Forschungszentrum Jülich

THE HOT CELL RADIOACTIVE WASTE CONCEPT OF FORSCHUNGSZENTRUM JUELICH

by

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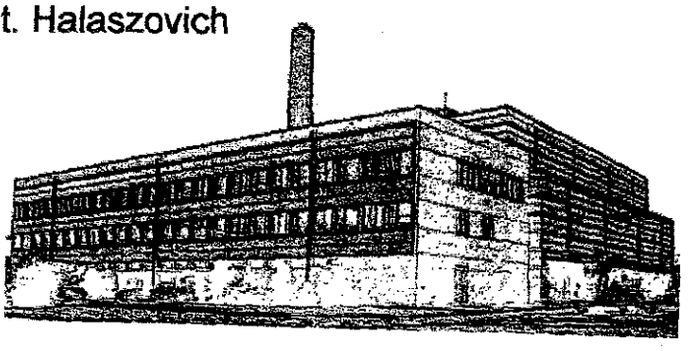

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The Hot Cell Radioactive Waste Concept of Forschungszentrum Juelich

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ABSTRACT

During the last 30 years extensive scientific examinations on radioactive metalls, ceramics and fuel elements have been carried out, so that a high volume of waste has resulted. Also from the dismantling of irradiated facilities metallice waste has o be handed. Prior for equipment repair the hot cell envolved has to be decontaminated and a large amount of lower active waste is produced. The waste is collected for conditioning and storing. There are different categories as: low active liquid waste, low active burnable waste, fuel waste, low and high active metallic waste. For each waste category special transport container are used. For the volume reduction our WASTE DEPARTMENT is equipped with special facilities e.g.: furnace for burning, dryer, liquids evaporator, hydraulic press for pelletising, decontamination box for the dismantling ad cleaning of components. After conditioning the waste will be stored on site. Special documentation has to be done for the acceptance of this waste.

WASTE FROM "HOT CELL FACILITY"

The "Großen Heißen Zellenlabors" (GHZ) consists of 2 large concrete cell blocks together with 2 smaller lead-shielded cells, offering 16 technicians access for investigating reactor fuel elements and activated structure components. The cells are gas-tight allowing destructive work like cutting, grinding and annealing to be performed on nuclear fuel elements. Each cell is equipped with master/slave manipulators, a running crane (30 kN) and with power manipulators. The cell shielding thickness is 0.85 resp. 1.10m heavy concrete, gamma sources with a strength of max. 3×10^{15} Bq at 1 MeV gamma energy can be handled safely. (fig. 1) The objects to be investigated are guided through an interlocking sluice which allows access for lorry-sized vehicles and a running crane with a capacity of 300 kN handles the transport containers.

Radioactive waste (1) will be produced during the examination of neutron irradiated fuel elements and structure materials due to

- Dismantling of components, irradiation facilities and sample manufacturing by sawing, cutting and grinding.
- Destructive mechanical tests on irradiated material (e.g. pressure vessel steel, austenitic steels, zircaloy/aluminium and ceramics).
- Preparation of fuel and structure material samples for metallography by drilling, grinding, cutting and polishing.
- Testing the crushing strength of HTR fuel elements in compressing tests.
- Core sampling from waste drums.

Prior to equipment repair the hot cell involved has to be decontaminated and a large amount of low active waste is produced. The waste from the various laboratories and specially the waste of the hot cell facility are collected and transported to our waste department for conditioning and storing. There are 5 different categories

- medium active liquid waste
- low active solid burnable waste
- fuel waste
- high active metallic waste
- contaminated components, tools etc..

During the last 25 years about 150 m^3 solid waste and 1 m^3 liquid waste of medium activity was produced every year with a total activity of 1-8x10^{14} Bq/year. Details of the content and containers are given in Tab. 1.

Tab. 1: Different Waste Categories of Hot Cell – Facilities

Burnable waste:	Paper, foils, overshoes etc. Collected in 200 ltr. board or tin barrels
Unburnable waste:	Low activity, collected in 180 ltr. tin barrels for compacting
High active waste:	Fuel element segments, metallic samples or components from irradiated facilities packed in 5 ltr. tin cans
Contaminated components:	Boxes, metallic components, tools. Packed in steel containers (e.g. 10 m^3), 5 cm concrete shielding.
Liquid waste (medium active):	Chemical treatment, 12 – 80 ltr. volume, shielded up to 20 cm Pb (Cendrillon), 200 ltr. barrels shielded with 5 cm Pb.

Further waste treatment is being done in the "waste department" of Forschungszentrum Juelich. But for each special transport to the waste department, the supplier has to prepare documents, indicating the different isotopes including the individual activity. This can be done by calculation, weighing, gamma spectrometry and α / β measurements. More details are given in Tab. 2,3.

For the declaration of the fuel some main subjects should be considered:

- The dimensions and weight of the fuel (elements, section);
- The composition of fission products and isotopes by calculation (burn up. decay time, enrichment, composition before irradiation);
- If necessary measurements by beta-, gamma spectrometry are done;
- There should be no loss of fuel;
- For declaration the knowledge of guiding nuclides are necessary.

Tab. 2: Typical Isotope content of fuel (10% U5 enrichment, 3-4 years decay time)

ISOTOPE		%
Sr 90	β	8
Y 90	β	8
Ru 106	β	5
Rh 106	γ	5

Cs 134	γ	2
Cs 137	β	9
Ba 137m	γ	8
Ce 144	γ	16
Pr 144	γ	16
Pm 147	β/γ	11
		88
all others		12

Also for the steel samples or for contaminated components data are necessary:

- The activity of activated materials are normally calculated if the composition of the material is known (data: material composition, neutron fluence, decay time, weight);
- If necessary the gamma spectrum could be measured;
- After a longer testing period the guiding nuclides are well known for different materials.

Tab. 3: Typical Guiding Nuclides (200 d irradiation neutron flux $1x10^{14}$ n/cm^2 and 200 d decay time)

	Half-live time	Gamma-activity MeV	PVS * %	Aust. Steel %
Cr 51	27,8 d	0,32	15	27
Mn 54	314 d	0,84	10	1
Fe 55	2,9 a	K	40	20
Co 58	71 d	0,81	6	5
Fe 59	45,1 d	1,99/1,29	25	1
Co 60	5,27 a	1,33/1,17	-	11
Ta 182	115 d	1,12/1,22	-	32
Total			96	97

* PVS: Pressure vessel steel

WASTE CONDITIONING

Treatment of burnable radioactive waste (2), (3)

As mentioned before, solid burnable waste is collected in 200 ltr. drums and fed into a preheated gasification chamber via a glove box and lock chamber without further pre treatment. (fig. 2) The waste sink down to a pair of flaps which serve as a grate separating the gasification chamber from the primary burner. While sinking, the waste is heated up to about 800°C. At the grate level it is partly burned partly gasified by adding understoichiometric air. An additional amount of air is blown through the flaps into the primary burner where the temperature rises to 1100°C. The gases pass sideways to the secondary burner. The ashes fall down to be collected on a second pair of flaps where combustion is completed. Finally the ashes are discharged into 180 ltr. drums. These drums fit into the FAKIR super compactor.

The main advantage of this process developed at Juelich Research Centre is that gasification and burning are performed under very stable constant conditions. Neither the temperature nor the off gas

composition does oscilate. The carbon monoxide content is very low. Comparatively small amounts of fly ash are generated.

The incinerator nominal throughput is 50 kilograms per hour based on a 20.000 KJ per kilogram energy content of the waste. The yearly throughput is limited by the license to 140 tons, 120 tons solid waste and 20 tons liquids. A smaller incinerator of this system was developed and constructed for the research centre INCHASS.

Treatment of radioactive waste water (Fig. 3,4)

Radioactive waste waters and sludges are evaporated. The waste water treatment line consists of storage tanks, evaporator and two stage drier. Waste water from the different laboratories of the Juelich research centre is collected and are be stored in 17 tanks with a total storage capacity of 825 m^3. The large number of storage tanks allows the separate treatment of different wastes.

The waste water is continuously pumped from two supply vessels into the evaporator. Acid is added into the transfer pipe in order to prevent foaming in the evaporator. Five tons of water are evaporated per hour until the solid content in the batch amounts to maximum 40 %. The distillate is pumped back into a clean vessel in the waste water store. It can be recycled into the evaporator for further decontamination if necessary.

All kinds of radioactive concentrates, sludges and precipitates are dried and heat treated in the two stage drier. In the supply vessel of the drier the feed will be homogenised and a sample will be taken in order to achieve radiological data for final documentation. The first stage is a fluidised bed drier operating with hot air. The dry granules produced in this stage will be heated to 350 °C in the second stage which is a pipe type fluidised bed drier. In the second stage residual organics and nitrates in the dry product will react.

Super compacting

During the operation and decommissioning of a nuclear facility, contaminated solid waste is generated, such as protective clothing, rags, concrete parts, insulation, cables and filters. Volume reduction of such waste results in considerable savings with regard to the costs of interim storage and final disposal. The solution is the FAKIR Hydraulic Super Compactor. It allows a waste volume reduction by a factor of 3-5.

Technical data of the super compactor

Dimensions,	length :	~ 10,0 m
	width :	~ 2,4 m
	height :	~ 2,8 m
Weight :		50 – 60 t
Compacting force :		1.200 – 1.500 t
Throughput :		up to 20 drums/h

Dismantling and decontamination (REBEKA)

REBEKA is the site for the treatment of radioactive solid materials which had been constructed throughout the years 1992 until 1995. It is a new concrete building and has been fitted inside the existing process building according to the latest requirements on earthquake resistance and on water and air pollution control. REBEKA contains the equipment for dismantling, cutting and decontamination of radioactive solids as well as the equipment for solid waste conditioning (see fig. 5). Large components up to 25 tons can be brought into two steel boxes (9 meters long, 7 meters wide and

5 meters high) and isolated treated. Decontamination is performed by pressurised water, by etching or using mechanical tools. Mainly thermal methods are used for cutting. A super compactor (FAKIR see 2.3) and a drier (PETRA) for drum drying of wet materials are used in the same location. During 1998 a new sandblasting cabin for surface decontamination has been taken into operation. As experience shows sandblasting is a quick and very effective method for the decontamination of metal and concrete surfaces. Apart from these main working areas there are transfer areas for containers and a shielded area for radiological control measurements on decontaminated objects.

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Fig. 1 : Hot Cell Laboratory - GHZ

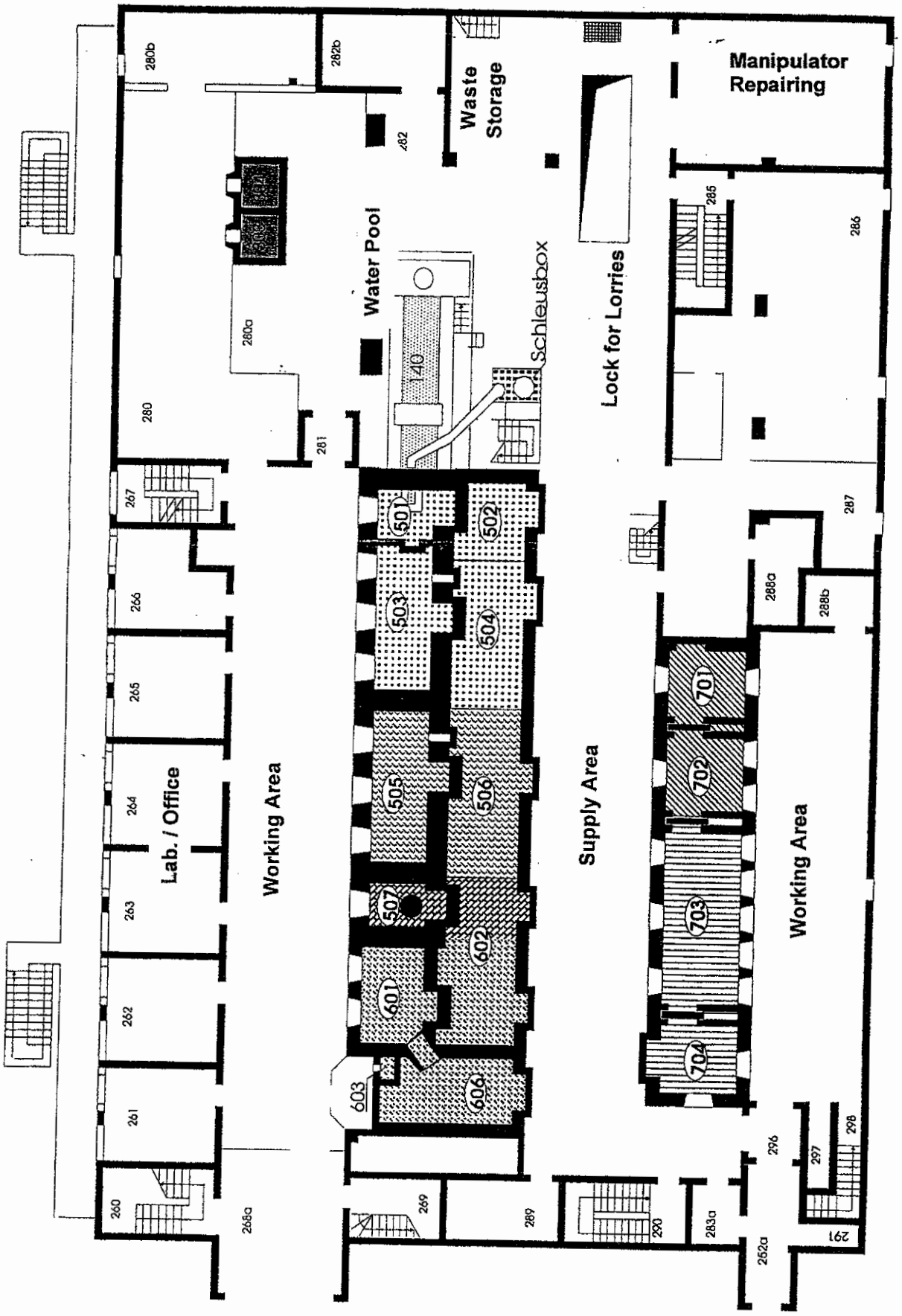


FIG. 2: PRINCIPLE FLOW SHEET OF INCINERATOR

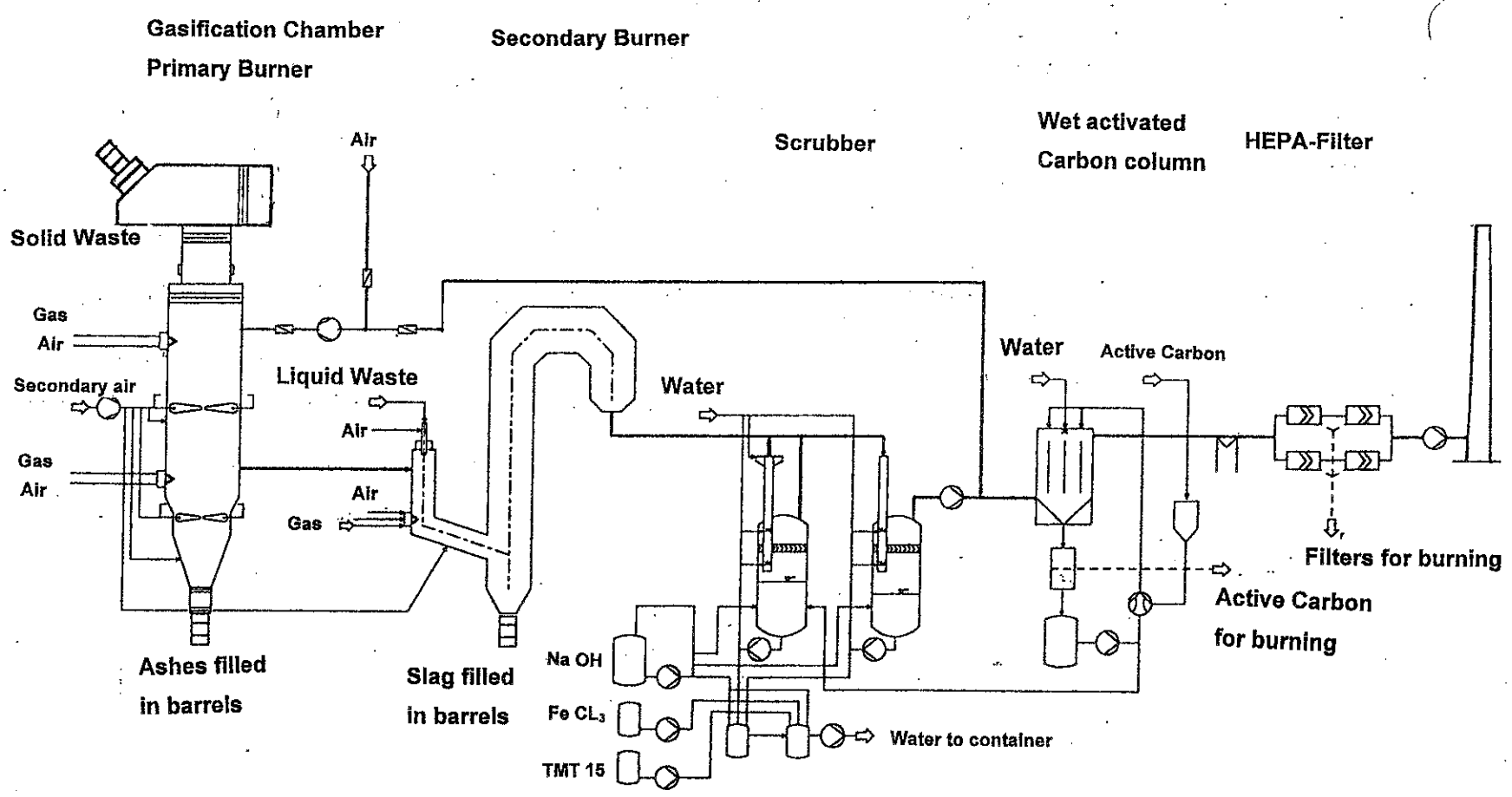


Fig. 3 : PRINCIPLE FLOW SHEET OF EVAPORATOR

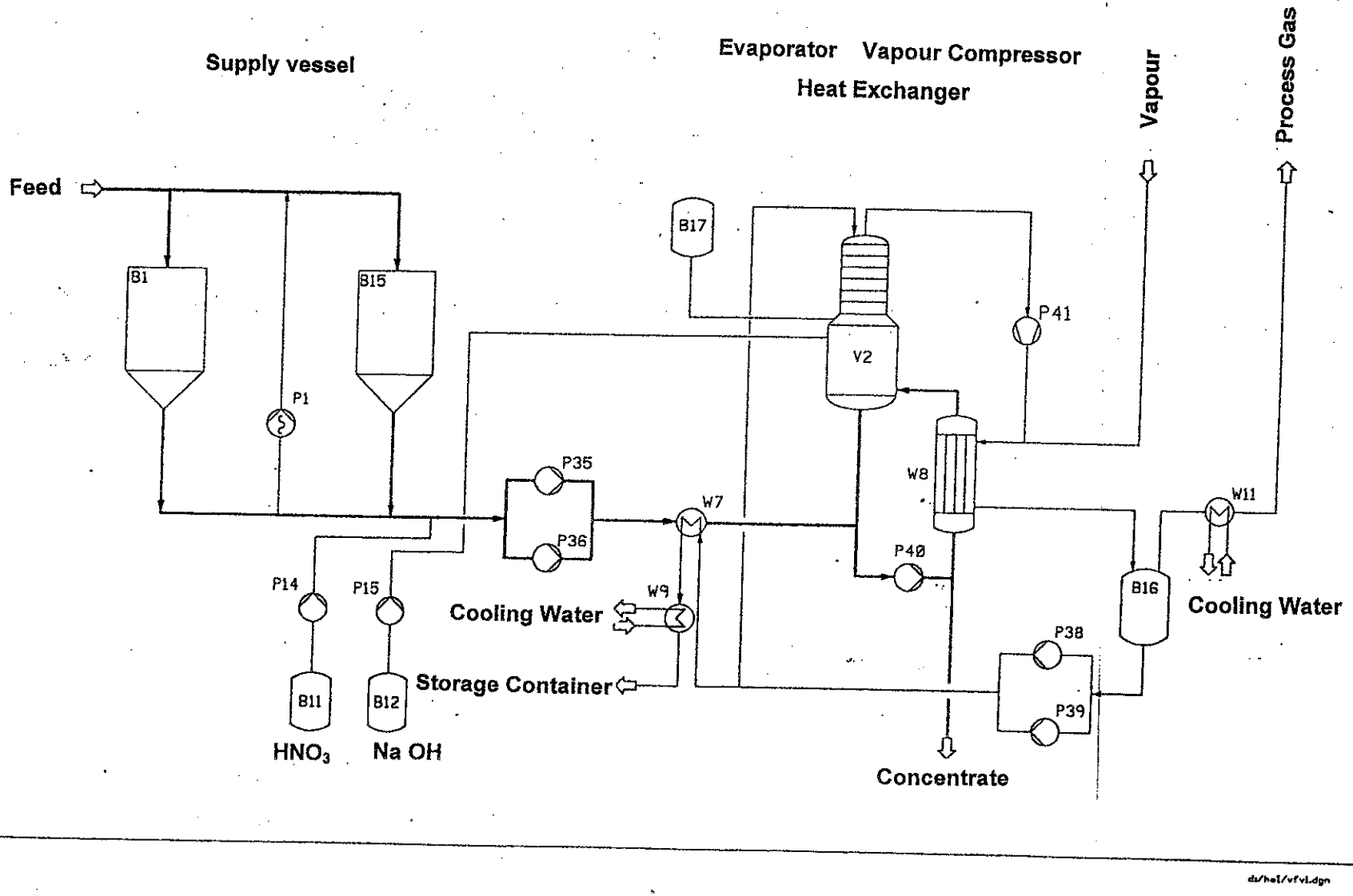


Fig. 4 : **PRINCIPLE FLOW SHEET OF DRYING FACILITY**

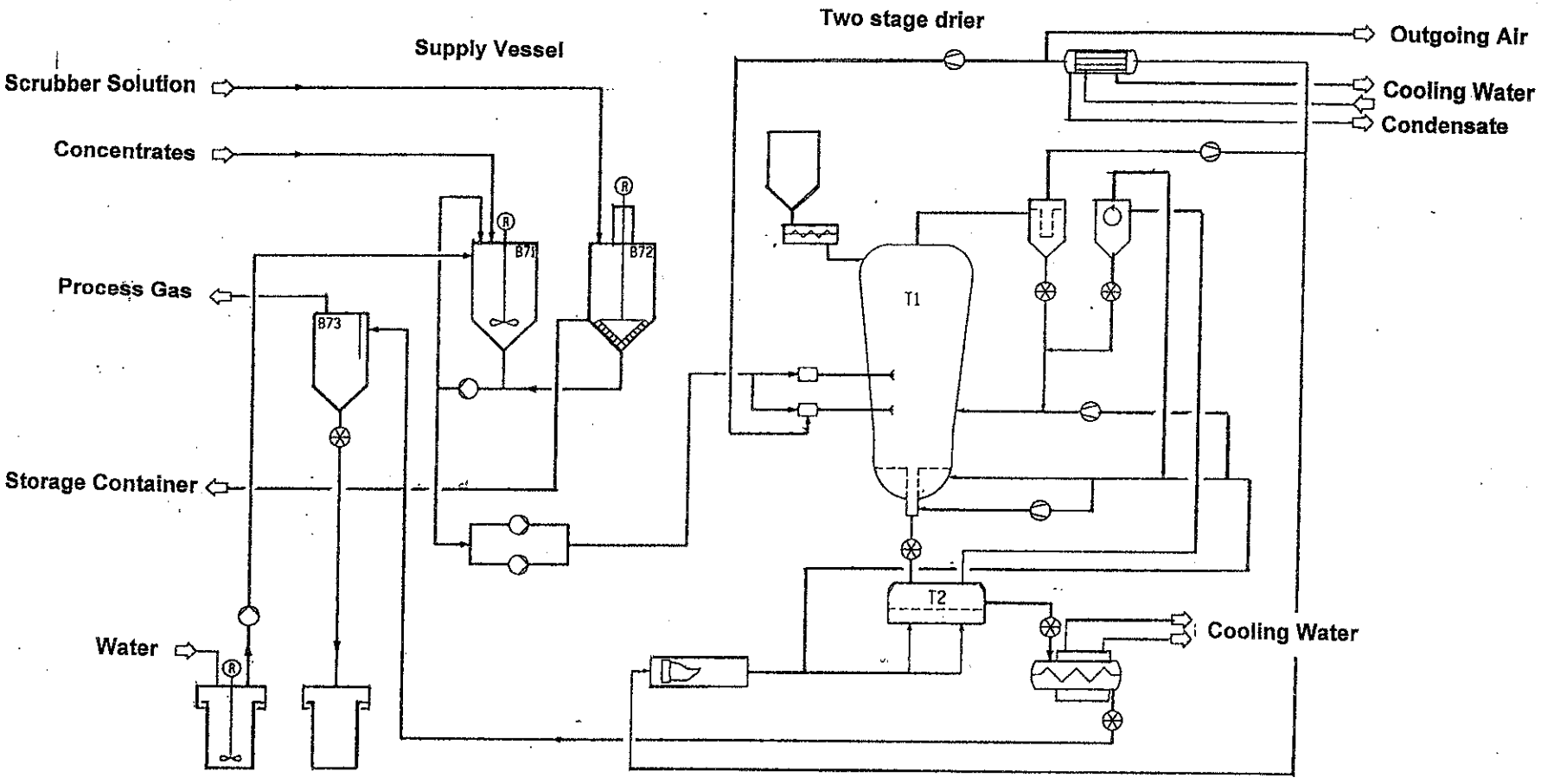
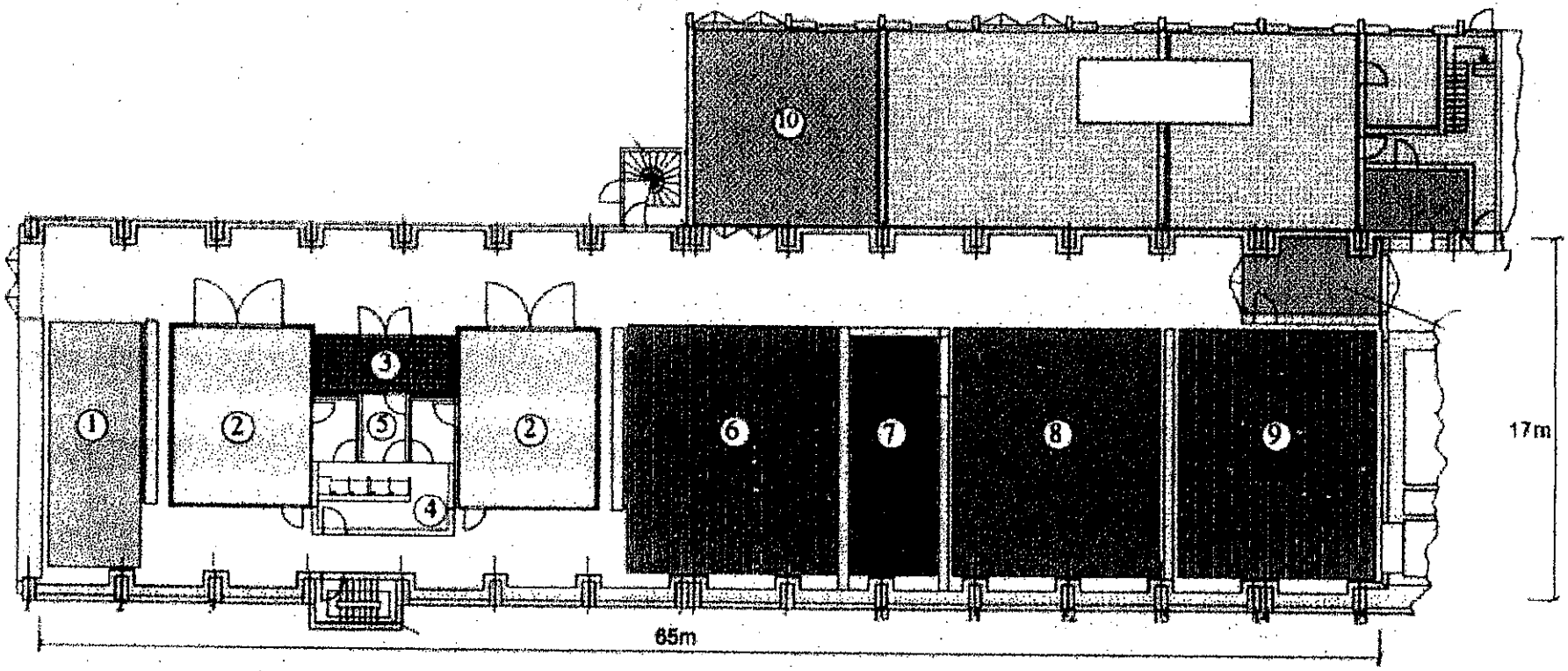


Fig. 5 : REBEKA - Facility for Recycling and Waste Treatment



1 Measurement of activity
2 Cutting, decontamination
3 Sand blasting
4,5 Lock, change rooms
6,7 Container area
8 Compacting, drying
9 Cementation
10 Material lock

Background

Epoxy mounted specimens

Ceramographic and other specimens contain epoxy resins in intimate contact with spent fuel

Epoxy can decompose due to radiolysis from α, β, γ activity in fuel

Over 50 kg of spent fuel in over 25 kg epoxy at Studsvik

Disposal of other spent fuel residues from Hot Lab

All other spent oxide fuel residues welded into stainless steel capsules, backfilled with He, in cell

Capsules taken to CLAB interim wet storage, later final disposal

Can spent fuel with epoxy be stored in CLAB for 40 years?
Can it be included in the Swedish KBS-3 final repository?

Design pressure of capsules

Capsule design

Based on gas evolution rates and material volumes:

-Anticipated pressure < 2,5 MPa after 50 years interim storage

Calculated design pressure: ≥ 3,0 MPa

Capsule pressure test

Two standard capsules, welded under identical conditions to in-cell

Hydraulic burst tested at room temperature

-Rupture pressure ≥ 40 MPa

➡ Capsule design has adequate design margin

Capsule design approved for interim and final storage

Gas evolution from epoxy

Review and initial prediction

Literature review of radiolysis of epoxies

Prediction assuming only β–radiation from whole pellets:

-10 litre/year total gas for 27 kg epoxy

-90 % H_2, 10 % CO, small amount of hydrocarbons, amines, ammonia

Experimental tests

Spent fuel samples in epoxy, backfilled with He

Short term test (~100 days, sampling, mass spectrometry)

Long term test (846 days, pressure monitoring)

-Only H_2, CO, CO_2 found; 0,35-0,60 litre/kg epoxy/year

~10-20 litre/year total gas for 27 kg epoxy

EPOXY MOUNTED SPECIMENS AS A WASTE FORM

Max Lundström
Leif Kjellberg
Bengt Sundström
Stefan Hammar

Studsvik Nuclear, Hot Cell Laboratory

How epoxy mounted metallographic specimens are incorporated into the Swedish national spent fuel disposal system

Studsvik™ *Nuclear*

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