$2,1\ 10^{19}$		$2,1\ 10^{19}$

* Concentration of tritium (as HTO) in the water resp. the residual moisture of the waste form ≤ $1,1 \cdot 10^{13}$ Bq/m³.
Total activity in the waste form without tritium activity: a) < 10^{10} Bq b) 10^{10} Bq to < 10^{12} Bq c) ≥ 10^{12} Bq.
Percentage (mass) of water resp. residual moisture in the waste form d) < 1 % e) ≥ 1 %.

TABLE 2.27 : ACTIVITY LIMITING VALUES FOR KEY NUCLIDES AND NON- SPECIFIED FURTHER α and β/ EMITTERS RESULTING FROM THE INCIDENT ANALYSIS

Data in Bq per waste package

Radionuclide/ Radionuclide group	Waste Class I					Waste Class II
	01 Bitumen and plastic products	02 Solid matter	03 Metallic solid matter	04 Compacted waste	05, 06 Cemented/ concreted wastes; concentrates	Waste form groups 01 to 06
Cl 36	$7,3\ 10^{8}$	$7,3\ 10^{8}$	$7,3\ 10^{8}$	$7,3\ 10^{8}$	$7,3\ 10^{8}$	$1,2\ 10^{11}$
J 129	$2,7\ 10^{9}$	$2,7\ 10^{9}$	$2,7\ 10^{9}$	$2,7\ 10^{9}$	$2,7\ 10^{9}$	$4,4\ 10^{11}$
Ra 226	$4,9\ 10^{8}$	$2,4\ 10^{10}$	$6,0\ 10^{10}$	$1,6\ 10^{11}$	$4,9\ 10^{11}$	$8,1\ 10^{12}$
Sr 90	$1,9\ 10^{9}$	$8,9\ 10^{10}$	$2,2\ 10^{11}$	$5,8\ 10^{11}$	$1,9\ 10^{12}$	$3,1\ 10^{13}$
Cm 248	$2,6\ 10^{9}$	$1,3\ 10^{11}$	$3,2\ 10^{11}$	$8,1\ 10^{11}$	$2,6\ 10^{12}$	$4,3\ 10^{13}$
Pb 210	$3,1\ 10^{9}$	$1,6\ 10^{11}$	$4,0\ 10^{11}$	$9,6\ 10^{11}$	$3,1\ 10^{12}$	$5,1\ 10^{13}$
Pa 231	$3,3\ 10^{9}$	$1,6\ 10^{11}$	$4,0\ 10^{11}$	$1,0\ 10^{12}$	$3,3\ 10^{12}$	$5,1\ 10^{13}$
Th 232	$4,4\ 10^{9}$	$2,2\ 10^{11}$	$5,6\ 10^{11}$	$1,4\ 10^{12}$	$4,4\ 10^{12}$	$7,3\ 10^{13}$
Ra 228	$4,9\ 10^{9}$	$2,4\ 10^{11}$	$6,2\ 10^{11}$	$1,6\ 10^{12}$	$4,9\ 10^{12}$	$8,1\ 10^{13}$
Pu 239	$6,7\ 10^{9}$	$3,3\ 10^{11}$	$8,1\ 10^{11}$	$2,1\ 10^{12}$	$6,7\ 10^{12}$	$1,1\ 10^{14}$
Ac 227	$9,6\ 10^{9}$	$4,7\ 10^{11}$	$1,2\ 10^{12}$	$2,9\ 10^{12}$	$9,6\ 10^{12}$	$1,6\ 10^{14}$
Tc 99	$1,1\ 10^{10}$	$5,6\ 10^{11}$	$1,4\ 10^{12}$	$3,6\ 10^{12}$	$1,1\ 10^{13}$	$1,9\ 10^{14}$
Am 242m	$1,5\ 10^{10}$	$7,4\ 10^{11}$	$1,9\ 10^{12}$	$4,7\ 10^{12}$	$1,5\ 10^{13}$	$2,4\ 10^{14}$
Cs 137	$4,0\ 10^{10}$	$1,9\ 10^{12}$	$4,9\ 10^{12}$	$1,3\ 10^{13}$	$4,0\ 10^{13}$	$6,7\ 10^{14}$
Co 60	$4,9\ 10^{10}$	$2,4\ 10^{12}$	$6,2\ 10^{12}$	$1,6\ 10^{13}$	$4,9\ 10^{13}$	$8,1\ 10^{14}$
further alpha-emitters	$6,7\ 10^{9}$	$3,3\ 10^{11}$	$8,1\ 10^{11}$	$2,1\ 10^{12}$	$6,7\ 10^{12}$	$1,1\ 10^{14}$
further beta/gamma-emitters	$4,9\ 10^{10}$	$2,4\ 10^{12}$	$6,2\ 10^{12}$	$1,6\ 10^{13}$	$4,9\ 10^{13}$	$8,1\ 10^{14}$

Waste form groups and packagings have to comply with the following quality characteristics :

TABLE 2.28 : QUALITY CHARACTERISTICS OF THE WASTE PRODUCT GROUPS DERIVED FROM INCIDENT ANALYSES

WASTE PRODUCT GROUP	QUALITY CHARACTERISTICS
Ø1	- Basic requirements
Ø2	- Basic requirements - In the waste form, limitation of the activity portions of flammable waste with a melting point < 3ØØ°C to ≤ 1°C - Immobilization of the above-mentioned flammable substances so that they do not emerge from the waste form when they become fluid under thermal load.
Ø3	- Basic requirements - Waste form is metal or consists of materials from a reactor core, graphite excepted
Ø4	- Basic requirements - Waste form has been compacted with a pressure ≥ 3Ø MPa so that it is solidly stable

TABLE 2.28
(ct'd)

WASTE PRODUCT GROUP	QUALITY CHARACTERISTICS
05	- Basic requirements - Waste is immobilized in cement or concrete - Embedded or solidifed wastes, e.g., ashes, powder or aqueous concentrates, is uniformly and completely distributed in cement or concrete - Cast waste, e.g., scrap, is distributed in the waste form as uniformly as possible - The compressive strength of the waste form is ≥ 10 N/mm² or the portion in the volume of the waste matrix is > 40 %
06	- Basic requirements - The waste itself consists of a non-powdery due of evaporated liquids or dried sludges - The waste is not combustible

TABLE 2.29 : QUALITY CHARACTERISTICS OF THE PACKAGINGS DERIVED FROM INCIDENT ANALYSES

PACKAGINGS	QUALITY CHARACTERISTICS
CLASS 1	- Basic requirements - Up to an impact speed of ≤ 4 m/s, the integrity of the packaging is preserved to the extent that in the case of subsequent thermal influence, the access of oxygen to the waste form is limited so that flammable waste forms with melting points > 300°C do not burn away with a naked flame, by pyrolyse
CLASS 2	- Basic requirements - The packaging resists a drop from a height of 5 m on to an inflexible surface so that the leak rate (related to standard conditions as in the leakage test by the vacuum method) after the drop does not exceed the value of 1×10^{-4} Pam^3/s. (This does not apply to waste immobilized in bitumen if the waste is uniformly and completely distributed in the bitumen). - The package wall has a thermal resistance (layer thickness times reciprocal thermal conductivity) of ≥ 0.1 m^2K/W (lost concrete shieldings). - The leak rate (related to standard conditions as in the leakage test by the vacuum method) is < 1×10^{-5} Pam^3/s. In the case of fire with a temperature of 800°C during one hour, the integral leakage of the gas released from the packaging during the fire and a cooling phase of 24 hours is < 1 mole.

For the long-term stability of the Konrad repository the thermal influence on the host rock can theoretically be of some influence. In order to find a limiting value it was concluded that the temperature influence would be limited if the temperature variation of the rock would be small compared with those caused by mine ventilation at natural temperature changes.

This marginal condition is considered to be met with a temperature variation of 3 K inside the rock. Thus it is requested that the average temperature rise at the side wall of the waste emplacement rooms does not exceed 3 K.

If a homogeneous distribution of the heat sources along an emplacement room is assumed, a limiting heat production can be determined for an individual radionuclide or a radionuclide mixture, so that a temperature rise of 3 K at the side wall of the room is not exceeded during a certain period of the time which was fixed at 100.000 years.

Thus the limiting heat production is defined as the maximum permissible heat output of a radionuclide per unit length of the emplacement room.

These values can be converted into length related activity limiting values.

TABLE 2.30 : ACTIVITY VALUES FOR KEY NUCLIDES AND OTHER UNSPECIFIED α - AND β/ - EMITTERS, RESULTING FROM THE SAFETY ASSESSMENT OF THE THERMAL INFLUENCE OF THE HOST ROCK

Data in Bq per waste package

Radio-nuclide	Activity limiting value rel. to drift length in Bq/m	Activity value Cylindrical concrete packaging type I	type II	type III	Cylindrical cast iron packaging type I	type II	type II*
Th 232	1.3E+11	6.8E+09	7.4E+09	1.6E+10	4.3E+09	7.4E+09	6.8E+09
U 235	1.5E+11	7.4E+09	8.1E+09	1.7E+10	4.7E+09	8.1E+09	7.4E+09
U 233	1.8E+11	9.0E+09	9.8E+09	2.1E+10	5.7E+09	9.8E+09	9.0E+09
Th 230	1.9E+11	9.7E+09	1.1E+10	2.2E+10	6.1E+09	1.1E+10	9.7E+09
Pa 231	2.0E+11	1.0E+10	1.1E+10	2.4E+10	6.5E+09	1.1E+10	1.0E+10
U 234	2.6E+11	1.3E+10	1.4E+10	3.0E+10	8.3E+09	1.4E+10	1.3E+10
Cm 248	3.0E+11	1.5E+10	1.7E+10	3.5E+10	9.7E+09	1.7E+10	1.5E+10
Np 237	3.3E+11	1.7E+10	1.8E+10	3.9E+10	1.1E+10	1.8E+10	1.7E+10
Cm 247	3.5E+11	1.8E+10	1.9E+10	4.1E+10	1.1E+10	1.9E+10	1.8E+10
Pu 244	4.7E+11	2.4E+10	2.6E+10	5.5E+10	1.5E+10	2.6E+10	2.4E+10
Ra 226	4.7E+11	2.4E+10	2.6E+10	5.5E+10	1.5E+10	2.6E+10	2.4E+10
U 238	5.2E+11	2.7E+10	2.9E+10	6.1E+10	1.7E+10	2.9E+10	2.7E+10
Cm 245	9.0E+11	4.6E+10	5.0E+10	1.1E+11	2.9E+10	5.0E+10	4.6E+10
Ac 227	2.5E+12	1.3E+11	1.4E+11	3.0E+11	8.1E+10	1.4E+11	1.3E+11
Am 242M	3.6E+12	1.8E+11	2.0E+11	4.2E+11	1.2E+11	2.0E+11	1.8E+11
Ra 228	3.7E+12	1.9E+11	2.1E+11	4.3E+11	1.2E+11	2.1E+11	1.9E+11
Nb 94	5.0E+12	2.5E+11	2.8E+11	5.9E+11	1.6E+11	2.8E+11	2.5E+11
Pu 238	8.9E+12	4.5E+11	4.9E+11	1.0E+12	2.8E+11	4.9E+11	4.5E+11
Pb 210	1.5E+13	7.5E+11	8.1E+11	1.7E+12	4.7E+11	8.1E+11	7.5E+11
Ca 41	1.7E+13	8.5E+11	9.2E+11	2.0E+12	5.4E+11	9.2E+11	8.5E+11
Ag 108M	2.6E+13	1.3E+12	1.4E+12	3.0E+12	8.3E+11	1.4E+12	1.3E+12
Cl 36	2.6E+13	1.3E+12	1.4E+12	3.0E+12	8.3E+11	1.4E+12	1.3E+12
Be 10	2.6E+13	1.3E+12	1.4E+12	3.0E+12	8.3E+11	1.4E+12	1.3E+12
Sn 126	3.3E+13	1.7E+12	1.8E+12	3.9E+12	1.1E+12	1.8E+12	1.7E+12
Rb 87	3.7E+13	1.9E+12	2.1E+12	4.4E+12	1.2E+12	2.1E+12	1.9E+12
Co 60	5.2E+13	2.6E+12	2.9E+12	6.1E+12	1.7E+12	2.9E+12	2.6E+12
Ar 39	5.3E+13	2.7E+12	2.9E+12	6.2E+12	1.7E+12	2.9E+12	2.7E+12
Cs 137	8.8E+13	4.5E+12	4.9E+12	1.0E+13	2.8E+12	4.9E+12	4.5E+12
Ni 63	7.5E+14	3.8E+13	4.1E+13	8.7E+13	2.4E+13	4.1E+13	3.8E+13
Fe 55	2.9E+16	1.5E+15	1.6E+15	3.4E+15	9.4E+14	1.6E+15	1.5E+15
Alpha	1.2E+12	6.2E+10	6.8E+10	1.4E+11	4.0E+10	6.8E+10	6.2E+10
Bet/Gam	6.6E+13	3.4E+12	3.7E+12	7.8E+12	2.1E+12	3.7E+12	3.4E+12

TABLE 2.30 (Ctd) : ACTIVITY VALUES FOR KEY NUCLIDES AND OTHER UNSPECIFIED α - AND β/ - EMITTERS, RESULTING FROM THE SAFETY ASSESSMENT OF THE THERMAL INFLUENCE OF THE HOST ROCK

Data in Bq per waste package

Radio-nuclide	Cylindrical cast iron packaging type III	Activity value Container type I	type II	type III	type IV	type V	type VI
Th 232	5.8E+09	2.0E+10	2.2E+10	4.8E+10	4.0E+10	5.6E+10	2.8E+10
U 235	6.3E+09	2.2E+10	2.4E+10	5.3E+10	4.4E+10	6.2E+10	3.1E+10
U 233	7.7E+09	2.7E+10	2.9E+10	6.4E+10	5.3E+10	7.5E+10	3.7E+10
Th 230	8.3E+09	2.9E+10	3.1E+10	6.9E+10	5.7E+10	8.0E+10	4.0E+10
Pa 231	8.7E+09	3.0E+10	3.3E+10	7.2E+10	6.0E+10	8.4E+10	4.2E+10
U 234	1.1E+10	3.9E+10	4.2E+10	9.3E+10	7.7E+10	1.1E+11	5.4E+10
Cm 248	1.3E+10	4.5E+10	5.0E+10	1.1E+11	9.1E+10	1.3E+11	6.3E+10
Np 237	1.4E+10	5.0E+10	5.4E+10	1.2E+11	9.9E+10	1.4E+11	7.0E+10
Cm 247	1.5E+10	5.3E+10	5.8E+10	1.3E+11	1.1E+11	1.5E+11	7.4E+10
Pu 244	2.0E+10	7.0E+10	7.7E+10	1.7E+11	1.4E+11	2.0E+11	9.8E+10
Ra 226	2.0E+10	7.1E+10	7.8E+10	1.7E+11	1.4E+11	2.0E+11	9.9E+10
U 238	2.3E+10	7.8E+10	8.6E+10	1.9E+11	1.6E+11	2.2E+11	1.1E+11
Cm 245	3.9E+10	1.3E+11	1.5E+11	3.2E+11	2.7E+11	3.8E+11	1.9E+11
Ac 227	1.1E+11	3.8E+11	4.1E+11	9.1E+11	7.6E+11	1.1E+12	5.3E+11
Am 242M	1.6E+11	5.4E+11	5.9E+11	1.3E+12	1.1E+12	1.5E+12	7.6E+11
Ra 228	1.6E+11	5.6E+11	6.1E+11	1.3E+12	1.1E+12	1.6E+12	7.8E+11
Nb 94	2.2E+11	7.5E+11	8.2E+11	1.8E+12	1.5E+12	2.1E+12	1.1E+12
Pu 238	3.8E+11	1.3E+12	1.5E+12	3.2E+12	2.7E+12	3.7E+12	1.9E+12
Pb 210	6.4E+11	2.2E+12	2.4E+12	5.3E+12	4.4E+12	6.2E+12	3.1E+12
Ca 41	7.2E+11	2.5E+12	2.7E+12	6.0E+12	5.0E+12	7.0E+12	3.5E+12
Ag 108M	1.1E+12	3.9E+12	4.2E+12	9.3E+12	7.8E+12	1.1E+13	5.4E+12
Cl 36	1.1E+12	3.9E+12	4.2E+12	9.3E+12	7.8E+12	1.1E+13	5.4E+12
Be 10	1.1E+12	3.9E+12	4.3E+12	9.3E+12	7.8E+12	1.1E+13	5.5E+12
Sn 126	1.4E+12	5.0E+12	5.4E+12	1.2E+13	1.0E+13	1.4E+13	7.0E+12
Rb 87	1.6E+12	5.6E+12	6.1E+12	1.3E+13	1.1E+13	1.6E+13	7.8E+12
Co 60	2.2E+12	7.8E+12	8.5E+12	1.9E+13	1.6E+13	2.2E+13	1.1E+13
Ar 39	2.3E+12	8.0E+12	8.7E+12	1.9E+13	1.6E+13	2.2E+13	1.1E+13
Cs 137	3.8E+12	1.3E+13	1.4E+13	3.2E+13	2.6E+13	3.7E+13	1.8E+13
Ni 63	3.2E+13	1.1E+14	1.2E+14	2.7E+14	2.2E+14	3.1E+14	1.6E+14
Fe 55	1.3E+15	4.4E+15	4.8E+15	1.1E+16	8.8E+15	1.2E+16	6.2E+15
Alpha	5.3E+10	1.8E+11	2.0E+11	4.4E+11	3.7E+11	5.2E+11	2.6E+11
Bet/Gam	2.9E+12	1.0E+13	1.1E+13	2.4E+13	2.0E+13	2.8E+13	1.4E+13

2.5.5. The basic dimensional requirements for the packages acceptable in the Konrad iron-mine

The Konrad repository is planned to handle 3400 transport units/a in one-shift operation. A transport unit is a standardized container or one pool pallet loaded with up to three standardized cylindrical packagings.

In order to facilitate the handling on the surface and underground and to reduce personnel exposure it was necessary to limit the packaging sizes in a way which would also give a maximum use factor of the emplacement chambers.

The corresponding optimization studies resulted in the following standard dimensions of the packagings given in the table following TABLE 2.31 :

TABLE 2.31 : STANDARDIZED PACKAGINGS FOR THE DISPOSAL OF RADIOACTIVE WASTES IN THE KONRAD RESPOSITORY

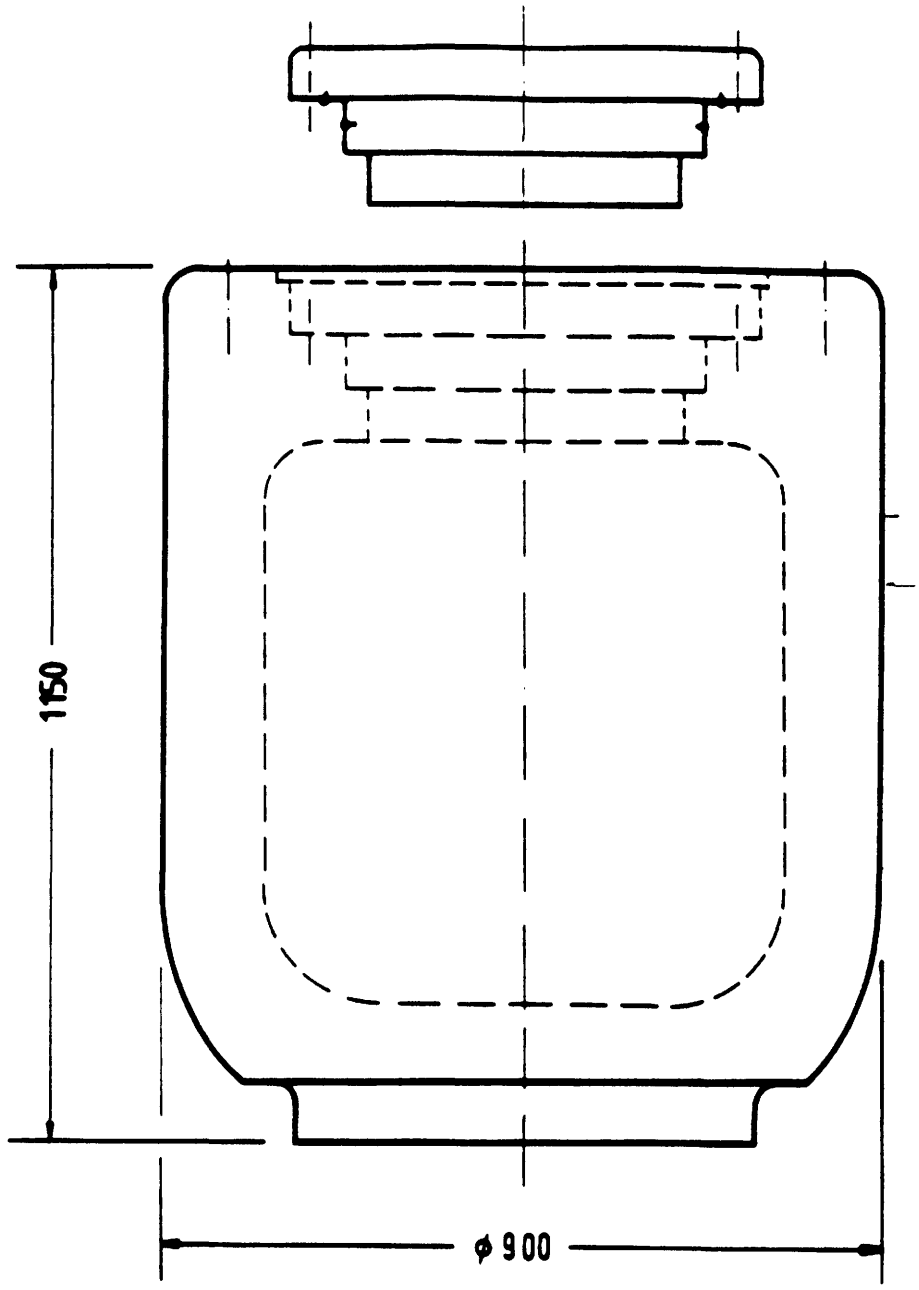
N°	DESIGNATION	EXTERNAL DIMENSIONS			GROSS VOLUME
		LENGTH/ DIAMETER mm	WIDTH mm	HEIGHT mm	m^3
	Concrete container (1)				
1	Type I	ø 1060	-	1370(3)	1.2
2	Type II	ø 1060	-	1510(4)	1.4
3	Type III	ø 1400	-	2000	3.1
	Cast iron container (1)				
4*	Type I	ø 900	-	1150	0.7
5*	Type II	ø 1060	-	1500(5)	1.3(1.2)
6	Type III	ø 1000	-	1240	1.0
	Large rectangular container(2)				
7	Type I	1600	1700	1450(6)	3.8
8	Type II	1600	1700	1700	4.6
9	Type III	3000	1700	1700	8.7
10*	Type IV	3000	1700	1450(6)	7.4
11*	Type V	3200	2000	1700	10.9
12	Type VI	1600	2000	1700	5.4

(1) Delivery on pool pallet
(2) Container materials : steel, cast iron, reinforced concrete
(3) Height : 1370 mm + 90 mm lifting lug = 1460 mm
(4) Height : 1510 mm + 90 mm lifting lug = 1600 mm
(5) And 1370 mm height, type KfK
(6) Stacking height : 1400 mm

* Packagings used for the study.
FIGURES 2.32, 2.33, 2.34, 2.35 show their main characteristics.

FIGURE 2.32 : CYLINDRICAL CAST IRON PACKAGING TYPE I (NR.4)

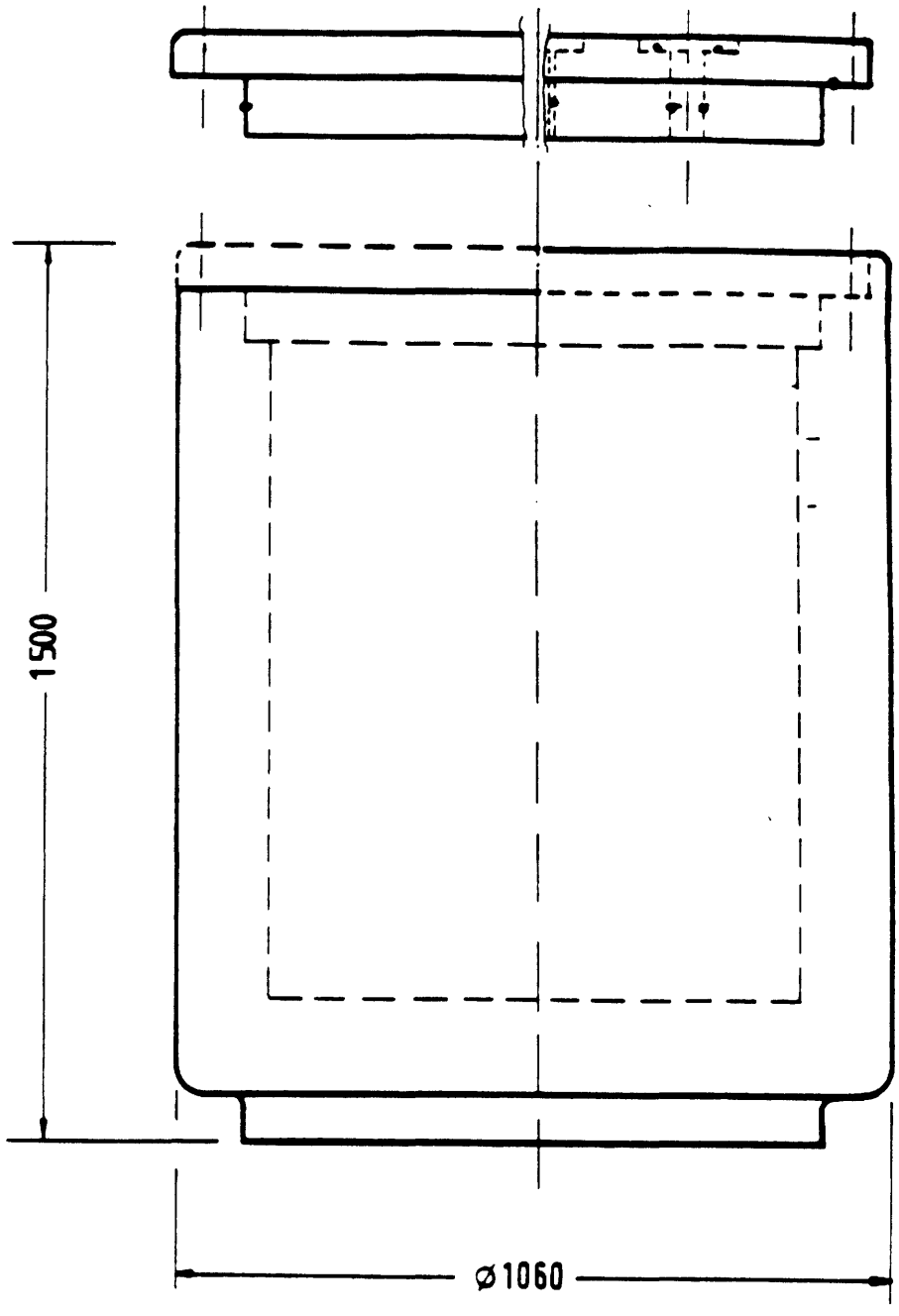
Example : Inserted and screwed lid



Height: 1150 mm
Diameter: 900 mm
Gross volume: 0,7 m³

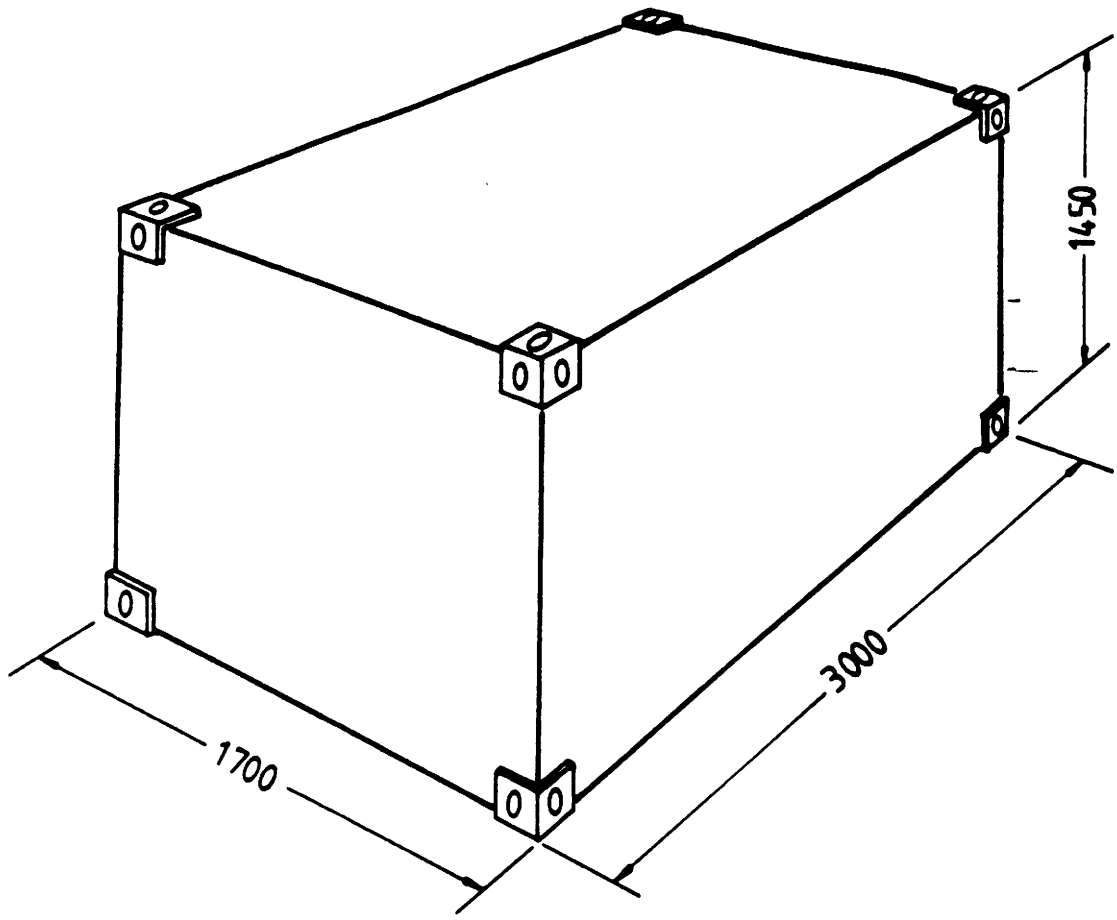
FIGURE 2.33 : CYLINDRICAL CAS IRON PACKAGING TYPE II (NR.5)

Example : Inserted/placed-on screwed lid with openings



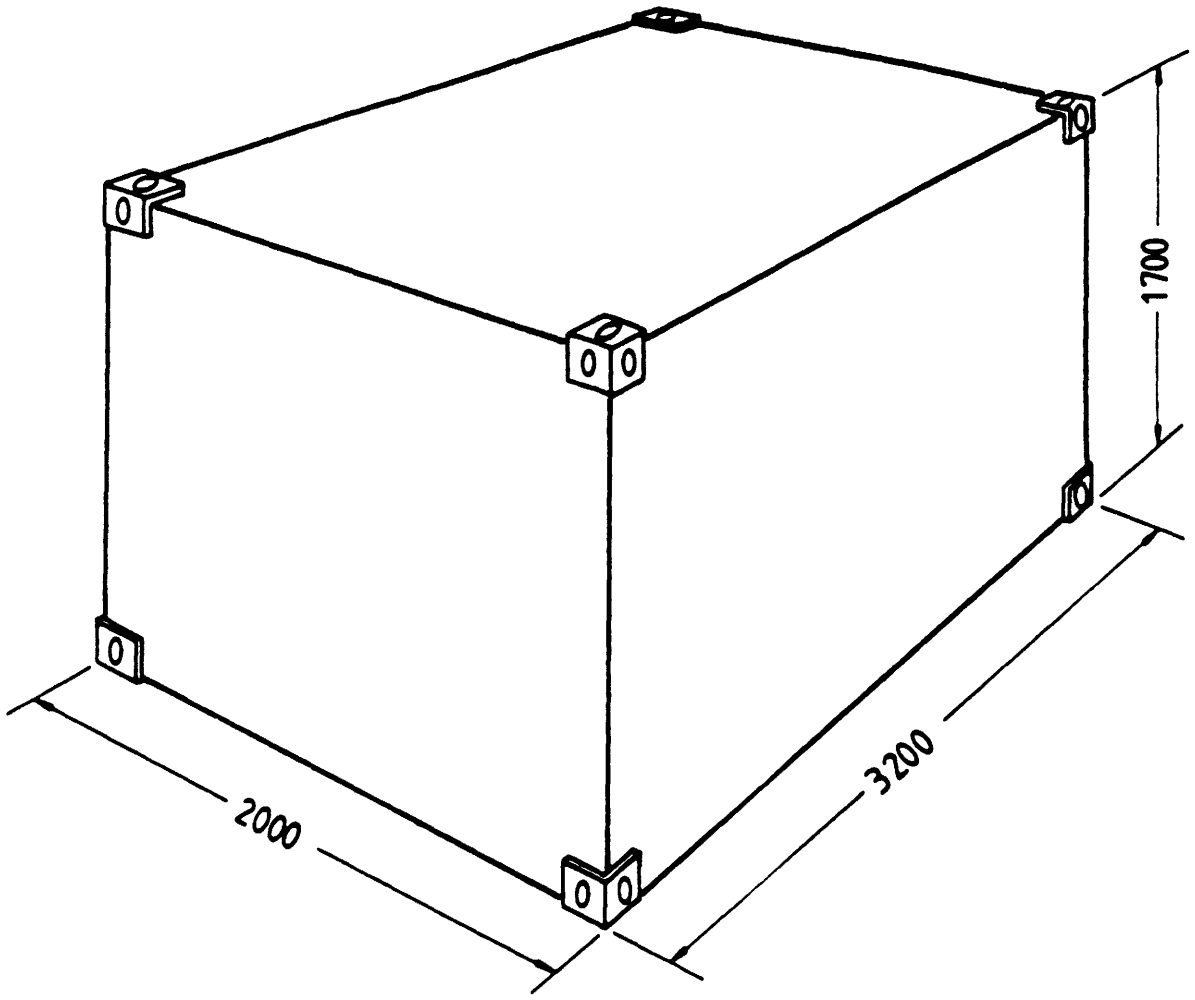
Height: 1500 mm (1370 mm for KfK type)
Diameter: 1060 mm
Gross volume: 1,3 m^3

FIGURE 2.34 : CONTAINER TYPE IV (NR. 10)



Length:	3000 mm
Width:	1700 mm
Height:	1450 mm (Stacking height: 1400 mm for KfK type)
Gross volume:	7,4 m^3
Attachment points:	ISO corner fittings accoriing to DIN 15 190, part 1 (preliminary standard)

FIGURE 2.35 : CONTAINER TYPE V (NR. 11)



Lenght:	3200 mm
Width:	2000 mm
Height:	1700 mm
Gross volume:	10,9 m^3
Attachment points:	ISO corner fittings according to DIN 15 190, part 1 (preliminary standard)

2.5.6. The purpose and extent of quality control

The goals fixed for the safe operations of the Konrad-mine can only be met, if all the specifications applying to the waste treatment processes and the packagings are carefully respected.

In order to assure this, PTB has charged the Nuclear Research Centre (KFA) Jülich to execute the quality control of all radioactive waste packages prior to their being shipped to the repository.

Radioactive waste packages for disposal must meet the waste acceptance requirements of the repository derived from safety considerations for the repository.

14 relevant properties (paragraphe 4.1) have been derived from the safety aspects.

The following institutions are involved in the Quality Control system :

The waste producers :

* are responsible for the quality of their waste package,
* have to take technical, organizational and administrative measures to fulfil the waste acceptance requirements (W.A.R.),
* have to demonstrate the observance of the W.A.R.

The PTB :

* must ensure that the waste package fulfil the W.A.R.,
* supervises the quality control group at the KFA and the DBE (see below),
* decides on process qualification, inspections and release of waste packages for disposal,
* decides in the case of faults and defects.

The quality control group of the PTB :

* checks existing documentations from the waste producers,
* performs random tests on waste packages,
* performs the qualification and inspection of waste conditioning processes.

The repository operating company (Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe mbH, DBE) :

* operates the repositories on behalf of the PTB,

* performs the entry control of waste packages (cheks on delivery documents and on waste packages with respect to radiation protection requirements).

The supervisory function of the Federal State (Länder) concerning waste conditioning processes is not affected by these regulations.

In thoses cases, where the waste is packaged without conditioning i.e. within a type B(U) cask, the QC is executed primarily during fabrication of the cask itself. Here the QC role implies the producer, the license holders and the licensing authority.

2.5.7. Sampling system

The following criteria have to be taken into account to establish the number of random tests.

* Documentation (extent and plausibility).

* Relevant properties, dependent on waste form group and waste class.

* Number of waste packages in a lot, number of batches, waste packages of the same type, waste packages from the same waste producer or conditioner.

* Radiological relevance according to three classes :

  a) ≤ 10 % of the guaranteed values for normal operation thermal and criticality limits.

     ≤ 1 % of incidents limits.

  b) ≤ 100 % of the guaranteed values for normal operation thermal and criticality limits.

     ≤ 10 % of incidents limits.

  c) > 100 % of the guaranteed values for normal operation thermal and criticality limits.

     ≤ 100 % of incidents limits.

Samples are selected by checking the documentation, visual inspections of waste packages, and statistical methods.

The quantity of samples and the necessary checks can be decreased if checks have been performed by independent authorities such as the IAEA or by post-qualification of the conditioning process.

There is also the possibility of decreasing the quantity of samples to be examined if after several examinations no waste packages with relevant faults were found.

Waste packages with relevant faults must not be shipped to the repository. If relevant faults are found during quality control, the respectibe batch will be identified and can be refitted by the waste producer. In addition, a second random test will be made and the respective waste package will be accepted if this test conforms to the waste acceptance requirements. The entire lot of waste packages is only accepted if no waste package from the lot has relevant faults.

3. GERMAN WASTE MANAGEMENT SCHEME FOR PWR'S

3.1. GENERAL

The treatment systems described are based on a 1300 MWe PWR. There is no significant difference of the equipment type and size for a 1300 MWe and a 900 MWe power plant. The German plants have got individual treatment systems for each unit which is different compared to the French and Belgian situation.

Primary coolant treatment system, liquid waste treatment system and off-gas system are located in the Reactor Auxiliary Building (RAB).

On FIGURE 3.1., a typical equipment arrangement in the RAB is shown.

The volume of the building for auxiliary systems is 53000 m^3.

On FIGURE 3.2., a general flow diagram of the nuclear auxiliary systems is shown. For the study, the following systems are required :

- Ventilation (KPV).

- Off-gas system (KPL).

- Primary coolant treatment system :

  . coolant degasification system (KRG),
  . coolant storage (KBB),
  . coolant treatment (KBF).

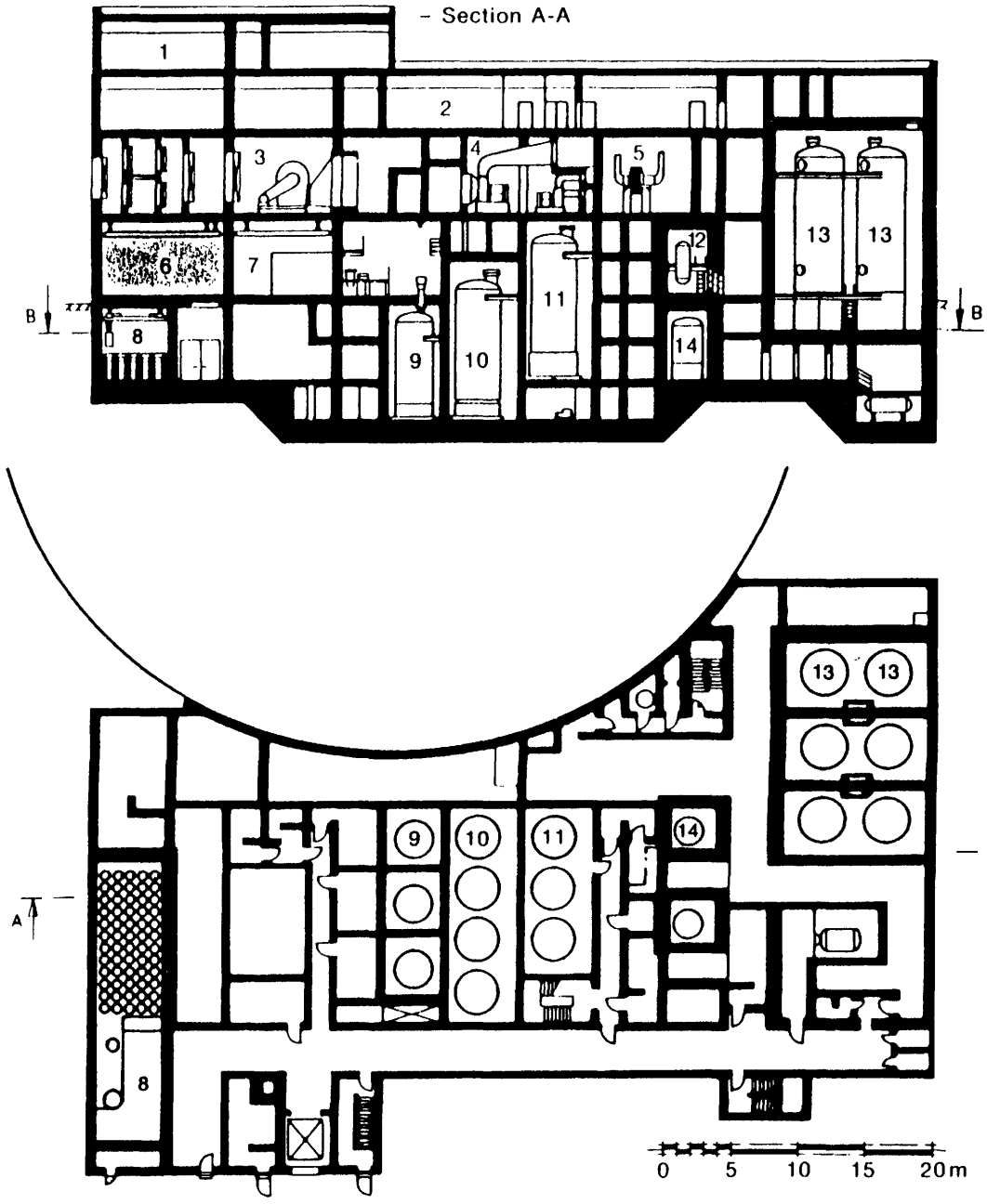
- Liquid waste treatment system (KPF) :

  . liquid waste storage and treatment,
  . control before release,
  . concentrate storage.

- Concentrate treatment system (KPC).

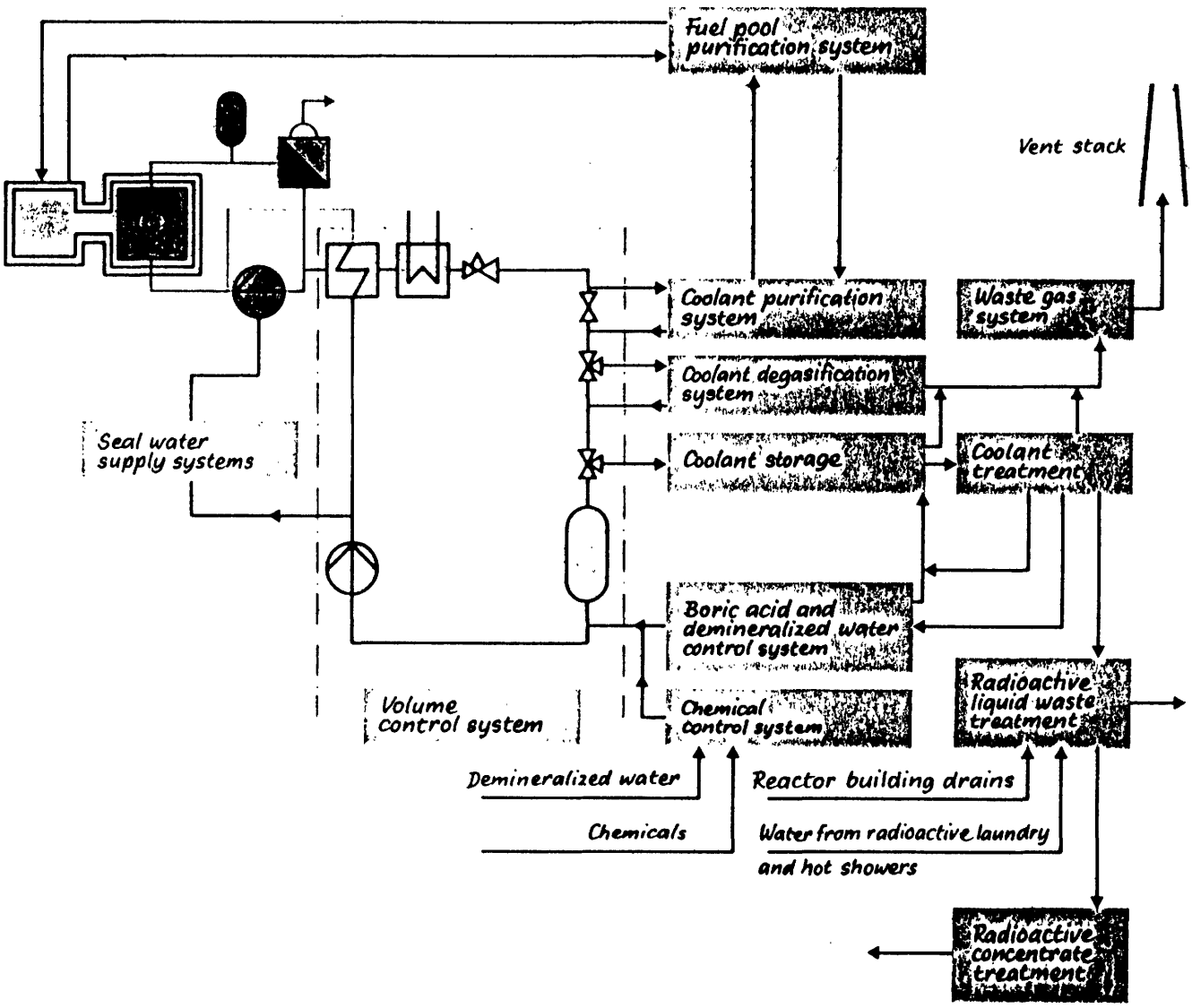
- Solid waste treatment systems (conditioning + packaging).

FIGURE 3.1 : REACTOR AUXILIARY BUILDING CROSS SECTIONS



1 Hot laboratory area
2 Personnel facilities area
3 Intake air system
4 Exhaust air system
5 Subatmospheric pressure ventilation system
6 Hot workshop
7 Decontamination room
8 Drum store for intermediate-level radioactive waste
9 Concentrate tank
10 Liquid waste storage tank
11 Monitoring tank
12 Mixed bed filter
13 Coolant storage tank
14 Spent resin tank

FIGURE 3.2 : THE NUCLEAR AUXILIARY SYSTEMS FLOW DIAGRAM



3.2. GASEOUS EFFLUENTS

3.2.1. Ventilation system

3.2.1.1. Process description

A flow diagram of ventilation (values issued from German PWR operation) is shown on FIGURE 3.3.

The air surrounding the radioactive zone may contain radioactivity which may escape through holes in the protective barriers or during maintenance work. Particles in the atmosphere surrounding the plant may also become radioactive.

Ventilation systems must therefore be provided to prevent concentration and uncontrolled release of radioactivity into the atmosphere.

Appropriate ventilation of regions of high and low radioactivity is ensured by a low pressure gradient of 1/10 mbar between limited access zones, unlimited access zones, surrounding areas in the reactor building and the atmosphere (these are known as pressure barriers). The exhaust air from the limited access areas is sent through aerosol and activated charcoal filters located in the subatmospheric pressure section, and is released through the stack after a control of iodine, noble gases and aerosol activity.

If, in the event of an accident, the pressure in the containment building rises, all the outlets in the ventilation system close automatically. The containment spray system is put into operation simultaneously, ensuring a low pressure barrier between the reactor building and the atmosphere outside the building. Uncontrolled radioactivity emissions are also prevented in the event of an external accident, e.g. plane crash.

Purification of the air in order to contain the radioactivity is based on two main principles :

- filtration (Fibreglass filters for aerosol removal),

- adsorption (Activated charcoal filters for iodine removal).

The size and shape of the particles to be removed determines the operational mode of the filter. Gas dynamics also enable small particles to be contained, depending on the size of the filter medium pores. The so-called deep bed filters are generally made out of many layers of small-pored mats, e.g. fibre-glass mats.

In order to contain radioactive iodine, activated charcoad is used. Activated charcoal has a very great internal surface area, ideal for the adsorption of iodine. The activated charcoal is impregnated with silver

nitrate, which is converted to silver iodide on contact with iodine. This stable compound chemically binds the iodine and enables a filter efficiency rating of 99.9 % to be achieved.

3.2.1.2. Block diagram

Following general assumptions have been taken into account :

- Iodine is filtered through activated charcoal
  (efficiency = 99.9 %)

- Aerosols are filtered through fibre-glass
  (efficiency = 99 %)

FIGURE 3.3 : VENTILATION FLOW DIAGRAM

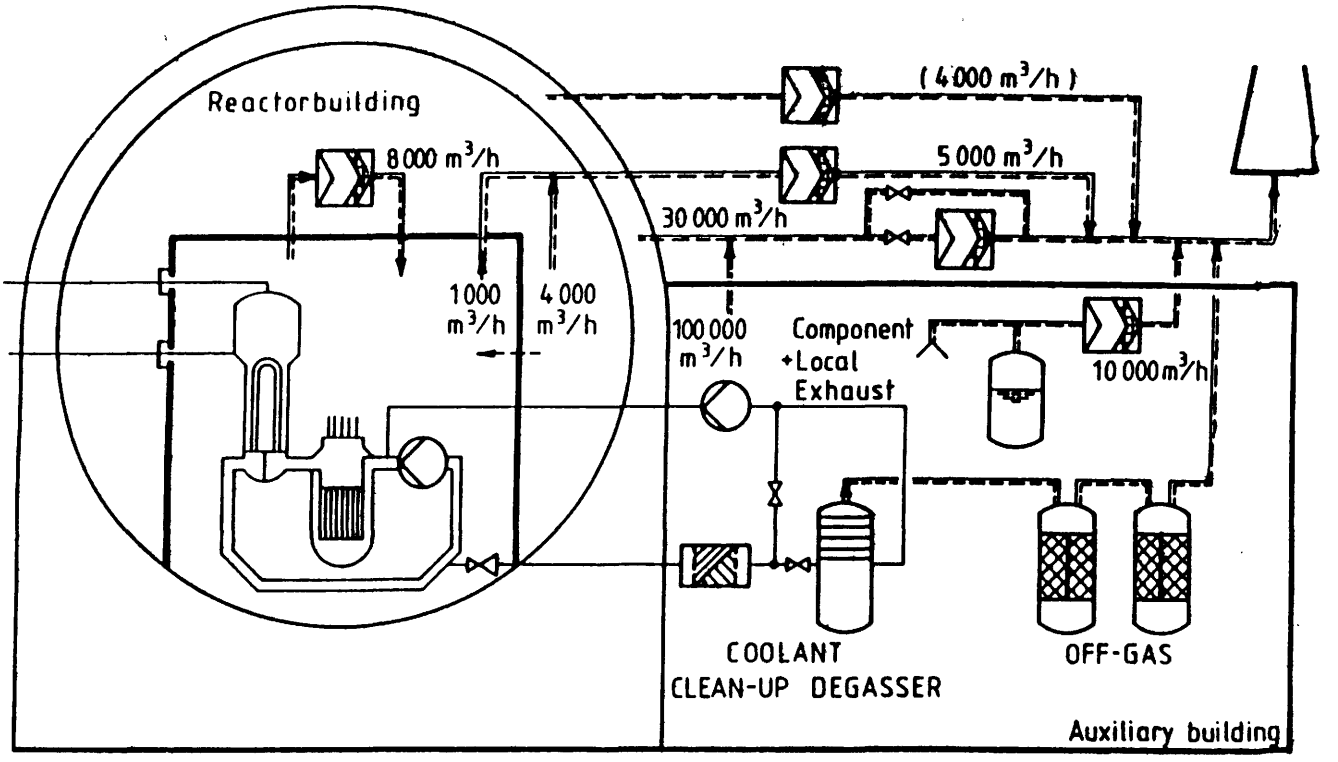


FIGURE 3.4 : VENTILATION BLOCK DIAGRAM

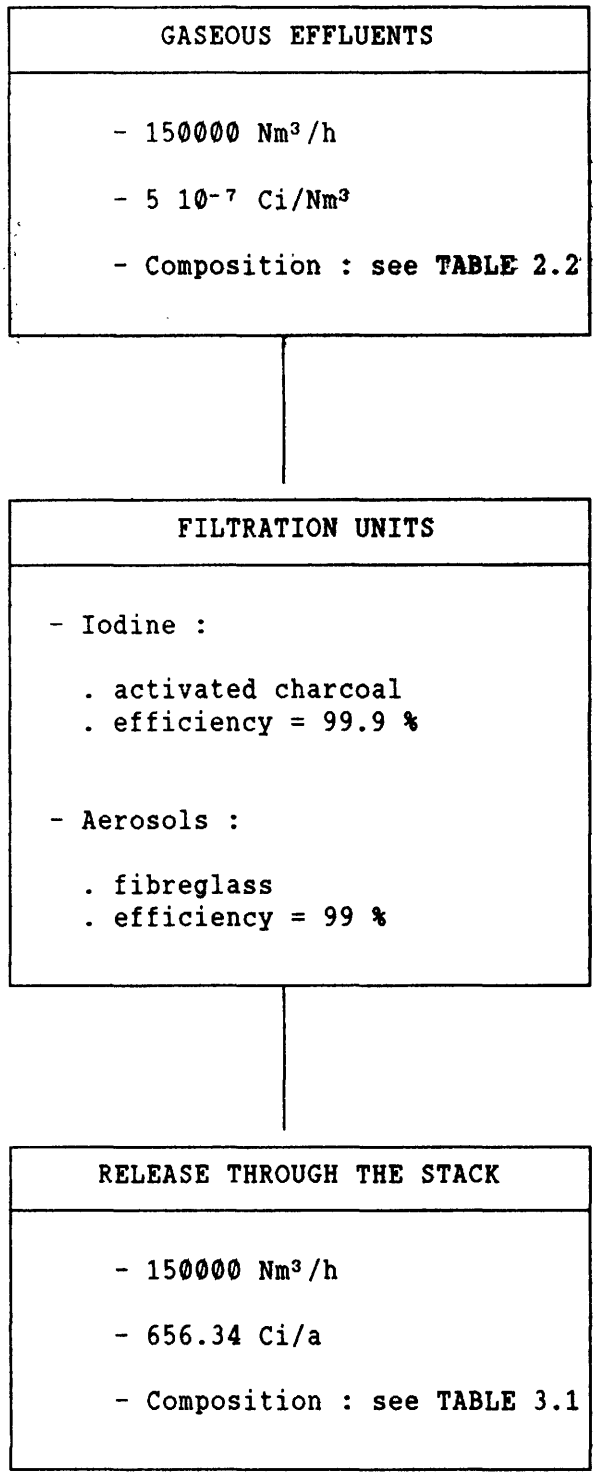


TABLE 3.1 : RADIONUCLIDE COMPOSITION AT VENTILATION EXIT

RADIONUCLIDES	INLET COMPOSITION (%)	INLET ACTIVITY (Ci/a)	FILTER EFFICIENCY (%)	OUTLET ACTIVITY (Ci/a)	OUTLET COMPOSITION (%)
C - 14	0.00001	$6.57\ 10^{-5}$	0	$6.57\ 10^{-5}$	0.00001
Kr - 85	0.03	0.20		0.20	0.03
Kr - 85 m	1.83	12.02	0	12.02	1.83
Kr - 87	1.25	8.21		8.21	1.25
Kr - 88	3.32	21.81		21.81	3.32
Xe - 133	80.41	528.29		528.29	80.49
Xe - 133 m	1.75	11.50		11.50	1.75
Xe - 135	11.31	74.31		74.31	11.32
I - 131	0.01	0.065		$6.57\ 10^{-5}$	0.00001
I - 132	0.02	0.13	99.9	$13.14\ 10^{-5}$	0.00002
I - 133	0.03	0.20		$19.71\ 10^{-5}$	0.00003
I - 134	0.01	0.065		$6.57\ 10^{-5}$	0.00001
I - 135	0.02	0.13		$13.14\ 10^{-5}$	0.00002
Aerosols	0.00001	$6.57\ 10^{-5}$	99	$6.57\ 10^{-7}$	0.0000001
TOTAL	100	657		656.34	100

3.2.2. Off-gas system

3.2.2.1. Process description

A flow diagram of oof-gas system is shown on **FIGURE 3.5.**

Radioactive gases evolve in the coolant under the influence of radiation release :

- fission gases from the fuel (Xenon-Krypton isotopes),

- activated gases from the coolant (oxygen-, hydrogen-, argon isotopes),

- hydrogen and oxygen from the radiation-included decomposition of water.

These gases are continuously drawn off or flushed out of the radioactive systems using nitrogen. The gaseous mixture is essentially help up in the off-gas system so that the activity can decline as much as possible. Hydrogen and oxygen are converted to water by means of a recombiner. Coolers and dryers remove the evaporated water, thereby reducing the gas volume. One part of gases is routed to a drying unit followed by an adsorption unit before release through the stack or recycling to the systems. The remaining gases are recycled without purification treatment to the systems.

3.2.2.1.1. Design objectives

Under normal operating conditions

The off-gas system must :

- prevent radioactive gas from escaping into the containment atmosphere by maintaining a negative pressure in the system,

- contain the relevant radioactive gases (Xe, Xr) before being released into the atmosphere, until the vast majority of radioactivity has been lost,

- keep the hydrogen content below 4 % by volume and the oxygen content below 0.1 % by volume in order to prevent formation of a combustible mixture, and also to prevent build up of oxygen in the coolant and avoid oxidation in the reactor coolant system.

In operation after a LOCA (Loss Of Coolant Accident)

The waste gas system must :

- recombine the hydrogen released into the reactor containment,

- supply air for ventilation of the reactor containment.

3.2.2.1.2.Equipment

The off-gas system is composed of the following main components :

- 1 gas dryer,

- 2 recombiners,

- 2 gas coolers,

- 3 off-gas compressors,

- 1 pre-dryer,

- 1 silica gel gas drying unit,

- 1 decay unit.

The components connected with the off-gas system are purged using nitrogen. These components mainly consist of :

- pressurizer relief tank,

- volume and control tank,

- reactor coolant purification system degasser,

- reactor coolant treatment system tanks, degasser and evaporator,

- drain tanks.

This purge or carrier gas is drawn by the off-gas compressor through the gas dryer, recombiner and the gas cooler (subatmospheric region of system). It contains H_2 and $\emptyset_2$ as well as radioactive gases (Xe, Kr).

The H_2 and $\emptyset_2$ concentrations are measured at the recombiner entry and corresponding amounts of $H \div 2 \approx$ and $\emptyset_2$ added, if necessary, to a obtain a stoichiometric ratio.

The H_2 and $\emptyset_2$ combine in the recombiner due to the action of a catalyst to form steam which is condensed in a gas cooler.

The hydrogen content is maintained below 4 % vol. by operating with one or two waste gas compressors, as necessary.

The purge gas is compressed by the off-gas compressor to 8 bar and then dried in the predryer.

The gas flow is then fed back through the reducing station into the subatmospheric section. The remaining gas flow is fed through the silica gel drying unit.

The dried gas is fed through the decay unit. The decay in the carrier gas N_2 of the radioactive gases (Xe, Kr) occurs in the decay tanks by adsorption on activated charcoal. The adsorption on active charcoal is a function of the weight of the gas molecules. The lighter hydrogen and oxygen molecules in the air pass through the decay unit faster than the much heavier Krypton and Xenon atoms (60 d retention time for Xe, 60 h for Kr).

After passing through the decay unit, the radioactivity of the gases is significantly reduced so that it can be released after reducing the pressure via the exhaust air system into the atmosphere or be fed back via a different pressure-reducing station to the subatmospheric section of the off-gas system.

In normal operation, only one recombiner train need be used for the waste gas system. The second, reserved for a loss of coolant accident, is used for a short time when necessary (fault in the operation of a recombiner train).

3.2.2.2. Block diagram

Following general assumptions have been taken into account :

- 120000 Ci/a are handled and released.

- Retention time in the decay unit :

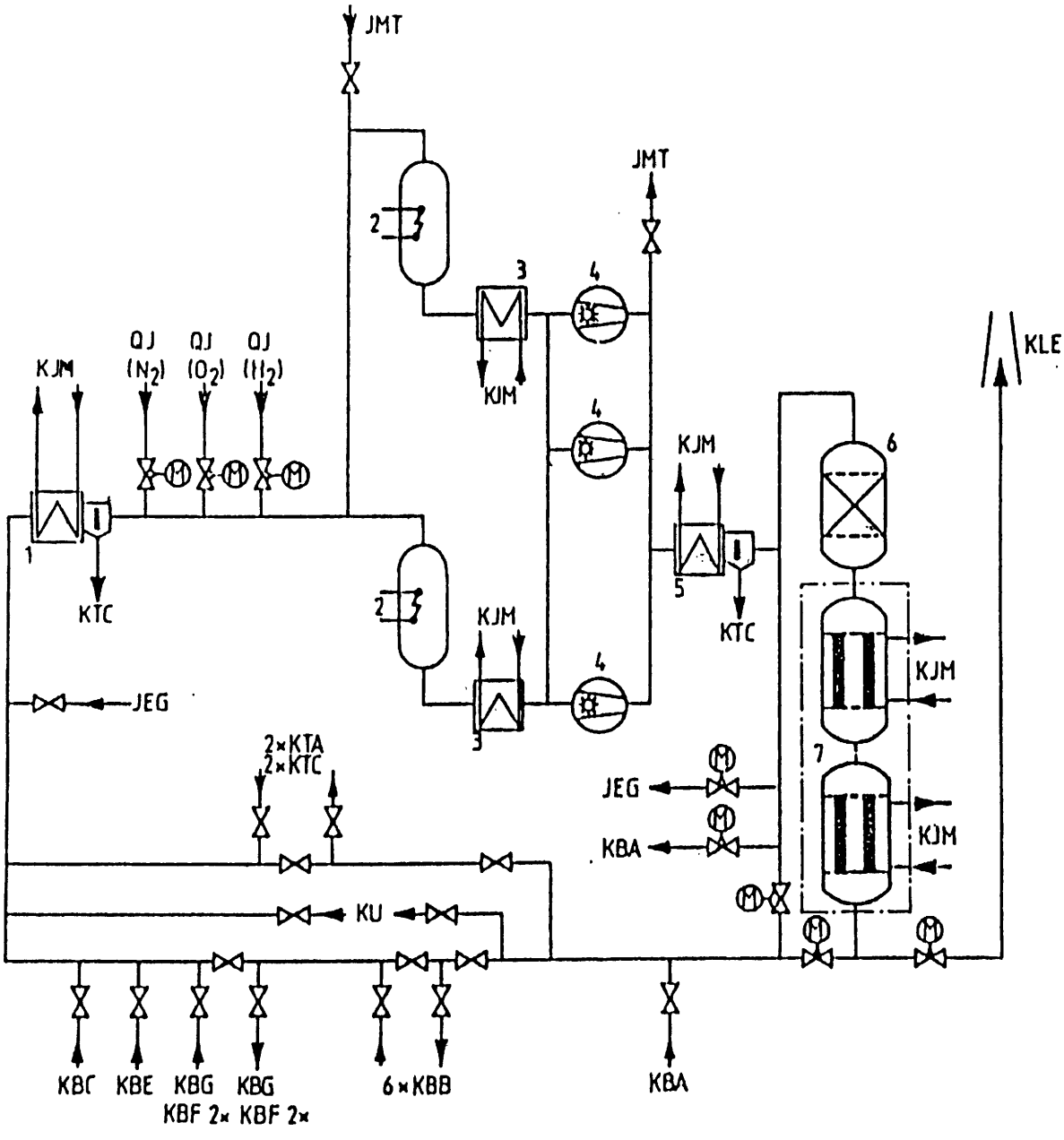
  60 days for Xenon
  60 hours for Krypton

- Iodine is retained and adsorbed in the decay unit (efficiency = 99.9 %).

- Aerosols are adsorbed in the decay unit
  (efficiency = 99 %)

- Activity : $A_t = A_o \, e^{\left(- \frac{\text{Log } 2}{T_{1/2}} t \right)}$

FIGURE 3.5 : SCHEME FOR OFF-GAS SYSTEM



1 - Gas dryer
2 - Recombiner
3 - Gas cooler
4 - Gas compressor
5 - Predryer
6 - Gel drying unit
7 - Decay unit

JEG : Primary coolant pressure relief system
JMT : Hydrogen reduction system
KBA : Volume control system
KBB : Primary coolant storage system
KBC : Boric acid and demineralized water feed system
KBE : Primary coolant purification system
KBF : Primary coolant treatment system
KBG : Primary coolant degassing system
KJM : Cooling water for off-gas system
KLE : Ventilation
KTA : Component drain sysem (reactor building)
KTC : Component drain system (reactor auxiliary system building)
KU : Sampling system
QJ : Gas feed system

FIGURE 3.6 : OFF-GAS SYSTEM BLOCK DIAGRAM

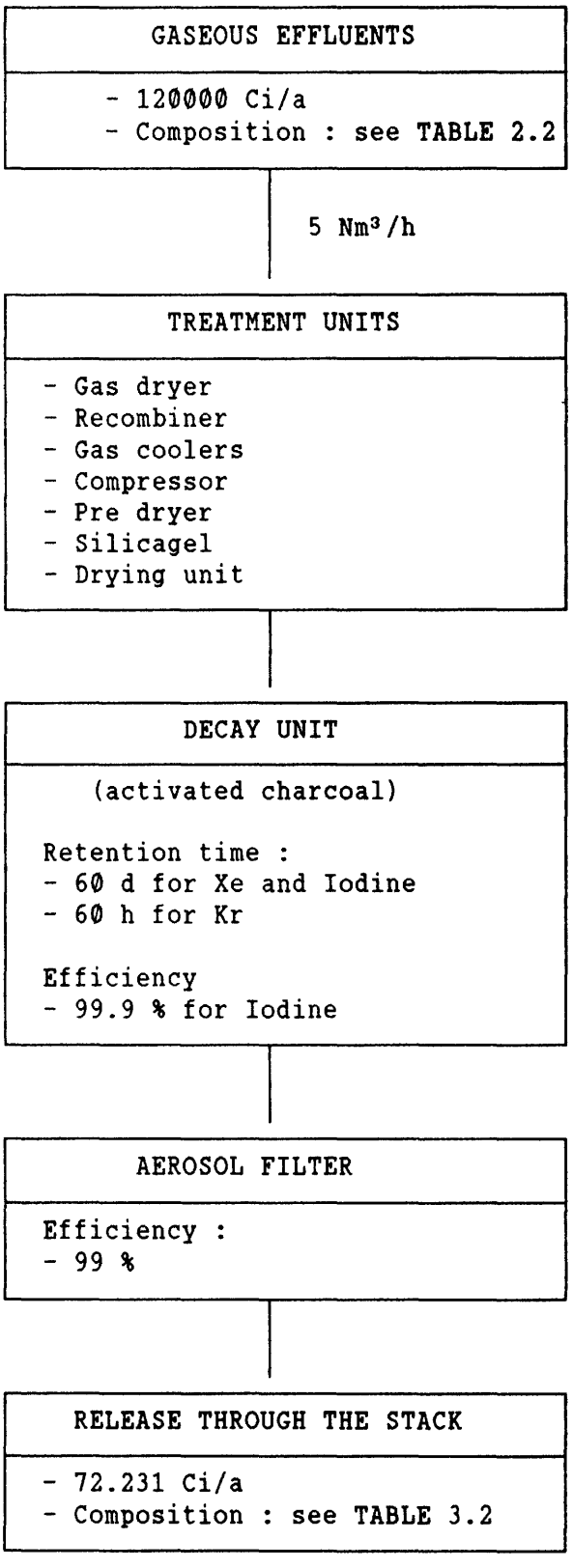


TABLE 3.2 : RADIONUCLIDE COMPOSITION AT OFF-GAS SYSTEM EXIT

RADIONUCLIDES	HALF LIFE	INLET COMPOSITION (%)	INLET ACTIVITY (Ci/a)	RETENTION TIME OR EFFICIENCY	OULET ACTIVITY (Ci/a)	OUTLET COMPOSITION (%)
C - 14	5700 y	0.00001	0.012	0 %	0.012	0.017
Kr - 85	10.6 y	0.03	36		35.984	49.818
Kr - 85 m	262 mn	1.83	2196	60 h	0.160	0.222
Kr - 87	78 mn	1.25	1500		0	0
Kr - 88	166 mn	3.32	3984		0.001	0.001
Xe - 133	5.27 d	80.41	96492	60 d	36.074	49.942
Xe - 133 m	2.3 d	1.75	2100		0	0
Xe - 135	552 mn	11.31	13572		0	0
I - 131	8.05 d	0.01	12		0	0
I - 132	138 mn	0.02	24	60 d	0	0
I - 133	1248 mn	0.03	36	99.9 %	0	0
I - 134	52.5 mn	0.01	12		0	0
I - 135	402 mn	0.02	24		0	0
Aerosols	-	0.00001	0.012	99 %	0.00012	0
TOTAL		100	120000		72.231	100

3.2.3. Comparison with discharge limits

The total released activity is as follows :

TABLE 3.3 : ACTIVITY RELEASED BY BOTH OFF-GAS SYSTEM AND VENTILATION

RADIONUCLIDES	ACTIVITY FROM OFF-GAS SYSTEM Ci/a	ACTIVITY FROM VENTILATION Ci/a	ACTIVITY RELEASED THROUGH THE STACK Ci/a	PER TYPE
C - 14	0.012	0	0.012	0.012
Kr - 85	35.984	0.20	36.184	
Kr - 85 m	0.160	12.02	12.18	
Kr - 87	0	8.21	8.21	
Kr - 88	0.001	21.81	21.811	728.56
Xe - 133	36.074	528.29	564.364	
Xe - 133 m	0	11.50	11.50	
Xe - 135	0	74.31	74.31	
I - 131	0	0	0	
I - 132	0	0	0	
I - 133	0	0	0	$< 10^{-3}$
I - 134	0	0	0	
I - 135	0	0	0	
Aerosols	0	0	0	$< 10^{-3}$
TOTAL	72.231	656.34	728.57	728.57

The activity released for each gaseous effluent through the stack is much lower than objective values :

TABLE 3.4 : COMPARISON WITH DISCHARGE LIMITS

GASEOUS EFFLUENTS	DESIGN VALUE Ci/a	OBJECTIVE VALUE Ci/a	RELEASE VALUE Ci/a
Noble gases	20000	2000	728.56
Halogens	0.3	0.02	$< 10^{-3}$
Aerosols	0.5	0.02	$< 10^{-3}$
Tritium	200	100	

3.3. LIQUID EFFLUENTS

3.3.1. Primary coolant treatment system

3.3.1.1. Process description

Primary Coolant degassing subsystem

A flow diagram of primary coolant degassing system is shown on FIGURE 3.7.

Before the reactor coolant system is opened up, for example to change a fuel element, the dissolved radioactive noble gases and hydrogen in the primary coolant must be removed in order to prevent them escaping into the atmosphere, and to reduce the direct radiation from the water.

The gases are removed in a vacuum degasser.

The degassing system is designed to take suction from two high pressure pumps (Chemical and Volume Control System), but the capacity can be increased using three high pressure pumps.

After purification, the primary coolant is fed to the degasser at a temperature of 50 °C. The boiling pressure corresponding to this temperature is maintained by means of a degasser vacuum pump.

The primary coolant is fed into the top of the degasser column and trickles over exchange plates into the bottom of the column in which approximately 3 % of the coolant is evaporated. This steam flows upwards towards the downward-trickling coolant, thereby stripping the gasses. The steam is condensed in the reflux condensor and is fed back to the top of the column. The gases are drawn from the gas cooler into the gaseous effluent system via the vacuum pump.

The degassed primary coolant in the bottom of the column is fed back to the chemical and volume control system using the degasser transfer pump.

Primary coolant storage and treatment subsystems

A flow diagram of primary coolant (storage and) treatment system is shown on FIGURE 3.8.

In the primary coolant storage subsystem, coolant removed during times of coolant exchnge - startup, compensation for burnup and load changes - and deborated primary coolant are temporarily stored, for reintroduction to the reactor coolant system as necessary.

In the primary coolant treatment subsystem, boric acid is removed from the coolant by evaporation and is collected in the boric acid tank as 4 % boric acid. It is also possible to degass the reactor coolant in a degasser if it is to be transferred to the liquid waste treatment system. The primary coolant make-up water is degassed using the same degasser.

The primary coolant is transferred to the evaporator because of the need to constantly adjust the amount of boric acid to compensate for burn-up and because of changes in load. The evaporator increases the boric acid concentration to 4 % ; the distillate has a maximum residual boron content of 1 ppm. Further treatment of the distillate using demineralizers to reduce the boron content is not necessary.

The degasser is designed to handle the same flow rate as the evaporator.

The primary coolant storage subsystem contains several coolant tanks, each of which is connected to the demineralized water or borated water. Motor-operated valves, actuated by an automatic control unit, permit changing the flow configuration to allow the tanks to handle either demineralized water or borated water. The level of the liquid can be read off the level indicators at the control station.

In order to be able to perform this operation, an empty coolant tank must be available for times when maximum amounts of coolant are produced for storage. The coolant to be treated is fed to the evaporator from the chemical and volume control system, temporarily stored in the coolant tanks and then pumped from the coolant tanks or sent directly from the chemical and volume control system into the coolant treatment system using the evaporator feed pump. The demineralized water is fed back into the chemical and volume control system using the transfer pump, which forms part of the chemical make-up system.

The coolant is then pumped through a demineralizer using the evaporator feed pump, before flowing into the evaporator. It is heated to approximately 90 °C by a steam-heated regenerating pre-heater before entering the evaporator column through a control valve. The solution is vaporized in the lower part of the evaporator by the steam rising through the plate column and is partly condensed in the regenerating pre-heater and finally fully condensed in the condenser.

In order to prevent steam from entering the off-gas system along with the gases released, the mixture of gas and steam in the condenser off-gas line is fed through a gas cooler. This condenses the steam still present after initial condensing and the gases are cooled to approximately 50 °C. The evaporator condensate passes through an after-cooler into the coolant storage subsystem. Some of the condensate is fed back into the top of the evaporator column. This reflux enters into contact with the rising steam, reducing the boric acid concentration in the steam. The steam vapor exits the column with a boron content lower than 1 ppm.

The boric acid concentration in the column bottom is maintained at 4 % using a flow regulator. The boric acid drawn off is fed into the boric acid tanks (which belong to chemical make-up system).

The condensate is fed into the degasser using the evaporator condensate pump. When degassing demineralized make-up water, the demineralized water is heated close to its boiling point and then fed into the degasser.

The coolant then trickles over baffles and is dispersed over as wide an area as possible. A small quantity is vaporized in the degassing evaporator and used as carrier gas for the fission gases drawn off. The steam is condensed in the degassing condenser. The gas drawn off is cooled to 50 °C in a gas cooler in the same way as for the evaporator column and then fed into the off-gas system. The degassed, demineralized water is fed into the coolant tank or into the radioactive liquid waste system using the degasser transfer pump.

3.3.1.2. Block diagram

The following assumptions have been taken into account :

Six storage tanks are available at the system inlet. Three of them are used to collect borated water from primary coolant system. The remaining ones store if necessary the distillate issued from evaporation before recycling.

The concentrate (4 % acid boric) is directed to the chemical make-up system boric acid tank.

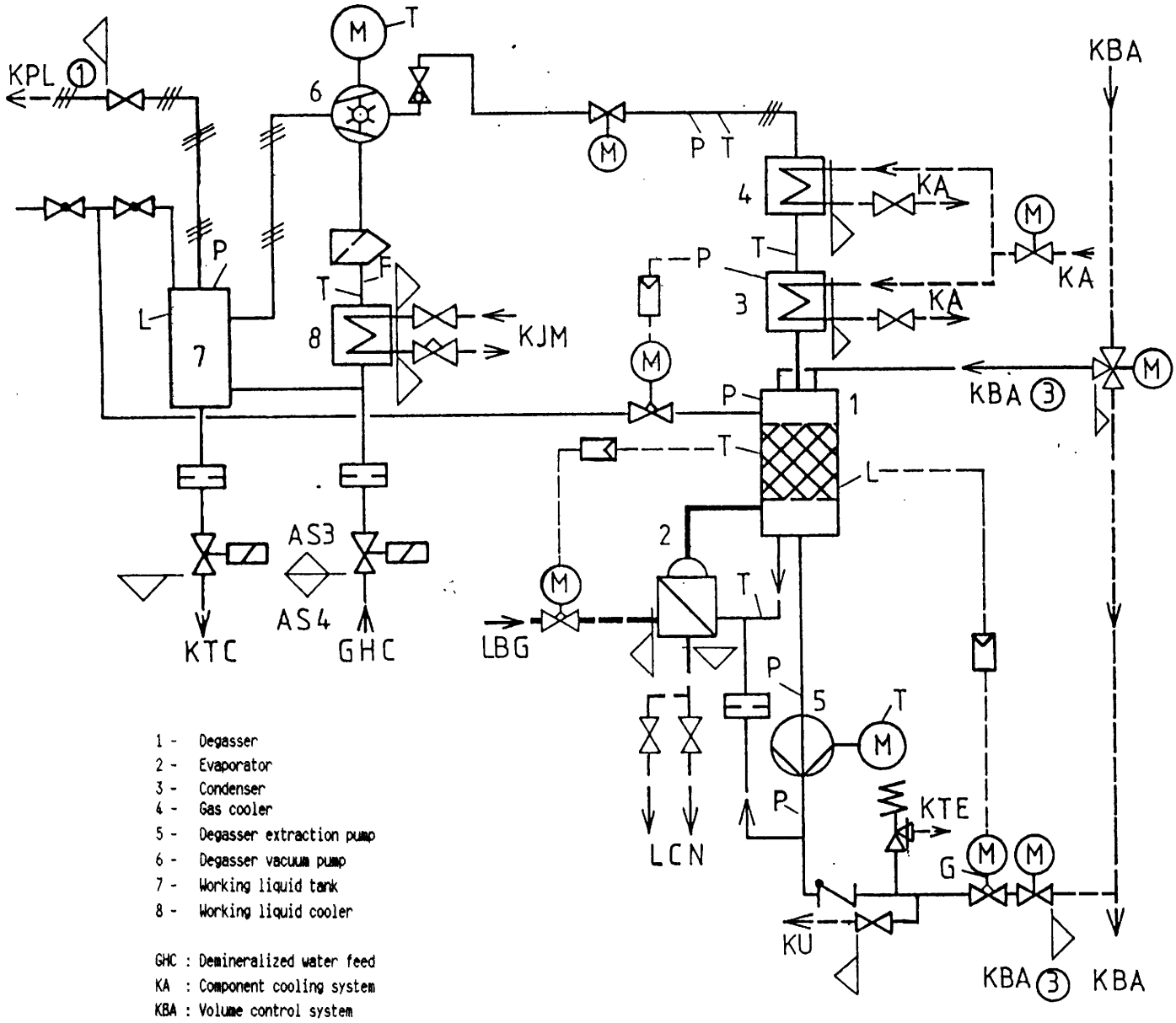
Primary coolant may be degassed by means of a degasser (see paragraph. 3.2.1.1., coolant degassing subsystem) directly connected to the chemical and volume control system (or primary coolant purification system) when the reactor is shut down. In this case, primary coolant is not directed to demineralization and evaporation but returns at once to the chemical and volume control system.

For our study, the amount of considered primary coolant (10000 m^3/a) is demineralized, evaporated and only degassed (by another degasser) if it is directed to the liquid waste treatment sytem.

To simplify the study, only the first degasser is considered, despite its greater capacity and even if it doesn't really belong to primary coolant treatment system.

Radionuclide composition is shown in TABLE 2.4.

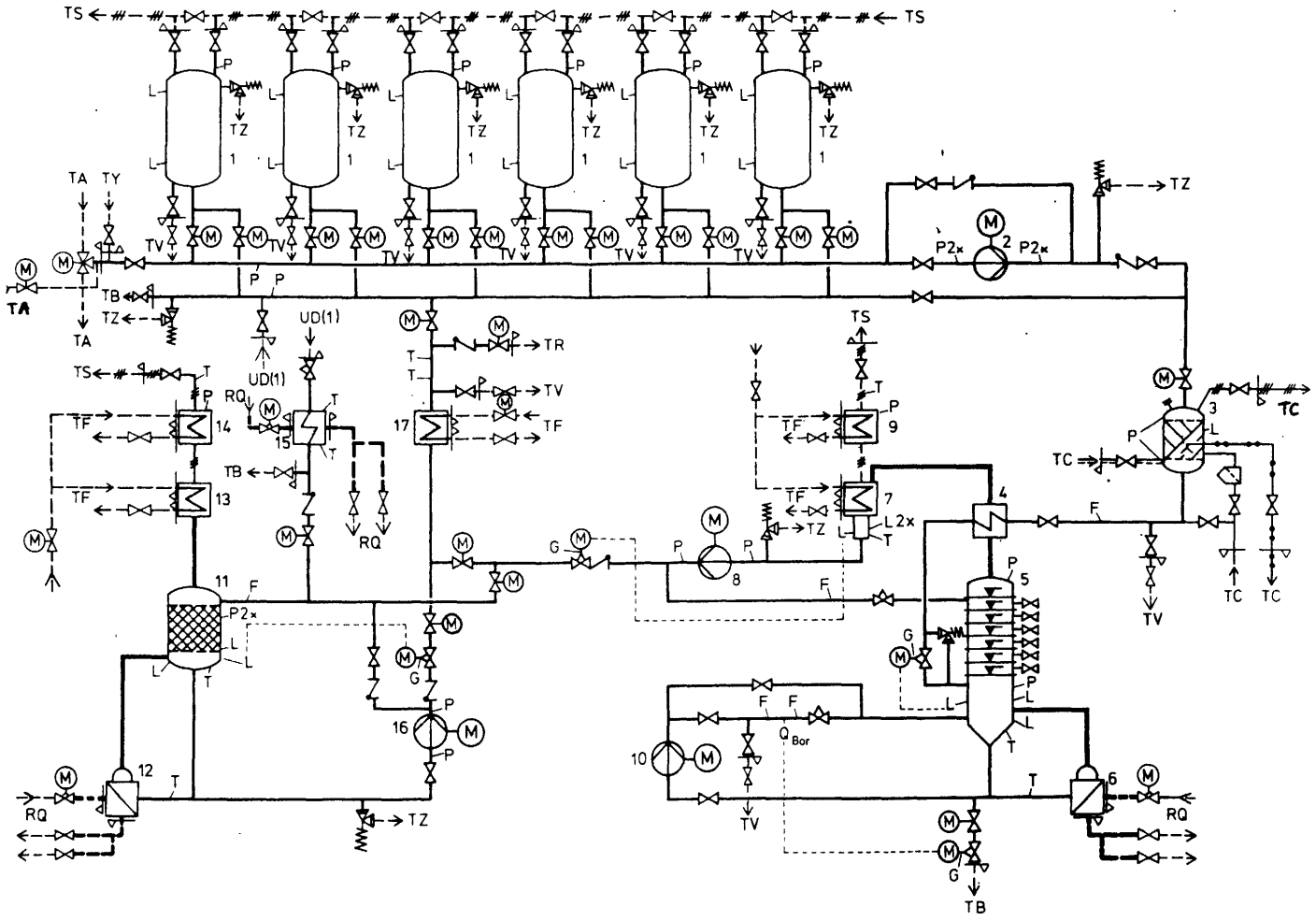
FIGURE 3.7 : PRIMARY COOLANT DEGASSING SYSTEM FLOW DIAGRAM



1 - Degasser
2 - Evaporator
3 - Condenser
4 - Gas cooler
5 - Degasser extraction pump
6 - Degasser vacuum pump
7 - Working liquid tank
8 - Working liquid cooler

GHC : Demineralized water feed
KA : Component cooling system
KBA : Volume control system
KJM : Cooling water for off-gas system
KTE : Disposal system for safety valves
KPL : Off-gas system
KU : Sampling system
KTC : Component drain system
(reactor auxiliary system building)
LCN : Condensation system
+++ : Fission gases + O_2 + H_2

FIGURE 3.8 : PRIMARY COOLANT TREATMENT SYSTEM FLOW DIAGRAM



1 - Primary coolant storage tank
2 - Evaporator feed tank
3 - Demineralizer

Boron preparation station (evaporator) :

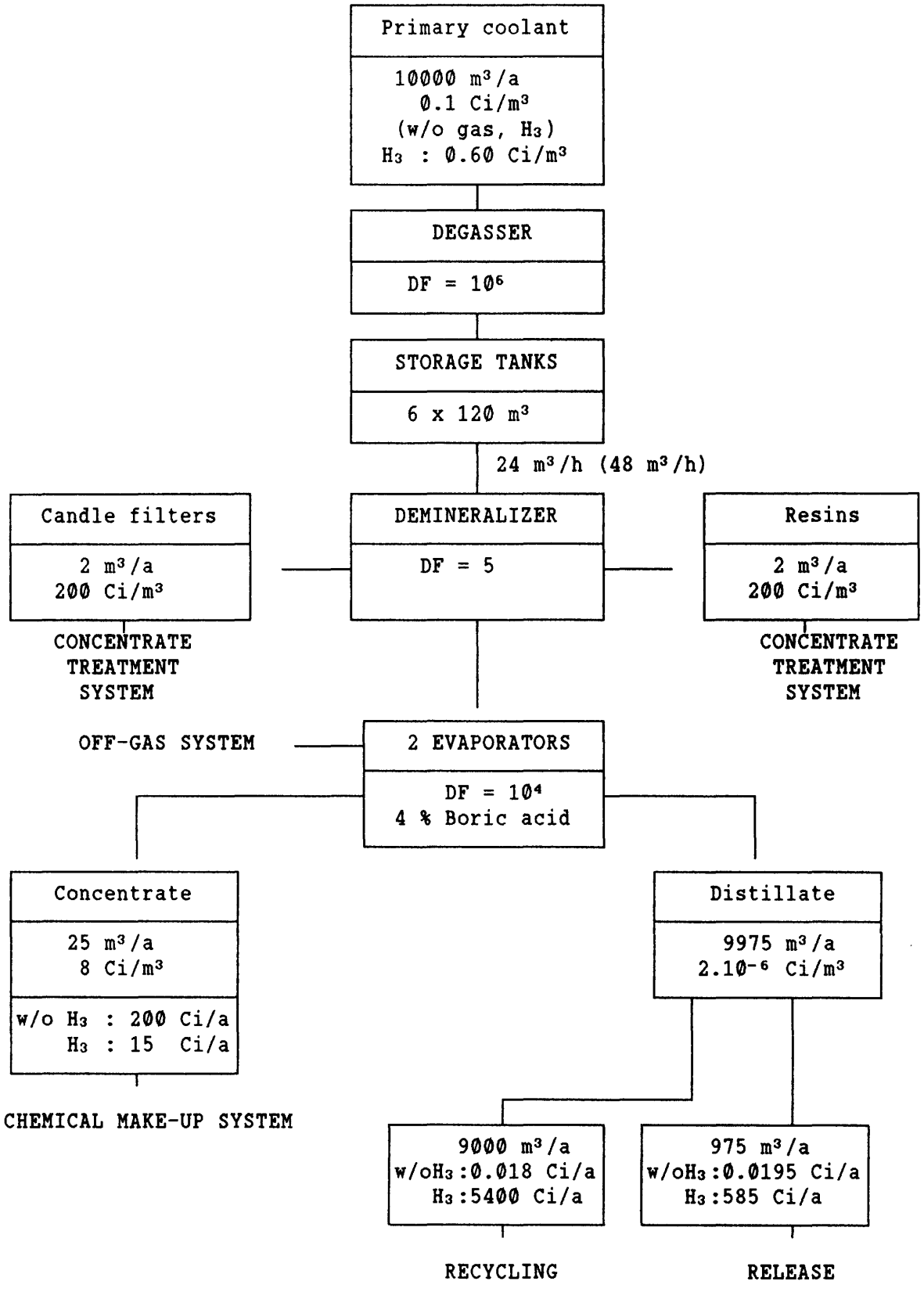
4 - Recuperative heat exchanger
5 - Stripping column
6 - Evaporator heating body
7 - Condenser
8 - Evaporator condensate pump
9 - Gas cooler
10 - Boron measurement pump

Degassing station :

11 - Degassing column
12 - Degasser heating body
13 - Degasser condenser
14 - Degasser gas cooler
15 - Feedwater preheater
16 - Degasser extraction pump

RQ : Auxiliary steam feed system
TA : Volume control system
TB : Chemicals
TC : Primary coolant purification system
TF : Cooling system
TR : Liquid waste treatment system
TS : Off-gas system
TV : Sampling system
TY : Waste water collection system (components)
TZ : Waste water collection system (rooms)
UD : Demineralized water feed system
+++ : Off-gas

FIGURE 3.9 : PRIMARY COOLANT TREATMENT SYSTEM BLOCK DIAGRAM



3.3.1.3. Releases

Primary coolant treatment system

Concentrate (recycled) :

- Volume : 25 m^3/a

- Activity (w/o H_3) : 8 Ci/m^3
  200 Ci/a

- Activity (H_3) : 0.6 Ci/m^3
  15 Ci/a

Distillate (recycled) :

- Volume : 9000 m^3/a

- Activity (w/o H_3) : 0.10^{-6} Ci/m^3
  0.018 Ci/a

- Activity (H_3) : 0.6 Ci/m^3
  5400 Ci/a

Distillate (released through liquid waste treatment system) :

- Volume : 975 m^3/a

- Activity (w/o H_3) : 2.10^{-6} Ci/m^3
  0.00195 Ci/a

- Activity (H_3) : 0.6 Ci/m^3
  585 Ci/a

3.3.2. Liquid waste treatment system

3.3.2.1. Process description

A flow diagram of liquid waste treatment system is shown on FIGURE 3.10.

Design objectives

The purpose of the radioactive liquid waste treatment system is to collect and treat all active and inactive liquid waste during normal plant operation and shutdown so that they can be discharged in conformity with the West German Nuclear Safety Authorities Regulation KTA 1504 "Measurement of radioactive fluids for the monitoring of radioactive discharge and the relevant plant water regulations".

Storage of radioactive liquid waste

Liquid waste collection tanks

A flow diagram of collection tanks is shown on FIGURE 3.11.

Such liquids are segregated into two groups according to their chemical properties and activity :

Group I - Active liquid waste

- Sump water from equipment rooms.
- Liquid waste from the chemical laboratories.
- Waste liquid from the decontamination room.
- Water from the spent fuel pit system.
- Decanted liquid from the liquid waste treatment system concentrate tanks.
- Concentrate treatment waste water.

Activity : 3.7 to $3.7.10^{3}$ Bq/cm^{3} (10^{-4} to 10^{-1} Ci/m^{3}).

Group II - Low active or inactive liquid waste

- Sump water from the operating rooms.
- Waste water from the laundry rooms.

- Waste water from the shower and wash rooms.

- Distillate from the primary coolant purification system evaporator.

- Regeneration and sluice water from the mixed-bed demineralizers.

- Regeneration and sluice water from the Steam Generator blowdown system demineralizers (inactive in normal conditions).

- Rinsing water from the electro-magnetic filter (inactive in normal conditions).

Activity : < 3.7 Bq/cm^3 ($< 10^{-4}$ Ci/m^3).

2 liquid waste collection tanks are provided for group I, and 3 for group II. As soon as one container is full it can be prepared for further processing.

A group II liquid waste collection tank is provided for the normally inactive liquid waste (regeneration and sluice water from the SG blowdown system demineralizers).

Liquids are now segregated according to new criteria : clear liquids are collected together and dirty ones are routed to the other tanks. The judgment is made by the operator based on his experience.

Five tanks are available. One of these tanks is used to store the usually inactive liquids coming from the regeneration and sluice water of S.G. blowdown system demineralizers and the rinse water of electromagnet filters. They are decanted/separated and then, only need a control before release.

Distillate from the primary coolant treatment system is either released to the discharge tanks or reused.

Concentrate tanks

The concentrate from the evaporation unit or filter residue from the pre-coat filters are collected in the concentrate tanks and prepared for further processing in the radioactive concentrate treatment system. There are 3 tanks for the evaporation unit concentrate. There is a further tank for the filtered matter from the pre-coat filters and the sludge from the collection tanks. Decanting nozzles are attached to the concentrate tanks for decanting after sedimentation of solid particles. The contents of the tanks are held in suspension during the transfer of the concentrate from the system using a motor-driven agitator and compressed air mixing.

Treatment of radioactive liquid waste

The following treatment processes are used according to the activity and chemical composition of the liquid waste :

- Evaporation.

- Pre-coat filtration or centrifuge.

- Mixed-bed demineralization.

- Chemical reduction in combination with another decontamination process.

Evaporation treatment

A flow diagram of an evaporation station is shown on FIGURE 3.12.

The highest degree of decontamination of all the treatment processes is achieved with the liquid waste evaporation units.

The liquid waste is fed from the liquid waste collection tanks using the evaporator feed pump into the evaporator column after analysis and, if necessary, after appropriate treatment with chemicals in the liquid waste collection tanks. During normal operation the feedrate to the evaporator is controlled on the basis of the liquid level in the evaporator column.

The liquid waste is vaporized during circulation on the shell side of the tube bundle heat exchanger. The steam produced separates from the circulating water in the evaporator column and flows through the gas stripper sieve plates from which it is released following the removal of entrained water droplets. The steam is entrained in the evaporator column through a demister in which any water droplets still present are separated. The water is returned to the evaporator bottom.

Some of the distillate condensed in the condensor is returned to the column as reflux while most of the distillate is degassed in a degasser, then cooled by the distillate cooler to below 45° C and collected in the control tanks.

The evaporator concentrate is fed discontinuously into the concentrate tanks. The degree of decontamination achievable with the evaporation unit is around 10^{-3} to 10^{-5} depending on the composition of the liquid waste. The concentrate produced has an average solid content of approximately 15 to 25 % by weight.

Steam generator blowdown sludge following a steam generator leak is radioactive and is normally treated in the evaporation unit.

Treatment using pre-coat filters

The purpose of the pre-coat filter is to reduce the radioactivity, mainly in suspension, in the liquid waste. Pre-coat filtering is performed after preliminary filtration to remove the large particles. Body feed, constantly added during filtration of the charged liquid, renews the filter coat. The remaining unfiltered water and the filter cake are sluiced into the concentrate tanks.

In the recent liquid waste processing systems, the pre-coat filter has been replaced by a centrifuge system.

A flow diagram of a centrifuge station is shown on FIGURE 3.13.

It consists of a decanter for premilinary cleaning followed by a separator. The solids separated in the decanter are removed of the equipment in dewatered conditions.

After pretreatment, the concentrate arising in the separator is routed back to the decanter for dewatering.

The amount of solid waste produced is about 1/10 of that produced using pre-coat filters.

Treatment by mixed-bed demineralizers

A flow diagram of a mixed bed demineralizer is shown on FIGURE 3.14.

Mixed-bed demineralization is used when radioactivity in the liquid waste in the control tanks is too great for it to be released. The mixed-bed demineralizers can only be used if the water to be decontaminated has a low conductivity (< 10 µ5cm⌠-1⌡). In this case, the following configuration is used : control tank - filter feed pump - mixed bed demineralizers - control tanks.

Chemical precipitation

With liquid waste containing activity slightly in excess of the allowable levels (due to dissolved activity), a chemical precipitation process can be performed before further treatment.

During chemical precipitation, the radionuclides form relatively insoluble precipitates with the reagents, and are difficult to dissolve. Other impurities are removed from the water by adsorption and the formation of solids, such as corrosion products.

The precipitation process occurs into the collection tanks whereby the decomposed sludge is deposited at the bottom of the conical section. The clarified water above can be decanted through the pre-coat filter and the deposited sludge released directly into the concentrates tank.

Operating modes

Mode I

The first group comprising active liquid waste without detergents is routed to the evaporation station.

The second group mainly comprising low activity liquid waste is sent either to the evaporation station or to the pre-coat filtering station, depending on the level of activity.

The third group comprising normally inactive effluents is routed to the filtration station to meet the specify limit for the release of metal impurities. In case of steam generator leakage, these effluents can be sent to the evaporation station.

Mode II

This way of liquid processing is now more and more carried out and will be used for the study.

The liquid waste (clear or dirty) stored in a collection tank are sampled. A chemical and radiochemical analysis is then carried out ; samples can be taken at various levels in the tank ; liquids are released at the corresponding levels and treated differently according to the results of the analyses.

Sludge (including precipitates) are collected at the bottom of the tank and routed to the sludge storage tank.

Treatment of radioactive concentrate

The evaporator or filter concentrate for treatment is fed from the liquid waste treatment system.

The remaining carrier liquid can be decanted off and replaced by evaporator concentrate.

In order to homogenize the concentrate and keep it in suspension, it is mixed using the agitator and simultaneously circulated by the concentrate circulation pump.

Bead resing metering

For conditioning, the bead resins can be added to the evaporator and filter concentrate. For this purpose, the bead resins from the reactor coolant purification system demineralizers for example, are sluiced into resin metering vessel out of the spent resin tank using the resin sluicing pump.

After the sluice water has been drawn off, the bead resins are fed into the concentrate hold-up tank and mixed into the concentrate using the agitator.

The bead resins may also be fed directly from purification system demineralizers into a mobile resin binding unit.

Measurement of solid content

The concentrate-bead resin suspension is fed into a graduated flask. After the feed inlet has been closed off, the sludge settles down to the bottom. The percentage of solids in suspension can then be read off from the height of the sludge.

Transfer of concentrate

The concentrate is prepared using a mobile conditioning unit inside the controlled access area in the main entrance to the reactor containment building. The mobile conditioning unit is connected to the transfer tanks by means of hoses. The concentrate prepared in the concentrate hold-up tank is circulated using the concentrate circulating pump as follows : concentrate hold-up tank - transfer tank - mobile preparation unit - transfer tank - concentrate hold-up tank. The amount of concentrate to be teated is removed from the system by means of a branch line.

3.3.2.2. Block diagrams

BLOCK DIAGRAM 1 : Typical German inventory
Operating mode I
See FIGURE 3.15
Radionuclide composition is shown in TABLE 2.5

BLOCK DIAGRAM 2 : Typical German inventory
Operating mode II
See FIGURE 3.16
Radionuclide composition is shown in TABLE 2.5

BLOCK DIAGRAM 3 : European inventory
Operating mode II
See FIGURE 3.17
Radionuclide composition is shown in TABLE 2.5

The liquid processing way detailed in this last block diagram is the one used for the study.

The following assumptions have been taken into account :

- Decontamination factors :

  . evaporation : DF = 10 000
  . centrifuge : DF = 20
  . demineralizer : DF = 10

- Liquid treatment :

  . Secondary drain wastes : 1000 m^3 --> evaporation
    1500 m^3 --> centrifuge

  . Chemicals : 1500 m^3 --> evaporation

  . Decontamination : 10 m^3 --> evaporation

  . Building wastes : 3000 m^3 --> centrifuge

  . Laundry wastes : 4000 m^3 --> centrifuge

Demineralization is not required. If it was used, very few amounts of resins would be produced and they would be together with concentrate and sludge.

Processed liquids are stored in 3 control tanks.

They are released if their activity is lower than 5.10^{-4} Ci/m^3.

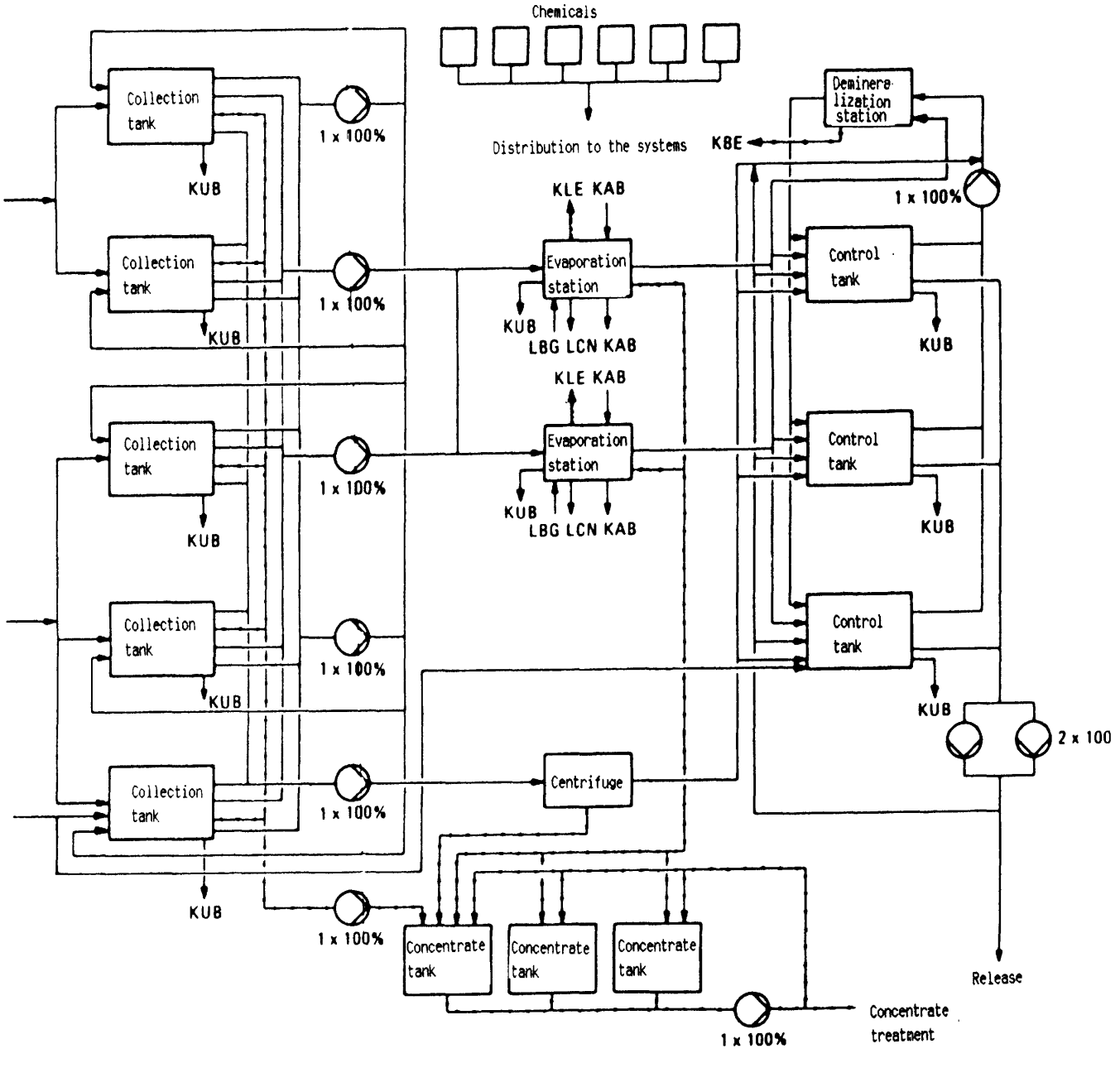
Their activity after dilution has to be lower than 10^{-7} Ci/m^3.

There are no other storage tanks before release.

Two of them collect water from treatment into, the discharge is ensured by the remaining one.

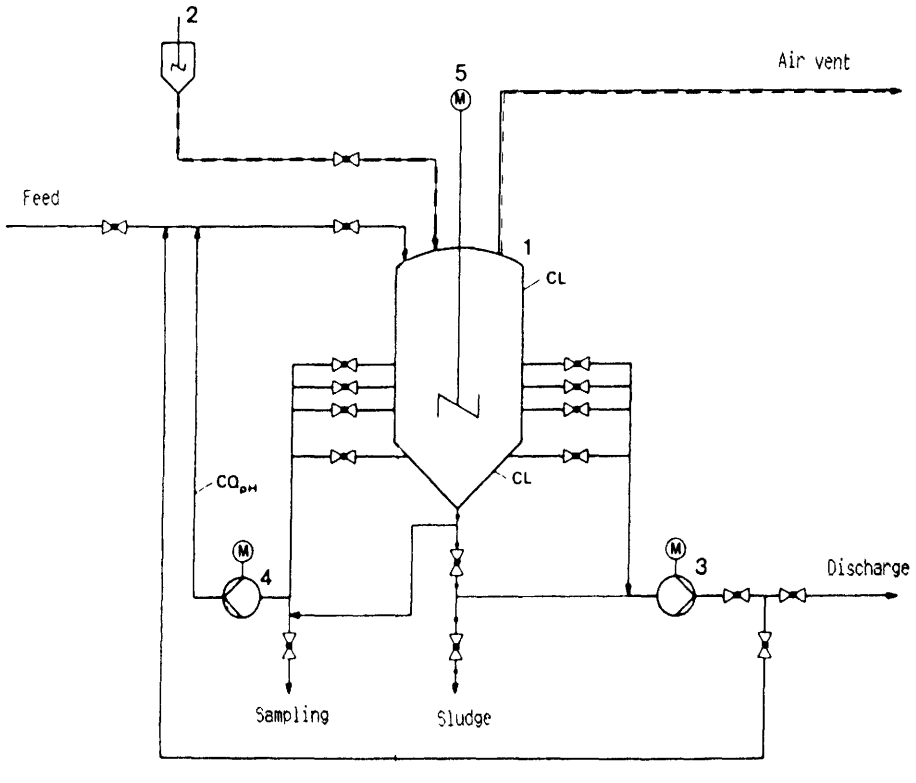
Inactive liquids coming from steam generator blowdown systems and liquids issued from primary coolant treatment system (975 m^3/a, 2.10^{-6} Ci/m^3) are collected into a fifth collection tank and are released (through one control tank) without further treatment.

FIGURE 3.10 : LIQUID WASTE TREATMENT SYSTEM FLOW DIAGRAM



KAB : Component cooling system
KBE : Primary coolant purification system
KLE : Ventilation
KUB : Sampling system
LBG : Auxiliary steam feed system
LCN : Steam condensation, collection and recirculation system

FIGURE 3.11 : COLLECTION TANK FLOW DIAGRAM



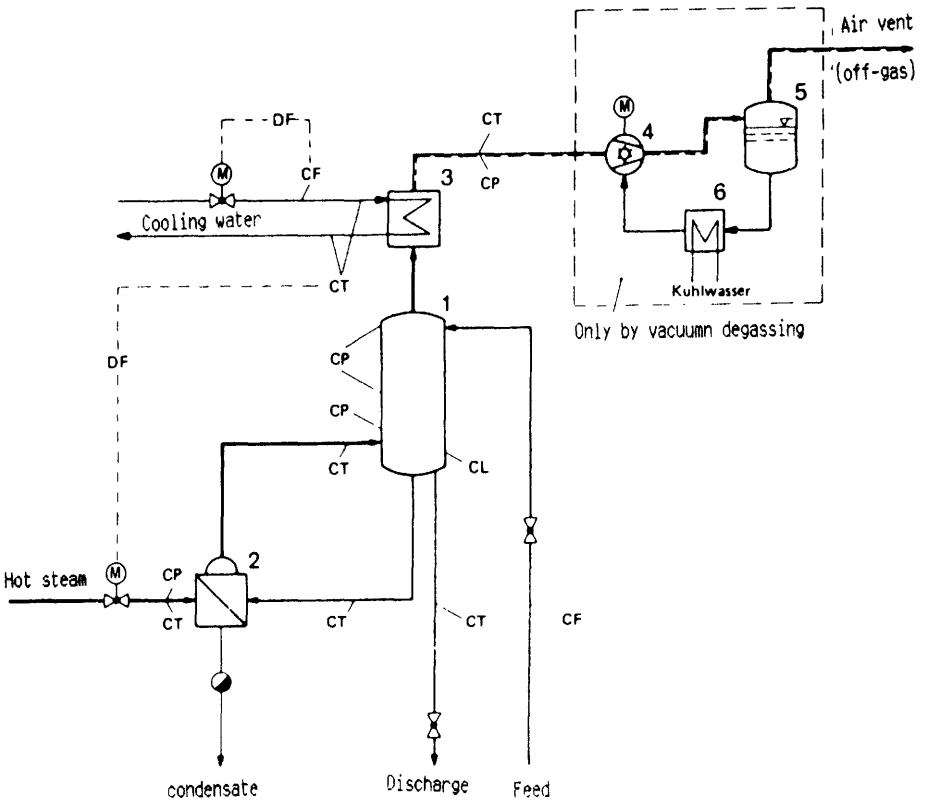
Legend :

1 - Collection tank
2 - Feed and dosing tank for chemicals and floculating agents
3 - Pump
4 - Pump
5 - Mixer

Measurements :

CL : Level measurement
CQ_{PH} : pH measurement

FIGURE 3.12 : EVAPORATION STATION FLOW DIAGRAM



Legend :

1 - Gas stripping column
2 - Evaporator heating body
3 - Condenser
4 - Liquide vacumn pump
5 - Condensate trap
6 - Heat exchanger

Measurements :

CF : Throughput measurement
CL : Level measurement
CP : Pressure measurement
CT : Temperature measurement
DF : Throughput control

FIGURE 3.13 : CENTRIFUGE STATION FLOW DIAGRAM

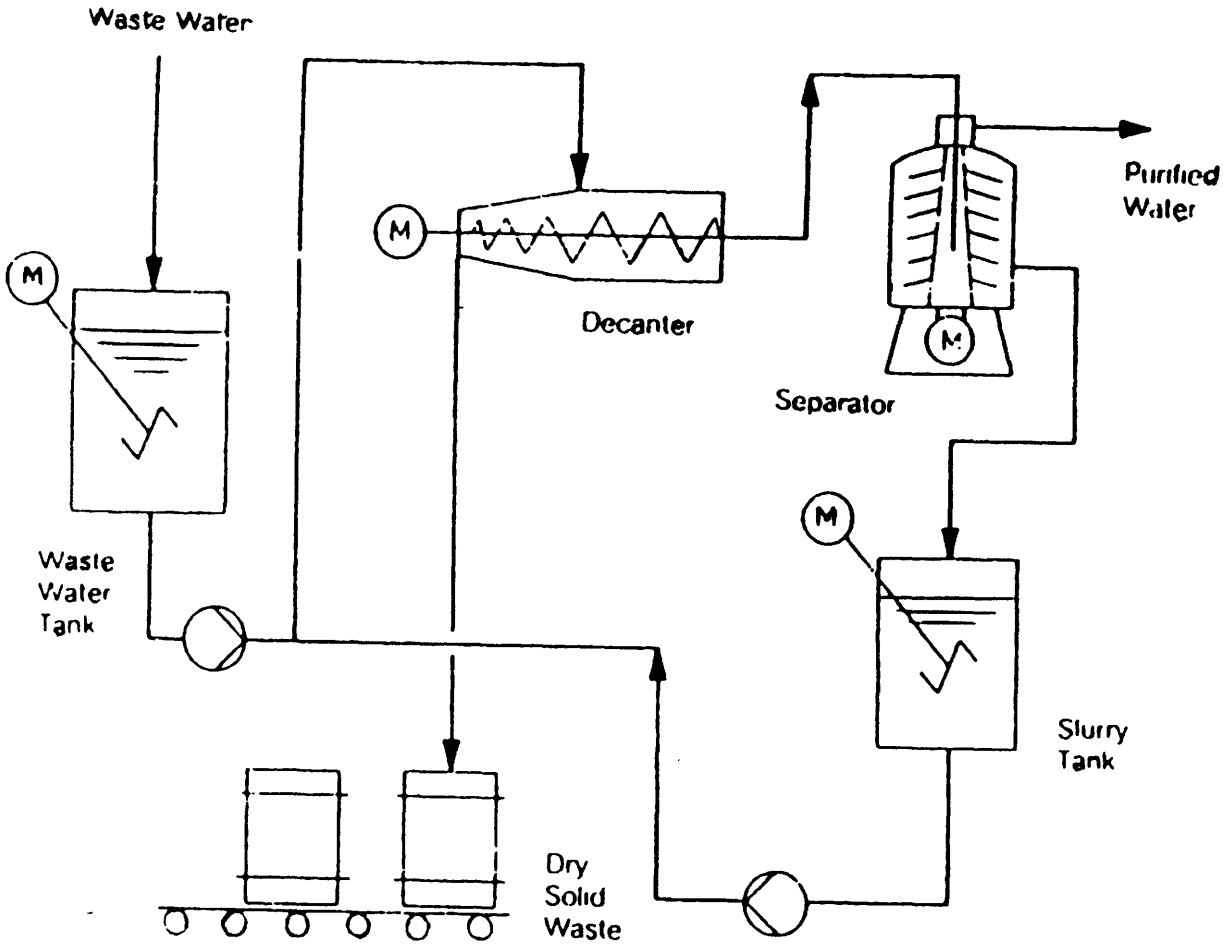
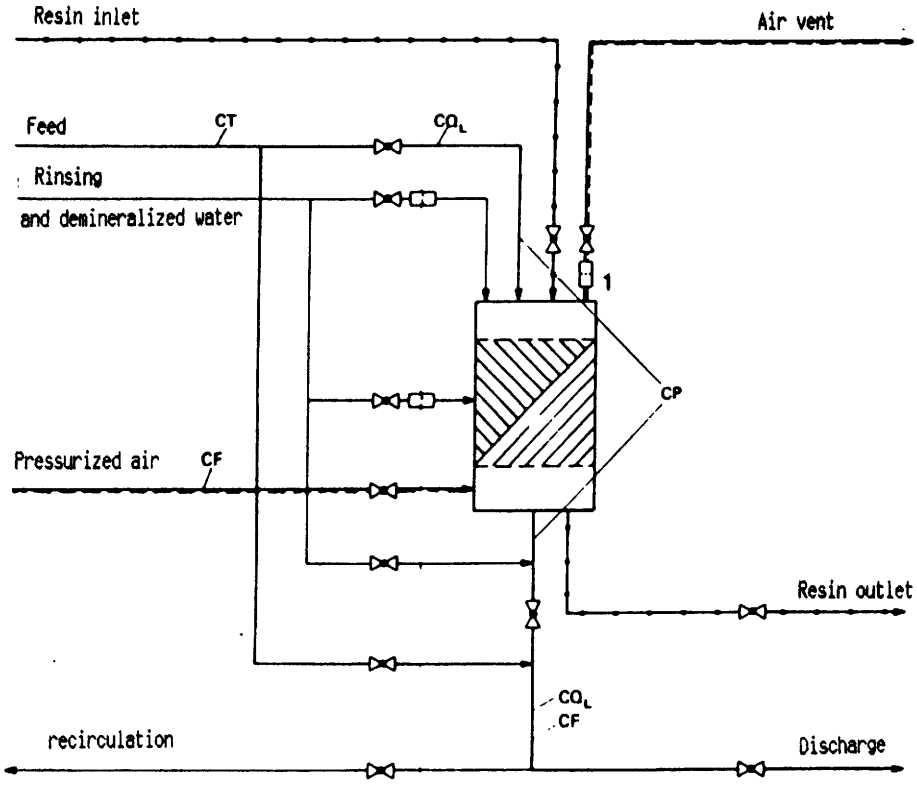


FIGURE 3.14 : MIXED-BED DEMINERALIZER FLOW DIAGRAM



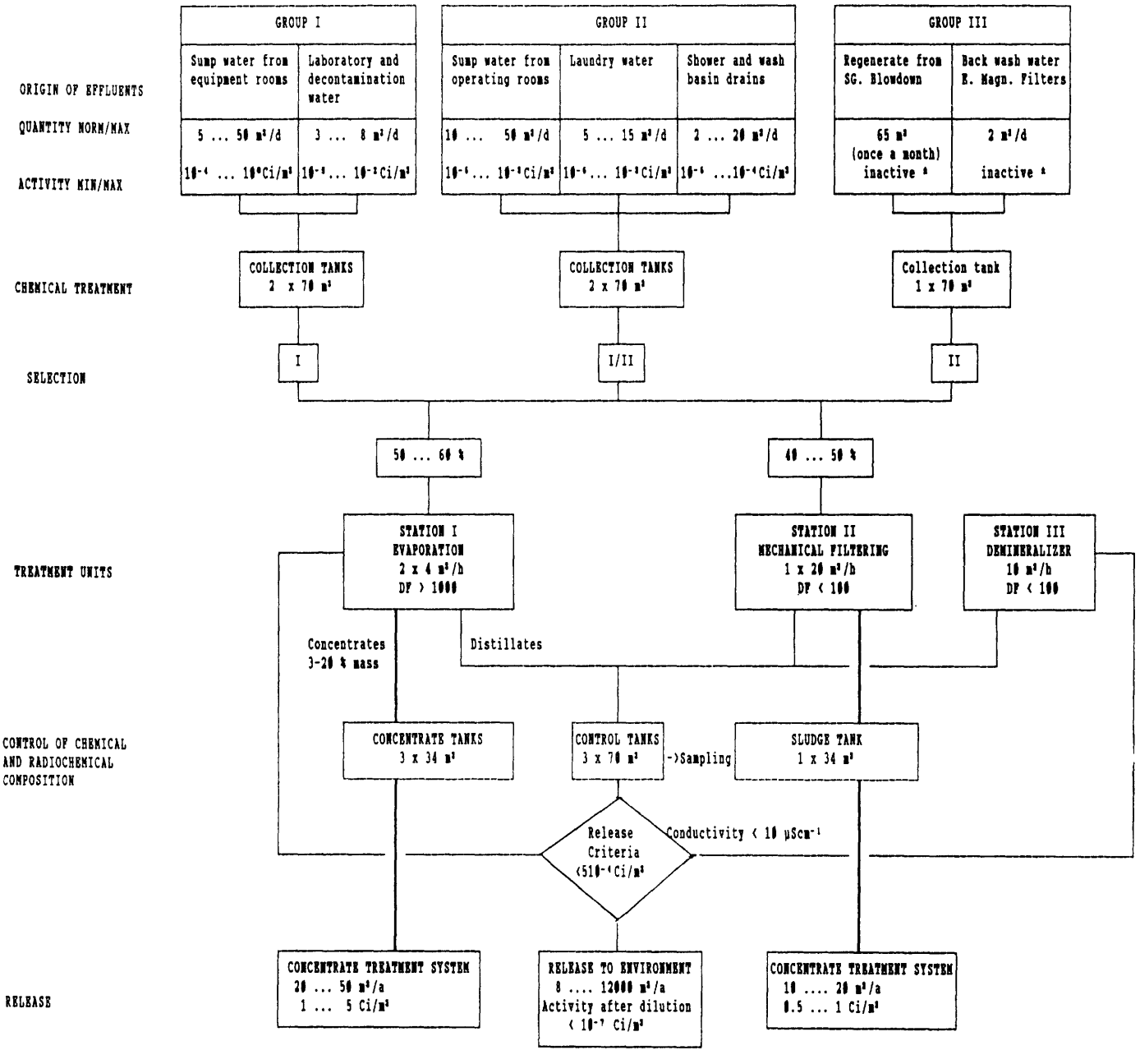
Legend :

1 - Demineralizer

Measurements :

CF : Throughput measurement
CP : Pressure measurement
$C\Omega_L$: Conductibility measurement

FIGURE 3.15 : LIQUID WASTE TREATMENT SYSTEM BLOCK DIAGRAM 1
TYPICAL GERMAN INVENTORY - OPERATING MODE I



* In case of steam generator leakage : activity : 10^{-4} ... 10^{-2} Ci/m³

FIGURE 3.16 : LIQUID WASTE TREATMENT SYSTEM BLOCK DIAGRAM 2
TYPICAL GERMAN INVENTORY - OPERATING MODE II

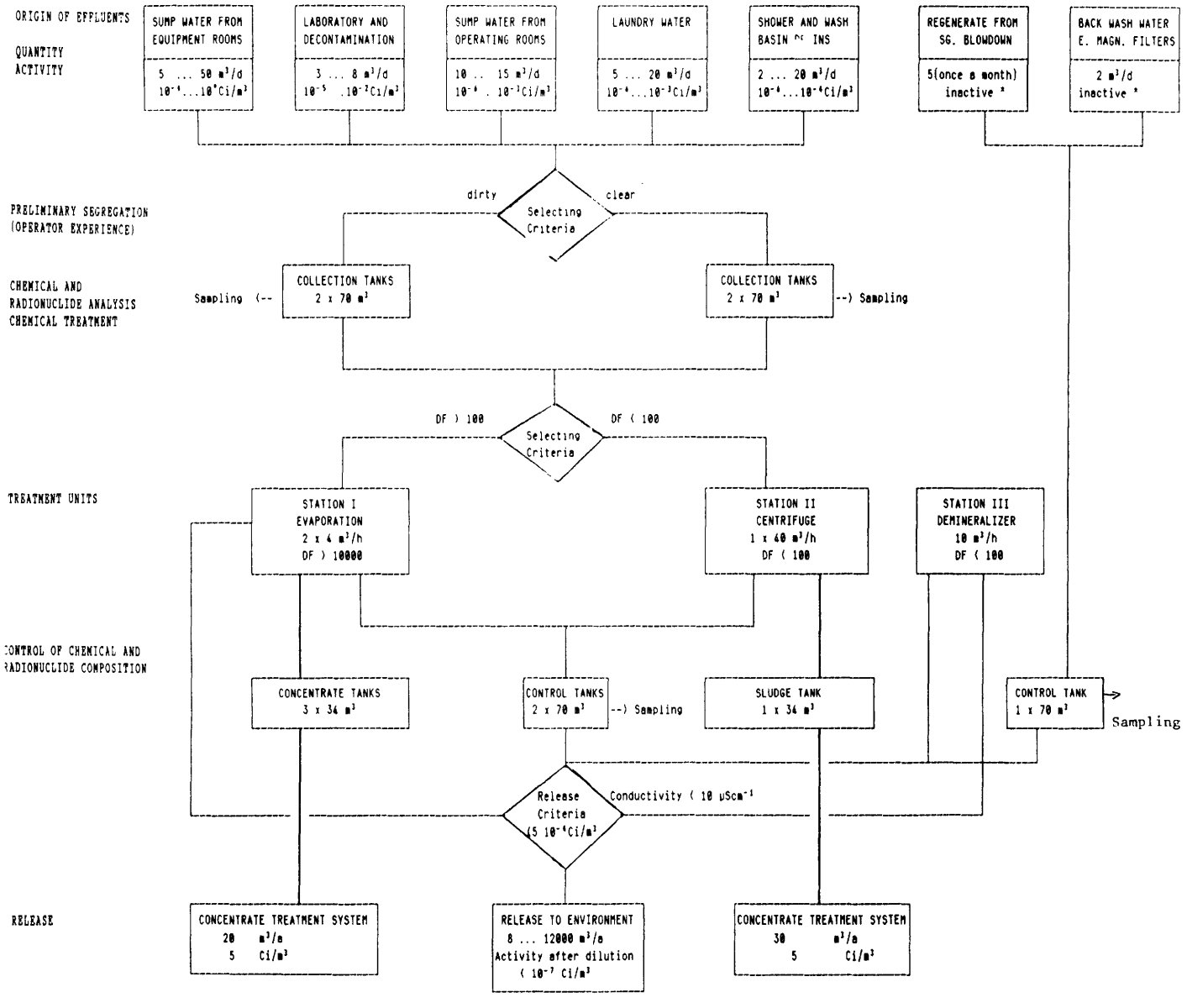
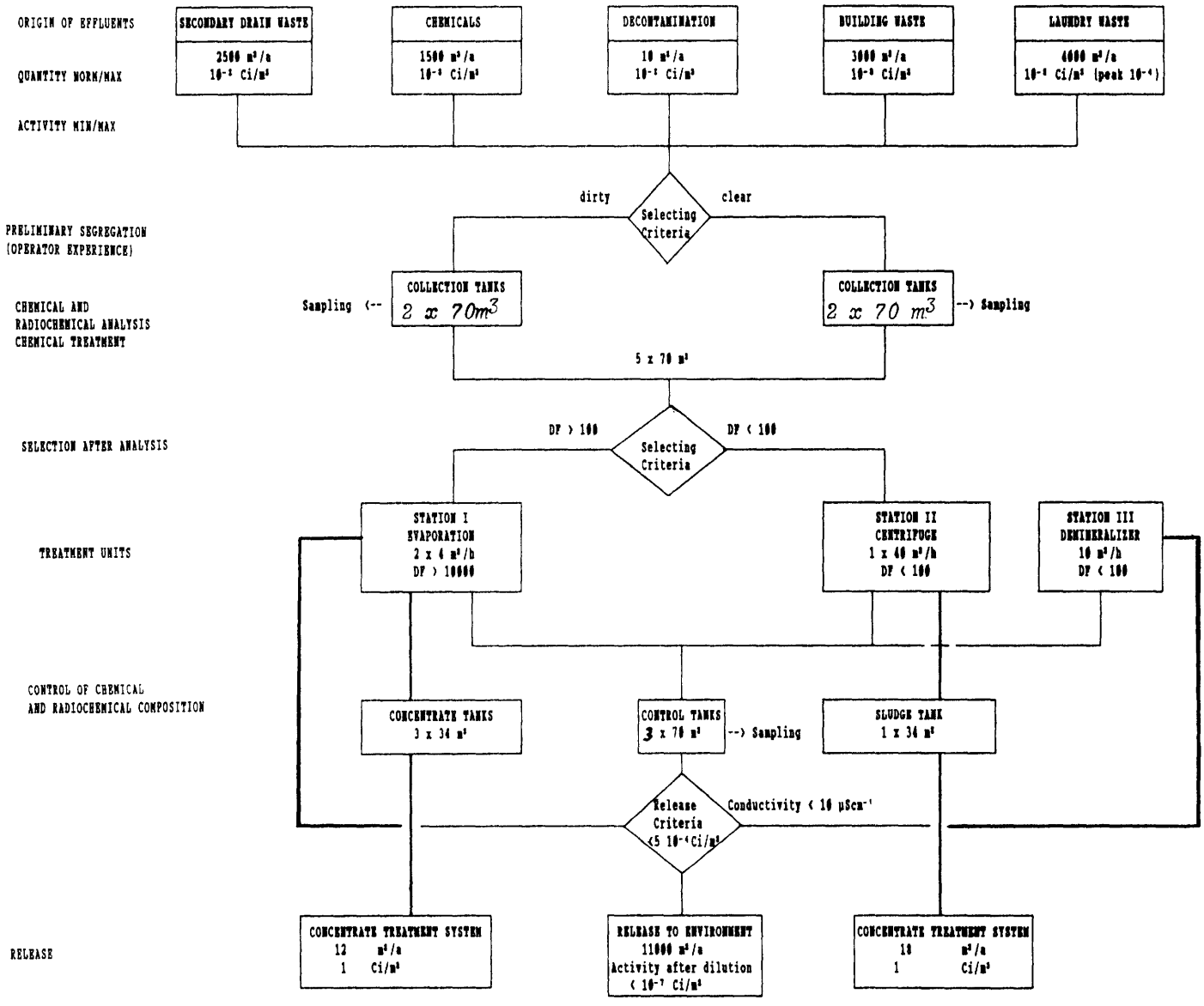


FIGURE 3.17 : LIQUID WASTE TREATMENT SYSTEM BLOCK DIAGRAM 3
EUROPEAN INVENTORY - OPERATING MODE II



3.3.2.3. Releases

TABLE 3.5 : AMOUNTS AND ACTIVITY OF RELEASED LIQUIDS

	VOLUME	INLET ACTIVITY	PROCESS	OUTLET ACTIVITY (w/o Hz)	H^3 INLET-OUTLET ACTIVITY
Secondary drain waste	1000 m^3/a	10^{-2} Ci/m^3 10 Ci/a	Evaporation DF = 10^4	10^{-6} Ci/m^3 10^{-3} Ci/a	10^{-4} Ci/m^3 0.100 Ci/a
	1500 m^3/a	10^{-2} Ci/m^3 15 Ci/a	Centrifuge DF = 20	5.10^{-4} Ci/m^3 0.74 Ci/a	10^{-4} Ci/m^3 0.150 Ci/a
Chemicals	1500 m^3/a	10^{-3} Ci/m^3 1.5 Ci/a	Evaporation DF = 10^4	10^{-7} Ci/m^3 0.5 10^{-3}Ci/a	10^{-5} Ci/m3 0.015 Ci/a
Decontamination	10 m^3/a	10^{-2} Ci/m^3 0.1 Ci/a	Evaporation DF = 10^4	5.10^{-6} Ci/m^3 10^{-5} Ci/a	10^{-4} Ci/m^3 0.001 Ci/a
Building waste	3000 m^3/a	10^{-3} Ci/m^3 3 Ci/a	Centrifuge DF = 20	5.10^{-5} Ci/m^3 0.15 Ci/a	10^{-5} Ci/m^3 0.030 Ci/a
Laundry waste	3600 m^3/a	10^{-5} Ci/m^3 0.036 Ci/a	Centrifuge DF = 20	5.10-7 Ci/m^3 1.80 10^{-3} Ci/a	10^{-7} Ci/m^3 36 10^{-5} Ci/a
	400 m^3/a	10^{-4} Ci/m^3 0.040 Ci/a	Centrifuge DF = 20	5.10^{-6} Ci/m^3 2.00 10^{-3} Ci/a	10^{-6} Ci/m^3 40 10^{-5} Ci/a
TOTAL	11010 m^3/a	29.676 Ci/a		0.905 Ci/a	0.297 Ci/a

3.3.3. Comparison with discharge limits

Total released activity

Tritium :

- Primary coolant treatment system : 585 Ci/a (975 m^3).
- Liquid waste treatment system : 0.30 Ci/a (11000 m^3).

Other radionuclides

- Primary coolant treatment system : negligible (975 m^3).
- Liquid waste treatment system : 0.91 Ci/a (11000 m^3).

Composition (without H_3)

TABLE 3.6 : RADIONUCLIDE COMPOSITION AT LIQUID TREATMENT SYSTEM EXIT

RADIONUCLIDE		Mn-54	Co-58	Co-60	Sr-90	Nb-95	Mo-99	Ag-110m
%		4.80	31.98	6.39	0.19	0.013	4.80	4.80
RADIONUCLIDE	Sb-124	I-131	I-132	I-133	I-134	I-135	Cs-134	Cs-137
%	4.80	0.46	0.80	1.39	0.46	0.93	19.19	19.19

The activity balances have been based on global decontamination factors.

Released activities remain below the objective values :

TABLE 3.7 : COMPARISON WITH DISCHARGE LIMITS

LIQUID EFFLUENTS	DESIGN VALUE Ci/a	OBJECTIVE VALUE Ci/a	RELEASE VALUE Ci/a
Total (H-3 excluded)	9	2	Ø.91
H-3	95Ø	75Ø	585

In the F.R.G., the authorized discharge limit is 1 Ci/a. The value reached by our treatment scheme corresponds to German practice.

3.4. SOLID WASTES

3.4.1. General description

3.4.1.1. Historical background

Since the middle of the seventies, the German PWR's have abandoned the installation or the use of already installed conditioning systems within their power stations.

The reasons herefore were operational and commercial : reliable mobile systems had been introduced into the market, which allowed the power station operator to call them in for a periodical conditioning campaign.

In this way, the utility is sure to have state-of-the-art conditioning equipment available and - perhaps most important - particularly skilled personnel due to the specification.

Furthermore in this way, the dose up-take of its own personnel is reduced.

It should not be neglected, that the previously used station-owned cementation or bitumization facilities had rather low reliability due to their batch operation and often ran into plugging problems.

Until 1978/1979, mobile cementation facilities were in use for sludges concentrate, resins (GNS - FAMA, TN-DEWA) and a polymer facility for resins only (GNS - FAMA, in France known under the name COMET and operated by STMI). The shielding for storage was assured by concrete over-packs.

Technological wastes were slightly compacted (compression force 16 - 300 t) and stored in 200 l drums for final disposal.

In 1979, the thus far available ASSE salt-mine was closed for final disposal of nuclear wastes and the utilities had to plan for an interim period of at least 10 years on-site storage.

Because of a lack of storage, volume reduction of wastes became a prime necessity and a value per se.

3.4.1.2. Inventories

Primary waste inventories : see TABLE 2.7.

Secondary waste inventories

During waste management operations, wet active wastes are generated :

- resins and filter candles in the primary coolant treatment system,
- concentrate from evaporators mixed with sludge coming from mechanical filtration (or centrifuge) and chemical precipitation in the liquid waste treatment system :

The characteristics of primary and secondary solid wastes that are to be treated (conditoned and packaged) are as follows :

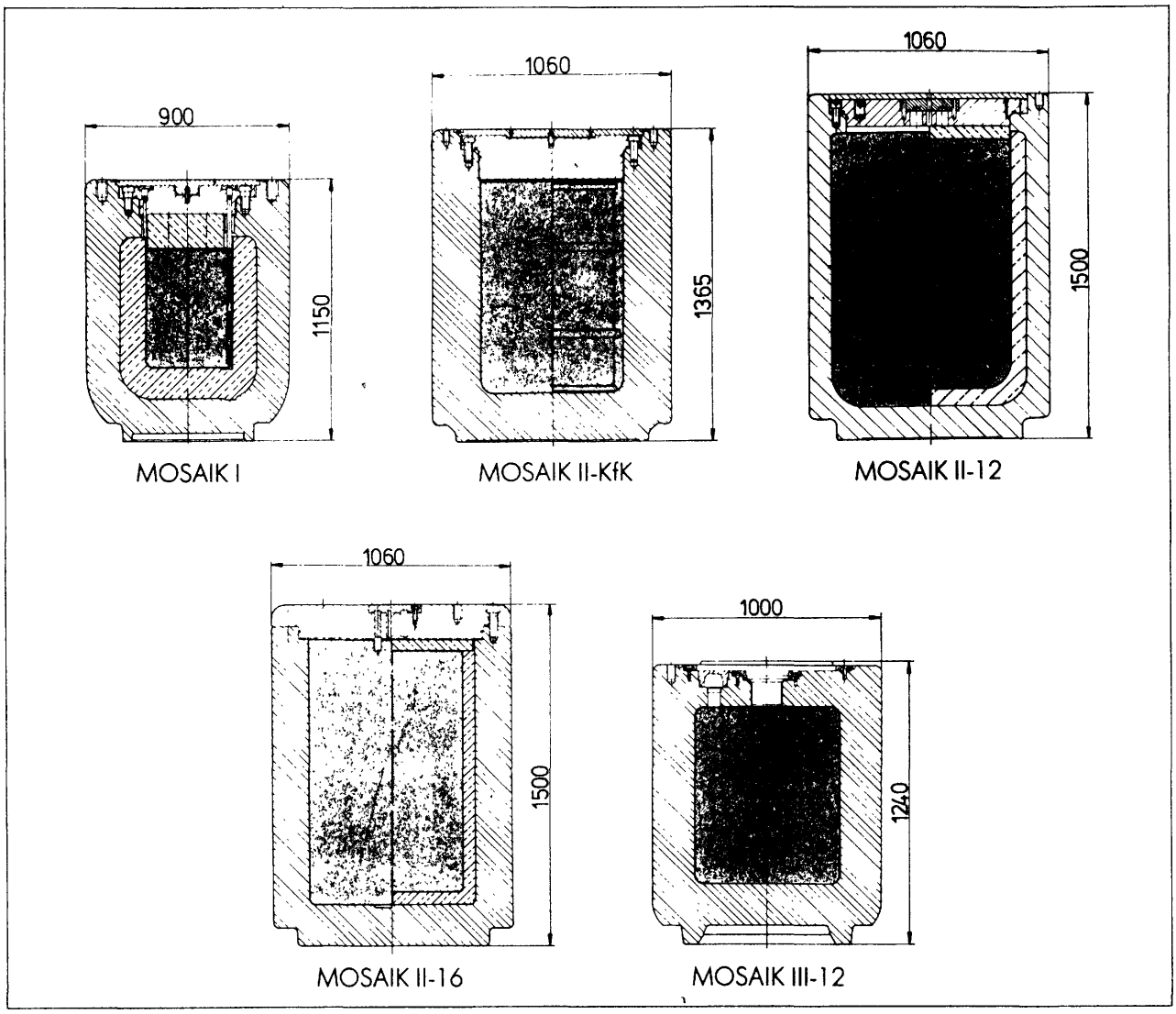
TABLE 3.8. : CHARACTERISTICS OF GENERATED SOLID WASTES

WASTES	PRIMARY WASTES		SECONDARY WASTES	
	VOLUME (m^3/a)	ACTIVITY (Ci/m^3)	VOLUME (m^3/a)	ACTIVITY (Ci/m^3)
WET ACTIVE WASTES				
. High active resins	1.3	500	2	200
. Low active resins	2.6	50		
. Filter candles			2	200
. Concentrate + Sludge			30	1
TECHNOLOGICAL WASTES				
. Combustible + Compactable	260	0.01		
. Non combustible + Compactable	100	0.01		
. Combustible + Non compactable	20	0.01		
. Non combustible + Non compactable	20	0.2		

3.4.1.3. Packagings

The packagings used to store the different solid wastes defined in TABLE 2.31. Characteristics of MOSAIK casks which can be used are detailed here after :

FIGURE 3.18 : MOSAIK CASK CHARACTERISTICS



Hauptkenndaten / Main Features	MOSAIK I	MOSAIK II-KFK	MOSAIK II-12	MOSAIK II-16	MOSAIK III-12
Außenhöhe Outer height	1150 mm	1365 mm	1500 mm	1500 mm	1240 mm
Außendurchmesser Outer diameter	900 mm	1060 mm	1060 mm	1060 mm	1000 mm
Innenhöhe max. Interior height max.	810 mm	930 mm	1145 mm	1170 mm	900 mm
Innendurchmesser max. Interior diameter max.	600 mm	630 mm	820 mm	740 mm	760 mm
Wanddicke Wall thickness	150 mm	215 mm	120 mm	160 mm	120 mm
Füllvolumen max. Filling volume max.	200 l	290 l	590 l	500 l	400 l
Gewicht min. Weight min.	3350 kg	6400 kg	5000 kg	6530 kg	4290 kg

3.4.2. Technological wastes

3.4.2.1. Process description

The general practice is as follows :

Technological wastes comprise essentially paper, cloth, metallic parts, etc... and are originally an undefined mixture of low-activity incinerable and non incinerable matter.

The reference concept for final disposal is the 800 m deep KONRAD iron-mine. After operation, the authorities request that there will be no empty spaces left, so the deduction is, that all compressible wastes have to be compacted with a force corresponding to the one which the convergence forces of the geological boundary will finally exert on the disposed wastes.

The long-term geological force corresponds to approximately 300 bar.

Thus, wherever compaction is chosen for volume reduction, this has to be done with presses of higher forces.

Characteristics of the supercompactor FAKIR (mobile unit) are shown on the FIGURE 3.11.

Depending upon their activity, the wastes are finally stored in containers without shielding or containers with concrete shielding.

In many cases, incinerable wastes are sorted out either within the power station or at some central location and directed towards incineration. The volume reduction factor remains between 20 and 40. The resulting ashes are either cemented or super-compacted (1000 - 1500 tons).

3.4.2.2. Block diagram

Figure 3.20 gives the block diagram for treatment of technological wastes.

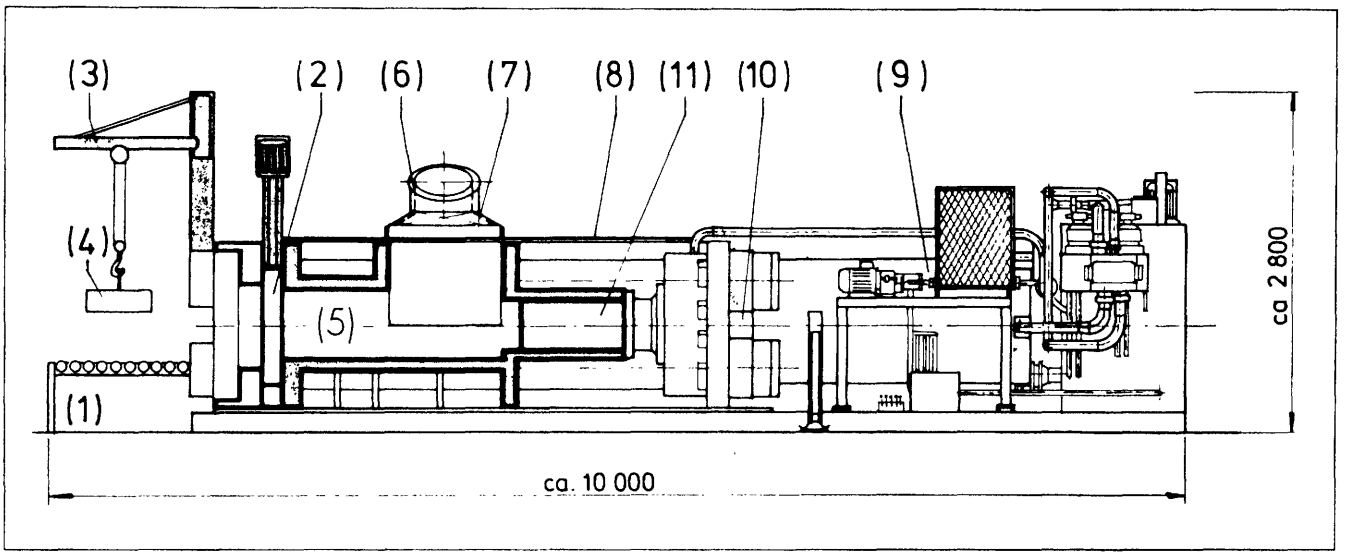
These ones are sorted and a relevant treatment is then applied.

- Compactable wastes (260 m^3/a) are precompacted, supercompacted (total VR = 9) and the final 200 metallic drums are stored in 8 containers (packagings Nr. 11 in TABLE 3.21) filled with concrete.

- Non compactable and combustible wastes (20 m^3/a) are incinerated (VR = 40) and the ashes are put into metallic drums which are stored in one of the 8 above mentioned containers.

- Non compactable and non combustible wastes (20 m^3/a) are put into 100 metallic drums which are stored into 13 containers (packagings Nr. 10 in TABLE 2.31).

FIGURE 3.19 : SUPERCOMPACTOR FAKIR CHARACTERISTICS



FAKIR
Hochdruckpresse zur Volumenreduzierung von leicht kontaminierten festen Abfällen

FAKIR
Hydraulic supercompactor for the volume reduction of slightly contaminated solid wastes

1) Auswurfseite
Exit

2) Schiebetür
Sliding door

3) Schwenkbarer Kran
Jib crane

4) Greifer
Grab

5) Preßkammer
Compaction chamber

6) Kippvorrichtung
Tipping device

7) Einfülltrichter
Charger

8) Arbeitsbühne
Working platform

9) Hydraulikaggregat
Hydraulic system and controls

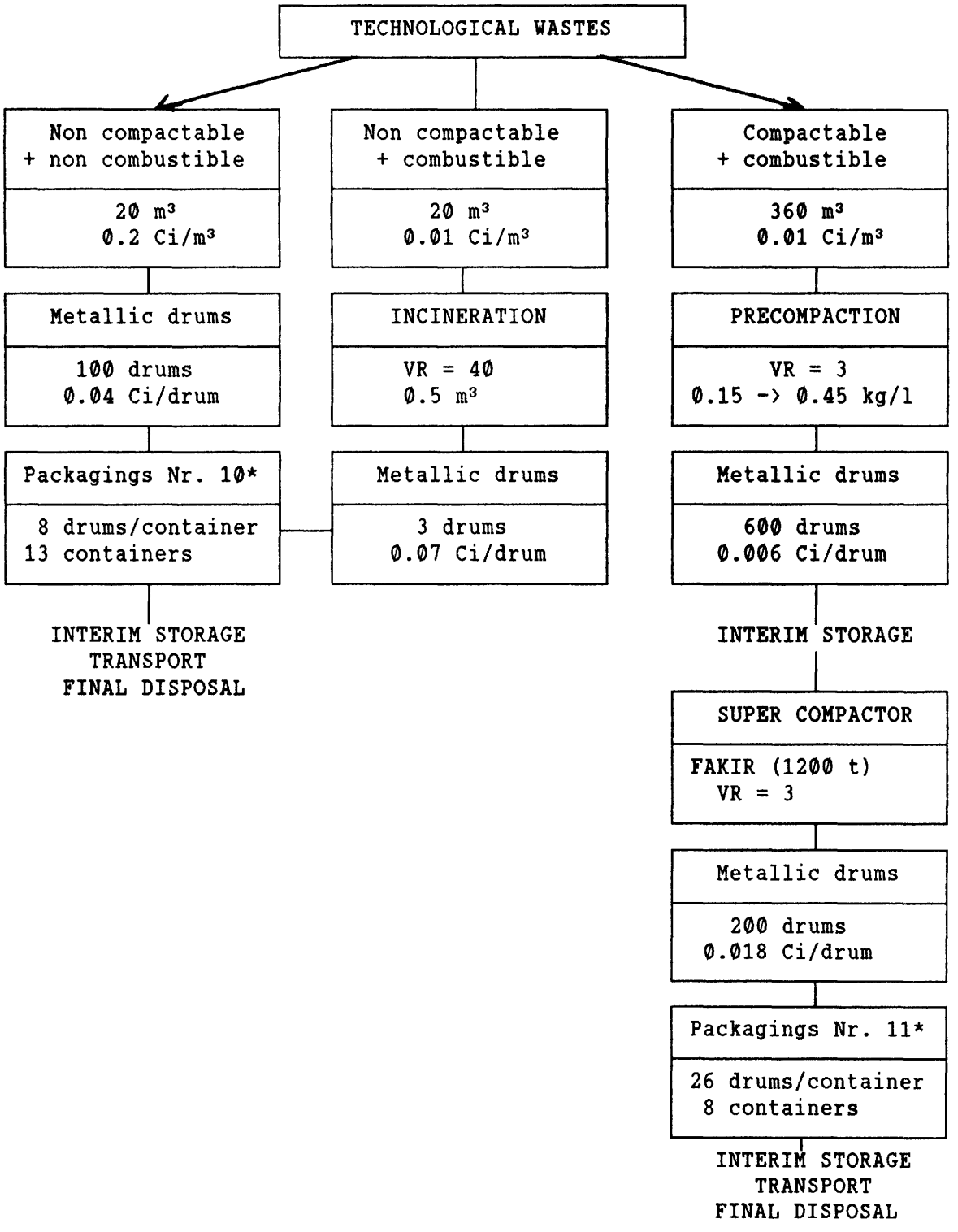
10) Hydraulikzylinder
Hydraulic cylinder

11) Preßstempel
Ram

Hauptkenndaten / Main Features	
Abmessungen/Dimensions:	Länge ca. / Length approx. 10000 mm
	Breite ca. / Width approx. 2400 mm
	Höhe ca. / Height approx. 2800 mm
Gewicht / Weight:	50 - 60 t / m.t.
Preßkraft / Compacting force:	1200 - 1500 t / m.t.
Durchsatz / Throughput:	bis 30 Fässer/h / up to 30 drums/h
Raumbedarf / Space requirement:	ca. / approx. 6,5 x 13 m

Abkürzungen / Abbreviations: m.t.: Metric tonnes

FIGURE 3.20 : TECHNOLOGICAL WASTE TREATMENT BLOCK DIAGRAM



* See TABLE 2.31.

3.4.3. Ion exchange resins and candle filters

3.4.3.1. Process description

The technology of the seventies was conditioning within polymers and cementation.

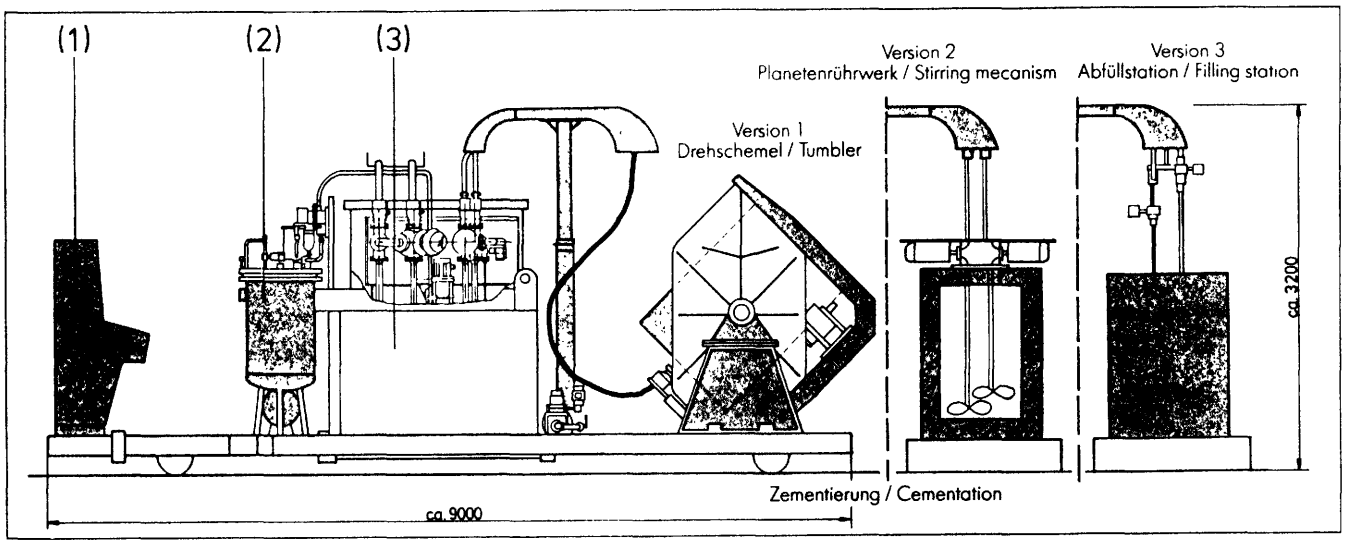
The new status is direct disposal after dewatering with a mobile unit, FAFNIR (VERSION 3) in so-called MOSAIK-casks. FAFNIR characteristics are shown on FIGURE 3.21.

These casks, here especially the cask II-15 (packaging Nr. 5), have useful volumes of 320 to 500 l depending on the shielding requirements.

3.4.3.2. Block diagram

Figure 3.22. gives the blocks diagram for treatment of resins and candle filters. 16 casks (packagings Nr. 5 in TABLE 2.31.) can be expected per year, each containing 320 l of wastes.

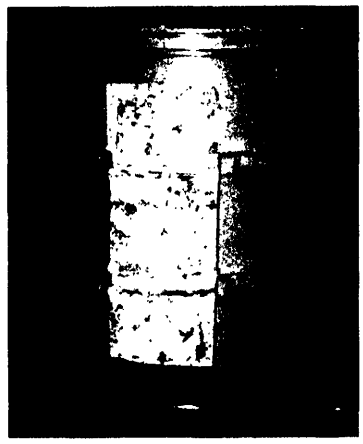
FIGURE 3.21 : FAFNIR MOBILE UNIT CHARACTERISTICS



FAFNIR
Mobile Konditionierungsanlage für radioaktive Flüssigabfälle und Ionentauscherharze

(1) Schalt- und Steuerpult / Switch and control board
(2) Vakuumvorlagebehälter / Vacuum container
(3) Dosierbehälter / Dosing tank

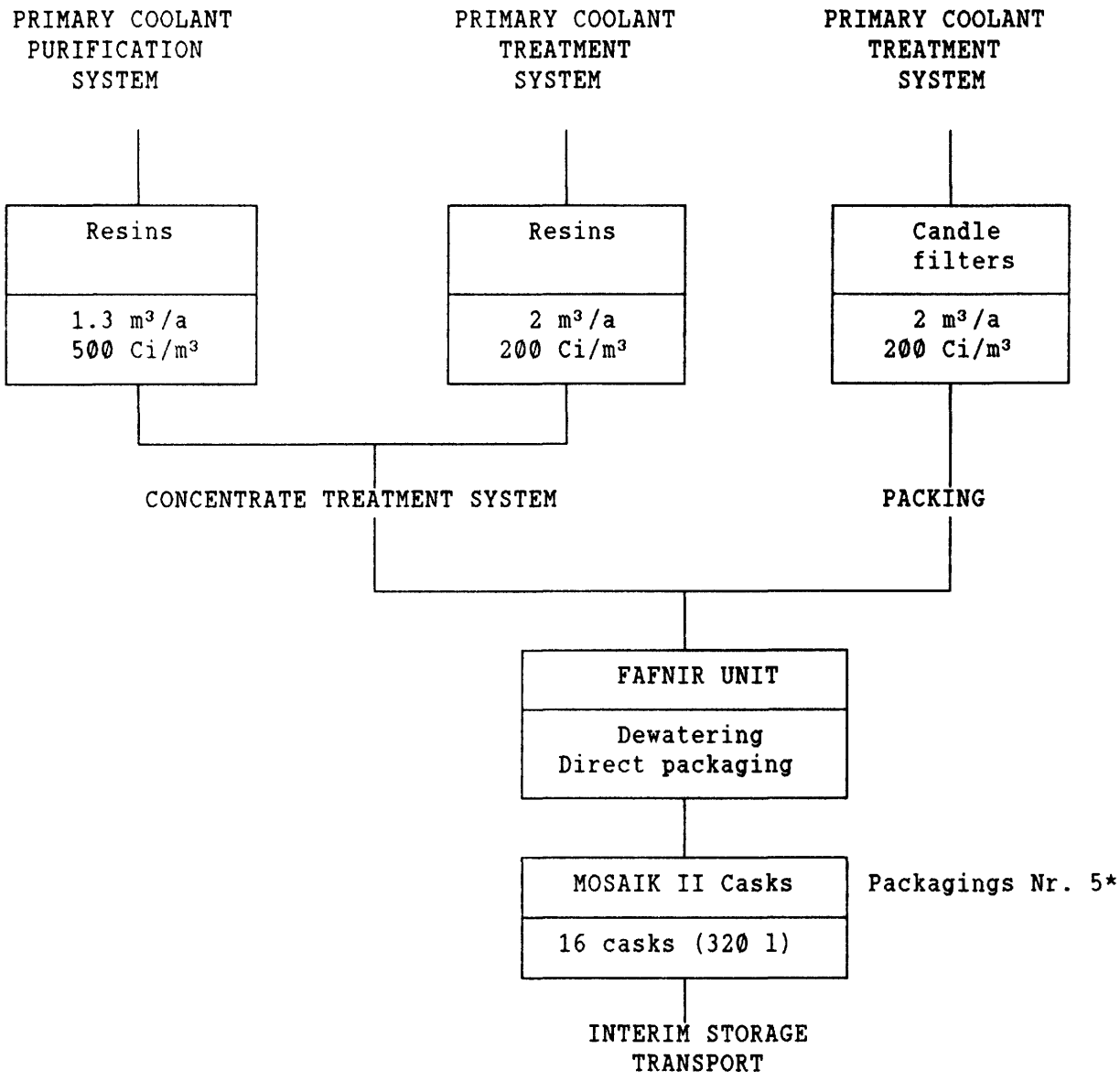
FAFNIR
Mobile conditioning facility for radioactive liquid wastes and ion exchange resins



FAFNIR-Abfallgebinde/ FAFNIR waste container

Hauptkenndaten / Main Features	
Abmessungen/Dimensions:	Länge ca. / Length approx. 9000 mm
	Breite ca. / Width approx. 2500 mm
	Höhe ca. / Height approx. 3200 mm
Gewicht / Weight:	ca. / approx. 20t / m.t.
Raumbedarf / Space requirement:	ca. / approx. 11 x 5 m
Strombedarf / Power consumption:	max. 7,5 kW
Kapazität / Capacity:	bis zu 15 Abfallbehälter pro Tag /
	up to 15 waste containers per day

FIGURE 3.22 : RESIN AND CANDLE FILTER TREATMENT BLOCK DIAGRAM



* See TABLE 2.31

3.4.4. Concentrate and sludge

3.4.4.1. Process description

The old technology was cementation (30 %) in drums and storage/disposal within concrete over-packs.

Now these liquid wastes are treated on-site with the mobile drying unit FAVORIT (a vacuum distillation unit) or on a central location.

FAVORIT characteristics are shown on FIGURE 3.23.

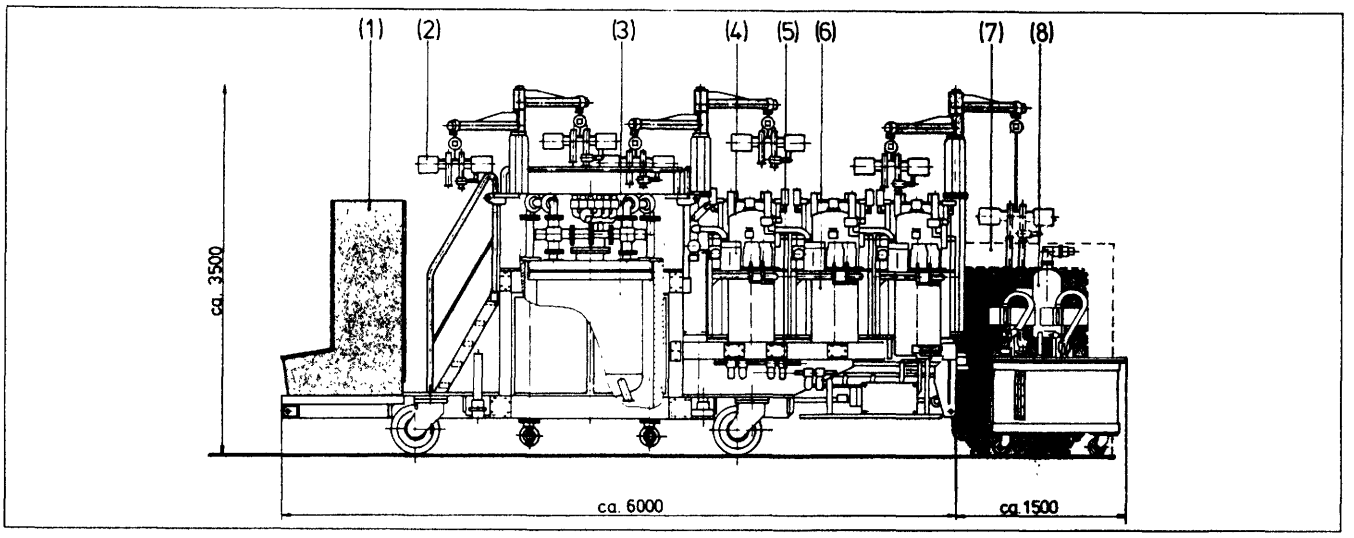
The objective is total drying in order to achieve the formation of a solid salt block inside the MOSAIK final disposal cask. The concentrate and sludge stored in tanks at the power plant are pumped through FAVORIT'S dosing tank into the final storage casks. Up to six casks can be connected to the facility at one time. The casks are electrically heated from the outside and the interior is simultaneously held under vacuum.

During the evaporation process, the entrainment of liquid droplets in the vapor stream is reduced by an inside, "lost", mechanical filter and another one on top of the cask. The condensate flows through a collecting tank and is directed to the power plant.

3.4.3.2. Block diagram

Figure 3.24. gives the block diagram for treatment of concentrate, sludge and low active resins. 17 casks (packagings Nr. 5 in TABLE 3.21) can be expected per year and 1300 MW PWR plant, each containing 500 l of wastes.

FIGURE 3.23 : FAVORIT MOBILE UNIT CHARACTERISTICS



FAVORIT
Trocknungsanlage zur Volumenreduzierung flüssiger radioaktiver Abfälle

(1) Schalt- und Überwachungspult
Switch and control board
(2) Behälter-Ankopplungsstation
Cask coupling station
(3) Dosiereinrichtung
Dosing tank
(4) Vakuumpumpe
Vacuum pump
(5) Kondensator
Condensor

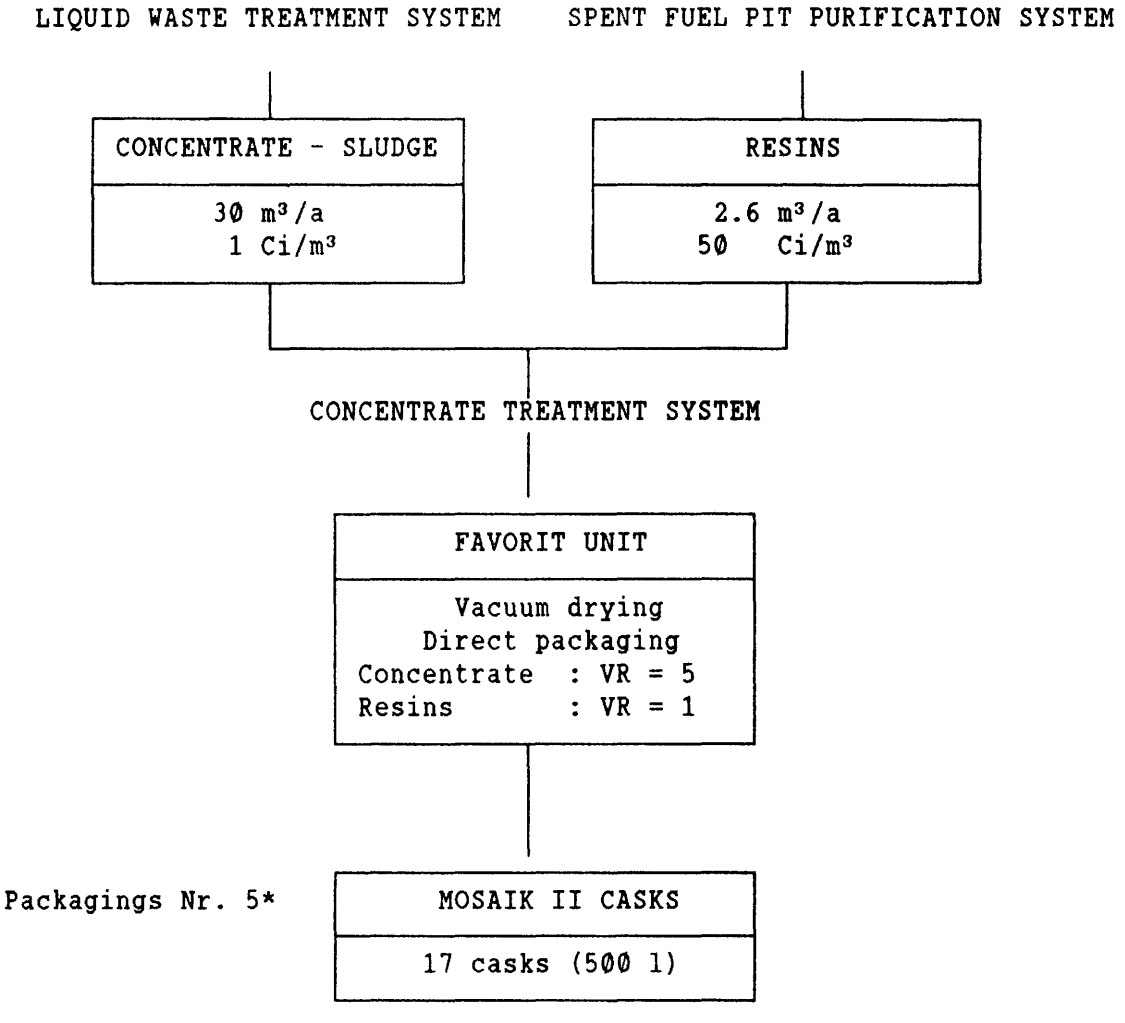
FAVORIT
Drying Facility for the Volume Reduction of Liquid Radioactive Wastes

(6) Drucklufteinspeisung
Compressed air supply
(7) MOSAIK-Behälter mit Heizung
MOSAIK cask with heating
(8) Kondensat-Sammelbehälter
Condensate collecting tank
Nicht abgebildet:
Kühlblock und Zwischenwärmetauscher
Not depicted:
Cooling block and intermediate heat exchanger

Hauptkenndaten / Main Features	
Abmessungen/Dimensions:	Länge ca. / Length approx. 6000 mm
	Breite ca. / Width approx. 5200 mm
	Höhe ca. / Height approx. 3500 mm
Gewicht / Weight:	**ca. / approx. 20 t / m.t.**
Raumbedarf / Space requirement:	ca. / approx. 9 x 6 m
Strombedarf / Power consumption:	**100 - 120 kW**
Kapazität / Capacity:	bis zu / up to 100 l/h
Containermaße / Container dimensions:	**Länge ca. / Length approx. 6060 mm**
	Breite ca. / Width approx. 2440 mm
	Höhe ca. / Height approx. 2990 mm

Abkürzungen / Abbreviations: m.t.: Metric tonnes

FIGURE 3.24 : CONCENTRATE, SLUDGE AND RESIN TREATMENT BLOCK DIAGRAM



* See TABLE 2.31.

3.4.5. Core components

3.4.5.1. Process description

Under this title we comprise also PWR-absorber rods or BWR water channels, etc...

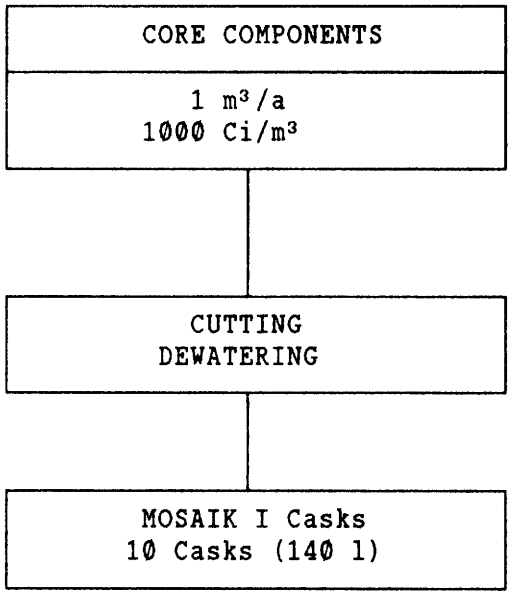
They can have dose-rates of 50.000 rem/h. They require therefore special shielding casks and here the MOSAIK I with an adaptable inner lead liner serves as reference.

The useful volume per cask lies between 120 and 140 l.

The packaging is preceded by underwater cutting and compacting.

3.4.5.2. Block diagram

FIGURE 3.25 : CORE COMPONENT TREATMENT BLOCK DIAGRAM



* See table 2.31.

3.4.6. Scrap metal, active carbon filters

During back-fitting and decommissioning, there are considerable arisings of contaminated scrap metal.

These are used (up to 74 Bq/g) without decontamination as raw material for the fabrication of the above-mentioned MOSAIK casks.

The active carbon (charcoal filters) is generally of very low activity and can be used as fill-up in order to produce the right carbon content of the cast nodular iron, which requires around 4 weight %, while the input, ferritic steel, has only around 0,30 weight %.

The dust, slag, etc... in a weight proportion of 5 % go to the waste stream.

Due to the quality requirements from metallurgy, this scrap can only be used as add-up to clean raw material for the fabrication of waste casks. Proportions up to 1 can be accepted for more shielding blocks.

3.4.7. Characteristics of conditioned and packaged wastes

TABLE 3.9. shows characteristics of conditioned wastes (without packaging) :

- conditioning type,
- annual produced volume,
- specific activity.

TABLE 3.10. shows characteristics of packaged wastes :

- type of package,
- number of packages, annual volume and specific activity,
- further packaging,
- final number of packages, annual volume.

TABLE 3.9 : CHARACTERISTICS OF CONDITIONED WASTES

WASTES	CONDITIONING OPERATION	VOLUME (*) m^3/a	ACTIVITY Ci/m^3
WET ACTIVE WASTES			
. High active resins	Dewatering + Direct packaging	1.3 + 2	318.2
. Low active resins	Vacuum drying (VR = 1) + Direct packaging	2.6	50 -> 18.6 (**)
. Filter candles	Dewatering + Direct packaging	2	200
. Concentrate + Sludge	Vacuum drying (VR = 5) + Direct packaging	6	5 -> 18.6 (**)
TECHNOLOGICAL WASTES			
. Combustible + Compactable	Precompaction (VR = 3) + Super compaction (VR = 3)	28.9	0.09
. Non combustible + Compactable	Precompaction (VR = 3) + Super compaction (VR = 3)	11.1	0.09
. Combustible + Non compactable	Incineration (VR = 40	0.5	0.4
. Non combustible + Non compactable	Packaging	20	0.2

(*) Without shielded cask

(**) Low active resins and concentrate/sludge are mixed together and the mean activity is 18.6 Ci/m3

TABLE 3.10 : CHARACTERISTICS OF PACKAGED WASTES

WASTES	TYPE OF PACKAGE	NR OF PACKAGES VOLUME (m3/a) ACTIVITY (Ci/m3)	FURTHER PACKAGING	NR OF PACKAGES VOLUME (m3/a)
WET ACTIVE WASTES				
. High active resins	MOSAIK II cask Vi = 320 l Ve = 1.3 m3	10 13 80.8	- - -	10 13
. Low active resins	MOSAIK II cask Vi = 500 l Ve = 1.3 m3	5 6.5 7.2 (*)	- - -	5 6.5
. Filter candles	MOSAIK II cask Vi = 320 l Ve = 1.3 m3	6 7.8 51.3	- - -	6 7.8
. Concentrate + Sludge	MOSAIK II cask Vi = 320 l Ve = 1.3 m3	12 15.6 7.2 (*)	- - -	12 15.6
TECHNOLOGICAL WASTES				
. Combustible + Compactable	Metallic drum (200 l)	144.5 28.9 0.09		
. Non combustible + Compactable	Metallic drum (200 l)	55.5 11.1 0.09	CONTAINER VE = 10.9 m3 26 drums/cont.	8 87.2
. Combustible + Non compactable	Metallic drum (200 l)	3 0.6 0.4	CONTAINER Ve = 7.4 m3 8 drums/cont.	13 96.2
. Non combustible + Non compactable	Metallic drum (200 l)	100 20 0.2		

(*) Low active resins and concentrate/sludge are mixed together.

3.5. COSTING OF GERMAN WASTE MANAGEMENT SCHEME

3.5.1. Assumptions

The costs provided in this section are based on the following assumptions :

- 1300 MWe PWR,

- one liquid waste treatment system per unit,

- prices of 1987 given in ECU.

3.5.2. Capital costs

The capital costs for the systems are turn-key prices and include costs for quality insurance, installation, start-off operation, planning and licensing.

The following fractional costs for major equipment can be assumed :

- Quality insurance : 35 %,

- Installation : 35 %,

- Start-off : 10 %.

The "base value" (BV) calculated by TASK/KAH has to be derived from major equipment cost (ME) provided in the next paragraphs as follows :

$$BV = ME/1.8$$

3.5.2.1. Ventilation system

The major equipment cost ("base value" in TASK cost calculation) is 11.6 Mill.ECU 88.

3.5.2.2. Off-gas system (KPL)

DIRECT COST

Major equipment (ME)

. Pumps, compressors	:	Mill. ECU	0.6
. Tanks	:	Mill. ECU	1.0

. Filter, heat exchangers, cooler, dryer, etc...	:	Mill. ECU	1.Ø
. Recombiners	:	Mill. ECU	1.Ø
. Adsorption system	:	Mill. ECU	2.6

Bulk materials

. Piping	:	Mill. ECU	1.3
. Valves	:	Mill. ECU	Ø.7
. Isolation	:	Mill. ECU	Ø.5
. Small parts	:	Mill. ECU	Ø.8
. El supply, control	:	Mill. ECU	1.5

INDIRECT COST

. Planning, licensing	:	Mill. ECU	2.Ø
TOTAL	:	Mill. ECU	13.Ø

BV = ME/1.8 = 6.2/1.8 = 3.4 Mill. ECU

3.5.2.3. Primary coolant treatment system (KBF)

Degassing system (KBG)

DIRECT COST

Major equipment (ME)

. Pumps, compressor	:	Mill. ECU	Ø.2
. Tanks	:	Mill. ECU	Ø.1
. Filter	:	Mill. ECU	Ø.1
. Heat exchanger	:	Mill. ECU	Ø.2
. Degasser	:	Mill. ECU	1.2

Bulk materials

. Piping	:	Mill. ECU	Ø.8
. Valves	:	Mill. ECU	Ø.4
. Isolation	:	Mill. ECU	Ø.5
. Small parts	:	Mill. ECU	Ø.2
. El. supply, control	:	Mill. ECU	Ø.5

INDIRECT COST

. Planning, licensing	:	Mill. ECU	Ø.6
TOTAL	:	Mill. ECU	4.8

BV = ME/1.8 = 1.8/1.8 = 1 Mill. ECU

Storage and treatment system

DIRECT COST

Major equipement (ME)

. Pumps	:	Mill. ECU	Ø.2
. Tanks	:	Mill. ECU	1.2
. Demineralizer	:	Mill. ECU	Ø.4
. Evaporator	:	Mill. ECU	2.Ø

Bulk materials

. Piping	:	Mill. ECU	1.5
. Valves	:	Mill. ECU	Ø.4
. Isolation	:	Mill. ECU	Ø.5
. Small parts	:	Mill. ECU	Ø.2
. El. supply, control	:	Mill. ECU	Ø.5

INDIRECT COST

. Planning, licensing	:	Mill. ECU	Ø.6
TOTAL	:	Mill. ECU	7.5

BV = ME/1.8 = 3.8/1.8 = 2.1 Mill. ECU

3.5.2.4. Liquid waste treatment system (KPF)

DIRECT COST

Major equipment (ME)

. Pumps	:	Mill. ECU	Ø.7
. Tanks	:	Mill. ECU	1.4
. Filters	:	Mill. ECU	Ø.1
. Evaporators	:	Mill. ECU	2.Ø
. Centrifuge	:	Mill. ECU	1.5
. Auxiliary equipment (stirrer etc.)	:	Mill. ECU	Ø.5

Bulk materials

. Piping	:	Mill. ECU	1.5
. Valves	:	Mill. ECU	Ø.8
. Isolation	:	Mill. ECU	Ø.5
. Small parts	:	Mill. ECU	Ø.8
. El. supply, control	:	Mill. ECU	1.5

INDIRECT COST

. Planning, licensing	:	Mill. ECU	2.Ø
TOTAL	:	Mill. ECU	13.3

BV = ME/1.8 = 6.2/1.8 = 3.4 Mill. ECU

3.5.2.5. Concentrate treatment system (KPC)

DIRECT COST

Major equipment (ME)

. Pumps	:	Mill. ECU	Ø.1
. Tanks	:	Mill. ECU	Ø.2
. Filters	:	Mill. ECU	Ø.1

Bulk materials

. Piping	:	Mill. ECU	Ø.8
. Valves	:	Mill. ECU	Ø.1
. Small parts	:	Mill. ECU	Ø.3
. El. supply control	:	Mill. ECU	Ø.5

INDIRECT COST

. Planning, licensing	:	Mill. ECU	Ø.5
TOTAL	:	Mill. ECU	2.6

BV = ME/1.8 = Ø.4/1.8 = Ø.2 Mill. ECU

3.5.2.6. Solid waste treatment system

Mobile units are bought and used for solid waste conditioning within the 20 GWe nuclear park :

- 2 supercompactors FAKIR (technological wastes) : 2 x 1.0 Mill. ECU
- 2 FAFNIR units (resin conditioning) : 2 x 1.0 Mill. ECU
- 3 FAVORIT units (concentrate conditioning) : 3 x 1.5 Mill. ECU

The price of the incinerator (turn-key) is 7.5 Mill. ECU but for the study we recommend to take no capital cost but cost for rented incineration service of 18 Ecu/kg of incinerated wastes (included shipment).

3.5.2.7. Total capital cost

KPV	:	Mille ECU	11.6
KPL	:	Mill. ECU	13.0
KBF	:	Mill. ECU	12.3
KPF	:	Mill. ECU	13.3
KPC	:	Mill. ECU	2.6
Solid waste treatment system	:	Mill. ECU	8.5
TOTAL	:	Mill. ECU	61.3

3.5.3. Building cost

3.5.3.1. Auxiliary building

All the above mentioned systems are located in the auxiliary reactor building. This auxiliary building is part of the power plant.

The building volume is approx. 53,000 m^3. On the agreed basis of 135 ECU/m^3 a total building cost of Mill. ECU 7.2 can be considered.

3.5.3.2. Storage building

The annual production of waste packages for one 1300 MWe power plant is a follows :

TABLE 3.10. : TOTAL EXTERNAL VOLUMES OF PACKAGES

PACKAGES	EXTERNAL PACKAGE Volume m^3
8 containers (Type V)	87.2
13 containers (Type IV)	96.2
33 casks (Type II)	42.9
10 casks (type I)	7
TOTAL/YEAR	233.3 m3

The annual storage volume necessary for 20 GWe is :

(87 x 20)/1.3 = 1.342 m^3/a

(96 x 20)/1.3 = 1.480 m^3/a

(43 x 20)/1.3 = 660 m^3/a

(7 X 20)/1.3 = 108 m^3/a

TOTAL 3.590 m^3/a (20 GWe)

The volume of interim storage capacity will be calculated by TASK, knowing that containers can be pilled up as follows :

- 6 containers Type V and IV,
- 3 casks Type I and II.

3.5.4. Operating costs

3.5.4.1. Waste treatment system operation

The operating costs are personal costs of power plant personnel working (part time) with the treatment systems. The following annual man-power (man-hours) for one 1300 MWe plant can be considered :

Plant systems	
Ventilation KBF-System KPF-System KPL-System KPC-System	4.100 man-hours
Technological waste collection and sorting	10.200 man-hours
Mobile systems	
Wet solid waste treatment	1.000 man-hours
Technological waste treatment	4.800 man-hours
TOTAL	20.100 man-hours

3.5.4.2. Material, energy consumption

The costs for material and energy consumption (steam, electricity, etc.) are not determined and negligible compared to all other costs.

3.5.4.3. Package costs

Prices for waste packages

The prices for waste packages given below are based on 1988 values. The prices per piece are :

- Container Type IV (N° 10) : ECU 6,000
- Container Type IV (N° 11) : ECU 6,500
- Cask Type II (N° 5) : ECU 13,000
- Cask Type I (N° 4) : ECU 15,000

TABLE 3.12 : TOTAL PRICES OF PACKAGES

Waste package	N° per year 1300 MWe	N° per year 20 GMWe	Price Mill ECU
Container IV	8	123	0.7
Container V	13	200	1.3
Cask Type II	33	508	6.6
Cask Type I	10	154	2.3
TOTAL			10.9

3.5.4.4. Transport cost

Transport will be done by rail only in Germany. The mean shipping costs (Round trip) of waste packages for distances given in this study (2 x 500 km) are :

- Cask Type I, II : ECU 1,000
- Container Type IV, V : ECU 2,000

The waste packages are shipped in 20'-containers. Three cylindrical casks (Type I, II) and 2 containers (Type IV, V) are necessary for one shipment.

The equipment costs for transport equipment of one 1300 MWe PWR plant is approximately ECU 200,000.

3.5.4.4.5. Interim storage and final disposal costs

The operating costs for interim storage of waste packages delivered to a central storage facility or to an on-site storage facility are 2500 man-hours per year for a 1300 MWe PWR.

The operating costs for final disposal are :

ECU 2,500 per m^3 of waste package.

For a 1300 MWe plant annual disposal costs of about ECU 583,000 can be considered.

3.5.4.4.6. Incineration service cost

20 m^3 of combustible waste are incinerated per year in a 1300 MWe plant. This corresponds to 10 t.

The service price for incineration including shipment is 18 Ecu/kg.

For a 20 GWe park, the following annual incineration costs arise :

10000 x (20000/1300) x 18 = 2.8 Mill. ECU.

4. GERMAN WASTE MANAGEMENT SCHEME FOR BWR'S

4.1. GENERAL

The waste treatment systems described are based on a 1300 MWe BWR.

The systems are located in the reactor building and in the turbine building. There is no separate building for these systems.

FIGURE 4.1 gives a cross section of the reactor building and the turbine building.

The total volume of the two buildings is :

- reactor building : 155 000 m^3,

- turbine building : 220 000 m^3.

The fractional volume of these buildings where the waste treatment systems are located is 67000 m^3.

On FIGURE 4.2, a general flow diagram of the nuclear auxiliary systems is shown. For the study, the following systems are required :

- ventilation (KPV),

- leak-off systems (KLS),

- off-gas system (KPL),

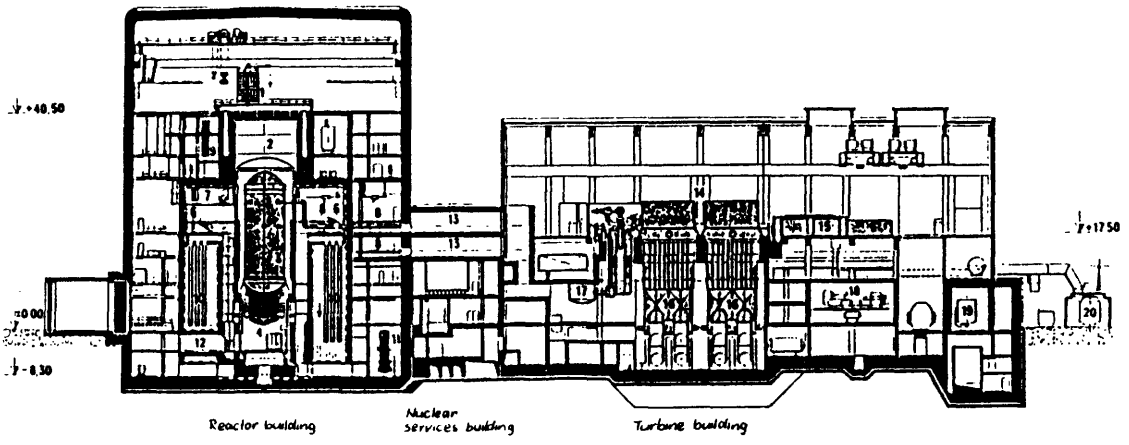
- liquid waste treatment system (KPF),

- solid waste treatment systems (conditioning + packaging).

FIGURE 4.1 : REACTOR AND TURBINE BUILDING CROSS SECTIONS

Longitudinal section and ground plan of the reactor building, nuclear services building and turbine building

1 Refueling machine
2 Reactor well
3 Reactor pressure vessel
4 Control rod drives
5 Internal recirculation pumps
6 Containment structure
7 Air recirculation system
8 Pipe galleries
9 Fuel pool cooling heat exchanger
10 Pressure suppression chamber
11 Residual heat exchangor
12 Air lock
13 Main steam and feedwater piping duct
14 Turbine
15 Generator
16 Condensers
17 Reheater
18 Gaseous waste system
19 Condensate polishing system
20 GSU transformer
21 Feedwater tank
22 HP feedwater heaters
23 LP feedwater heaters



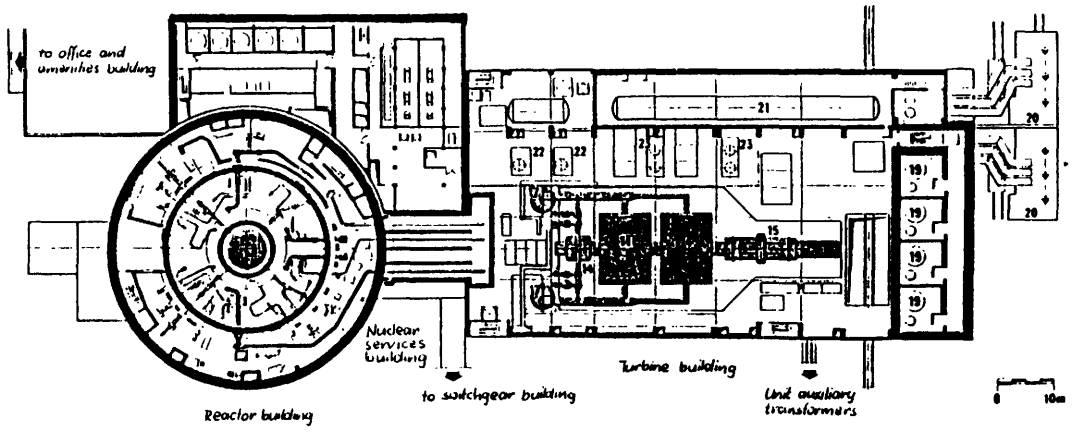
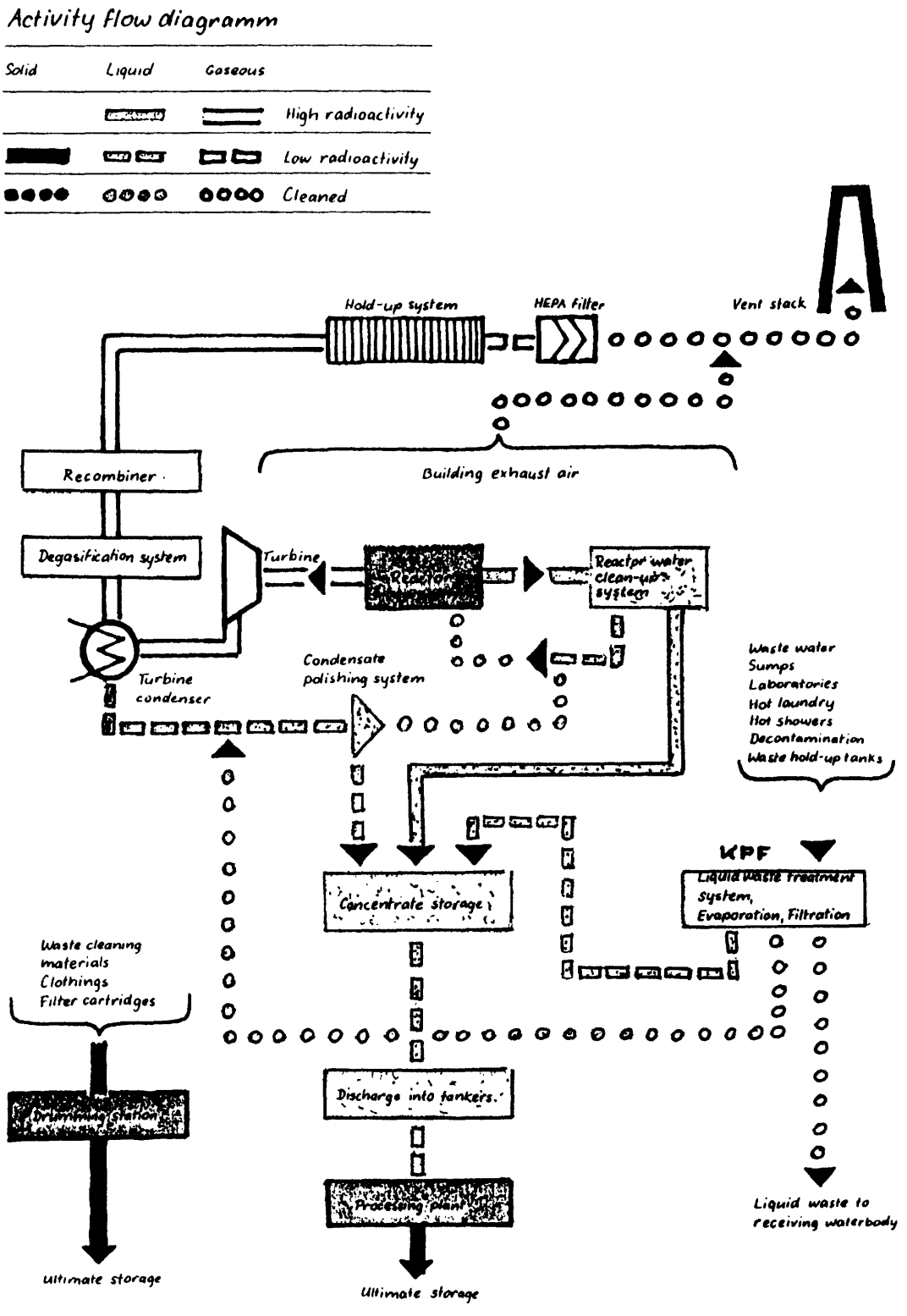


FIGURE 4.2 : NUCLEAR AUXILIARY SYSTEM FLOW DIAGRAM



4.2. GASEOUS EFFLUENTS

4.2.1. Ventilation system

4.2.1.1. Process description

A flow diagram of ventilation is shown on FIGURE 4.3.

In the controlled area of a nuclear power station the possibility always exists that small quantities of radioactivity become airborne due to leakages from the systems or evaporation from open sumps. Furthermore the activation of airborne particles in the vicinity of the reactor must be considered. Therefore an unacceptable build-up of radioactivity in the air and a spread of radioactivity within the controlled areas of nuclear power station must be prevented by technical means.

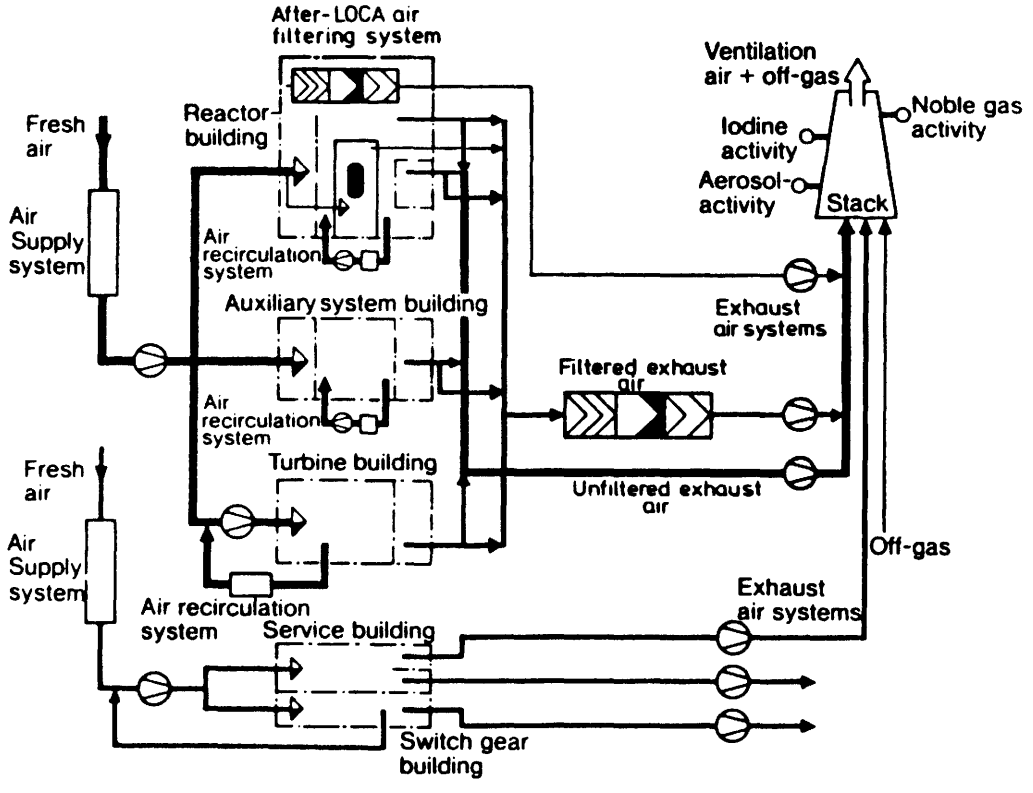
Fresh air is treated in a central air supply plant and routed via a duct system to the operating alleys and floors of the reactor building, turbine building and auxiliary building such that defined fresh-air zones are created for control and operating tasks. From these zones the air then flows as secondary air to the accessible equipment rooms in which the majority of the apparatus, machinery and piping is located and where most of the heat losses occur. These rooms or groups of rooms where normally no disturbing nuclides are to be expected are cooled by removing the heat from the recirculated air in coolers as required. A partial air stream flows from the relatively large area of the accessible equipment rooms to the remaining restricted areas where equipment could lead to an unacceptably high level of airborne activity in these areas. Therefore, the exhaust air from these areas is handled by a central exhaust air unit which is common to the three buildings and normally discharged through the stack without filtration. A central exhaust air unit with a filter arrangement can be brought into service in case filtration for iodine and airborne particles of the exhaust air from the equipment rooms of one of the three buildings should become necessary. The filter arrangement consists of a prefilter, a particulate filter, a iodine filter and a micro filter. Its retention efficiency is 99.95 percent for particles and 99 percent for iodine even with a methyl iodine content of 10 percent.

The equipment rooms within the safety containment where the reactor, recirculation pumps and the pressure suppression system are located are cooled by an air recirculation system. A small quantity air is used as a flushing stream to avoid concentration of radioactive nuclides within the safety containment caused by activated airborne particles and leakages from the nuclear steam supply system. This way the accessibility of the safety containment is assured. The exhaust air is routed to the central exhaust air filter arrangement.

The service and switch gear building containing the main control room, laboratories, laundry and showers as well as offices and rest rooms is supplied with air from a fresh air system. This system is designed for air

renewal and air conditioning as well as maintaining the controlled areas in the service building at below-atmospheric pressure. The exhaust air from the controlled areas is routed to the stack. The exhaust air from the other rooms of the service and switch gear building is partly recirculated to the fresh air intake and the rest released directly to the atmosphere.

FIGURE 4.3 : VENTILATION FLOW DIAGRAM



4.2.1.2. Block diagram

Following general assumptions have been taken into account :

- Iodine is filtered through activated charcoal
  (efficiency = 99.9 %)

- Aerosols are filtered through filbre-glass
  (efficiency = 99 %)

FIGURE 4.4 : VENTILATION BLOCK DIAGRAM

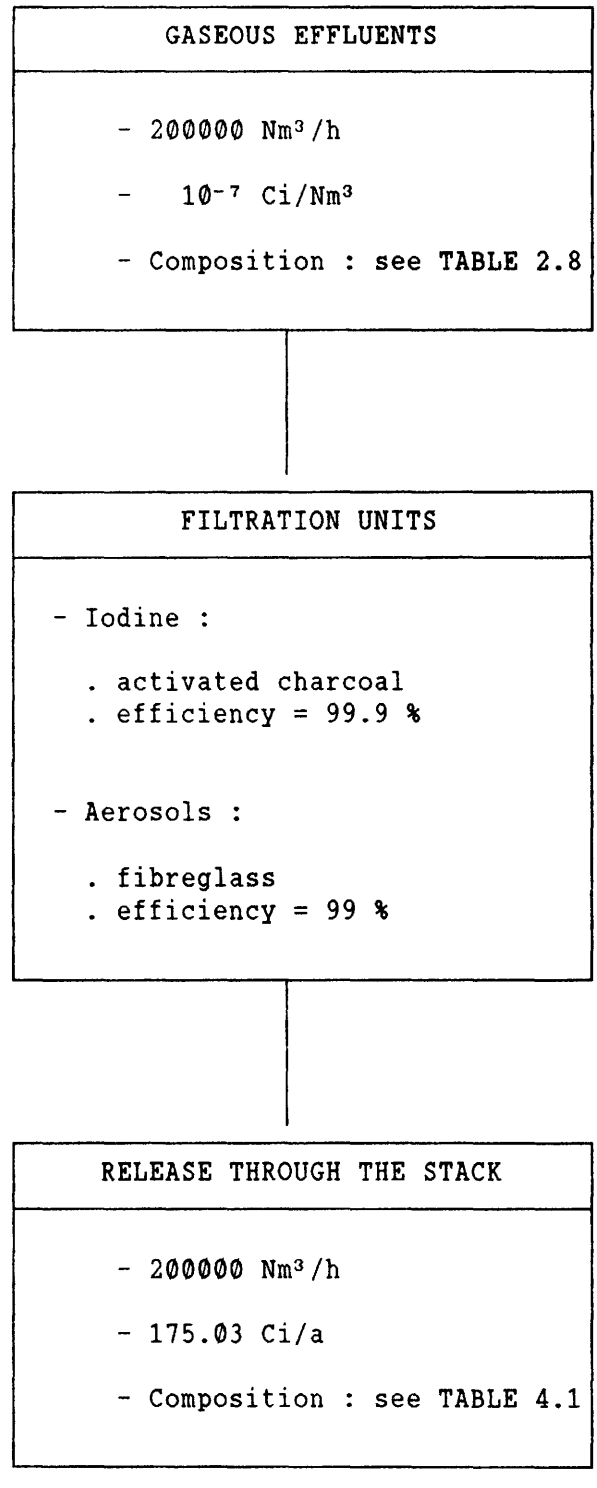


TABLE 4.1 : RADIONUCLIDE COMPOSITION AT VENTILATION EXIT

RADIONUCLIDES	INLET COMPOSITION (%)	INLET ACTIVITY (Ci/a)	FILTER EFFICIENCY (%)	OUTLET ACTIVITY (Ci/a)	OUTLET COMPOSITION (%)
C - 14	0.00001	$1.75\ 10^{-5}$	0	$1.75\ 10^{-5}$	0.00001
Kr - 85	0.03	0.05		0.05	0.03
Kr - 85 m	1.83	3.20	0	3.20	1.83
Kr - 87	1.25	2.19		2.19	1.25
Kr - 88	3.32	5.82		5.82	3.32
Xe - 133	80.41	140.88		140.88	80.49
Xe - 133 m	1.75	3.07		3.07	1.75
Xe - 135	11.31	19.82		19.82	11.32
I - 131	0.01	0.02		$1.75\ 10^{-5}$	0.00001
I - 132	0.02	0.04	99.9	$3.50\ 10^{-5}$	0.00002
I - 133	0.03	0.05		$5.25\ 10^{-5}$	0.00003
I - 134	0.01	0.02		$1.75\ 10^{-5}$	0.00001
I - 135	0.02	0.04		$3.50\ 10^{-5}$	0.00002
Aerosols	0.00001	$6.57\ 10^{-5}$	99	$1.75\ 10^{-7}$	0.0000001
TOTAL	100	175.2		175.03	100

4.2.2. Leak-off systems

4.2.2.1. Process description

Besides the ventilation systems different leak-off-systems are provided. The purpose of these systems is to prevent leakage of radioactve substances into the air of the controlled areas. Leakage can result from components which by nature are subject to frequent leaks and from degasification of radioactive liquids from tanks. The activity carried by these systems is reduced to allowable values by appropriate processes.

Gland leak-off system

A flow diagram of gland leak-off system is shown on FIGURE 4.5.

To the gland leak-off system all those valves of the primary steam system, reactor water clean-up system, feed water system are connected which are equipped with stuffing boxes and which carry media with high activity content at pressures above 1 bar and temperatures above 99 °C. Furthermore the shaft sealing arrangement of he reactor water recirculation pumps are also connected to the gland leak-off system.

The leakages which are picked up in the stuffing boxes and in the shaft sealing arrangements are carried away in pipes. Leakages consist first of all of steam with small quantities of inert gas mostly H_2, O_2 and N_2 and the carrier of radioactivity such as iodine, xenon and krypton. The steam atmosphere is condensed in a condense and cooler and the rest gas is routed to a noble gas delay bed. If necessary air is added to reduce the H_2 content. If noble gases are present the xenon and krypton isotopes can be delayed in the decay bed to values which are equal or below the values at the condenser off-gas system exit. Filtration of the gas stream because of iodine can be managed by routing the gas stream through the exhaust air filter arrangement prior to release to the atmosphere.

Tank leak-off system

A flow diagram of tank leak-off system is shown on FIGURE 4.6.

The air volume of all tanks of the liquid and solid rad waste systems whose liquid content can produce a high gaseous nuclide concentration is connected to the tank-leak-off system. Mostly xenon which results from the decay of iodine is released to the tank atmosphere above the liquid level from the stored coolant or exhausted ion echange resins. To avoid the release of xenon with expelled air directly to the atmosphere in case of increasing liquid levels the tank leak-off system is provided.

The interconnection of the air volumes of the tanks is done in series and in the order of increasing radioactivity. A continuous small purge air stream flows through the tanks from low activity to high activity and is then routed to the off-gas system. This way the activity level of the xenon in the tanks above the liquid phase is kept low. In case of a liquid

volume increase in the tanks an equivalent amount of air is expelled. If this air flow is larger than the purge air flow then it is routed as a counter flow to the building ventilation exhaust air system. There iodine is retained in the ventilation air filter arrangement prior to release of the air volume to the atmosphere. The xenon carried with this air volume is retained in the charcoal filter. If during plant operation the liquid levels in the tanks are lowered because of discharge of water then the air volume above the liquid level increases and an appropriate quantity of fresh air will flow back to the tanks through the charcoal filter. During this process the radioactive xenon which is retained in the charcoal filter is carried back to the water tanks provided it has not decayed. This way the xenon release to the environment is negligible in comparison with releases from other sources.

FIGURE 4.5. : GLAND LEAK-OF SYSTEM FLOW DIAGRAM

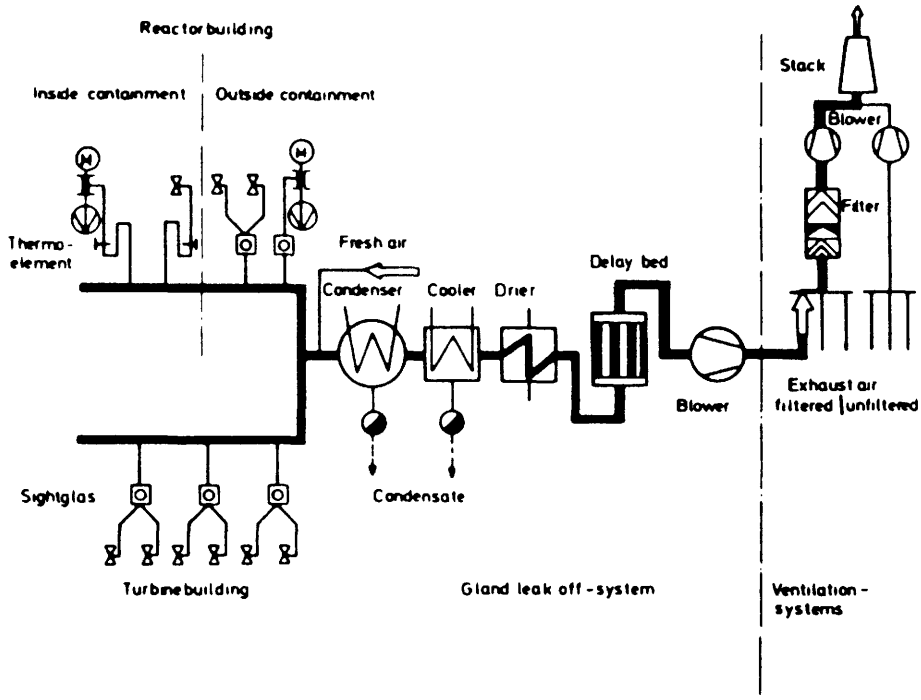
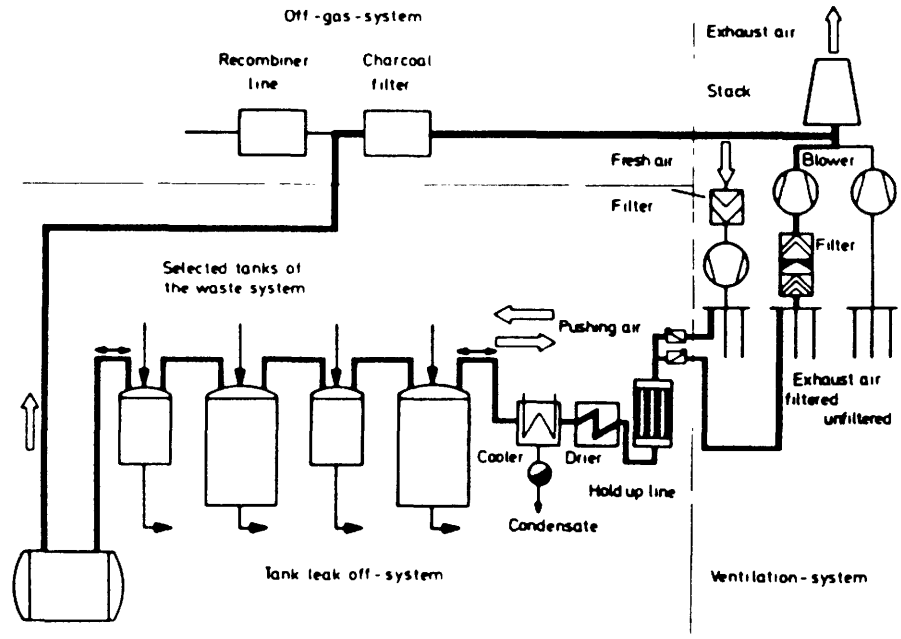


FIGURE 4.6. : TANK LEAK-OFF SYSTEM FLOW DIAGRAM



4.2.3. Off-gas system

4.2.3.1. Process description

A flow diagram of off-gas system is shown on FIGURE 4.7.

The gaseous substances formed in the reactor include noble fission gases from the fuel and activated gases from the coolant as well as free hydrogen and oxygen which originate from the radiolytical decomposition of the coolant. This gas mixture is carried with the generated steam from the reactor to the turbine. Air inleakage into the turbine condenser takes place despite of careful sealing and is added to the gas volume coming from the nuclear reactor. This gas mixture is extracted continuously from the turbine condenser to maintain the operating vacuum in the condenser. This way, the majority of the radioactive isotopes which were formed in the reactor are removed from the primary circuit via the turbine condenser for treatment in an off-gas system.

The off-gas mixture is evacuated from the turbine condenser by steam jet-air ejectors, diluted with driving steam and routed to the off-gas system. The steam-gas mixture is slightly superheated in a preheater and passed through a recombiner to combine the free hydrogen and oxygen at the surface of a catalyst. This process is exothermic and the heat release results in a temperature increase of the gas mixture up to 399 °C. The steam gas mixture is then directed into a condenser where the dilution steam and steam formed from the radiolytic gas composition is condensed and drained. The remaining gas stream represents only 10 percent of the original gas flow and consists mainly of air. This air is cooled to ambient temperature in an after cooler and then dried in a chiller system by freezing out the water. The required delay of the gas stream for decay of the radioactivity takes place in a charcoal adsorption system which operates in two steps and at ambient temperature. A pread-sorber is used for decay of the short half life xenon and krypton isotopes which include the main portion of the total activity. In the preadsorber most of the decay heat and most of the solid daughter products are generated. The remaining long-half life fission gases and activation gases decay in the main adsorbers to values below permissible release rates. A micro filter is used to filter the off-gas stream coming from the adsorbers before it is diluted with the building ventilation exhaust air and released to the atmosphere. Because of the relatively high activity level in the off-gas system leaks can seriously affect the activity levels in the rooms. Therefore the off-gas system is operated at a slight vacuum which is maintained by water ring pumps.

The decay times of the various isotopes in the adsorption system are selected such that at the outlet only the isotop Kr 85 is governing the rest activity. For a activation and noble gas mixture of a BWR this will call for a decay time for xenon of about 50 days. The resulting activity release rates to the environment are then far below permissible limits related to the design release rates from the core. They allow the reactor to be kept in operation at higher than design release rates from the core

unless other criteria call for a shutdown. To achieve the above mentioned decay time with an economically design system, the air inleakage in the turbine and main condenser must be limited by appropriate means.

4.2.3.2. Block diagram

Following general assumptions have been taken into account :

- 438000 Ci/a are handled and released.

- Retention time in the decay unit :

  60 days for Xenon
  60 hours for Krypton

- Iodine is retained and adsorbed in the decay unit (efficiency = 99.9 %).

- Aerosols are adsorbed in the decay unit
  (efficiency = 99 %)

- Activity : $At = Ao\ e^{\left(- \frac{\text{Log } 2}{T_{1/2}} t \right)}$

FIGURE 4.7 : OFF-GAS SYSTEM FLOW DIAGRAM

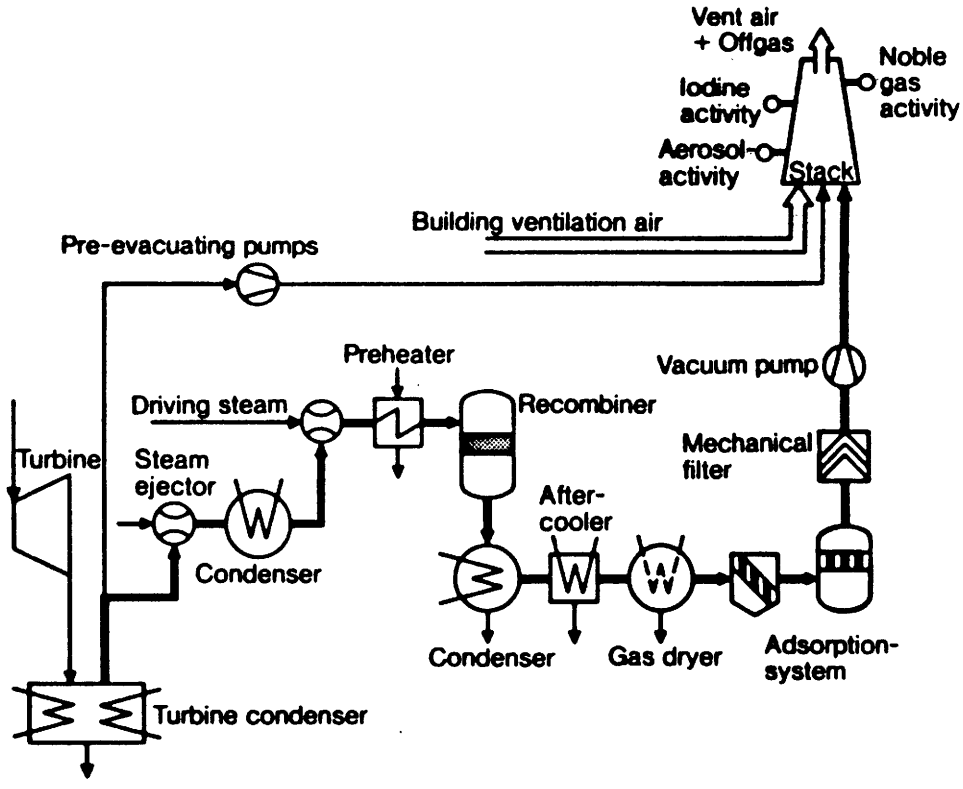


FIGURE 4.8 : OFF-GAS SYSTEM BLOCK DIAGRAM

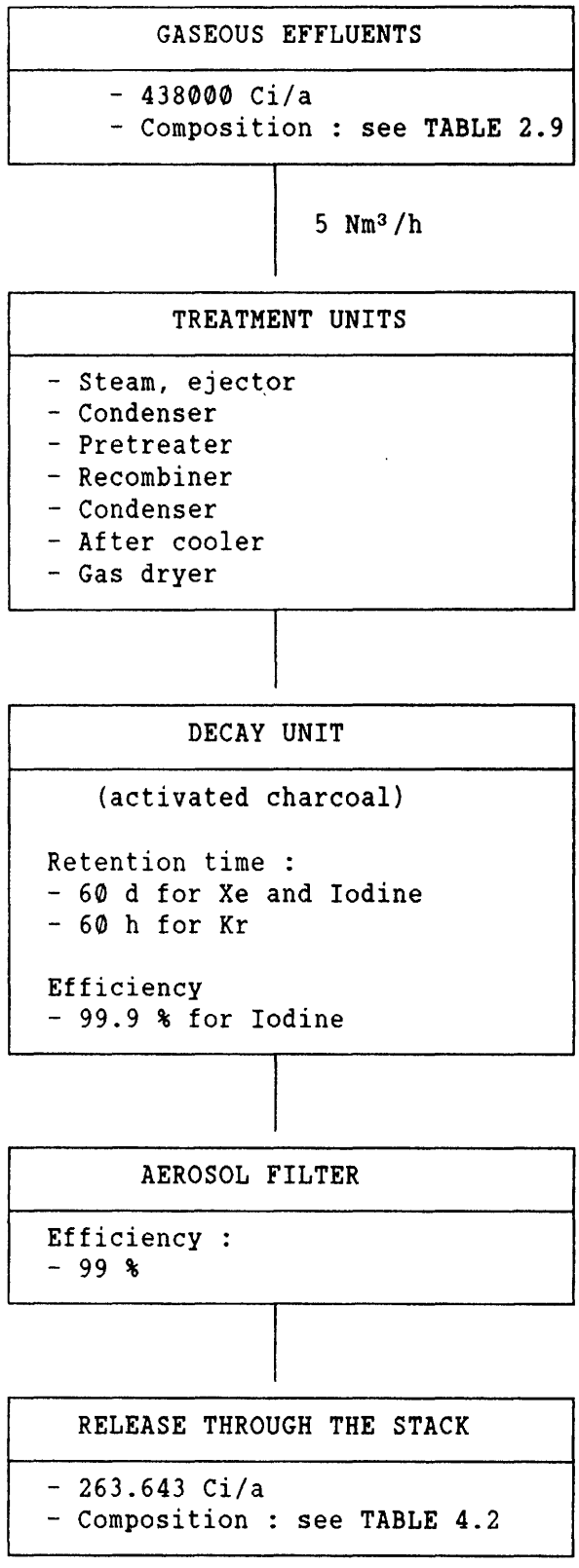


TABLE 4.2 : RADIONUCLIDE COMPOSITION AT OFF-GAS SYSTEM EXIT

RADIONUCLIDES	HALF LIFE	INLET COMPOSITION (%)	INLET ACTIVITY (Ci/a)	RETENTION TIME OR EFFICIENCY	OULET ACTIVITY (Ci/a)	OUTLET COMPOSITION (%)
C - 14	5700 y	0.00001	0.043	0 %	0.043	0.017
Kr - 85	10.6 y	0.03	132	60 h	131.342	49.818
Kr - 85 m	262 mn	1.83	8015		0.584	0.222
Kr - 87	78 mn	1.25	5475		0	0
Kr - 88	166 mn	3.32	14542		0.004	0.001
Xe - 133	5.27 d	80.41	352196	60 d	131.670	49.942
Xe - 133 m	2.3 d	1.75	7665		0	0
Xe - 135	552 mn	11.31	49538		0	0
I - 131	8.05 d	0.01	44	60 d	0	0
I - 132	138 mn	0.02	88		0	0
I - 133	1248 mn	0.03	132	99.9 %	0	0
I - 134	52.5 mn	0.01	44		0	0
I - 135	402 mn	0.02	88		0	0
Aerosols	-	0.00001	0.043	99 %	0.00043	0
TOTAL		100	438000		263.643	100

4.2.4. Comparison with discharge limits

The total released activity is as follows :

TABLE 4.3 : ACTIVITY RELEASED BY BOTH OFF-GAS SYSTEM AND VENTILATION

RADIONUCLIDES	ACTIVITY FROM OFF-GAS SYSTEM Ci/a	ACTIVITY FROM VENTILATION Ci/a	ACTIVITY RELEASED THROUGH THE STACK Ci/a	PER TYPE
C - 14	0.012	0	0.043	0.043
Kr - 85	131.342	0.05	131.392	
Kr - 85 m	0.584	3.20	3.784	
Kr - 87	0	2.19	2.19	
Kr - 88	0.004	5.82	5.824	438.63
Xe - 133	131.670	140.88	272.55	
Xe - 133	0	3.07	3.07	
Xe - 135	0	19.82	19.82	
I - 131	0	0	0	
I - 132	0	0	0	
I - 133	0	0	0	$< 10^{-3}$
I - 134	0	0	0	
I - 135	0	0	0	
Aerosols	0	0	0	$< 10^{-3}$
TOTAL	263.643	175.03	438.67	438.67

The activity released for each gaseous effluent through the stack is much lower than objective values :

TABLE 4.4 : COMPARISON WITH DISCHARGE LIMITS

GASEOUS EFFLUENTS	DESIGN VALUE Ci/a	OBJECTIVE VALUE Ci/a	RELEASE VALUE Ci/a
Noble gases	20000	3000	438.63
Halogens	0.05	0.01	$< 10^{-3}$
Aerosols	1.5	0.03	$< 10^{-3}$
Tritium	500	150	

4.3. LIQUID EFFLUENTS

4.3.1. Liquid waste treatment system

4.3.1.1. Process description

A flow diagram of liquid waste treatment system is shown on FIGURE 4.9.

Any water which is collected in sumps or tanks within the controlled area of the nuclear power station is considered waste water. Such water is not allowed to be discharged to the environment for reasons of environmental protection and must therefore be treated within the nuclear power station. The radioactivity of the waste water is caused by fission products and activated impurities which are dissolved or suspended in the water together with the non-radioactive products. The waste water is decontaminated by purification to the best practicable extend. If adequate purification methods are used it becomes possible to convert most of the waste water into water of feed water quality and hence most of the water can remain within the water household of the power station. This is important from the economical point of view because less make-up water is required and especially from the environmental point of view because the discharge of radioactivity is drastically reduced.

The waste water can be divided into 3 groups depending on the type and quantity of the impurities :

- Leakages from the primary system and the connected systems having a very low content of ional and solid impurities.

- Floor drain waters, laboratory drains and decontamination drains having higher contents of ional and solid impurities.

- Washing waters from washing machines, showers and sinks which are only slightly radioactive but contain high levels of solid and ional impurities.

For handling the different types and quantities of waste water a standard system was developed which consists of three independent treatment chains with one group of waste water being assigned to each chain for treatment.

The waste water from the primary cooling system and from the auxiliary systems containing very low impurities is collected in the liquid waste tanks and subsequently filtered in a precoat filter to remove undissolved impurities. The remaining dissolved impurities are removed in a demineralizer which is located down-stream of the filter. The treated water is collected in a clean water storage tank. The water is of reactor feed water quality and can therefore be reused in the reactor coolant systems.

The waste water originating from sumps, laboratories and decontamination stations containing higher impurities are collected in various evaporator feed tanks, sampled and prepared for the evaporation process. The pre-treated water is fed continuously into an evaporator. The evaporator distillate is collected in distillate storage tanks. The quality of the

distillate is such tant it can eitheir be converted into water of feed water quality by additional treatment in the demineralizer chain or it can be diluted with cooling water and discharged.

The laundry and washing water contains large amounts of impurities but are only slightly radioactive. These wastes are collected in the laudry and washing water tank and are then passed through the pre-coat filter in the laundry and washing water chain for removal of undissolved impurities. The filtered wastes are collected in a discharge tank and will be diluted with cooling water and released.

Wet solid waste pretreatment :

A flow diagram of wet solid pretreatment system is shown on FIGURE 4-10.

Effluent purification processes produce wet solid wastes :

- concentrates from evaporation,

- powder resins from demineralization,

- sludge from filtration.

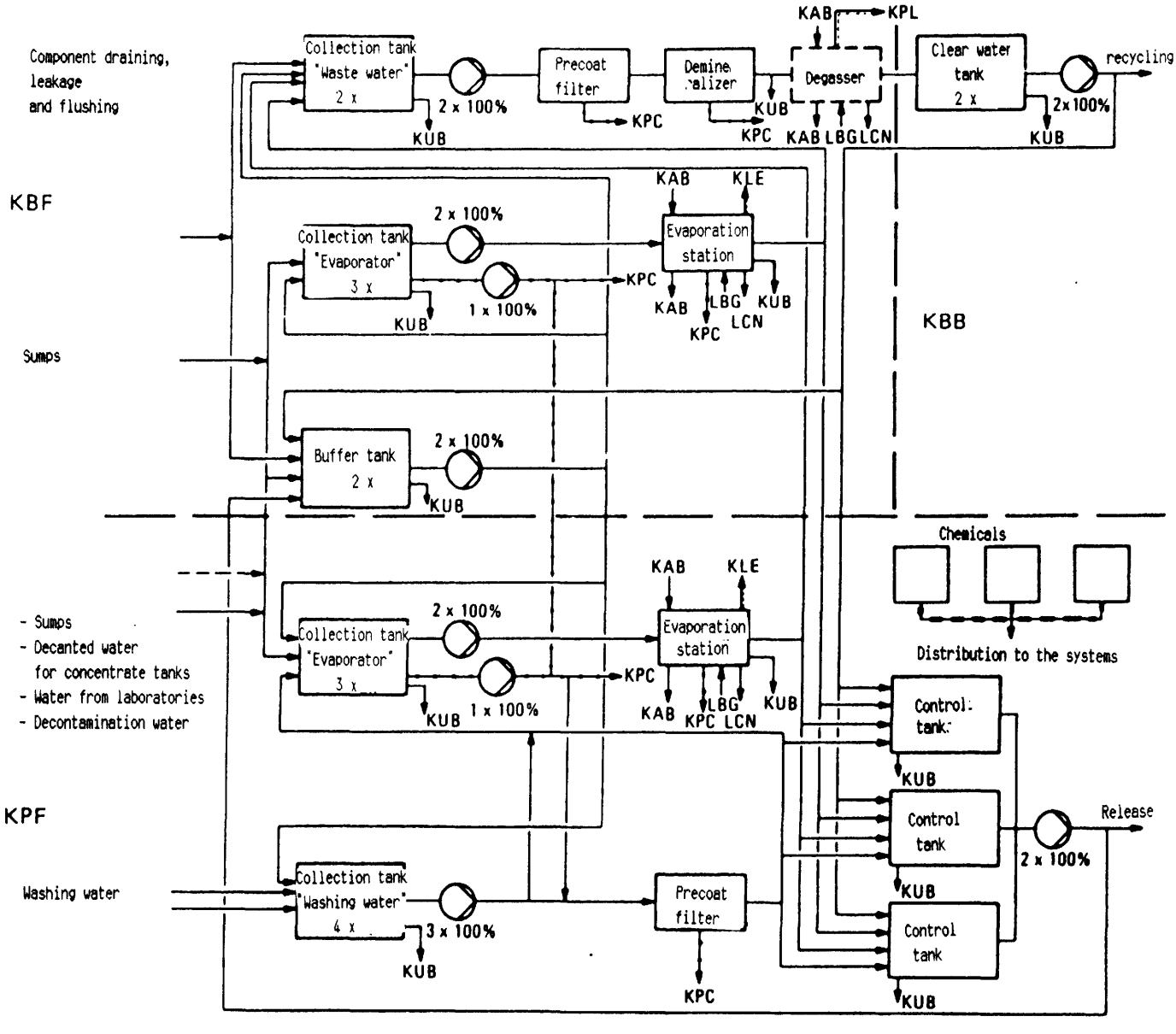
Evaporator concentrates are stored usually in 4 tanks (4 x 90 m^3).

Powder resins are generated by primary coolant and spent fuel pit purification systems. They are stored in 6 tanks (4 x 90 m^3, 2 x 45 m^3).

Bead resins are generated by condensate cleaning. They are stored in 2 tanks (2 x 30 m^3).

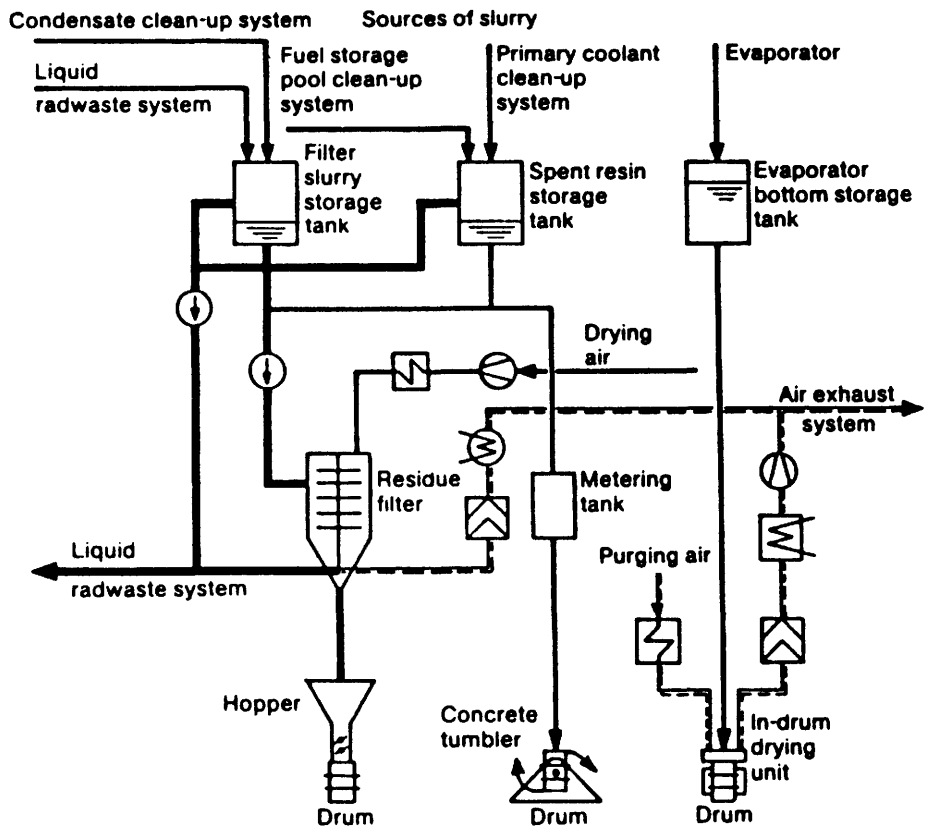
Solid wastes are pretreated and then conditioned into mobile units which are connected to the storage tanks. Processes are described in chapter 4.4.

FIGURE 4.9. : LIQUID WASTE TREATMENT SYSTEM FLOW DIAGRAM



KAB : Component cooling system
KBB : Primary coolant storage system
KBF : Primary coolant treatment system
KLE : Ventilation
KUB : Sampling system
KPC : Concentrate treatment system
KPF : Liquid waste treatment system
KPL : Off-gas system
LBG : Auxiliary steam feed system
LCN : Steam condensation, collection and recirculation system

FIGURE 4.10 : WET SOLID WASTE PRETREATMENT SYSTEM FLOW DIAGRAM



4.3.1.2. Block diagram

FIGURE 4.11. shows a typical German liquid treatment flow chart.

The block diagram with European inventory is shown on FIGURE 4.12.

The following assumptions have been taken into account :

- Decontamination factors :

  . evaporation : DF = 10^4

  . demineralization : DF = 100

  . filtration : DF = 50

- Liquid treatment :

  . Feed water, leaks ... : 15000 m^3 --> demineralization

  . Drain waters from building : 5000 m^3 --> evaporation + demineralization

  . Laundry waste : 3000 m^3 --> evaporation
    2000 m^3 --> filtration

  . Laboratory, decontamination : 500 m^3 --> evaporation

  . Waste from decantation : 2000 m^3 --> demineralization

Processed liquids are stored in 2 purified water tanks (recycling) and 3 discharge tanks (release).

They are released if their activity is lower than 5.10^{-4} Ci/m^3.

Their activity after dilution has to be lower than 10^{-7} Ci/m^3.

- Secondary solid wastes (produced and stored in the liquid waste treatment system) :

  . concentrates (evaporation),
  . powder resins (demineralization)
  . sludge (filtration)

  V = 198.5 m^3/a

  A = 3.75 Ci/m^3

They are stored in specific tanks and sent to the same conditioning unit.

- Primary solid wastes (only stored in the liquid waste treatment system)

  . Powder resins from primary coolant and spent fuel pit purification systems : V' = 1,5 m^3/a, A' = 40 Ci/m^3.

They are stored in the same tanks as above and sent to the same conditioning unit.

. Bead resins from condensate cleaning : V" = 4 m^3/a, A" = 50 Ci/m^3.

They are stored in a separate tank and sent to another conditioning unit.

FIGURE 4.11 : TYPICAL LIQUID TREATMENT FLOW CHART

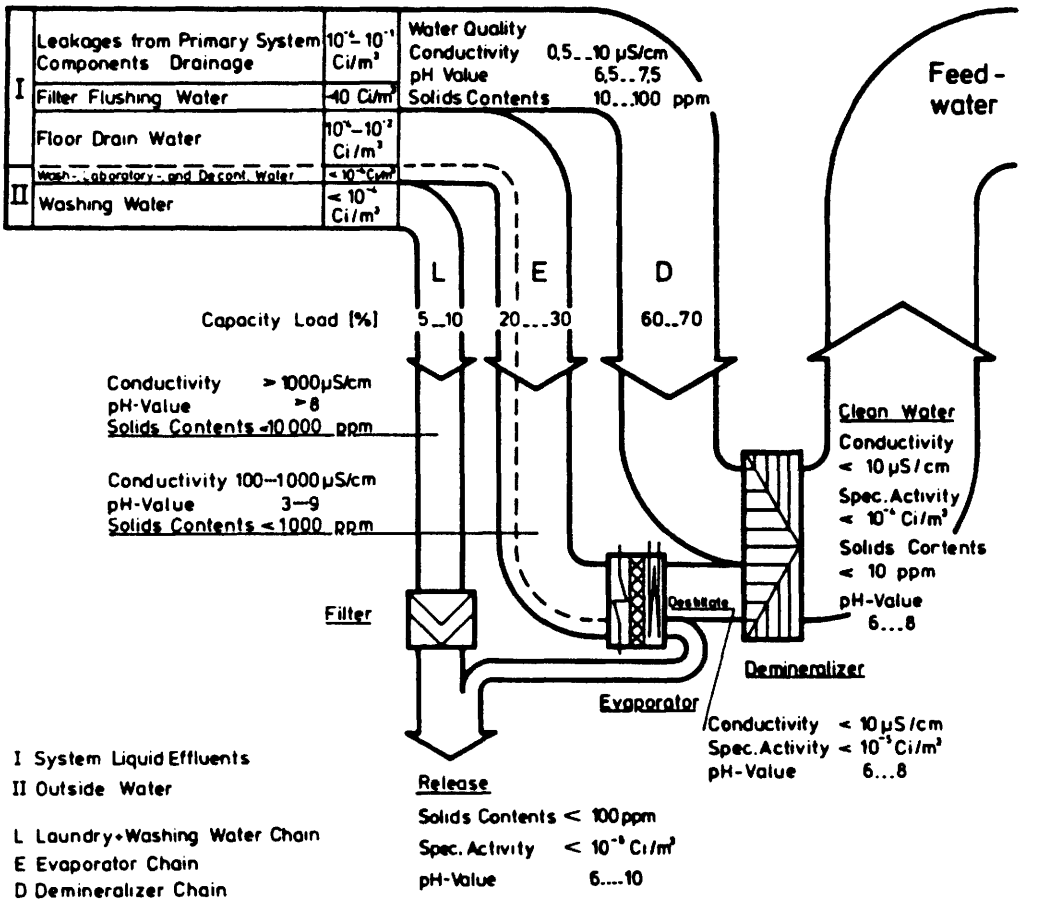
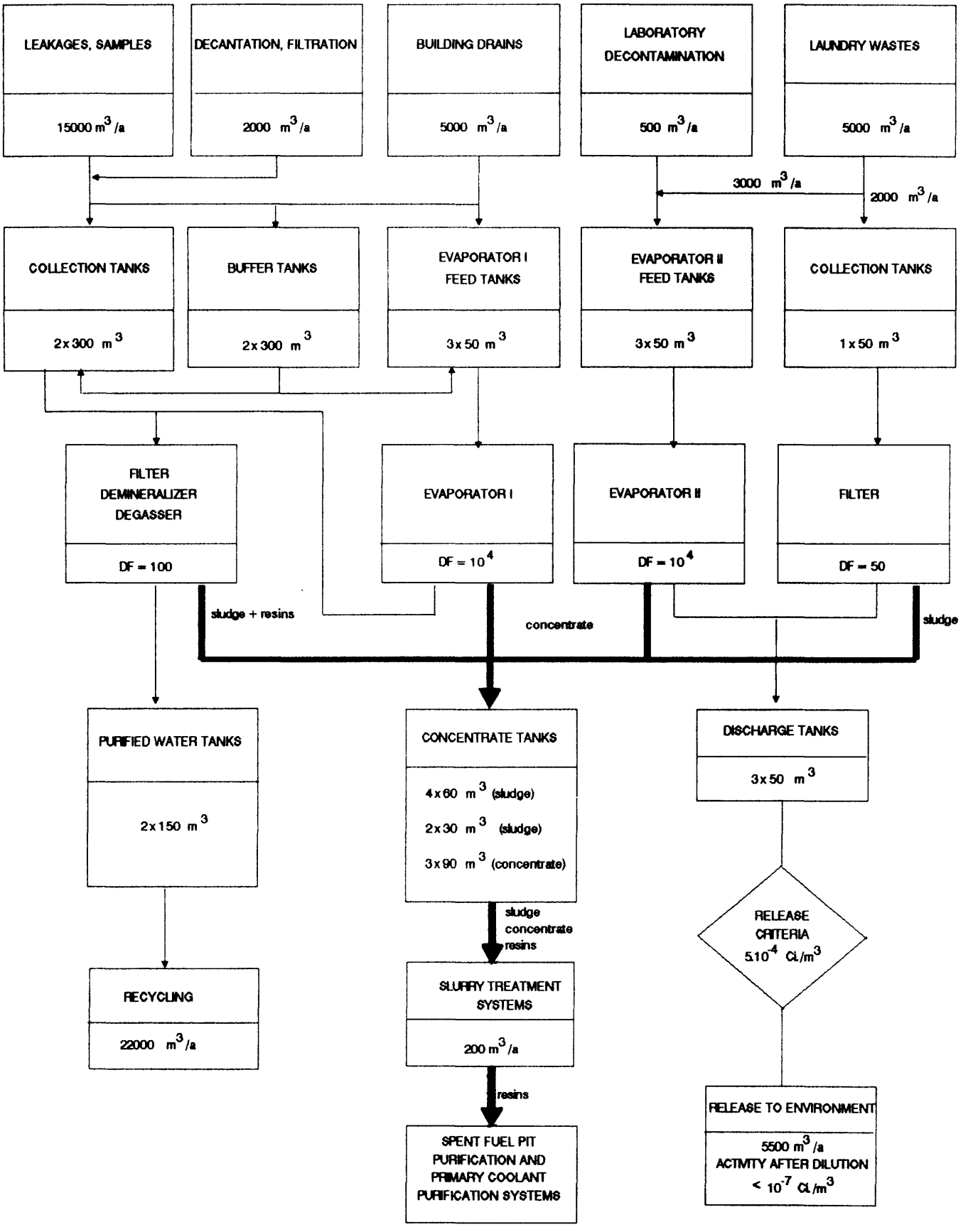


FIGURE 4.12 : LIQUID WASTE TREATMENT SYSTEM BLOCK DIAGRAM



4.3.1.3. Releases

TABLE 4.5 : AMOUNTS AND ACTIVITY OF RELEASED/RECYCLED LIQUIDS

	VOLUME	INLET ACTIVITY	PROCESS DF	OUTLET ACTIVITY (w/o H_3)	H_3 INLET-OUTLET ACTIVITY
Leakages, samples ...	15000 m^3/a	$5 10^{-2}$ Ci/m^3 750 Ci/a	Demine-ralization DF = 100	$5 10^{-4}$ Ci/m3 7.5 Ci/a	$5 10^{-4}$ Ci/m^3 7.5 Ci/a
Building drains	5000 m^3/a	10^{-3} Ci/m^3 5 Ci/a	Evap.+Demin. DF = 10^6	10^{-9} Ci/m^3 $5 10^{-6}$ Ci/m^3	10^{-5} Ci/m^3 0.05 Ci/a
Decantation, filtration	2000 m^3/a	10^{-4} Ci/m^3 0.2 Ci/a	Demineraliz. DF = 100	10^{-6} Ci/m^3 $2 10^{-3}$ Ci/a	10^{-6} Ci/m^3 $2 10^{-3}$ Ci/a
TOTAL RECYCLED	22000 m^3/a	755.2 Ci/a		7.502 Ci/a	7.552 Ci/a
Laboratory decontamination	500 m^3/a	10^{-3} Ci/m^3 0.3 Ci/a	Evaporation DF = 10^4	10^{-7} Ci/m^3 $5 10^{-5}$ Ci/a	10^{-5} Ci/m^3 $5 10^{-3}$ Ci/a
Laundry wastes	3000 m^3/a	10^{-4} Ci/m^3 0.3 Ci/a	Evaporation DF = 10^4	10^{-8} Ci/m^3 $3 10^{-5}$ Ci/a	10^{-6} Ci/m^3 $3 10^{-3}$ Ci/a
	2000 m^3/a	10^{-5} Ci/m^3 0.02 Ci/a	Filtration DF = 50	$2 10^{-7}$ Ci/m^3 $4 10^{-4}$ Ci/a	10^{-7} Ci/m^3 $2 10^{-4}$ Ci/a
TOTAL RELEASED	5500 m^3/a	0.82 Ci/a		$4.8 10^{-4}$ Ci/a	$8.2 10^{-3}$ Ci/a

4.3.2. Comparison with discharge limits

Total released activity

Tritium :

- Liquid waste treatment system : 8.2 10^{-3} Ci/a (5500 m^3).

Other radionuclides

- Liquid waste treatment system : 4.8 10^{-4} Ci/a (5500 m^3).

Composition (without H_3)

TABLE 4.6 : RADIONUCLIDE COMPOSITION AT LIQUID TREATMENT SYSTEM EXIT

RADIONUCLIDE		Mn-54	Co-58	Co-60	Sr-90	Nb-95	Mo-99	Ag-110m
%		4.80	31.98	6.39	0.19	0.013	4.80	4.80
RADIONUCLIDE	Sb-124	I-131	I-132	I-133	I-134	I-135	Cs-134	Cs-137
%	4.80	0.46	0.80	1.39	0.46	0.93	19.19	19.19

The activity balances have been based on global decontamination factors.

Released activities remain below the objective values :

TABLE 4.7 : COMPARISON WITH DISCHARGE LIMITS

LIQUID EFFLUENTS	DESIGN VALUE Ci/a	OBJECTIVE VALUE Ci/a	RELEASE VALUE Ci/a
Total (H-3 excluded)	8	1	$4.8\ 10^{-4}$
H-3	500	100	$8.2\ 10^{-3}$

In the F.R.G., the authorized discharge limits are the design values (TABLE 2-18). The value reached by our treatment scheme corresponds to German practice.

4.4. SOLID WASTE

4.4.1. General description

4.4.1.1. Historical background

German BWR background is the same as the one described for German PWR's (see sec. 3.4.11).

4.4.1.2. Inventories

Primary waste inventories : see TABLE 2.12.

Secondary waste inventories

During waste management operations, wet active wastes are generated and mixed in the liquid waste treatment system :

- powder resins (demineralization),

- sludge (filtration),

- concentrate (evaporation).

They are mixed with powder resins produced in spent fuel pit purification system and primary coolant purification system.

The characteristics of primary and secondary solid wastes that are to be treated (conditioned and packaged) are as follows :

TABLE 4.8. : CHARACTERISTICS OF GENERATED SOLID WASTES

WASTES	PRIMARY WASTES		SECONDARY WASTES	
	VOLUME (m^3/a)	ACTIVITY (Ci/m^3)	VOLUME (m^3/a)	ACTIVITY (Ci/m^3)
WET ACTIVE WASTES				
. Bead resins	4	50		
. powder resins	1.5	40	198.5	3.75
. Concentrate + Sludge				
TECHNOLOGICAL WASTES				
. Combustible + Compactable	350	0.01		
. Non combustible + Compactable	100	0.01		
. Combustible + Non compactable	50	0.01		
. Non combustible + Non compactable	20	0.2		

4.4.1.3. Packagings

The packagings used to store the different solid waste defined in TABLE 2.31.

4.4.2. Technological (or mixed solid) wastes

4.4.2.1. Process description

The general practice is as PWR's one.

3.4.2.2. Block diagram

Figure 4.13 gives the block diagram for treatment of technological wastes.

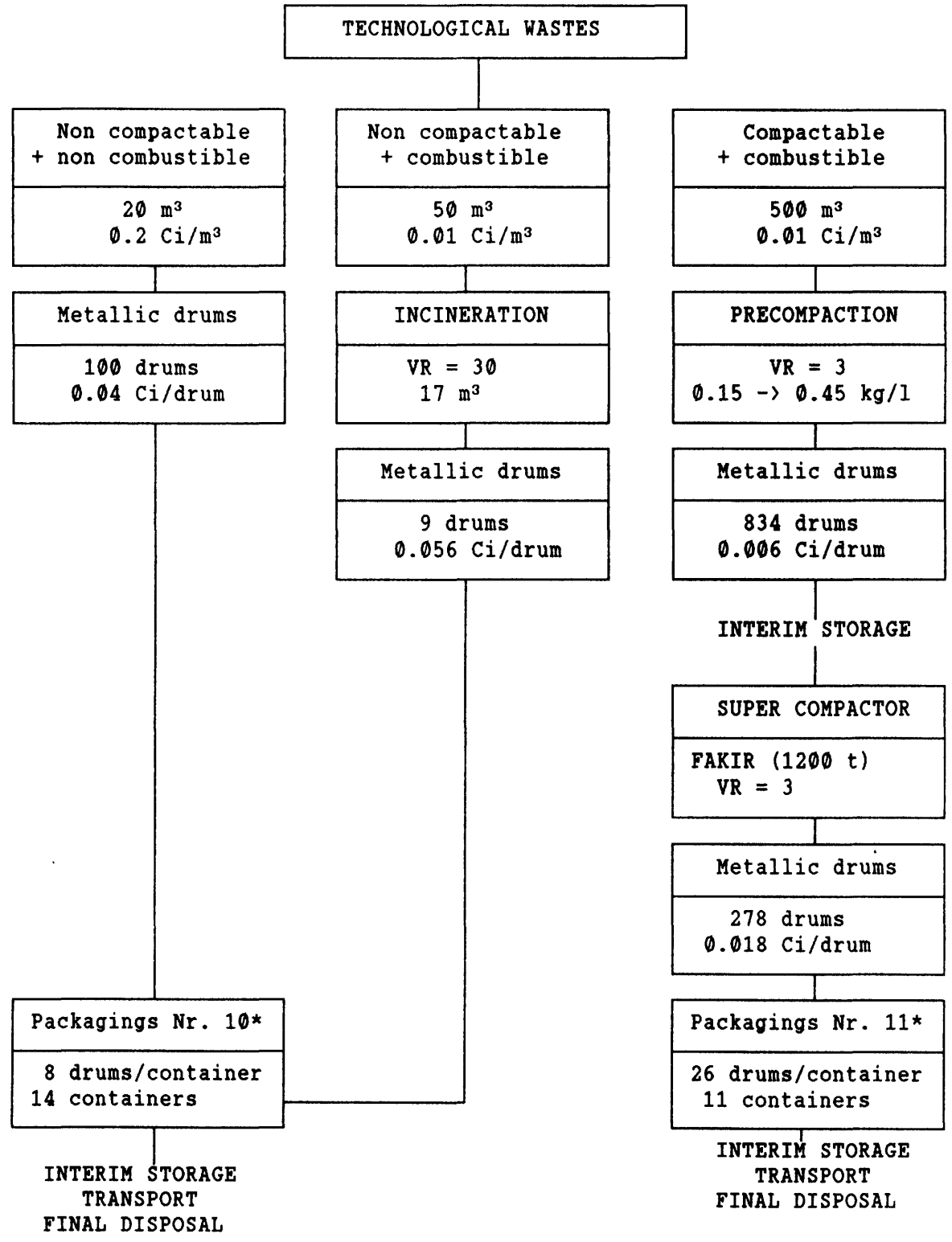
These ones are sorted and a relevant treatment is then applied.

- Compactable wastes (500 m^3/a) are precompacted, supercompacted (total VR = 9) and the final 200 metallic drums are stored in 11 containers (packagings Nr. 11 in TABLE 3.21) filled with concrete.

- Non compactable and non combustible wastes (20 m^3/a) are put into 100 metallic drums which are stored into 13 containers (packagings Nr. 10 in TABLE 2.31).

- Non compactable and combustible wastes (50 m^3/a) are incinerated (VR = 30) and the ashes are put into metallic drums which are stored in one of the 11 above mentioned containers.

FIGURE 4.13 : TECHNOLOGICAL WASTE TREATMENT BLOCK DIAGRAM



* See TABLE 2.31.

4.4.3. Ion exchange resins

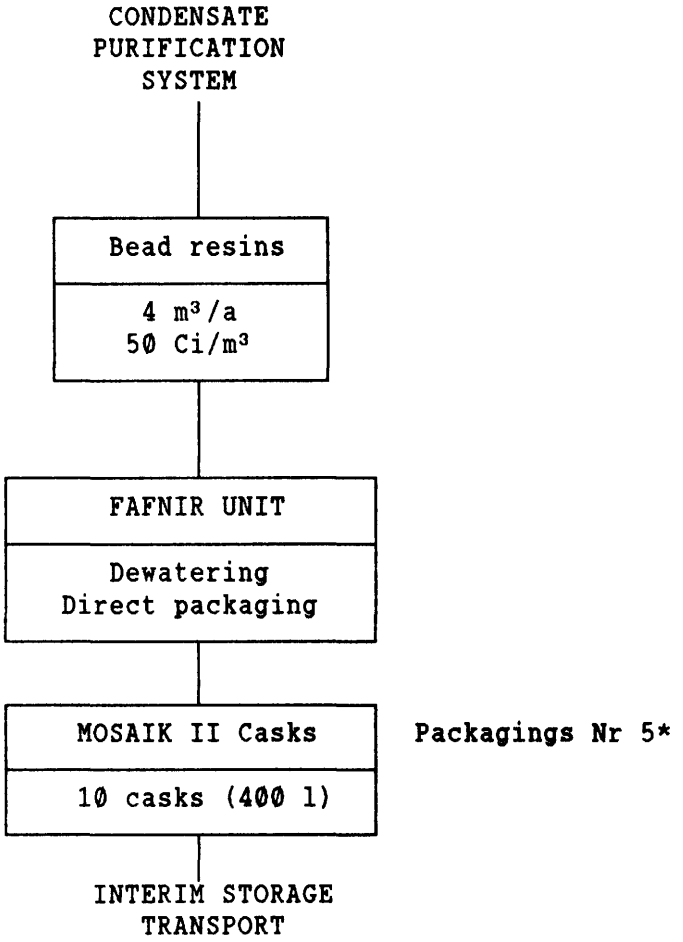
4.4.3.1. Process description

A mobile FAFNIR unit is connected to the spent resin tanks of the BWR. The resins are pumped by means of vacuum into MOSAIK II casks. After filling the casks, the resins are dewatered.

4.4.3.2. Block diagram

Figure 4.14. gives the block diagram for treatment of resins. 10 casks (packagings Nr. 5 in TABLE 2.31.) can be expected per year, each containing 400 l of wastes.

FIGURE 4.14 : RESIN TREATMENT BLOCK DIAGRAM



* See TABLE 2.31

4.4.4. Concentrate sludge and powder resins

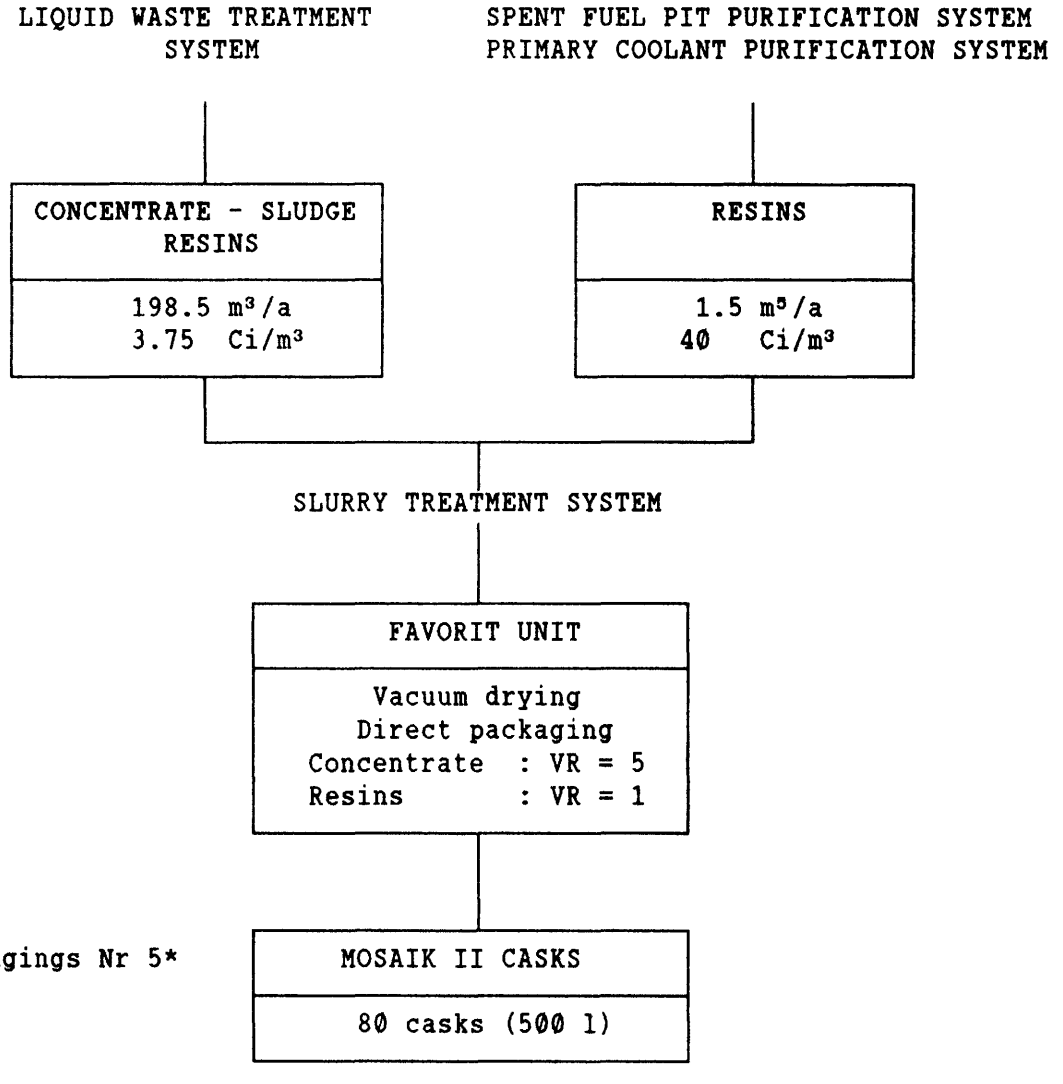
4.4.4.1. Process description

A mobile FAVORIT unit is connected to the slurry tanks. The slurry is filled into MOSAIK II casks. By means of vacuum and slight heat, the concentrate is dried to a solid block inside the cask.

4.4.3.2. Block diagram

Figure 4.15. gives the block diagram for treatment of concentrate, sludge and powder resins. 80 casks (packagings Nr. 5 in TABLE 3.21) can be expected per year, each containing 500 l of wastes.

FIGURE 4.15 : CONCENTRATE, SLUDGE AND RESIN TREATMENT BLOCK DIAGRAM



* See TABLE 2.31.

4.4.5. Core components

4.4.5.1. Process description

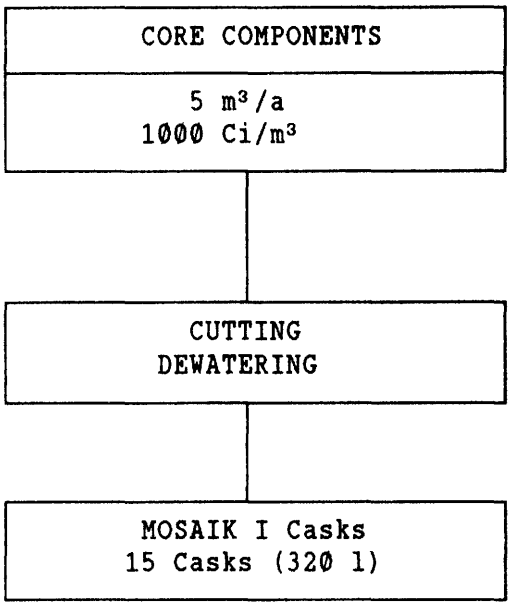
The packaging is preceded by underwater cutting and compacting.

The general practice is as PWR's one. The main core components of a BWR are water channels and control blades.

After cutting the pieces are packed into heavy shielded MOSAIK II casks. The useful volume per cask is about 320 l.

4.4.5.2. Block diagram

FIGURE 4.16 : CORE COMPONENT TREATMENT BLOCK DIAGRAM



* See table 2.31.

4.4.6. Scrap metal, active carbon filters

During back-fitting and decommissioning, there are considerable arisings of contaminated scrap metal.

These are used (up to 74 Bq/g) without decontamination as raw material for the fabrication of the above-mentioned MOSAIK casks.

The active carbon (charcoal filters) is generally of very low activity and can be used as fill-up in order to produce the right carbon content of the cast nodular iron, which requires around 4 weight %, while the input, ferritic steel, has only around 0,30 weight %.

The dust, slag, etc... in a weight proportion of 5 % go to the waste stream.

Due to the quality requirements from metallurgy, this scrap can only be used as add-up to clean raw material for the fabrication of waste casks. Proportions up to 1 can be accepted for more shielding blocks.

4.4.7. Characteristics of conditioned and packaged waste

TABLE 4.9. shows characteristics of conditioned waste (without packaging) :

- conditioning type,
- annual produced volume,
- specific activity.

TABLE 4.10. shows characteristics of packaged waste :

- type of package,
- number of packages, annual volume and specific activity,
- further packaging,
- final number of packages, annual volume.

TABLE 4.9 : CHARACTERISTICS OF CONDITIONED WASTES

WASTES	CONDITIONING OPERATION	VOLUME (*) m^3/a	ACTIVITY Ci/m^3
WET ACTIVE WASTES			
. Bead resins	Dewatering + Direct packaging (VR = 1)	4	50
. Concentrate + Sludge + Powder resins	Vacuum drying + Direct packaging (VR = 5)	40	20
TECHNOLOGICAL WASTES			
. Combustible + Compactable	Precompaction (VR = 3) + Super compaction (VR = 3)	38.9	0.09
. Non combustible + Compactable	Precompaction (VR = 3) + Super compaction (VR = 3)	16.7	0.09
. Combustible + Non compactable	Incineration (VR = 30	1.7	0.3
. Non combustible + Non compactable	Packaging	20	0.2

(*) Without shielded cask

TABLE 4.10 : CHARACTERISTICS OF PACKAGED WASTES

WASTES	TYPE OF PACKAGE	NR OF PACKAGES VOLUME (m^3/a) ACTIVITY (Ci/m^3)	FURTHER PACKAGING	NR OF PACKAGES VOLUME (m^3/a)
WET ACTIVE WASTES				
. Bead resins	MOSAIK II cask V_i = 400 l V_e = 1.3 m^3	10 4 50	- - -	10 13
. Concentrate + Sludge + Powder resins	MOSAIK II cask V_i = 500 l V_e = 1.3 m^3	80 40 25	- - -	80 104
TECHNOLOGICAL WASTES				
. Combustible + Compactable	Metallic drum (200 l)	194.5 38.9 0.09	11 CONTAINERS Ve = 10.9 m^3 26 drums/cont.	7.7 83.9
. Non combustible + Compactable	Metallic drum (200 l)	83.5 1.8 0.09		3.3 36.0
. Combustible + Non compactable	Metallic drum (200 l)	9 1.8 0.3	14 CONTAINERS Ve = 7.4 m^3 8 drums/cont.	1.2 8.9
. Non combustible + Non compactable	Metallic drum (200 l)	100 20 0.2	8 drums/cont.	12.8 94.7

4.5. COSTING OF GERMAN WASTE MANAGEMENT SCHEME

4.5.1. Assumptions

The costs provided in this section are based on the following assumptions :

- 1300 MWe BWR,

- one liquid waste treatment system per unit,

- prices of 1987 given in ECU.

4.5.2. Capital costs

The capital costs for the systems are turn-key prices and include costs for quality insurance, start-off operation, planning and licensing.

The following fractional costs for major equipment can be assumed :

- Quality insurance : 35 %,

- Installation : 35 %,

- Start-off : 10 %.

The "base value" (BV) calculated by TASK/KAH has to be derived from major equipment cost (ME) provided in the next paragraphs as follows :

$$BV = ME/1.8$$

4.5.2.1. Ventilation system (KPV)

The ventilation system of a BWR is part of the operational system of the plant and not only related to waste treatment functions. Only those parts are listed which have major functions for the gaseous waste treatment.

DIRECT COSTS

Major equipment (ME)

. Auxiliary systems buidlings : 3 Ventilators, 3 coolers	Mill. ECU	1.6
. Reactor building : Engines, heater, filtersystems, air recirculation system	Mill. ECU	4.2

. Turbine building : 2 ventilators air supply 1 heater 1 filtersystem 3 ventilators exhaust air	Mill. ECU	3.6
. Service building : 2 ventilators 1 filtersystem 1 heater	Mill. ECU	3.2
. Stack : iodine activity measuring system, aerosol activity measuring system, noble gas activity measuring system, exhaust air filter (prefilter, particulate filter, iodine filter, microfilter)	Mill. ECU	4.0
	Mill. ECU	16.6

Bulk materials

. channels	Mill. ECU	2.0
. valves	Mill. ECU	0.7
. isolation	Mill. ECU	0.7
. small parts	Mill. ECU	0.5
. El. supply, control	Mill. ECU	1.5

INDIRECT COSTS

. Planning, licensing	Mill. ECU	4.0
TOTAL	Mill. ECU	26.0
BV = ME/1.8 = 16.16/1.8 =	Mill. ECU	9.2

4.5.2.2. Leak-off systems (KLS)

DIRECT COSTS

Major equipment (ME)

. Gland leak-off system condenser, cooler, drier engines (blower), filtersystem	Mill. ECU	2.2
. Tank leak-off system condenser, cooler, drier, hold-up line, charcoal filter	Mill. ECU	1.9
TOTAL ME	Mill. ECU	4.1

Bulk materials

. Piping	:	Mill. ECU	Ø.3
. Valves	:	Mill. ECU	Ø.3
. Isolation	:	Mill. ECU	Ø.2
. Small parts	:	Mill. ECU	Ø.2
. El supply, control	:	Mill. ECU	Ø.4

INDIRECT COSTS

. Planning, licensing	:	Mill. ECU	1.Ø
TOTAL	:	Mill. ECU	6.5
BV = ME/1.8 = 4.1/1.8	:	Mill. ECU	2.3

4.5.2.3. Off-gas system (KPL)

DIRECT COST

Major equipment (ME)

. Pumps, compressors	:	Mill. ECU	Ø.8
. Tanks	:	Mill. ECU	1.Ø
. Filter, heat exchangers, cooler, dryer, etc...	:	Mill. ECU	1.2
. Recombiners	:	Mill. ECU	1.Ø
. Adsorption system	:	Mill. ECU	2.6
TOTAL ME	:	Mill. ECU	6.6

Bulk materials

. Piping	:	Mill. ECU	1.4
. Valves	:	Mill. ECU	Ø.8
. Isolation	:	Mill. ECU	Ø.6
. Small parts	:	Mill. ECU	Ø.8
. El supply, control	:	Mill. ECU	1.6

INDIRECT COST

. Planning, licensing	:	Mill. ECU	2.2
TOTAL		Mill. ECU	14.Ø
BV = ME/1.8 = 6.6/1.8	:	Mill. ECU	3.7

4.5.2.4. Liquid waste treatment system (KPF)

Including concentrate, resins and sludge storage and circulating devices.

DIRECT COST

Major equipment (ME)

Coolant treatment system

. Pumps	:	Mill. ECU	0.3
. Tanks	:	Mill. ECU	2.2
. Demineralizer	:	Mill. ECU	0.4
. Precoat filter	:	Mill. ECU	0.2

Waste water and concentrate treatment

. Pumps	:	Mill. ECU	0.8
. Tanks	:	Mill. ECU	2.2
. Filters	:	Mill. ECU	0.4
. Evaporators	:	Mill. ECU	2.0

Laundry + washing water treatment

. Pumps	:	Mill. ECU	0.7
. Tanks	:	Mill. ECU	1.0
. Precoat filter	:	Mill. ECU	0.2
. Centrifuge	:	Mill. ECU	1.5
TOTAL ME		Mill. ECU	11.9

Bulk materials

. Piping	:	Mill. ECU	2.3
. Valves	:	Mill. ECU	1.0
. Isolation	:	Mill. ECU	0.8
. Small parts	:	Mill. ECU	1.0
. El. supply, control	:	Mill. ECU	1.8

INDIRECT COST

. Planning, licensing	:	Mill. ECU	2.Ø
TOTAL		Mill. ECU	2Ø.8
BV = ME/1.8 = 11.9/1.8		Mill. ECU	6.6

4.5.2.5. Solid waste treatment system

Mobile units are bought and used for solid waste conditioning within the 2Ø GWe nuclear park :

- 2 supercompactors FAKIR (mixed solid wastes)	:	Mill. ECU	2.Ø
- 2 FAFNIR units (resin conditioning)	:	Mill. ECU	2.Ø
- 3 FAVORIT units (concentrate conditioning)	:	Mill. ECU	4.5

The price of the incinerator (turn-key) is 7.5 Mill. ECU but for the study we recommend to take no capital cost but cost for rented incineration service of 18 ECU/kg of incinerated wastes (included shipment).

4.5.2.6. Total capital cost

KPV	:	Mill. ECU	26.Ø
KLS	:	Mill. ECU	6.5
KPL	:	Mill. ECU	14.Ø
KPF	:	Mill. ECU	2Ø.8
Solid waste treatment systems	:	Mill. ECU	8.5
TOTAL	:	Mill. ECU	75.8

4.5.3. Building cost

4.5.3.1. Auxiliary building

All the above mentioned systems are located in the auxiliary reactor building. This auxiliary building is part of the power plant.

The building volume is approx. 67,000 m³. On the agreed basis of 135 ECU/m³ a total building cost of Mill. ECU 9.1 can be considered.

4.5.3.2. Storage building

The annual production of waste packages for one 1300 MWe plant is a follows :

TABLE 4.11 : TOTAL EXTERNAL VOLUME OF PACKAGES

PACKAGES	EXTERNAL PACKAGE Volume m³
14 containers (Type IV)	103.6
11 containers (Type V)	119.9
90 casks (Type II)	117.0
15 casks (Type I)	10.5
TOTAL/YEAR	351.0 m³

The annual storage volume necessary for 20 GWe is :

(103.6 x 20)/1.3 = 1594.3 m³/a

(119.9 x 20)/1.3 = 1845.7 m³/a

(117.0 x 20)/1.3 = 1800 m³/a

(10.5 x 20)/1.3 = 162 m³/a

TOTAL 5401 m³/a (20 GWe)

The volume of interim storage capacity will be calculated by TASK, knowing that containers can be pilled up as follows :

- 6 containers Type V and IV,

- 3 casks Type I and II.

4.5.4. Operating costs

4.5.1. Waste treatment system operation

The operating costs are personal costs of power plant personnel working (part time) with the treatment systems. The following annual man-power (man-hours) for one 1300 MWe plant can be considered :

Plant systems	
Ventilation KPL-System KPF-System	5.600 man-hours
Technological waste collection and sorting	14.300 man-hours
Mobile systems	
Wet solid waste treatment	2.000 man-hours
Mixed dry solid waste treatment	6.700 man-hours
TOTAL	28.600 man-hours

4.5.4.2. Material, energy consumption

The costs for material and energy consumption (steam, electricity, etc.) are not derermined and negligible compared to all other costs.

4.5.4.3. Package costs

Prices for waste packages

The prices for waste packages given below are based on 1988 values. The prices per piece are :

- Container Type IV (N° 10) : ECU 6,000
- Container Type IV (N° 11) : ECU 6,500

- Cask Type II (N° 5) : ECU 13,000

- Cask Type I (N° 4) : ECU 15,000

TABLE 4.12 : TOTAL PRICES OF PACKAGES

WASTE PACKAGE	N° per year 1200 MWe	N° per year 20 GWe	Price Mill ECU
Container IV	14	234	1.4
Container V	11	184	1.2
Cask Type II	90	1500	19.5
Cask Type I	15	250	3.8
TOTAL			25.9

4.5.4.4. Transport cost

Transport will be done by rail only in Germany. The mean shipping costs (Round trip) of waste packages for distances given in this study (2 x 500 km) are :

- Cask Type I, II : ECU 1,000

- Container Type IV, V : ECU 2,000

The waste packages are shipped in 20'-containers. Three cylindrical casks (Type I, II) and 2 containers (Type IV, V) are necessary for one shipment.

The equipment costs for transport equipment of one 1300 MWe PWR plant is approximately ECU 200,000.

4.5.4.5. Interim storage and final disposal costs

The operating costs for interim storage of waste packages delivered to a central storage facility or to an on-site storage facility are 3500 man-hours per year for a 1300 MWe PWR.

The operating costs for final disposal are :

ECU 2,500 per m^3 of waste package.

For a 1300 MWe plant annual disposal costs of about ECU 877,500 can be considered.

4.5.4.6. Incineration service cost

50 m^3 of combustible waste are incinerated per year in a 1300 MWe plant. This corresponds to 25 t.

The service price for incineration including shipment is 18 ECU/kg.

For a 20 GWe park, the following annual incineration cost arises :

25000 x (20000/1200) x 18 = 6.9 Mill. ECU.

5. SENSITIVITY STUDIES

In the first part of this section, the replacement of incineration by super-compaction for dry solid wastes is discussed. The changes in waste volumes and the related cost impact is shown.

In the second part of this section, the differences for wet waste treatment using cementation versus drying are analyzed.

5.1. INCINERATION AND COMPACTION

5.1.1. Different modes of waste treatment

Regarding dry solid wastes, two different treatment processes can be applied, namely compaction and incineration, which may complement one another. The possible treatment modes for dry solid wastes are :

a) No compaction, no incineration.

b) Pre-compaction, no incineration
   Pre-compaction force : 16 tons to 30 tons.

c) Pre-compaction and incineration, cementation of ashes.

d) Super-compaction, no incineration
   Super-compaction force : > 1000 tons.

e) Super-compaction and incineration, supercompaction of ashes.

There are technical and enonomical reasons for the selectiosn of the different possible treament modes.

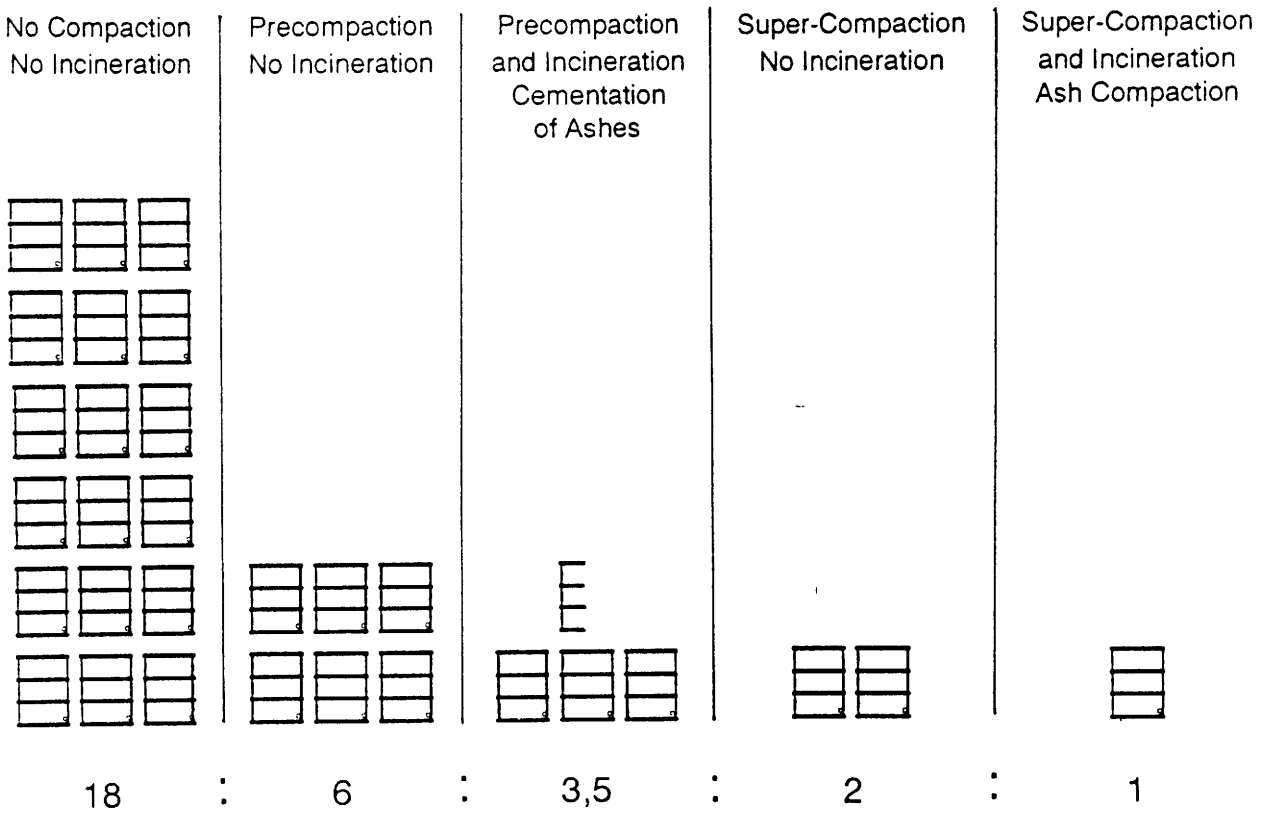
In German nuclear power plants modes (a), (b) and (c) are not used. All power plants used pre-compaction with a compaction force of about 20 tons. If incineration is carried out, always super-compaction will be applied as well. For the German situation modes (d) and (e) are of importance.

5.1.2. WASTE VOLUMES

It can be assumed that about 40 % to 50 % of the total dry solid wastes is combustible. Only this portion can be incinerated. The remaining portion, 50 % to 60 %, can be compacted. In FIGURE 5-1 the volume reduction ratios for different treatment modes of dry solid waste are shown. The figures are based on the assumption of 50 % combustible and 50 % uncombustible parts in compactable mixed solid wastes.

The combination of high-force compaction (> 1000 tons) and incineration with additional compaction of the incineration residues instead of cementation of ashes gives the maximum volume reduction of waste products to be stored or disposed. Super-compaction without using incineration seems to be the second best option with regard to volume reduction.

Pre-compaction with simple drum-compactors (16 tons to 30 tons compaction force) is a very efficient first step to reduce drum handling operations by decreasing the waste volumes (factor of 3).



Precompaction force : 20 tons
Supercompaction force : 1500 tons

FIGURE 5-1 : VOLUME REDUCTION RATIOS FOR DIFFERENT TREATMENT MODES OF SOLID RADIOACTIVE WASTES

5.1.3. OPERATIONAL COSTS

5.1.3.1. Influence of safety aspects

The decision whether super-compaction or/and incineration is used for treatment of mixed solid wastes is not only based on an economical analysis. There are additional safety-related and legal aspects which make it difficult to evaluate and compare the different treatment modes. Examples for safety-related reasons are :

Supercompaction :

- Reduction of potential fire risk during storage, because the waste is compacted into sheet metal cartridges. The compacted waste is nearly uncombustible.
- Reduction of corrosion risk due to dewatering the waste during compaction.
- Compaction > 300 bar in Germany required for final disposal in deep geological formations.

Incineration :

- Elimination of all organic material and combustible material from waste. No biologic activity in the waste. No fire risk during storage inside a facility.

The decision what kind of method has to be applied for waste treatment is in many cases derived from legal aspects and licensing conditions, such as occupational dose exposure limitations, which sometimes do not allow to make an extensive sorting of combustible and uncombustible parts of mixed solid wastes.

5.1.3.2. Costs for waste handling

The annual operating costs are personal costs for waste collection, sorting and treatment. For handling and treatment of solid wastes about 15.000 man-hours are needed in a 1300 MWe PWR. This assumption is valid for treatment of 380 m^3 of waste, where 360 m^3 super-compacted and only 20 m^3 are prepared for incineration. If incineration has to be applied for all combustible waste parts, the effort for waste sorting increases. The operating costs given in related man-hours for the different treatment modes discussed in Sec. 5.1.1. are as follows :

a) No compaction, no incineration : 11.500 man-hours less sorting, but more drums and related handling.

b) Pre-compaction, no incineration : 10.000 man-hours.

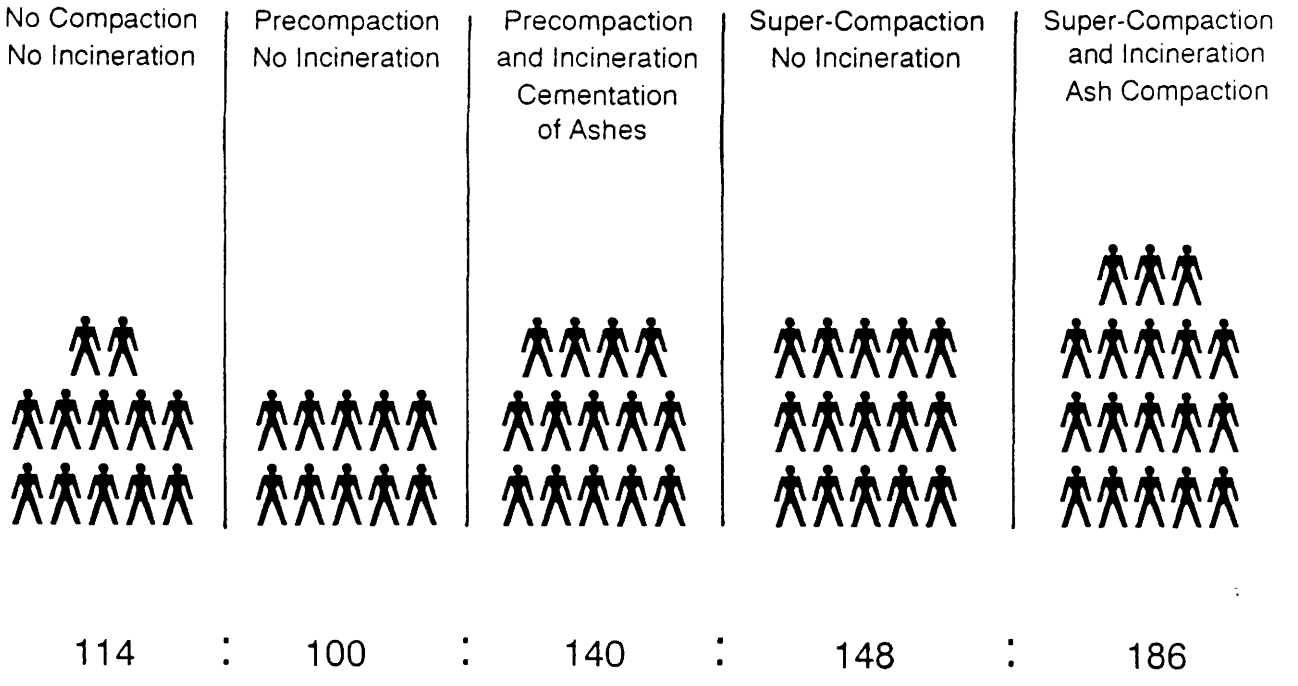
c) Pre-compaction, no incineration : 14.000 man-hours.

d) Supercompaction, no incineration : 14.800 man-hours.

e) Supercompaction and incineration : 18.600 man-hours.

FIGURE 5-2 shows the waste handling effort for the corresponding treatment modes.

If no compaction of wastes is applied (case "a"), the number of waste packages is higher by a factor of 3 and the related handling operations need more man-hours. Case "b" seems to be the case with the lowest operating costs. Incineration of waste needs a good sorting in combustible and uncombustible parts prior to incineration. In general 10 man-hours are needed for sorting of 1 m^3 of mixed solid waste.



Precompaction force : 20 tons
Super-compaction force : 1500 tons

FIGURE 5-2 : WASTE HANDLING EFFORT FOR THE DIFFERENT TREAMENT MODES OF MIXED SOLID WASTE

5.1.3.3. Treatment costs

The costs for waste treatment result from :

- personal costs (man-hours),
- equipment costs,
- service prices,
- package costs,
- interim storage costs,
- transport costs,
- final disposal costs.

The personal costs are given in Sec. 5.1.3.2. The following assessments can be made for a cost evaluation :

Personal costs : 75 ECU per man-hour.

Equipment costs : Precompactor ECU 50.000
Super-compactor ECU 1.0 Mill

Incineration : Service price 18 ECU / kg
1 m^3 = 500 kg

Package costs : see Sec. 3.5.4.3.

Transport costs : see Sec. 3.5.4.4.

Interim storage costs : ECU 600 / m^3 per year.

Final disposal costs : ECU 2500 ECU / m^3
(estimated for German situation)

The following cost comparison can be based on the German situation :

Supercompaction :

Assumptions :

. 1 kg mixed solid waste = 1,4 liter conditioned waste incl. package

. All values in ECU / kg

Transport	:	Mobile equipment, waste packages to interim storage and to final disposal	1.3
Compaction	:	incl. metal cartridges	3.0
Drying	:	drying of wet compacts in Germany requested 50 % wet compacts	2.0
Package	:		1.0
		ECU / kg :	8.8

Interim storage : ECU/kg 0.85 per year

Final disposal : ECU/kg 3.5

Incineration :

Assumption :

. 1 kg combustible waste = 0.2 liter conditioned waste incl. package.

. All values in ECU / kg.

Service price incl. transport	18.0
Package	0.2
Transport to interim storage and disposal	0.1
Documentation / management	1.0
ECU / kg :	19.3

Interim storage : ECU/kg 0.12 per year

Final disposal : ECU/kg 0.5

For a 1300 MWe, PWR about 400 m^3 per year of mixed solid wastes are generated. This corresponds to about 200.000 kg.

Treatment costs (1300 MWe PWR)

Compaction w/o incineration : ECU Mill. 1.8

Compaction and incineration : ECU Mill. 2.8

Conclusion :

Compaction without incineration seems to be more economical, if final disposal is immediately available. If a longer period of intermim storage prior to final disposal is necessary, which is the case in Germany, the economical advantage of compaction without incineration decreases. This judgement is made neglecting all other licensing and legal conditions which may influence all economical statements.

5.2. DRAINING, DRYING AND CEMENTATION

5.2.1. Different treatment of wet wastes

Regarding wet and liquid waste treatment, two different treatment processes can be applied, namely solidification of liquids in a matrix material, i.e. cements, bitumen or plastic, or drying of liquids to a solid body and filling the dry, solid product in a waste containe. The possible treatment modes for liquids are :

Evaporator concentrates :

a) Solidifiction of liquids by direct cementing.

b) Dehydration and subsequent cementation of the dewatered raw waste.

c) Drying, crystalization of liquids to a solid body in shielded waste container.

Spent (wet) resins :

a) Solidification of high active resins with organic binders (i.e. polystyrol).

b) Solidification of low active resins with cement.

c) Draining of resins and filling into shielded waste containers (high and low active resins).

5.2.2. Waste volumes

In FIGURE 5-3, the volume reduction ratios for the different treatment modes of liquid wastes are shown. One m^3 of liquid raw waste (Ø.6 Ci/m^3) generates 15 m^3 (incl. packages) of cemented waste product if the liquids are directly solidified with cement without any pretreatment. Pretreatment by dehydration and subsequent cementation leads to an increase of the disposal volumes by factor of 7 for boron-containing liquids and a factor of 3 for non-boron-containing liquids.

Using in-drum drying processes (i.e. FAVORIT with vacuum drying system) gives a volume reduction of about 25 % of disposal volumes. During the drying process, all liquids are removed and the remaining solids stay inside the waste container. About Ø.2 m^3 solids are usually in 1 m^3 of evaporator concentrate which corresponds to a volume reduction of the waste product of factor 5 and an increase of the spec. activity of the same amount. Due to shielding requirements a heavy shielded waste container is needed to get the same dose rates at the cask surface as for other packages required.

High active spent resins (PWR) have been solidified with poslystyrol divenylbenzol (COMET, FAMA systems). For 1 m^3 of resins abouts 15 m^3 of disposal volume are generated.

Solidification of high active resins with cement is not carried out.

Low active resins (BWR, PWR) can be solidified with cement. For 1 m^3 of powder resins about 8 m^3 of disposal volume are generated.

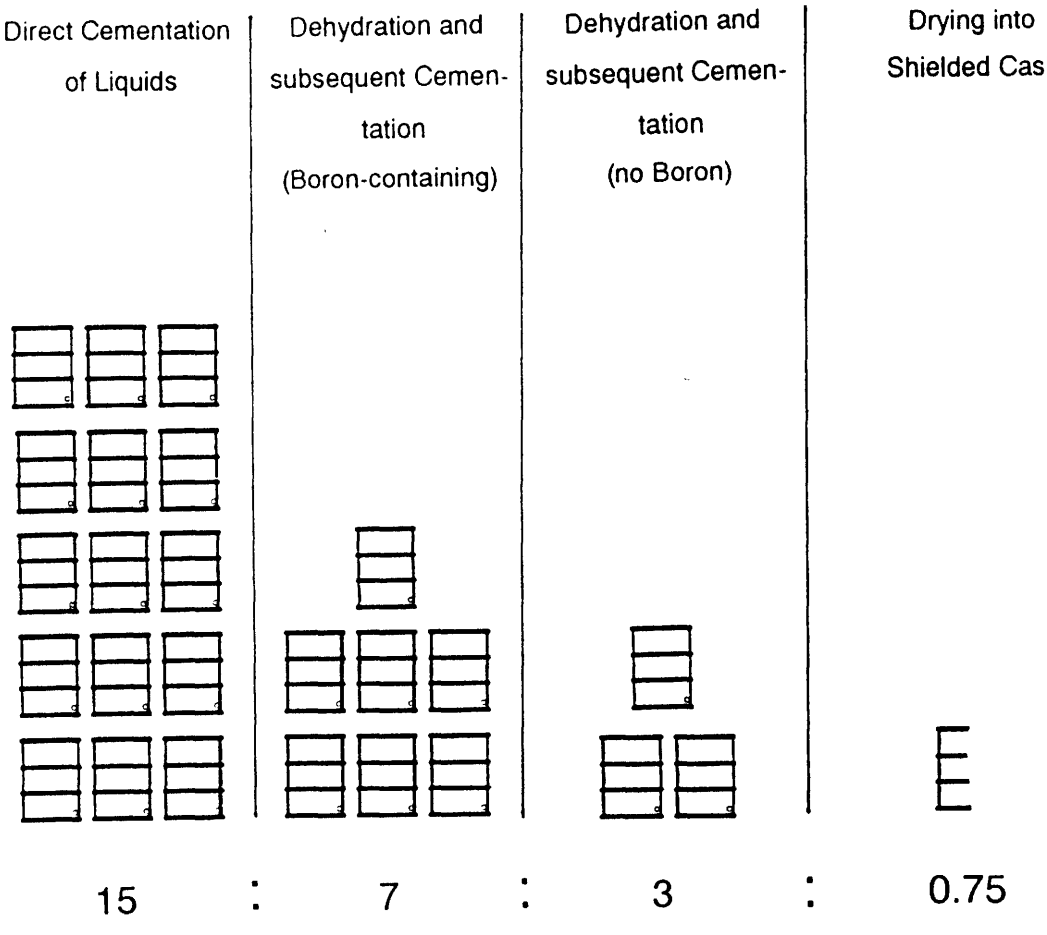


FIGURE 5-3

DISPOSAL VOLUMES FOR DIFFERENT TREATMENT MODES OF LIQUID CONCENTRATES
(Volumes incl. shielded packages corresponding to 1 m^3 of raw waste)

Instead of solidification with cement, draining or/and vacuum drying of high active and low active resins can be applied. The drying preferably takes place inside the waste container (FAVORIT system). Shielded, high integrity containers are used for the resins. The disposal volume corresponding to 1 m^3 of resins is about 3 m^3 for low active resins and 4 m^3 for high active resins.

FIGURE 5-4 gives the disposal volumes for the different treatment modes of spent resins.

5.2.3. Operational costs

5.2.3.1. General

Influences of safety and licensing aspects are not considered in this study, but they may change all economical considerations. For a cost evaluation of the different treatment modes the following parameters are of importance :

a) Acceptable level of dose rate at outer surface of the waste package.

b) Availability of volume for interim storage or final disposal.

For the German situation all waste packages are one-way-packages and have the same (low) limits for the allowable surface dose rate (max. 200 mrem/h).

Limited volume for interim storage (on site or in central storage facilities) is available in Germany, therefore treatment modes giving a low storage volume are necessary.

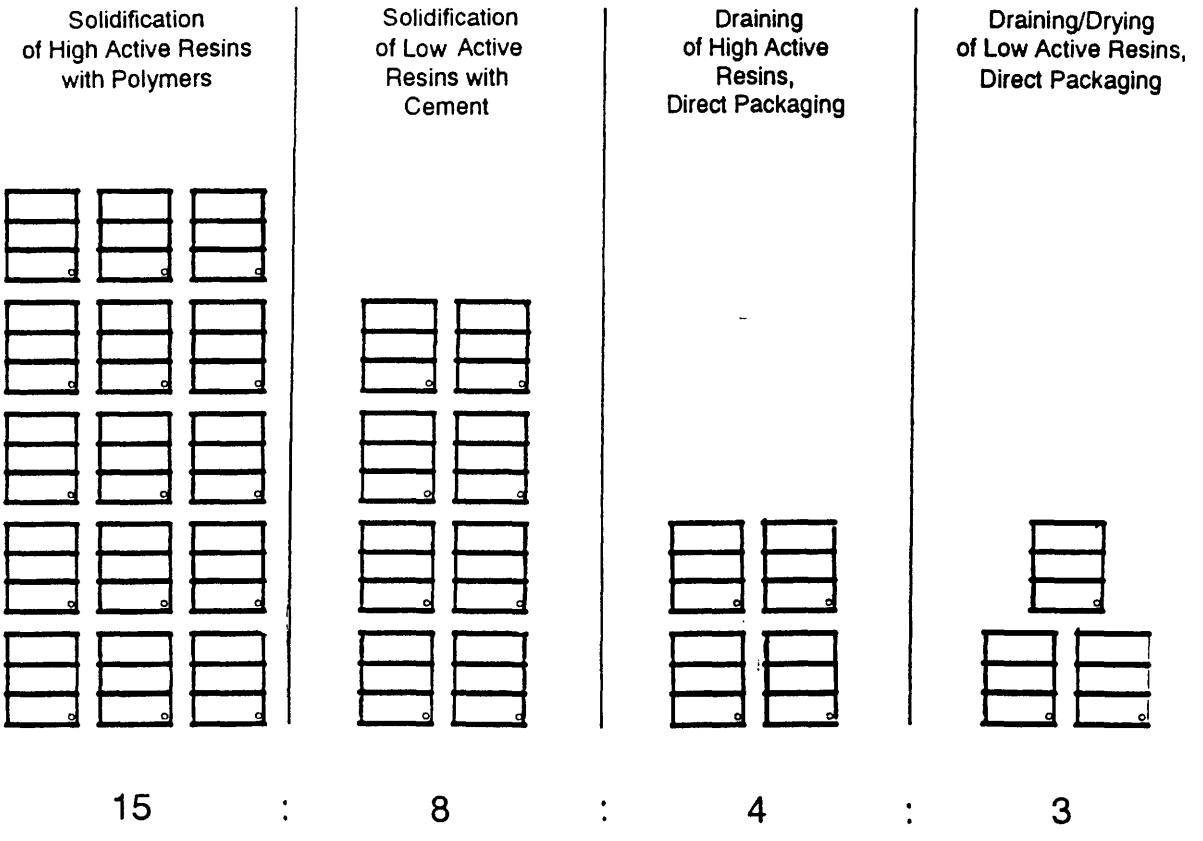


FIGURE 5-4

DISPOSAL VOLUMES FOR DIFFERENT TREATMENT MODES OF SPENT RESINS
(Volumes incl. shielded packages and correspond to 1 m^3 untreated resin)

5.3.2.2. Treatment costs

The costs for treatment result from the same boundary conditions as mentioned in Sec. 5.1.3.3.

The operational costs for the different mobile or stationary treatment systems differ not very much and give no advantage or disadvantage to any specific treatment mode. About 1000 man-hours are needed for treatment of liquids and resins anually generated in 1300 PWR.

Cost differences can only derive from :

- prices for waste packages (see Sec. 3.5.4.3.),

- transport costs (see Sec. 3.5.4.4),

- interim storage and disposal costs (see Sec. 3.5.4.5.).

The following specific costs for the different treatment modes can be given (based on 1 m^3 untreated liquid) :

Liquid concentrates

a) Direct cementation

Package price	ECU	8300
Transport	ECU	4000
Final disposal	ECU	37500
Interim storage (per year)	ECU	9000

b) Dehydration/cementation (boron-containing)

Package price	ECU	5200
Transport	ECU	2000
Final disposal	ECU	17500
Interim storage (per year)	ECU	4200

c) Dehydration/cementation (no boron)

Package price	ECU	2300
Transport	ECU	750

Final disposal	ECU	7500
Interim storage (per year)	ECU	1800

d) Drying into shielding casks

Package price	ECU	9750
Transport	ECU	750
Final disposal	ECU	1900
Interim storage (per year)	ECU	450

Treatment of Resins

a) Solidification of High Active Resins

Package price	ECU	8300
Transport	ECU	4000
Final disposal	ECU	37500
Interim storage (per year)	ECU	9000

b) Solidification of Low Active Resins

Package price	ECU	6000
Transport	ECU	2000
Final disposal	ECU	20000
Interim storage (per year)	ECU	4800

c) Draining and Direct Packaging (High Active Resins)

Package price	ECU	39000
Transport	ECU	3000
Final disposal	ECU	10000
Interim storage (per year)	ECU	2400

d) Draining and Direct Packaging (Low Active Resins)

Package prices	ECU	30000
Transport	ECU	2300
Final disposal	ECU	7500
Interim storage (per year)	ECU	1800

Conclusion

Drying of liquids or draining of resins with direct packaging into shielding containers without further additives gives the lowest possible disposal volumes. If a longer period of interim storage prior to final disposal is necessary, which is the case in Germany, the economical advantage of drying/draining of liquids or resins is obvious.

6. ASSESSMENT OF THE RELATED OCCUPATIONAL EXPOSURE

6.1. GENERAL

Experience from operating LWR's shows that activated corrosion products constitute the main source of occupational radiation exposure, being responsible for some 80 per cent of the total collective dose. The principal nuclides of concern are Co-60 and Co-58.

In some case leaking of the fuel pins has to be considered and the pressure of fission products (mainly Cs-134 and Cs-137) in the waste streams are observed as well. The occupational radiation exposure to the operating personnel varies with reactor size, age and extend of remote techniques used for waste treatment.

About 75 per cent of exposure is received during reactor shutdown when inspection and maintenance creates the largest amount of solid waste. In many cases, specialist contract workers for waste maintenance and handling are brought in during reactor shutdown periods. On average, two thirds of the total annual collective dose are received by this personnel.

6.2. PERSONNEL DOSE EXPOSURE AT THE NUCLEAR POWER PLANT

In the Federal Republic of Germany, the average annual collective dose per reactor is about 4 man-Sv/year (400 man-rem/year). The pictures are nearly the same for PWR and BWR. There is no distinction made in the power plant between personnel working with waste treatment or general maintenance work. But from the collective dose, an individual dose of 2.5 mSv/person and year (250 mrem/person and year) can be derived for the personnel working permanently for waste treatment.

Considering 14300 man-hours for waste treatment at a 1300 MWe-power plant the total annual collective dose is 24 mSv (2.4 rem).

6.3. PERSONNEL EXPOSURE AT MOBILE SYSTEMS

GNS has recorded the occupational dose exposure for own personnel over a period of more than 10 years. The records show that 70 % of the personal receives and individual dose of less then 1.0 mSv/year, 85 % less than 2.5 mSv/year and 95 % less than 5 mSv/year.

The total annual collective dose for the waste treatment with mobile units is about 10 mSv (1.0 rem). This collective dose is calculatd under the assumption of an individual annual dose exposre of 2.5 mSv (250 mrem) and 5.800 man-hours for the operation of mobile units in a 1300 MWe PWR.

6.4. TOTAL OCCUPATIONAL EXPOSURE

The total annual occupational exposure during waste treatment is the sum of exposure at the power plant with the stationary equipment and the exposure during opeation of mobile waste treatment units. Considering an individual dose exposure of 2.5 mSv/year (250 mrem/year), the total annual occupational exposure for waste treatment in a 1300 MWe PWR is about 34 mSv (3.4 rem). This occupational exposure is derived from routine operation. In the case of a special waste treatment campaign, maintenance, repair of other problems the occupational exposure can be higher.

European Communities – Commission

EUR 14043 – Assessment of management alternatives for LWR wastes (Volume 3)
Description of German scenarios for PWR and BWR wastes

S. Santraille, K. Janberg, H. Geiser

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This report deals with the description of a management route for PWR waste relying to a certain extent on German practices in this particular area. This description, which aims at providing input data for subsequent cost evaluation, includes all management steps which are usually implemented for solid, liquid and gaseous wastes from their production up to the interim storage of the final waste products.

This study is part of an overall theoretical exercise aimed at evaluating a selection of management routes for PWR and BWR wastes based on economical and radiological criteria.

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