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THE MANAGEMENT OF INTERMEDIATE LEVEL WASTES IN SWEDEN

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Prepared for the Swedish Nuclear Power Inspectorate

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1 INTRODUCTION

Current practices and research in Sweden on the management of intermediate level wastes are given a brief overview in the present paper. Intermediate level wastes include spent resins, filters and core components from the six power reactors in operation; radioactive wastes from nuclear fuel development at Studsvik and from non-nuclear applications are a minor contribution.

At the nuclear power stations current practices include incorporation of spent resins into concrete or bitumen with on-site storage of conditioned waste. A computerized register of accumulated waste has been initiated by the Nuclear Power Inspectorate (SKI). Waste inventories are reported in detail annually with complementary reporting quarterly, to SKI and to the National Institute of Radiation Protection (SSI).

Development work on improvements of immobilization technology and on storage and disposal are sponsored by the National Council for Radioactive Waste (Prav) and by the Board for Energy Production Research (NE). Utility sponsored work on intermediate level waste is coordinated in the Nuclear Fuel Safety project (KBS) with a focus on disposal aspects. Safety oriented research is sponsored by SKI and SSI. Studsvik Energiteknik AB (STUDSVIK) is presently engaged in a programme to improve conditioning, storage and disposal of wastes received or originating from operations at Studsvik. Nordic cooperation in the field is coordinated by the Nordic liaison committee for atomic energy (NKA).

Contracts signed for the reprocessing of spent nuclear fuel from Swedish power reactors contain options for return of wastes of a number of categories. This has initiated efforts in Sweden on immobilization and disposal technology for alpha bearing wastes from reprocessing. These efforts are, however, not dealt with in this paper.

2 CHARACTER OF WASTE

Besides the radioactive waste produced at the light water reactor plants in Sweden, minor amounts arise also from research and development activities. A brief characterization of both types of waste is given in this chapter.

2.1 Reactor wastes

The radioactive wastes produced at the nuclear power plants may be divided into two categories:

- Wet wastes: Spent ion exchange resins, filter sludges, evaporator concentrates, etc.
- Solid wastes: Spent core components, miscellaneous trash, etc.

Wet wastes

Typical characteristics of wet wastes produced in Swedish nuclear power plants are given in table 1. Spent ion exchange resins from the reactor water clean-up system and powder resins from pool water clean-up systems dominate, containing more than 95 % of the total activity of wet wastes before conditioning. Spent resins are stored in tanks as slurries before conditioning.

The waste facilities at all Swedish nuclear power stations are equipped with evaporators for treatment of waste water from drainage. At the PWRs evaporation is also used to regulate the boron concentration of the reactor water. So far, however, it has been sufficient to treat waste waters by filtration and no evaporator concentrates have been produced.

Table 1. BWR and PWR wet waste categories

Category	Source	Production [1] m^3/y per reactor (by experience)	Specific activity [2] Bq/m^3 (Ci/m^3)	Typical radio-nuclides [2]
BWR-wastes				
Granular resins, mixed-bed	- Reactor water clean-up system	10-20	$(2-11)\cdot 10^{12}$ (50-300)	Cs-137 40 % Cs-134 15 % Sr-90 2 % Co-60 20 %
	- Liquid waste system	5-10	$< 4\cdot 10^{10}$ (< 1)	
Powder resins	- Pool water clean-up system	2-5	$(4-11)\cdot 10^{12}$ (100-300)	Cs-137 40-60%
	- Condensate clean-up system	30-50	$< 4\cdot 10^{11}$ (< 10)	
PWR-wastes				
Granular resins, mixed-bed	- Primary system	5-10 [3]	$(4-8)\cdot 10^{13}$ (1000-2000)	Mostly Cs-137
	- Steam generator clean-up system	10-15 (calc.)	$< 4\cdot 10^{10}$ (< 1)	

Remarks:

1) Volumes refer to dewatered resins, where granular resins contain 70 % water and powder resins 85 % water.

2) Assuming 0.1 % fuel leakage (Leakage is normally much less.)

3) Contains about 60 g H_3BO_3 per kg resin.

Solid wastes

Solid wastes are normally of quite low level activity, with the exception of core components of high induced activity and filter cartridges used for water cleaning in the PWR primary circuit.

The arisings of spent core components are about 10 m^3/y for a BWR and less than 1 m^3/y for a PWR. Total activity of these wastes will be 400-4000 TBq (10^4-10^5 Ci) per reactor annually, dominated in the short term by Co-60, Zr/Nb-95 and Fe-55, in the long term by Ni-63, Ni-59 and Nb-94.

Cartridges for mechanical filtration of particulates are used only in the PWRs. About 500 filter cartridges are spent annually per reactor, with a storage volume in concrete of about 20 m^3. The dominating nuclide is Co-60, giving a surface dose rate of up to 1.5 Sv/h (150 rem/h) from the filters.

About 300 m^3 of solid low level waste arise from each reactor annually, containing in total less than 40 GBq (1 Ci) mainly from corrosion products. Most of these wastes are burnable and are sent to Studsvik for incineration.

2.2 Wastes from research and development and from non-nuclear applications

Small amounts of intermediate level wastes arise in research and development at the Studsvik Research Center and the universities, and from non-nuclear applications at industries and hospitals. These wastes (spent radiation sources, radioactive equipment etc) are received on a service basis at Studsvik for treatment and storage. In 1979 about 3 m^3 of this waste category was received at Studsvik, together with a much larger volume of various low level wastes.

3 CONDITIONING

Two different techniques for the immobilization of resins and sludges are currently used in Sweden, cementation and bituminization.

Spent core components are now stored in water pools on-site at the reactors. It is foreseen that they will be cut into pieces and immobilized in concrete. PWR filter cartridges are put into drums or concrete blocks and immobilized in concrete.

R & D work includes characterization of cemented and bituminized wastes, volume reduction of spent resins by incineration and use of inorganic ion exchangers for the immobilization of long-lived nuclides, primarily Cs-137 and Sr-90.

A brief account is given in the following of currently used solidification systems, and of the development work on resin incineration and the use of inorganic ion exchangers.

3.1 Cementation

The process used at Ringhals is illustrated in fig. 1. The spent resins are metered into concrete moulds (1.2 x 1.2 x 1.2 m) and mixed with cement. The mixer is left in the filled-up mould. After 1-2 days of setting a lid of concrete is poured on top of the concrete block.

The mixing ratios used depend on the activity level of the resin, in order to meet the criteria of the safety authorities that surface dose rates are not to exceed 10 mSv/h. Up to now moulds with a wall thickness of 0.1 m have been used, but thicker walls may become actual in the future.

In general experiencies from the process at Ringhals have been good. Great care has to be taken, however, to avoid deterioration from the swelling of the resins. An essential factor is to keep a low water-to-cement ratio. The Swedish Cement and Concrete Institute (CBI) has carried out a thorough study of an optimized process and recommends that this ratio should be kept below 0.4.

Cementation of borated resins is another problem that calls for special attention. Borated resins at Ringhals from PWR operation have not yet been solidified on a regular basis. Development work is, however, in progress.

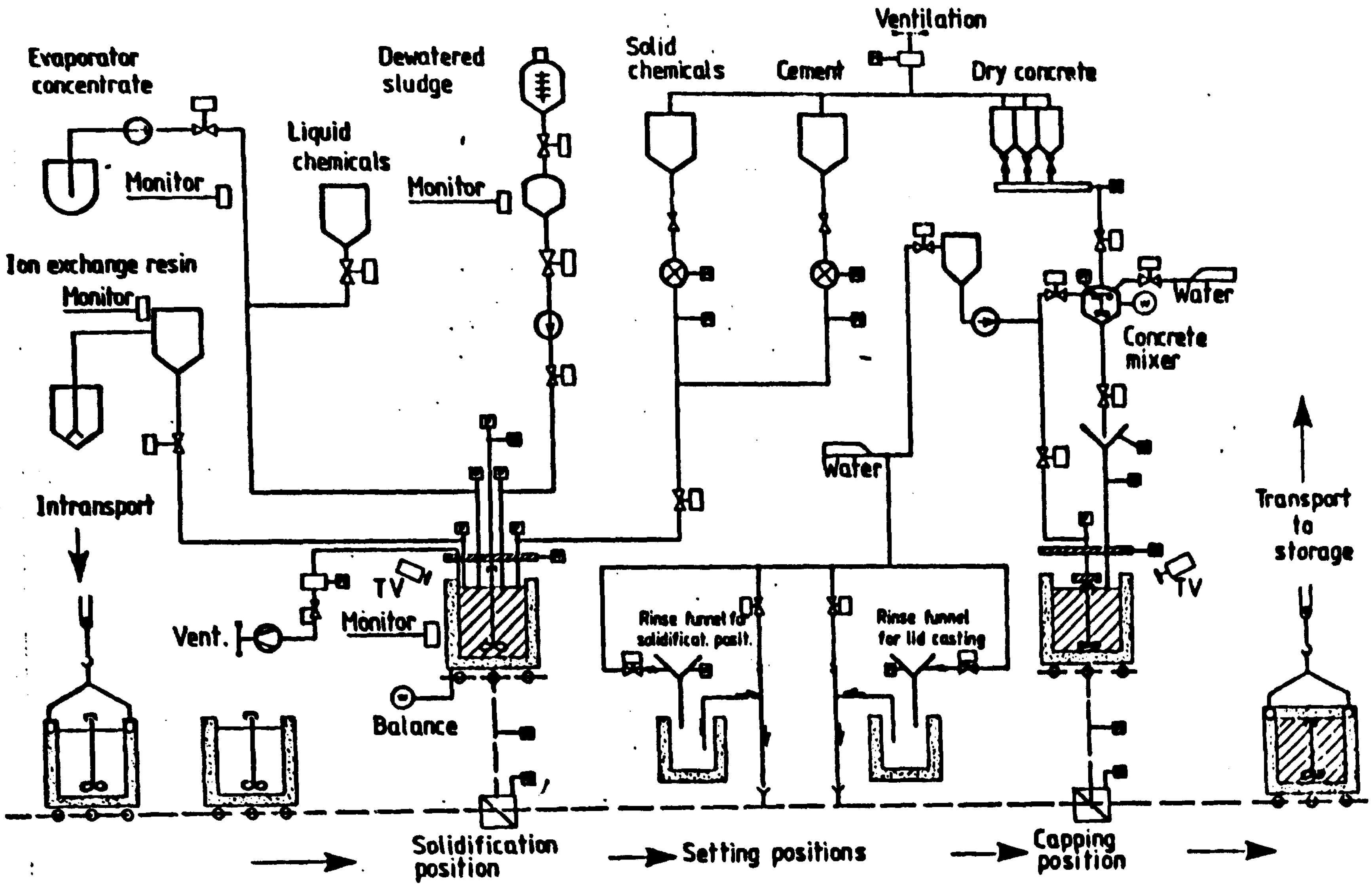

Fig. 1. Cementation process at the Ringhals Nuclear Power Plant.

3.2 Bituminization

Bituminization of spent resins in a thin film evaporator is used at the Barsebäck nuclear pcwer station. Better economy during storage and transport was important for the choice of bitumen for waste solidification rather than cement. The thin film evaporator was found suitable for the treatment of concentrates, resins and filter sludges or mixtures of these.

Figure 2 illustrates the bituminization process at Barsebäck. Wastes to be solidified, primarily spent resins, are collected in a tank equipped with stirrer and a coil for steam heating. A slurry with 15-20 % dry weight is prepared. Granular resins are ground to powder before transfer to this tank. The slurry is heated to 60°C. Sodium sulphate and an emulsifying agent are added. Slurry and bitumen (Mexphalte 40/50) are fed into the thin film evaporator, which is heated by an electrically heated thermofluid. The bituminized waste flows from the bottom of the evaporator into a 220 liter drum.

There is now 5 years of operating experience with this process at Barsebäck. About 1500 drums have been produced. Activity concentrations in the waste before solidification have been in the range 4 MBq - 4 GBq/kg dry weight (0.1-100 mCi/kg). The contents of fission products such as Cs-137 and Sr-90 are very low and below detection limits, due to the excellent condition of the reactor fuel, and waste activity is almost exclusively from activated corrosion products. The ratio bitumen/dry weight in the bituminized waste is 1.7 - 2.5.

A program to establish quality control of the product on a routine basis is underway. Water content is usually < 0.5 % and swelling is unnoticeable visually after 3 months. Leach rates are about $2 \cdot 10^{-6}$ $g/cm^2 \cdot d$, both in deionized water and in salt water. Thermal analysis of the bituminized waste has shown no exothermic reaction below 500°C.

Waste bituminization is also planned for the Forsmark nuclear power plant. A batch process has been chosen, where the spent resins are dried before mixing with bitumen in a special vessel and filled into 200 liter drums. This plant is now being tested during inactive operation.

Arisings of bituminized resins are estimated to be 100-150 drums from granular resins and about 140 drums from powder resins, annually per BWR unit.

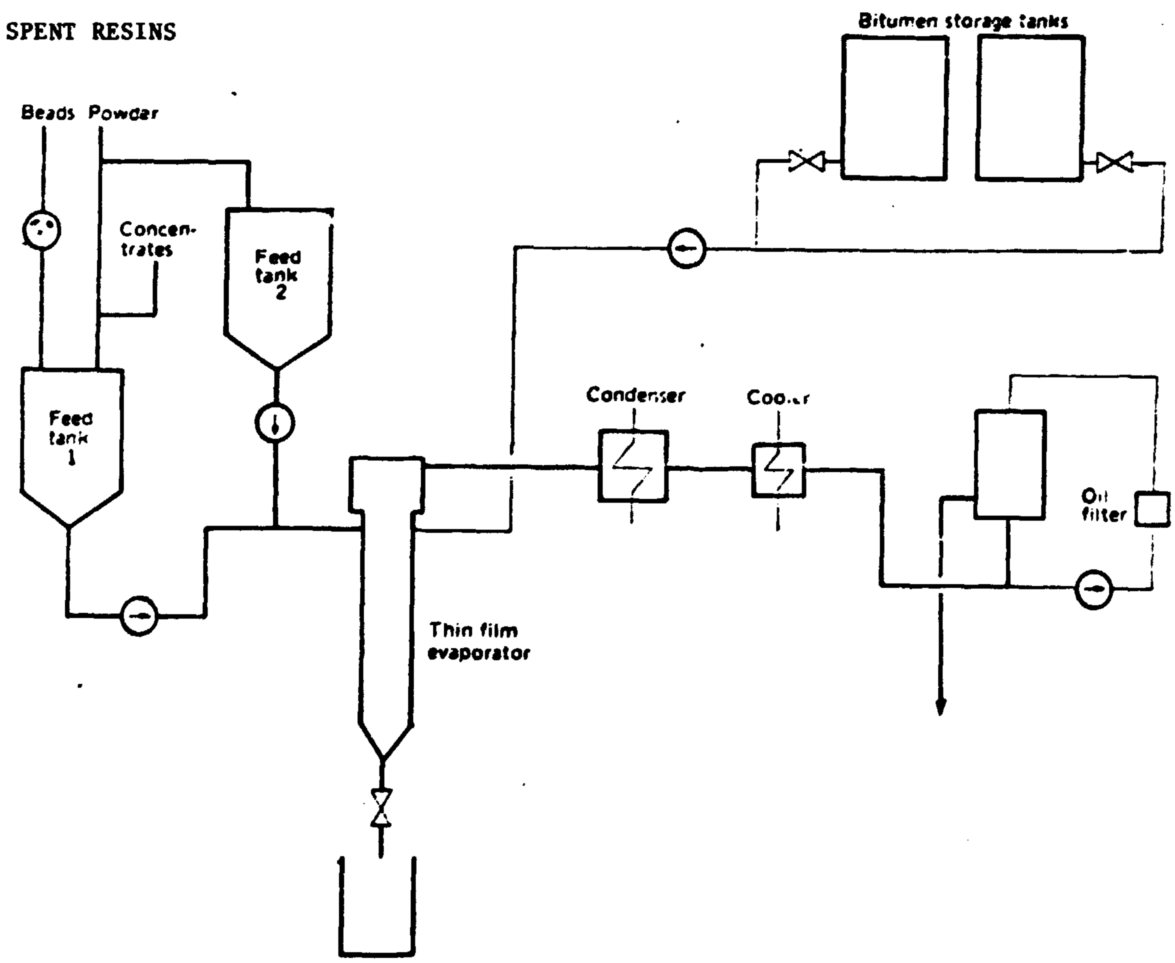

Fig 2. Waste bituminization at the Barsebäck Nuclear Power Plant

3.3 Research and development

The National Council for Radioactive Waste

An alternative route for the treatment of spent resins is under development, sponsored by the National Council for Radioactive Waste (Prav). The basic idea is to transfer the long-lived activity of significance, mainly Cs-137 and Sr-90, from the resins to inorganic ion exchangers (zeolites and titanates) that can be sintered to stable end products. A less demanding incineration of the spent resins may then be possible. A tentative flowsheet is presented in fig 3. A solution of a complexing agent (such as sodium tartrate) is circulated through columns, containing spent resin, zeolite and titanate. Radioactive elements are eluted from the resin and sorbed on the zeolite (cesium) and the titanate (strontium and corrosion products). Bench scale tests with spent resins from the Oskarshamn power plant have verified that at least 99.9 % of Cs and Sr in the resins can be eluted and sorbed on zeolite-titanate. An engineering design study of a pilot plant with a nominal capacity of 10 m^3/y spent resins is now near completion.

Parallel studies on drying-sintering of loaded zeolite and titanate are in progress including hot isostatic pressing-sintering. Fluidized-bed incineration of spent resins is studied in an inactive pilot plant, as part of the overall Prav programme. One aim is to attain a complete incineration at a temperature sufficiently low to avoid cesium volatilization. The ashes may then be mixed with e.g. glassforming additives and sintered to stable materials.

The integrated process could reduce initial waste volumes sent to storage/disposal by a factor of 10, to compare with a volume increase by a factor of 2 for bituminization or 4-20 for cementation. One additional advantage is the low leach rates of Cs and Sr for the ceramic end products, comparable to those of glass.

During the spring of 1980, a Swedish-Norwegian cooperation, supported by Prav, will demonstrate the use of titanate for the immobilization of high-level waste from the reprocessing of low burn-up fuel at Kjeller, Norway.

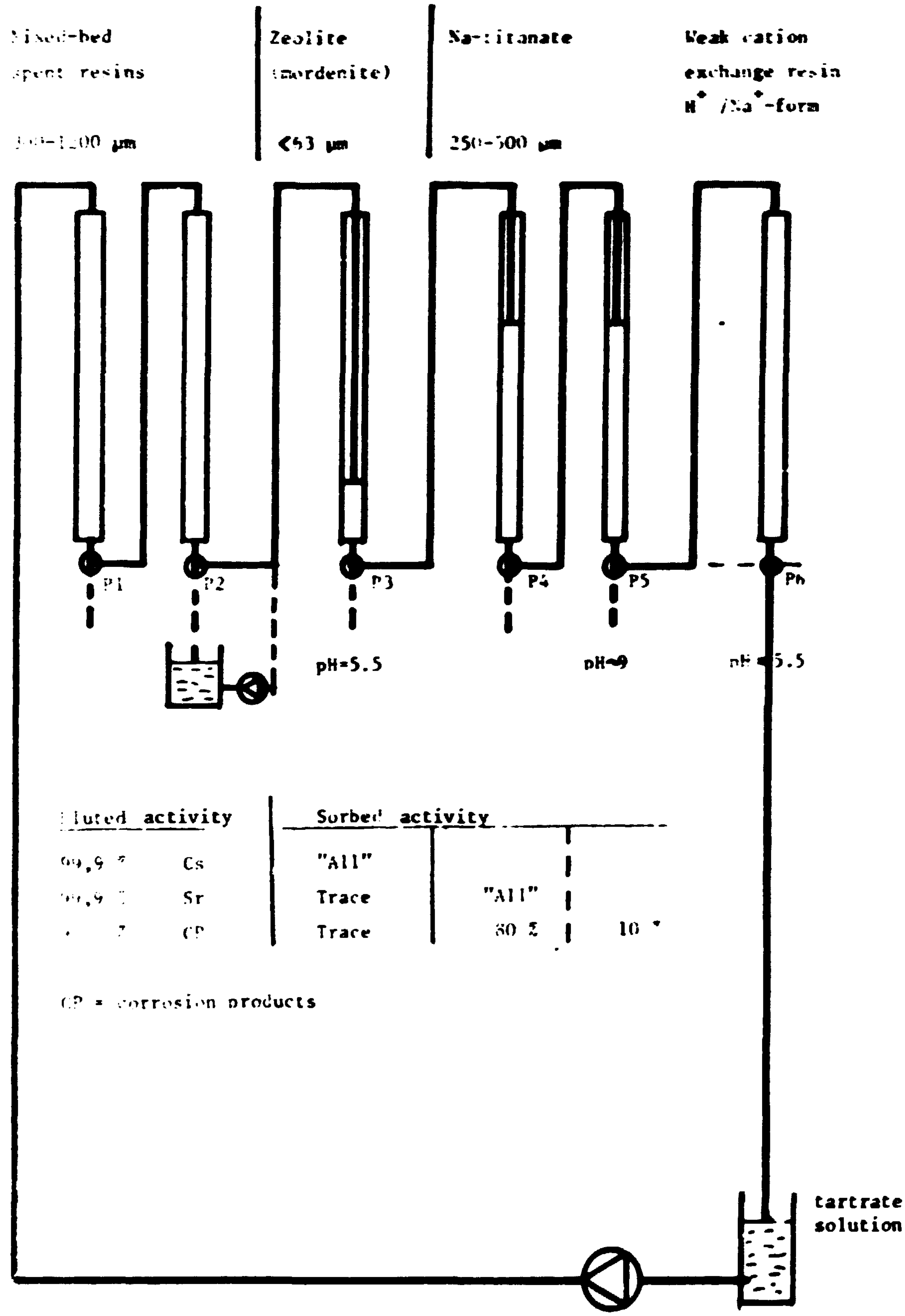

Fig 3. Tentative flowsheet for activity transfer from spent resins to zeolite-titanate

Studsvik Energiteknik AB

Development work at Studsvik, sponsored by the Board for Energy Production Research (NE) is focused on materials testing and waste product characterization. STUDSVIK has also developed a powder sintering process for nuclear waste immobilization. Calcines or ashes are here micro encapsulated in a glass matrix after mixing with glass powder and sintering at high pressure.

Utility programme

The utilities have within the KBS project during 1979 started a comprehensive evaluation of the present management of low and medium level wastes and future options in this field. The objective is to develop a safe and cost-effective scheme for the management of these wastes from their arising to their disposal.

4 STORAGE AND DISPOSAL

4.1 On-site

Storage facilities for low and medium level waste with a capacity to store wastes from 5 years of operation have been available from the beginning at the Swedish nuclear power plants. Since waste arisings have been larger than expected and the resolution of the question of a central waste facility has been delayed, additional storage facilities have been constructed at the 3 nuclear power plants in operation, i.e. Barsebäck, Oskarshamn and Ringhals. These facilities cover waste storage requirements until 1985-87.

At Barsebäck, the 670 m^2 storage building contains 112 concrete cells, each with a capacity of 24 drums of bituminized waste. Three larger concrete cells are reserved for spent core components. The ventilation system keeps the air humidity at 50 %. Special attention has been given fire detection and fighting in the storage building.

The storage building at Ringhals has an area of 2880 m^2, of which 2390 m^2 are for concrete moulds. Remaining area is for service and the storage of waste drums.

At Oskarshamn the new waste storage plant has been constructed in granite rock with its floor 19 m below ground level. Seven 150 m^2 tunnels linked at right angle by a transport tunnel, and a long storage tunnel parallel to the transport tunnel, have been excavated. The long tunnel is reserved for spent components and concrete moulds. Components of only medium level activity will be stored in this facility. The shorter tunnels will be used for the storage of concrete tanks with dewatered resins, and for trash. One of them will house facilities for service, control, and personnel. A drainage system prevents flooding of the plant. The ventilation system keeps temperature above + 5^{o}C and air humidity below 60 %.

At Studsvik sludges and spent resins are stored in 2 concrete tanks, each of 150 m^3 capacity. Part of the ashes from future operations of the incineration plant for low level waste may also be of intermediate level activity. All ashes are immobilized in concrete. The solid intermediate level waste is stored in a concrete shielded facility, used also for the storage of solid high level waste, and in a small separate facility, shielded by concrete.

A study of the design and possible siting of a repository at Studsvik for low and medium level waste produced and treated there was commissioned during 1979. Location in rock is a basic assumption. The design will be for retrievability with later in-place disposal of part of the waste possible.

4.2 Central repository

A conceptual study of a central repository for low and intermediate level waste (ALMA) was started in 1977. It included two alternatives: a surface storage facility and a rock cavern. The rock cavern alternative appeared most attractive after the first stage of the study since, with certain precautions, it should allow final disposal of the waste. The second stage of the ALMA project has thus been focused on the disposal aspect.

The repository is dimensioned for a 30 year operation of a 10 000 MWe nuclear power programme, giving rise to a waste volume of 120 000 m^3, 80 000 m^3 of which are intermediate level waste. Inventories of different waste packages and radioactive nuclides in the filled repository are presented in tables 2 and 3, respectively.

Table 2. Waste volumes and disposal area in ALMA

Package type	Total volume m^3	Intermediate level waste	
		Volume, m^3	Surface area, m^2
Concrete blocks	70 000	55 000	275 000
Bitumen blocks	20 000	16 000	160 000
Ash drums	6 000	2 000	20 000
Compacted solids	24 000	7 000	65 000
Total	120 000	80 000	520 000

Table 3. Inventories of radioactive nuclides in ALMA

Nuclide	Half life, years	Activity, Bq (Ci)
H-3	12.3	$1 \cdot 10^{13}$ (300)
C-14	5 735	$7 \cdot 10^{10}$ (2)
Co-60	5.3	$7 \cdot 10^{14}$ (20 000)
Ni-59	80 000	$2 \cdot 10^{12}$ (50)
Ni-63	92	$2 \cdot 10^{13}$ (500)
Sr-90	28	$4 \cdot 10^{14}$ (10 000)
I-129	17 000 000	$2 \cdot 10^{10}$ (0.5)
Cs-135	3 000 000	$2 \cdot 10^{10}$ (0.5)
Cs-137	30	$4 \cdot 10^{15}$ (100 000)
Pu-239	24 400	$2 \cdot 10^{10}$ (0.5)

Crystalline rock of good quality can be found in many places in Sweden and was chosen as host rock. Three different designs of the repository were evaluated:

- horizontal vaults
- vertical, cylindrical silos
- horizontal tunnels

A leakage barrier, i a a sand-bentonite mixture, 1.5 m thick, between the rock and the concrete walls of the repository, was introduced in the three designs, to improve containment and to reduce dependence on the quality and control of the waste packages. The increased containment efficiency also justifies the construction of the repository at more shallow depth, which should be, however, at least 75 meter.

The tunnel alternative was found to be about 25 % more expensive than the other two; it puts lower demands, however, on the rock quality. A preliminary estimate indicates a construction cost of 500 $\$/m^3$ or $3 \cdot 10^{-5}$ \$/kWh.

4.3 Radioactive waste register

A computerized register of radioactive waste inventories was initiated by SKI during 1979 and is now operating on a trial basis. Waste is recorded at the time it has received its final form for storage. Numerical codes have been worked out for different categories of waste and containers and for different modes of conditioning. Information recorded includes also volumes, weights, activity contents, dose rates and the times at conditioning, transfer, activity monitoring and dose rate estimations.

For process waste (spent resins etc), activity is monitored by gamma spectrometry. From this analysis a number of nuclides are recorded, mainly Co-60, Zn-65, Cs-134, Cs-137, Mn-54, Co-58, Zr-95, Nb-95 and Ag-110m. Amounts of Sr-90 and eventual actinides, such as U-238, Pu-239 and Cm-242, are recorded annually, based on radiochemical analysis and alpha spectrometry.

For non-process waste (components etc) only dose rates at 1 m are presently monitored. This is complemented by an annual account of nuclide distribution in contaminated components.

Main reports, to SKI and SSI, are issued annually by April 1st, completed by less detailed reports issued quarterly. In the reports the recorded volume of data is reduced to a manageable level, without losing representativity. Only contents of significant nuclides are reported separately with the rest reported summarized.

A detailed print-out of all the actual waste data is readily available with this type of register to support management decisions in the case of abnormal events.

5 TRANSPORTATION

The system necessary for the transportation of radioactive waste to a central repository has been studied in connection with the ALMA programme. The system capacity includes the movement during a 10-year period of waste presently stored on-site in addition to new waste arisings, exclusive of 5 years production. The need of waste package standardization to a few categories became evident, and pre-fabricated steel or concrete containers for drums or box-shaped waste units are therefore assumed. Strong industrial packages are considered sufficient, dimensioned to give surface dose rates below 2 mSv/h from the waste contained.

All the Swedish nuclear power stations are sited on the coast and the main alternative is a containerized roll on/roll off sea transportation system. A special vessel was projected, based on 170 days per year operation and a crew of 8 man.

The displacement of the vessel is about 3700 tonnes and the payload about 1100 tonnes with the radiation shields water filled. The vessel is constructed for service during ice conditions. A deep double bottom and wide side tanks are expected to protect the cargo during all possible cases of grounding and in 85-90 % of all statistically expected serious collisions. The salvage of the vessel in the actual waters would be possible using conventional technology.

The cargo is located in the middle of the vessel, with all machinery remotely controlled. The cargo area may be washed after deloading with automatic sprinklers, if required. Normally, the cargo is handled by only one man.

The consideration of barges showed these to be of less interest, both from safety and economic points of view. A truck transportation system was found to have a greater flexibility to the variation of waste volumes, but more complicated to manage and supervise. Sea transportation by ship is therefore the preferred alternative.

6 RISK ANALYSIS

Most of the Swedish risk analysis work in the radioactive waste area has been devoted to high level waste within the utility efforts to fulfil the Stipulation Law. For intermediate level waste Prav has initiated work for the ALMA-programme. Some work on model development is sponsored by SKI. A joint Nordic project started in 1977 includes the overall safety analysis of the handling and disposal scheme for reactor wastes.

6.1 The ALMA programme

Potential environmental impacts from the sea transportation of waste to a central repository have been assessed. Under normal conditions no radioactive material is released to the air or to the sea. In the event of a severe ship collision, part of the cargo may be damaged and lost to the sea. The frequency for the ALMA-ship to be involved in a serious accident while carrying radioactive cargo at sea is estimated to be about 10^{-3} per year. The risk to loose a few concrete containers is about $2 \cdot 10^{-5}$ per year. Estimates of resulting individual and collective dose commitments are in progress and will be published by Prav during spring 1980.

The unlikely event of a total loss of the vessel without any retrieval will cause very low releases and most of the activity will decay before its release. In view of the low probabilities for accidents releasing radioactive materials and their limited environmental impacts, a sea transportation system is found to meet very high safety standards.

An assessment of the post-operational long-term safety of the repository is also in progress. Permissible leak rates for significant nuclides from waste containers have been calculated assuming no man-made barrier around the repository, to indicate the requirements on waste containment quality without such a barrier. The migration of significant nuclides to surrounding rock in the presence of a peripheral barrier will be reported shortly. The most important nuclide turned out to be I-129.

6.2 Nordic cooperation

The Nordic liaison committee for atomic energy (NKA) initiated in 1977 a joint project in the radioactive waste area as part of a safety research programme, financed by the Nordic council of ministers. The project analyses the management scheme for reactor waste from solidified product to its disposal. As reference waste, spent resin from a BWR is selected, immobilized in cement or bitumen. Transportation by truck and by ship from intermediate storage to a repository is considered. Reference disposal is by shallow-land burial, in a near-surface concrete facility and in a rock cavern repository. The purpose of the project is to define the important evaluation parameters, to establish interrelations between system conditions and required containment quality of packaged waste, and to indicate where further R&D would be required. The project will be terminated by the end of 1980.

7 CONCLUSION

The conditioning and on-site storage of intermediate level wastes have been practised for a number of years in Sweden. Considerable efforts are currently devoted to improved technology for their immobilization and disposal. It may be assumed that by the end of the 80's an improved management system, including disposal, will be available and in operation.

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