

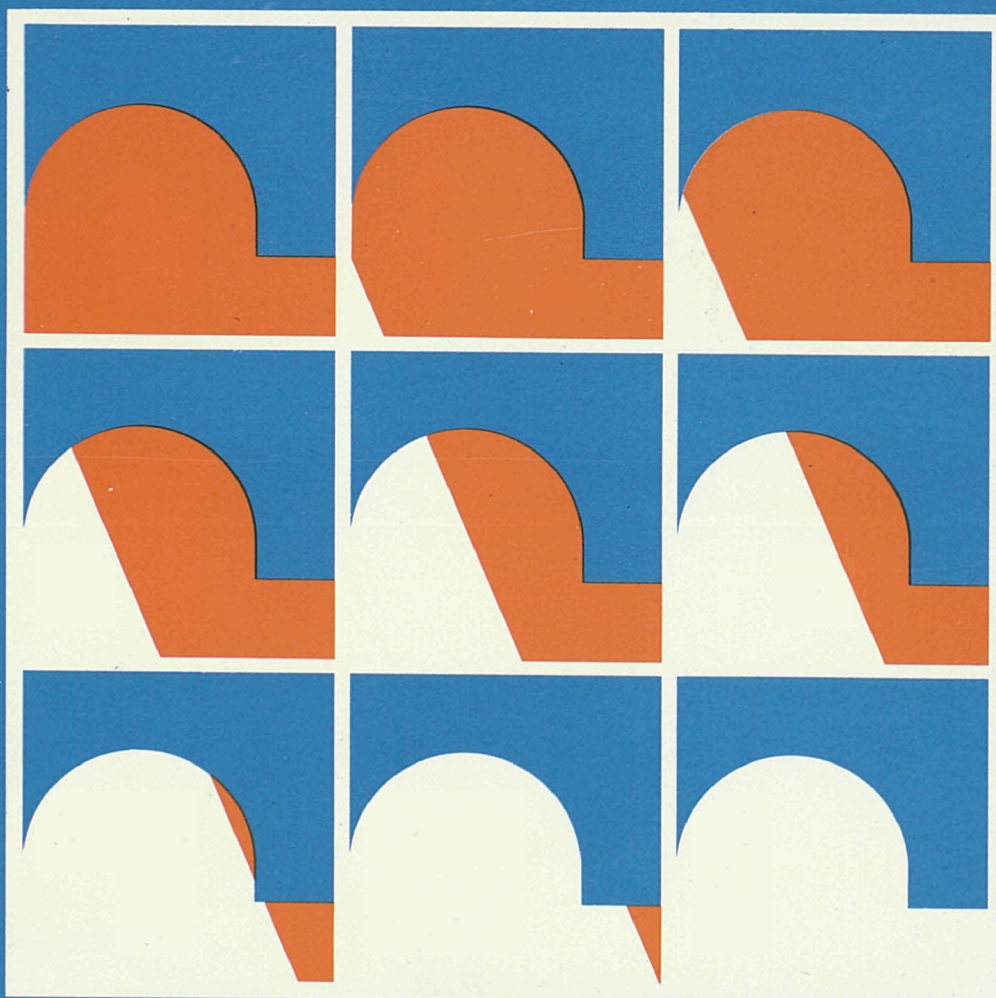


Commission of the European Communities

nuclear science and technology

The Community's
research and development programme
on decommissioning of nuclear installations

First annual progress report (year 1985)



Report

EUR 10740 EN

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Directorate-General
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FOREWORD

This is the first Annual Progress Report of the European Community's 1984-88 programme of research on the decommissioning of nuclear installations. It shows the status of implementation reached on 31 December 1985.

The Council of the European Communities adopted the programme in January 1984 /1/, considering: "Certain parts of nuclear installations inevitably become radioactive during operation; it is therefore essential to find effective solutions which are capable of ensuring the safety and protection of both mankind and the environment against the potential hazards involved in the decommissioning of these installations".

Also, the Council recognized that the 1979-83 programme of research on the decommissioning of nuclear power plants, of which the current programme is a follow-up, "has yielded positive results and opened up encouraging prospects". The main publications relating to the results of this first programme are listed in Annex I.

The 1984-88 programme has the following contents:

- A. Research and development projects concerning the following subjects:
 - Project No 1: Long-term integrity of building and systems;
 - Project No 2: Decontamination for decommissioning purposes;
 - Project No 3: Dismantling techniques;
 - Project No 4: Treatment of specific waste materials: steel, concrete and graphite;
 - Project No 5: Large containers for radioactive waste produced in the dismantling of nuclear installations;
 - Project No 6: Estimation of the quantities of radioactive wastes arising from the decommissioning of nuclear installations in the Community;
 - Project No 7: Influence of installation design features on decommissioning.
- B. Identification of guiding principles, namely:
 - certain guiding principles in the design and operation of nuclear installations with a view to simplifying their subsequent decommissioning,
 - guiding principles in the decommissioning of nuclear installations which could form the initial elements of a Community policy in this field.
- C. Testing of new techniques under real conditions, within the framework of large-scale decommissioning operations undertaken in Member States.

The research is carried out by public organisations and private firms in the Community under cost-sharing contracts with the Commission of the European Communities. The Commission budget planned for this five-year programme amounts to 12.1 million ECU.

The Commission is responsible for managing the programme. In this task, the Commission has been assisted till end of 1984 by the Advisory Committee on Programme Management in the field of the decommissioning of nuclear installations (see Annex II) and since the beginning of 1985 by the Management and Coordination Advisory Committee "Nuclear fission energy - Fuel cycle/processing and storage of waste" (see Annex III).

At the beginning of the 1984-88 programme, the Commission issued a call for research proposals /2/ with a first closing date of 9 May 1984, and in 1985 it announced a second closing date (30 June 1985), for complementary proposals /3/. Over 200 research proposals have been received in total and only a fraction thereof could be accepted.

By 31 December 1985, 27 research contracts had been concluded - they form the subject of the present report - and 35 contracts were at the stage of negotiation. Progress achieved in 1984, the starting year of the programme, was not important enough to form the subject of a separate report and has, therefore, been included in the present report.

This first progress report, covering the period of putting the programme into action, describes the work to be carried out under the 27 research contracts concluded, as well as initial work performed and first results obtained.

For each contract, the Paragraph "C. Progress of Work and Obtained Results" has been prepared by the contractor, under the responsibility of the Project Leader. The Commission wishes to express its gratitude to all scientists of the contractors who have contributed to this report.

The Commission staff having edited the report are: E. Skupinski, R. Bisci and K. Pflugrad.

B. Huber
Head of the Programme

References

- /1/ Council Decision of 31 January 1984 adopting a research programme concerning the decommissioning of nuclear installations. OJ N° L 36, 8.2.1984, p. 23.
- /2/ Commission Communication concerning the research programme on the decommissioning of nuclear installations (1984 to 1988). Call for research proposals. OJ N° C 68, 9.3.1984, p. 2.
- /3/ Commission Communication concerning the research programme on the decommissioning of nuclear installations (1984 to 1988). Closing date for research proposals. OJ N° C 101, 23.4.1985, p. 2.

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1. PROJECT N°1:
LONG-TERM INTEGRITY OF BUILDINGS AND SYSTEMS

A. Objective

It has been proposed that the dismantling of nuclear installations be delayed for periods ranging from several decades to about a hundred years. Thereupon, the radioactivity having largely died away, dismantling would be easier and the radiation exposure of the dismantling personnel would be less. The objective of this project is to determine the measures required for maintaining shut-down plants in a safe condition and to assess the radiological consequences and costs.

B. Research performed under the 1979-83 programme

The work performed under the previous programme relates mainly to the following aspects:

- mode and pace of degradation of various materials as they exist in nuclear power plants;
- measures for maintaining plants in a safe condition and for keeping the necessary ancillary systems operable;
- monitoring and inspection procedures;
- radiological consequences and costs of maintaining the plants.

C. 1984-88 programme

The work performed under the first five-year programme should be complemented by further tests and the study of control methods relating to the aging of relevant plant materials and by exploitation of additional experience with shut-down nuclear installations.

D. Programme implementation

At the end of 1985, three research contracts relating to Project N°1 were at the stage of negotiation.

2. PROJECT N°2:
DECONTAMINATION FOR DECOMMISSIONING PURPOSES

A. Objective

The objective of this project is to develop and assess techniques for decontaminating surfaces of components and structures of nuclear installations that are past use. The main purpose of decontamination would be reduction of the occupational radiation exposure during dismantling of the contaminated item and/or reduction of the volume of radioactive waste.

B. Research performed under the 1979-83 programme

The following decontamination techniques have been developed and assessed:

- techniques based on the use of chemically aggressive decontaminants in liquid and gel-like form;
- electrochemical techniques;
- hydromechanical techniques (high-pressure water lance, erosion by cavitation);
- decontamination of concrete walls by flame spraying.

Other activities were:

- investigation of the characteristics and distribution of contamination in nuclear power plants that are past use;
- economic assessment of decontamination for unrestricted release;
- collection of information on the particular decontamination problems posed by accidental contamination, as in the case of the TMI-2 nuclear power plant.

C. 1984-88 programme

Selected aggressive decontamination methods should be further developed with a view to their industrial application. Increased effort should be paid to the conditioning of spent decontaminants, where suitable techniques do not yet exist, and to the reduction of secondary waste arisings. Physical methods that limit the production of liquid effluents might be considered.

An important new topic of the second programme would be the decontamination of hot cells and equipment contaminated with plutonium and other transuranics for purposes of the decommissioning of fuel-cycle installations. The specific features of such installations (chemical nature of the liquids used during their operation, dimensions of the components, etc.) would be taken into account.

D. Programme implementation

At the end of 1985, six research contracts were at the stage of execution, and three contracts were at the stage of negotiation.

2.1. Complete Decontamination of a Primary Steam Piping of the Lingen BWR

Contractor: Kernkraftwerk Lingen GmbH, Lingen, Germany
Contract N°: FI1D-0001
Working Period: January 1985 - March 1986
Project Leader: W. Ahlfänger

A. Objectives and Scope

A foregoing research contract (DE-B-004-D), aimed at the investigation of the composition of contamination layers and of the effectiveness of possible decontamination procedures of primary circuit steam lines, was concluded by following main results:

- the surface contamination is to an extent of 99% of oxide composition, the remainder is located at a penetration depth of up to 90 μm in the base material. For a successful decontamination, it is necessary to dissolve, besides the oxide layer deposited on the surface, also a small layer of the base material;
- the best way of decontamination (using solutions with less than 2% concentration) is to strip the deposited oxide layer by a LOMI reactive and a part of the base material by a mixture of hydrochloric and nitric acid.

These results have been obtained by laboratory-scale tests on representative samples.

The objective of this research contract is to demonstrate that the above decontamination procedure is also appropriate for a large-scale application to a steam line of the Lingen Nuclear Power Station.

B. Work Programme

- B.1. Manufacturing of the decontamination rig comprising the sample steam pipe and all needed components for decontamination.
- B.2. Preliminary laboratory decontamination tests of representative samples including determination of the composition and activity level of the contaminated layer.
- B.3. Main test programme using the decontamination rig.
- B.4. Assessment on optimal treatment of the generated radioactive secondary waste.
- B.5. Evaluation of experimental results with respect to man-dose, quantities of secondary waste and cost analyses, with extrapolation to a 1200 MWe BWR.

C. Progress of Work and Obtained Results

Summary

An average value of 3.1×10^4 Bq/cm² for the surface activity (Co-60) of the primary steam line resulted from the studies. As shown by the results of the depth profile measurement, the Co-60 content in the base material amounts to $< 4 \times 10^{-3}$ Bq/mg concerning a layer of 40 μm. The lowering of the pH value to 2.5 concerning the LOMI treatment steps was an essential change with respect to the different chemical treatment solutions. By doing this, the loosening of the oxidic layers could be clearly improved. After the completion of the process, the residual activity (Co-60) amounted to 0.33 Bq/cm².

Progress and Results

1. Set-up of the decontamination circuit (B.1.)

A pipeline piece was separated from the remaining system for the decontamination test and bound into a decontamination circuit with pump, buffer container etc. (Fig. 1).

2. Pre-examinations in the laboratory (B.2.)

As was shown by the pre-examinations, the concentrations of the oxidation and oxalic acid solutions could be reduced by a factor of 4 without a deterioration of the decontamination effect. The examination was of special importance as the removal of the oxidic layers could be clearly improved by the lowering of the pH value to 2.5 with respect to the LOMI reagent. The results from the studies concerning the initial surface activity (Co-60) showed different activity values:

$$6.1 \times 10^3 \text{ to } 5 \times 10^4 \text{ Bq/cm}^2.$$

From the activities removed by the decontamination, an average surface activity of

$$3.1 \times 10^4 \text{ Bq/cm}^2$$

was calculated.

The results of the depth profile measurement are shown in Fig. 2.

3. Execution of the decontamination (B.3.)

The entire process course and the decontamination effect of the individual treatment steps are summarized in tables I to III. The following particularities were remarked:

- Contrary to the observations during the laboratory tests, pit corrosion was clearly noted after the first acid treatment.
- The Cr₂O₃ layer, precipitated during the acid treatment with the also precipitated activities, was not sufficiently removed by the subsequent treatments with the oxidation and oxalic acid solution, so that activity was carried into the next treatment step. This is shown by the too high surface activities of the pipe samples f and g compared to the activities measured in the base material in these layers (Fig. 2).

After the last treatment step, the residual activity (Co-60) amounted to 0.33 Bq/cm² and was therefore smaller than the limit for unrestricted release (0.37 Bq/cm²).

TABLE I: Treatment to remove the oxide layer

Solution	Chemical agents	Duration (h)	Temperature (°C)
LOMI	VSO ₄ 0.75 g/l	0.5	90
	H ₂ SO ₄ 0.3 g/l		
	Formic acid 1.0 g/l		
	Picoline acid 2.5 g/l		
	Hydrazine (15%) 1.2 ml/l		
	pH 2.5		
Oxidation	KMnO ₄ 1.0 g/l	1.5	90
	HNO ₃ - pH 2.5		
Acid	Oxalic acid 1.0 g/l	1.5	88

TABLE II: Treatment to remove the base material

Treatment		1	2	3	4
<u>Acid</u>					
HCl	g/l	13	22	6.5	
HNO ₃	g/l	4.7	4.7	2.35	
duration	h	6.5	7	8	
temperature	°C	60	60	60	
dissolved material (Fe)	g	552	592	194	
removed layer	µm	15.3	16.4	5.4	
<u>Oxidation</u>					
KMNO ₄	g/l	0.5	1	1	2
NaOH	g/l	0.5	0.5	0.5	2
duration	h	2	2	2	2
temperature	°C	90	90	90	95
<u>Oxalic acid</u>					
oxalic acid	g/l	0.5	1	1	2
duration	h	0.5	0.5	0.5	1
temperature	°C	33	30	32	60

TABLE III: Decontamination course

Treatment steps	Dissolved activity (Bq)	Activity determined by electrolitic removal (Bq/cm ²)	Samples from a pipe rample
Before treatment			a
<u>Removal of the oxide layer</u>			
LOMI-solution	2.0×10^8		b
oxidation solution	3.1×10^8		c
oxalic acid solution	4.1×10^8		d
<u>Removal from the base material</u>			
1. acid treatment			
acid solution	7.4×10^6		e
oxidation solution	1.8×10^6		
oxalic acid solution	1.4×10^6	16	f
2. acid treatment			
acid solution	2.1×10^5		
oxidation solution	1.2×10^5		
oxalic acid solution	1.0×10^5	5	g
3. acid treatment			
acid solution	1.9×10^4		
oxidation solution	3.4×10^4		
oxalic acid solution	3.3×10^4		
4. subsequent solution			
oxidation solution			
oxalic acid solution	6.5×10^4	0.33	h

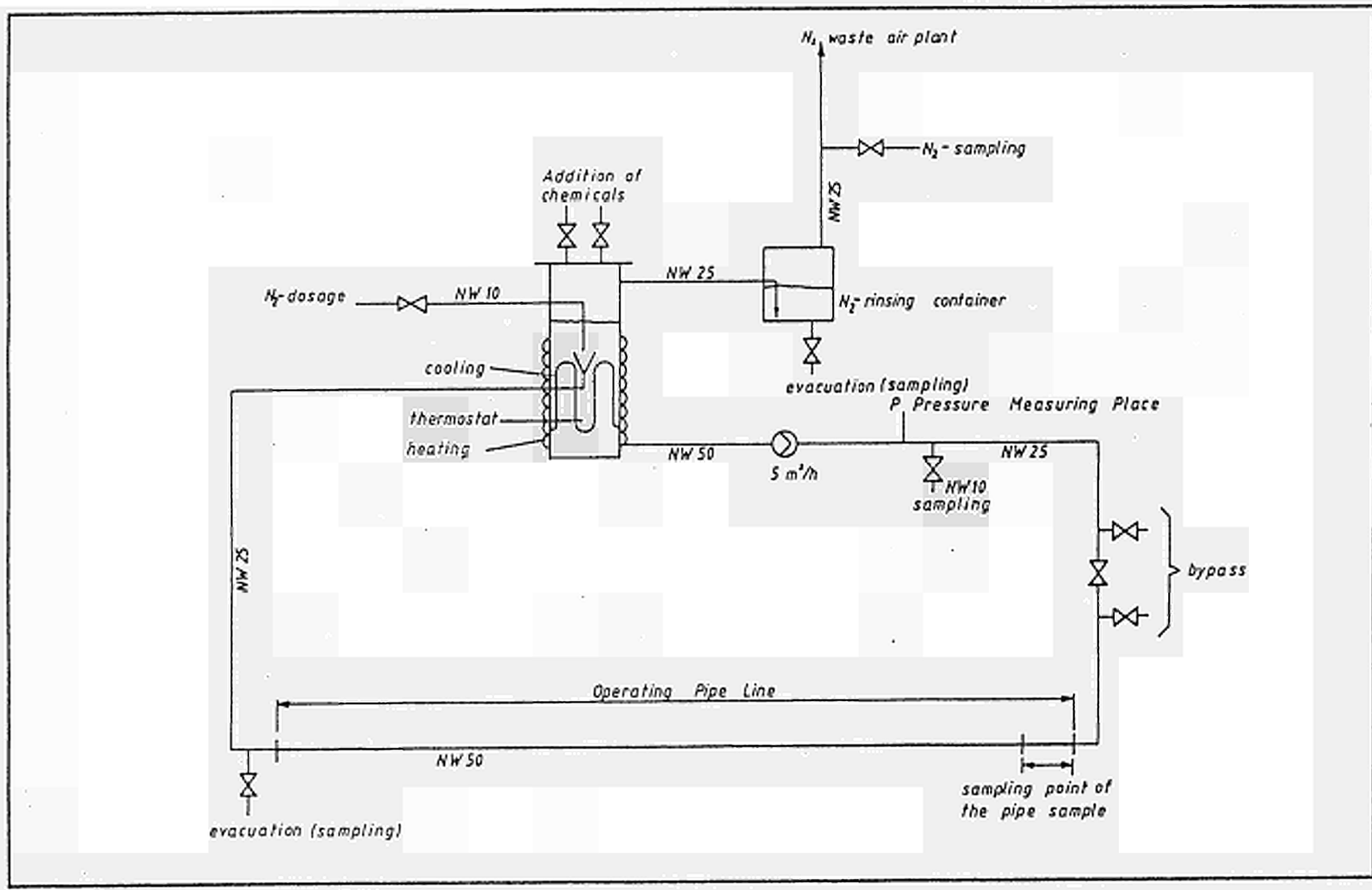


Figure 1. Basic scheme of the decontamination circuit

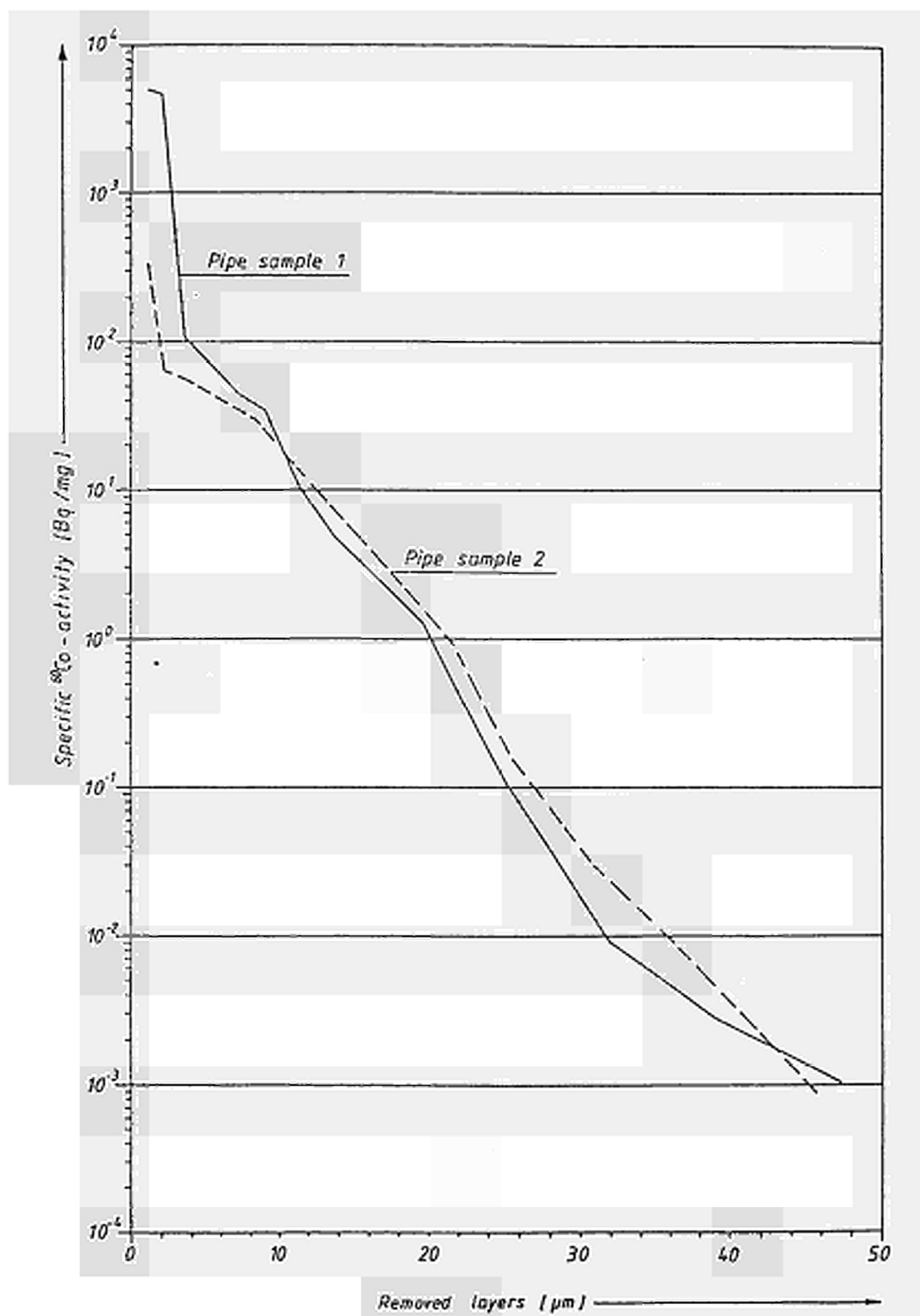


Figure 2. Specific Co-60 activity in the removed layers

2.2. Hard Chemical Decontamination Tests on Valves and Treatment of the Waste Arising from the Process

Contractor: Ente Nazionale per l'Energia Elettrica, Roma, Italy
Contract N°: FIID-0002
Working Period: January 1985 - December 1986
Project Leader: F. Bregani

A. Objectives and Scope

The aggressive chemical decontamination methods, whose effectiveness has been proved both in many laboratory tests and in pre-industrial applications, appear to need further investigations regarding both the decontamination of complex systems, such as valves, and spent decontaminant treatment in view of the limitation of the secondary wastes arising.

The scope of the research is both to check the effectiveness of hard chemical decontamination on used components, such as small valves, and to search and develop a suitable and safe procedure to treat spent solutions, arising from aggressive chemical decontamination.

The advantages of this research are the possible demonstration of the decontamination effectiveness on complex components and the minimization of the total wastes produced.

This proposed research will be carried out in collaboration with CISE in the framework of a specific multi-annual agreement already in force. The experiments will be performed in DECO laboratory at Ispra, JRC.

Regarding the application of chitosan, specific agreements with the University of Ancona have already been undertaken.

B. Work Programme

- B.1. Hard chemical decontamination tests on valves (2-3 inches) of the primary cooling system of the Garigliano BWR in DECO loop.
- B.2. Identification and qualification of a simple procedure to condition the spent decontaminant.
- B.3. Neutralization and flocculation tests in order to select and evaluate the best neutralizing agent and specific chemical agents, such as chitosan, as supporter in flocculation.
- B.4. Cost evaluation of the process and assessment of the possibility of reprocessing and reutilizing of specific agents.

C. Progress of work and obtained results

Summary

The results of two decontamination tests of small valves from the primary system of the Garigliano power station are reported.

The valves are in stainless steel base material and have been in operation for about fifteen years in high temperature and high pressure BWR conditions.

The runs were performed in the DECO experimental loop. The first run used a hydrochloric solution (4.1%) while the second one was carried out with a hydrofluoric and nitric mixture (1.5% + 5%). Many measurements on the valves after decontamination in order, to discover the exact localization of the residual radioactivity, were performed. To a first approximation, the results show that the hard chemical decontamination, to ensure total cleaning of all the internal surfaces, is very difficult; traces of residual contamination remain in some particular areas such as crevices, welds, dead zones, etc.

Preliminary tests about the treatment of the hard chemical decontaminant by flocculation and neutralization were performed too.

Progress and results

1. Introduction

Following the experimental studies performed in the period from October 1980 to September 1983 on the chemical decontamination with highly aggressive solution /1/, ENEL-CRTN is continuing with the activity in this field.

In particular, as part of a qualification programme in hard chemical decontamination, some small valves from the Garigliano power plant are testing in the DECO experimental loop at JRC-Ispra. Four contaminated valves from 1 to 2 inches nominal size, were selected for performing the tests. The valves were cut from auxiliary lines of the BWR Garigliano power plant in the Secondary Steam Generators (SSG) houses during the scheduled replacement work in 1979. The valves are in stainless steel base material and their internal surfaces are covered with deposits or oxide layers. The morphological characteristics of the contaminated films appear to vary largely for each valve.

2. Description of the decontamination tests (B.1.)

To perform the run, the valves were put on the DECO loop, on the main recirculation line after the flow-meter and before the large regulation valve. The on-line radioactivity is measured by means of Sodium-Iodine detector suitably shielded with lead and located near the contaminated ring of the electrochemical test section. The corrosion rate has been monitored by on-line the measurement of the inverse of the linear polarization resistance. For the test valves, the decontamination effectiveness has been measured by evaluating the radioactivity profiles before and after the run, both in the direction of the flow and in the direction along the obturator (trim). Before and after the run

in the DECO loop, the test valves were treated by ultrasounds in demineralized water, in order to remove the loose residual deposits and to produce a reference surface. At the end of the hard chemical decontamination the external surfaces of the valves were cleaned sometime by common detergents in order to make them as clean as possible and to have a clear localization of the residual radioactivity. The two first tests on the valves are detailly listed in table I and II.

3. Decontamination test results (B.1.)

At the end of the tests the valves were cut and sectioned in order to sample specimens for radiometric measurements. The results are given in Figure 1 and 2.

After the two first tests the following preliminary considerations can be made:

- the full scale decontamination of small valves (1-2 in. as nominal size) in order to reach residual contamination levels less than 1 Bq/cm² on all the internal surfaces, also as small as possible, is very difficult;
- whatever the decontamination process may be, some surfaces which are not totally decontaminated, so-called "hot-spots", remain in the valve (conventionally, for us, a hot-spot is an area, also as small as possible, in which the beta-gamma emitters contamination is greater than 1 Bq/cm²);
- there are generally hot-spots in dead areas where the recirculation of the decontaminant solution is very poor: such as crevices in weld regions, screw threds and so on;
- it is not possible to establish "a priori" the number of hot-spots in a valve because this depends on both the features of the valve and the characteristics of the contamination;
- up to now the only way to decontaminate a small valve totally is to cut it point in order to open and separate all the possible contaminated surfaces; of course this involves a considerable amount of time and expense and does not appear comparable for technical reason with the other ways (such as melting or direct disposal);
- one lesson learned is that is possible to recommend that in the design of nuclear valves, or any other nuclear components, much care must be taken in order to avoid the presence of crevices, screw threads, corners, edges and so on.

4. Treatment of spent decontaminants (B.2. and B.3.)

More than 25 batch tests have been performed mainly in order to establish the experimental ranges of precipitation of Co and Fe in varying the most significant parameters such as: nature of the spent solution, initial concentration of iron and/or cobalt, kind of neutralization agent, presence of flocculant (such as chitosan) and so on.

The main considerations from the tests are the following:

- Fe precipitates as ferric hydroxide at pH about 3.5 while Co precipitates

- as cobalt hydroxide at pH 6-9;
- no substantial differences exist in changing the base solution and the initial concentration of iron and cobalt;
- sodium hydroxide appears to be better than calcium oxide as a neutralizing agent because it permits better pH control;
- about chitosan the results show that it definitely has a beneficial effects reducing the residual radioactivity on the solution.

References

- /1/ F. BREGANI, R. PASCALI and R. RIZZI;
 Chemical decontamination for Decommissioning Purposes;
 EUR 9393 EN, 1984.

Table I - Description of the test with valve n° 1 (Dickers globe-valve, 1 in., ASA 1500).

-
- a) ULTRASOUND: 15 min in demineralized water;
 - b) HARD CHEMICAL DECONTAMINATION: 103 min; 4.1% HCl; 40°C; 1m/s;
 - c) HARD CHEMICAL DECONTAMINATION: 380 min; 4.1% HCl; 40°C; 1m/s;
 - d) ULTRASOUND: 15 min in demineralized water;
 - e) DECONTAMINATION WITH A COMMERCIAL DETERGENT: Brushing on Outside Surfaces;
 - f) ULTRASOUND: 30 min; Moving on to Close and to Open the Valve.
-

Table II - Decontamination steps of valve No. 2 (Edward Y-type valve, 1½ in., ASA 1500).

-
- a) ULTRASOUND: 15 min in demineralized water
 - b) HARD CHEMICAL DECONTAMINATION: 24 hours of which 8 hours in dynamic condtions;
 1.5% vol. HF + 5% vol. HNO₃; 40°C; 1 m/s
 - c) ULTRASOUND: 15 min in demineralized water
 - d) ELECTROCHEMICAL DESCALING: 4 steps of 20 min each. 1.5% vol. HF + 15% vol. HNO₃;
 25°C (start each step); 5-7 A/dm²
 - e) ELECTROPOLISHING
 - e-1) 3 steps of 20 min each;
 - e-2) 1 step of 120 min;
 - e-3) 2 steps of 60 and 330 min respectively;
 75% H₃PO₄; 25°C (start each step); 0.1 mA/cm²
 - f) HARD CHEMICAL DECONTAMINATION (only the trim): 4.5 hours in static conditions. 1.5% vol. HF + 5% vol. HNO₃, 25°C; and 15 min in static conditions: 4.1% vol. HCl 25°C.
 - g) HARD CHEMICAL DECONTAMINATION WITH ULTRASOUNDS (only the trim): 2 hours in static conditions; 4.1% vol. HCl; 25°C (start); 20 kHz
-

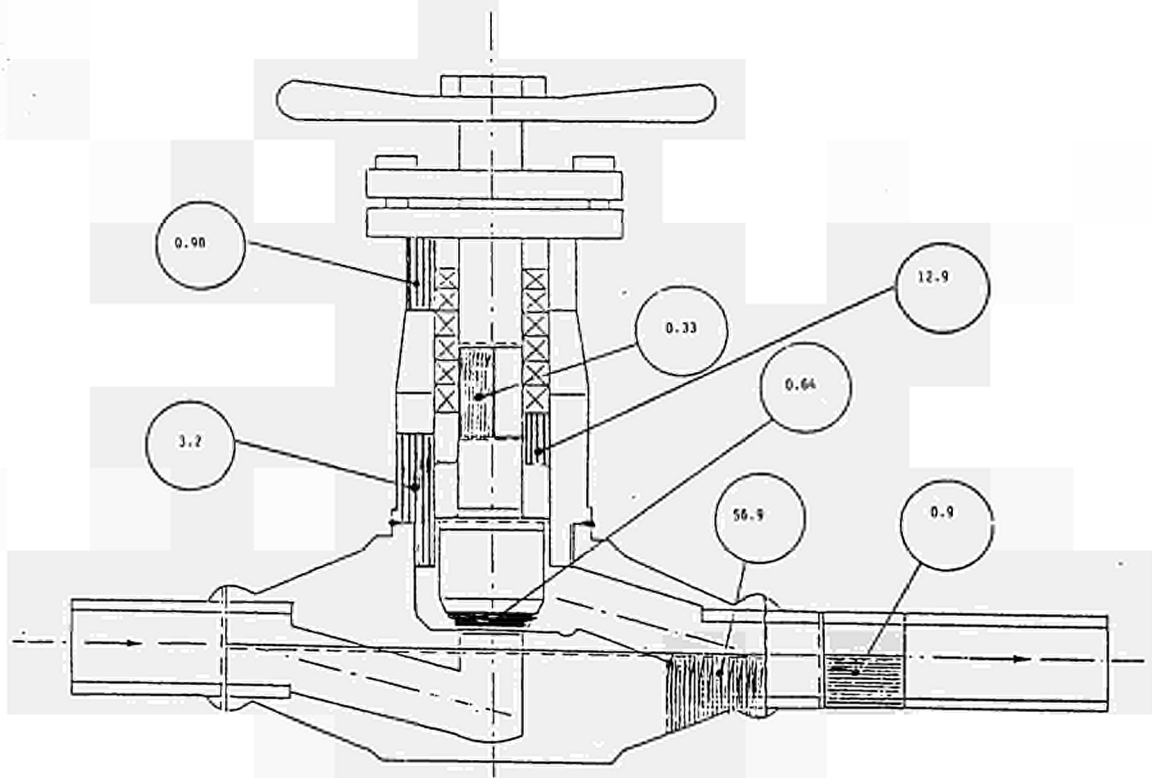


Figure 1. Co-60 data measurements (Bq/cm²) for the samples of valve N° 1

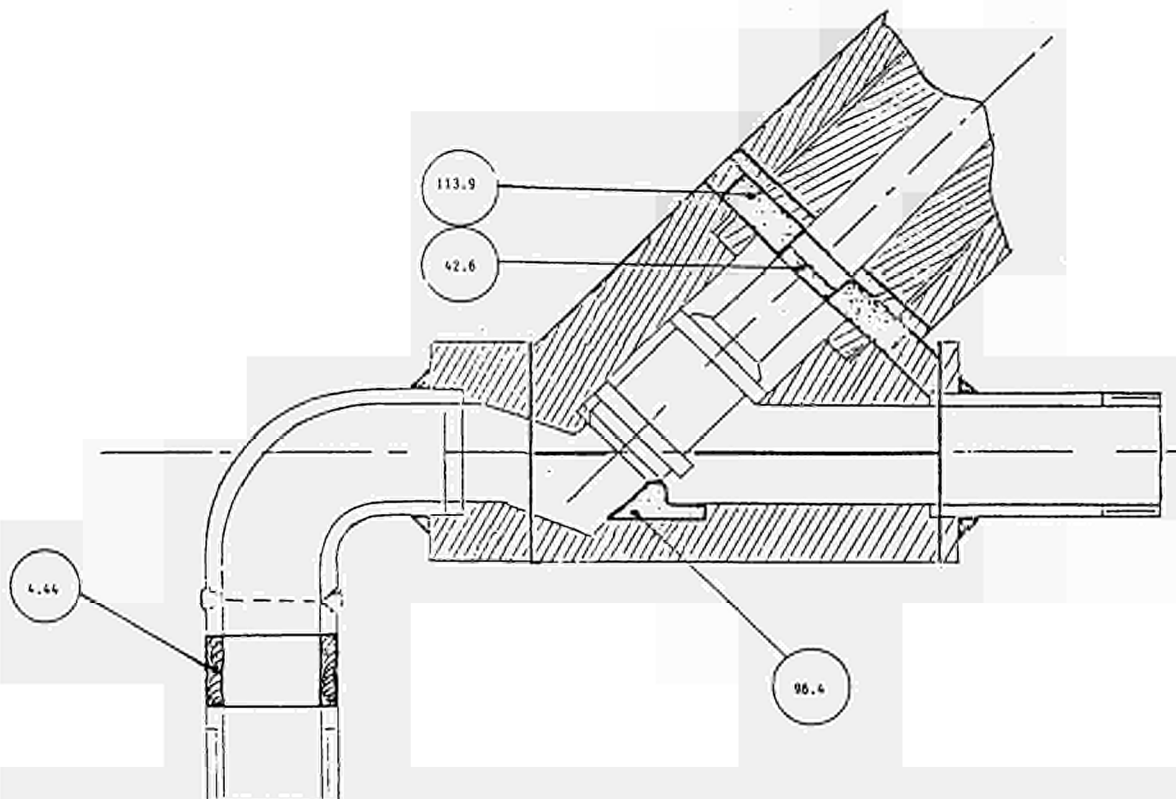


Figure 2. Co-60 data measurements (Bq/cm²) for the samples of valve N° 2

2.3. Decontamination Using Chemical Gels, Electrolytical Swab or Jet, Abrasives

Contractor: Commissariat à l'Energie Atomique, CEN-Cadarache, France
Contract N°: FI1D-0003
Working Period: January 1985 - December 1986
Project Leader: F. Josso

A. Objectives and Scope

As part of the dismantling of a nuclear installation, it is necessary to dispose of rapid and efficient decontamination procedures (high decontamination factor), which are simple to apply and lead to a low volume of wastes easy to treat.

The aim of this research is to study the following new decontamination techniques with a view to their application in the dismantling of nuclear installations:

- spraying of gels,
- electrolytical swab or jet,
- abrasive water blasting.

These techniques are expected to usefully complement the established methods (immersion in chemical bath, electrolytical bath, high-pressure jet) developed in a previous study (contract N° DE-B-008-F).

B. Work Programme

B.1. Optimization of the decontamination processes, i.e. abrasive water blasting, electrolytical tampon or jet and chemical gels, on non-radioactive samples of stainless steel, mild steel and aluminium.

B.2. Application on contaminated samples from various types of plant (graphite-gas reactor, PWR, LMFBR, fuel fabrication plant and reprocessing plant).

B.3. Implementation of these techniques with remote control and in the nuclear facilities before dismantling.

B.4. Assessment of quantity of secondary waste and its treatment.

B.5. Cost evaluation and assessment of radiological consequences of each process, including the treatment of secondary waste.

C. Progress of work and results obtained

Summary

This period was used for testing 3 decontamination methods (spraying of gels associated with decontaminating agents, swab electropolishing, abrasive water blasting) on non radioactive samples. Parameters to be determined covered the requirements for application of each method, the erosion velocity and the influence of each method on the surface condition of materials (stainless steel and mild steel).

Industrial application of gel spraying was demonstrated. The purpose of this operation was to decontaminate 17 tons of steel coming from a boiling water reactor. These parts consisted of frames and pipes.

Gel spraying was applied after decontamination in a chemical bath so as to obtain a high decontamination factor without producing a great quantity of secondary liquid wastes.

Progress and results

1. Tests on non radioactive samples (B1)

Tests were performed by applying the 3 decontamination methods specified on the following non radioactive samples :

- bright 304L stainless steel (initial roughness : Ra - 0.05 μm)
- glazed 304L stainless steel (initial roughness : Ra - 0.13 μm)
- 316L stainless steel (initial roughness : Ra - 0.17 μm)
- mild steel (initial roughness : 1.0 μm).

Parameters were defined for each method (application requirements erosion velocity, roughness).

Gel spraying

The volume of sprayed gel was 100 ml/m² to 200 ml/m².

With a hydrofluoric acid content of 1.6 mol/l, the erosion obtained by gel spraying on stainless steel is 0.1 $\mu\text{m}/\text{h}$ to 0.3 $\mu\text{m}/\text{h}$. The erosion is 15 times higher for an immersion with the same chemical agent, which is due to :

- the inhibiting power of the gel,
- the low quantity of chemical reagent contained in the gel,
- drying of the gel,
- the lack of natural convection.

On stainless steel, roughness after treatment with this type of gel increases by a factor of 2. It decreases on mild steel.

Swab electropolishing

The requirements for the application of this method are as follows :

- intensity : 125 A/dm²
- electrolytic flowrate : 5 l/h to 20 l/h
- electrolyte used : phosphoric acid.

For stainless steel, the erosion is 60 $\mu\text{m}/\text{h}$ to 100 $\mu\text{m}/\text{h}$. Final roughness obtained is 0.11 μm for 5 l/h flowrate, and 0.15 μm for 75 l/h flowrate.

With mild steel, the erosion velocity is 300 $\mu\text{m}/\text{h}$ to 600 $\mu\text{m}/\text{h}$ and roughness is not affected by the treatment.

Abrasive water blasting

The abrasive used is alumina (65 μm) , and the application requirements are as follows :

- Piece-to-nozzle distance : 7 cm
- Specific flowrate : 1 m³/m²
- Angle of incidence : 40

The water contains 10 % of abrasive.

The abrasive was tested on mild steel only. The erosion velocity was 5 $\mu\text{m}/\text{h}$ to 10 $\mu\text{m}/\text{h}$ and roughness slightly increased (Ra increased from 1.1 μm to 1.6 μm) (cf Fig. 1 and Fig. 2).

2. Application of gel spraying on a BWR (B2)

Gels were sprayed at Cadarache on parts coming from the cooling system of a BWR (11 tons of frames and 6 tons of pipes) (cf Table 1)

Processes involved were previously tested in laboratory. The most efficient reagents are :

- hydrofluoric acid at room temperature and hydrochloric acid at 80°C with a corrosion inhibitor,
- sulphuric acid up to 40°C,
- formol /formic acid mixture at 80°C.

The following process was selected :

Immersion in a sulphuric acid bath (2 mol/l) followed by spraying of gel containing nitric acid (2 mol/l) and hydrofluoric acid (2 mol/l)

A decontamination factor of 100 was obtained through the immersion stage.

A decontamination factor of 10 to 15 was obtained through the gel spraying stage.

Wastes produced by the decontamination of pipes (6 tons) were as follows :

- 5340 l liquid wastes with an activity of about 740 Bq/l
- 180 kg solid wastes with an activity of about 11 kBq/g
- 15 kg wastes with an activity of about 580 kBq/g.

The activity of pipes after decontamination was 1.4 Bq/g (cf Table 2).

Table I

Main specifications of parts of BWR to be decontaminated
(pipes only)

Mass	6 tons	
Surface	120 m ²	
Diameter (cm)	30 < ϕ < 50	
Thickness (cm)	0,7 < e < 2,5	
Operating temperature (°C)	45 < t < 135	
Dose rate (mrad/h)	10 < D < 300	70 %
	1 < D < 10	30 %
Gamma activity (kBq/cm ²)	2 < A _γ < 37	
Co 60 (%)	59	
Mn 54 (%)	40	

Table II

Pipes activity after decontamination

Masse (kg)	Activity (Bq/g)
851	1,1
448	3,2
552	1,3
346	2,2
408	1,2
420	0,5
295	0,7
464	2,7
394	0,8
389	1,0
282	0,3
230	0,7
234	1,2
Total 5313	1,4

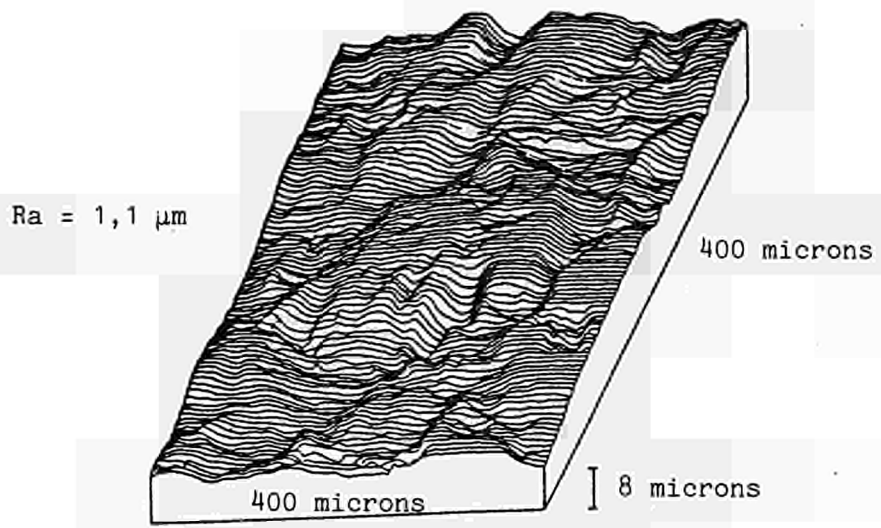


Fig. 1 - Mild steel (initial state)

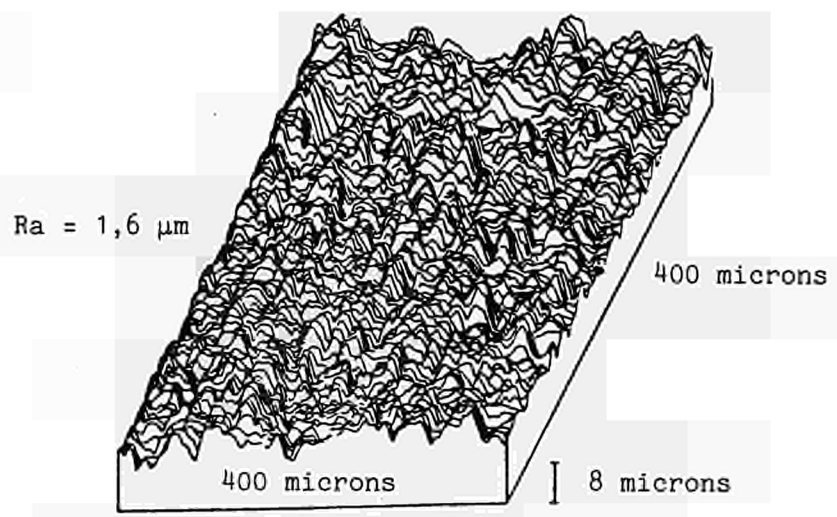


Fig. 2 - Mild steel after abrasion
with AVB 220 in water medium

2.4. Development of an Easy-to-Process Electrolyte for Decontamination by Electropolishing

Contractor: Kraftanlagen Aktiengesellschaft, Heidelberg, Germany
Contract N°: FIID-0004
Working Period: November 1984 - June 1986
Project Leader: A. Steringer

A. Objectives and Scope

Electropolishing has become an approved and suitable decontamination process achieving high decontamination factors. However, the spent electrolyte is hard to process and convert into a waste form suitable for disposal. For example, in order to solidify phosphoric acid at a concentration above 60% in cement, it must be neutralized and heavily diluted. As a result, the waste volume for disposal is much higher than the initial electrolyte volume.

The aim and objective of this research is to find an easy-to-process electrolyte with high decontamination factors, suitable for disposal, which would give a much wider range of application to electropolishing as a decontamination process. This means that it should be possible to condition the spent electrolyte in simple process steps, such as filtration, sedimentation and thermal decomposition, to produce a waste form that is easy to fix in cement.

The specified requirements with a view to easy processing of the electrolyte are fulfilled by a number of organic acids. In 1983, the contractor carried out various tests and experiments on organic acids. Whereas decontamination factors were satisfactory, unsatisfactory results were obtained for the electropolishing time, the service life and thermal stability of the electrolyte, current density etc. These process parameters must be optimized. This work will be carried out in collaboration with TEAM, Italy.

B. Work Programme

- B.1. Literature survey for identification of the available information on already existing experience.
- B.2. Selection of electrolytes other than phosphoric acid, promising easier conditioning and waste disposal.
- B.3. Test series on contaminated and non-contaminated samples in order to optimize the electrolytes with regard to decontamination efficiency (effect of chemical additives, of modifying process parameters,...).
- B.4. Optimization of the process to minimize the final waste volume.
- B.5. Development of procedures to extend the lifetime of electrolytes, in particular by continuous filtration.
- B.6. Processing of selected electrolytes (sediment elimination, salt precipitation, solidification of sludges, volume reduction of the residual liquid, solidification of electrolyte residues).
- B.7. Investigations about "on-the-job-safety": chemical aggressiveness, formation of toxic products, explosion hazards, ...

C. Progress of work and obtained results

Summary

During 1985 the literature review (B.1.) allowed the selection of two adequate electrolytes: formic acid and oxalic acid.

To increase the conductivity of the electrolytes, salts were added to the acids (B.2.). The addition of potassium bromide gave the best results: the value of the conductivity is higher than that of the conventional electrolytes: phosphoric acid/sulphuric acid (B.3.).

Progress and Results

1. Literature survey (B.1.)

This item of the programme covered the establishment of the theoretical, basic ideas and to get an idea of the work already done in this field with similar objectives. Decontamination trials with organic acid have been made so far in nuclear research facilities in Germany and France. The objective was to find out and evaluate the respective decontamination.

2. Selection of adequate electrolyte (B.2.)

Based on the findings obtained from item 1., two adequate electrolytes were chosen: formic acid and oxalic acid.

Formic acid has a high avidity related to iron. Oxalic acid forms hardly soluble salts, which settle out and facilitate filtration.

To increase the conductivity of the electrolytes, potassium chloride (KCl) and potassium bromide (KBr) were added.

3. Test series on non-contaminated samples in order to optimise the electrolyte (B.3.)

A test facility with a cooling bath was used to get a constant temperature (Fig. 1). Two series of tests were carried out.

In the preliminary tests, the removed surface thickness was measured: the addition of KCl decreases the electropolishing time (Fig. 2). In the second series of tests, temperature, pH-value, current, voltage and the discaled mass were measured and the anode-current-yield (ACY) was analysed.

In order to get comparable reference data, a measuring series was made with a conventionally used electrolyte being a mixture of 14% phosphoric acid and 47% sulphuric acid. For the tests, the following organic electrolytes were used:

E1: Oxalic acid (5%)

E2: Formic acid (10%)

E3: Oxalic acid (5%) / KCl (0,5 m)

E4: Oxalic acid (5%) / KBr (0,5 m)

E5: Formic acid (10%) / KBr (0,5 m).

On tests using organic acid (E1 and E2) low ACY were obtained, except formic acid on carbon steel. When KBr was added to the organic acids (E4, E5), a high improvement on the ACY could be obtained: 90% on stainless steel and > 90% (E5) and 75% (E4) on carbon steel. If KCl is added to the acid (E3), there would be a 10% decrease in ACY.

The comparison with the phosphoric/sulphuric acid-electrolyte (ACY 85% on stainless steel, 60% on carbon steel) indicates better ACY's for the organic acid/KBr-electrolytes (Fig. 3).

4. Development of the process to extend the electrolyte lifetime (B.5.)

The oxalic acid forms metal oxalates which are hardly soluble and therefore easily settle out and facilitate filtration. If KBr was added to the

oxalic acid a high ACY was detected, i.e. low current loss. Therefore the temperature of the electropolishing bath did not increase. No thermal decomposition of the electrolyte could be detected (no bubbling, no foam formation).

If using oxalic acid/KBr-electrolyte, only the loss of the chemical conversion has to be replenished.

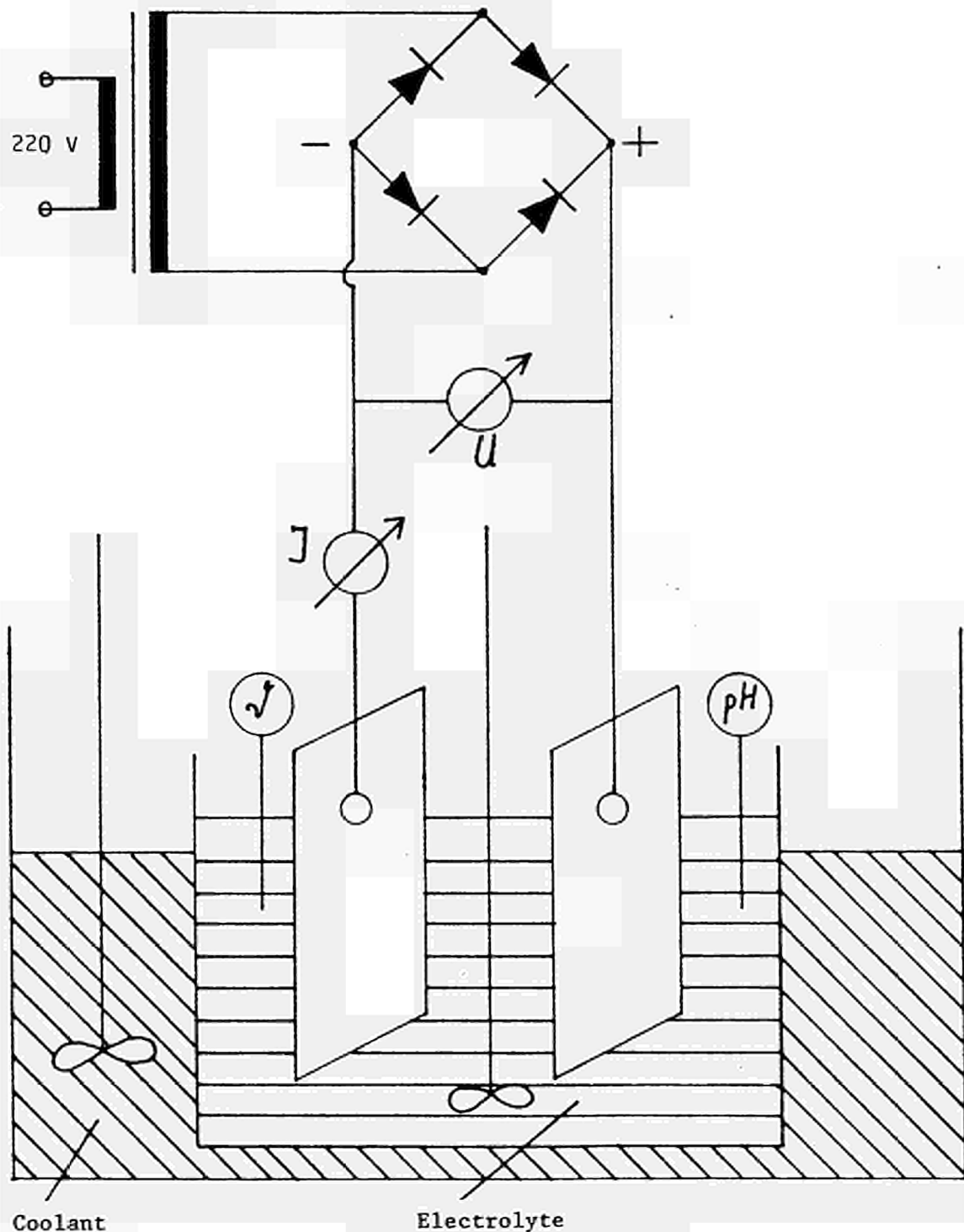


Figure 1. Test facility

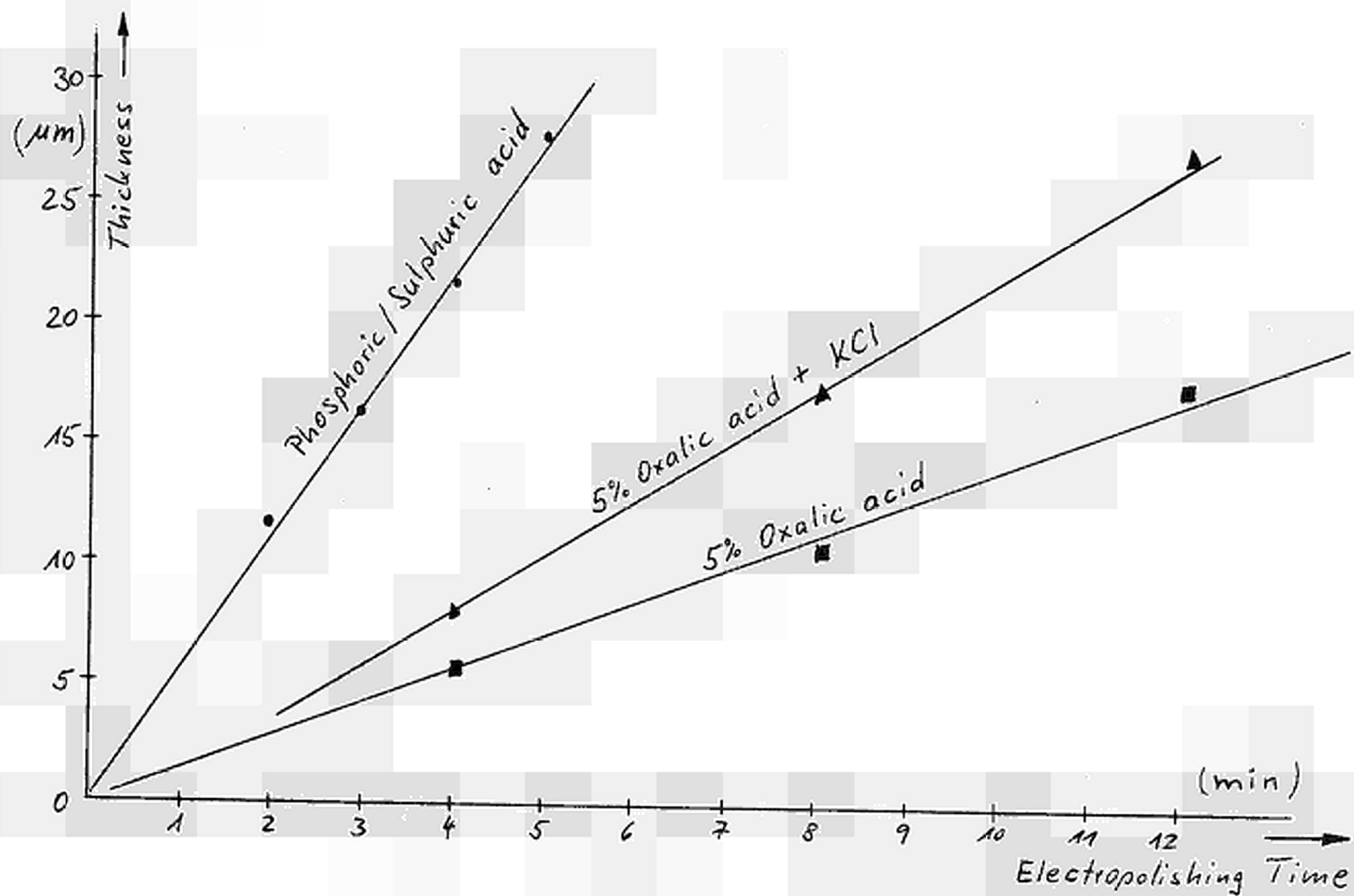


Figure 2. Removed surface thickness

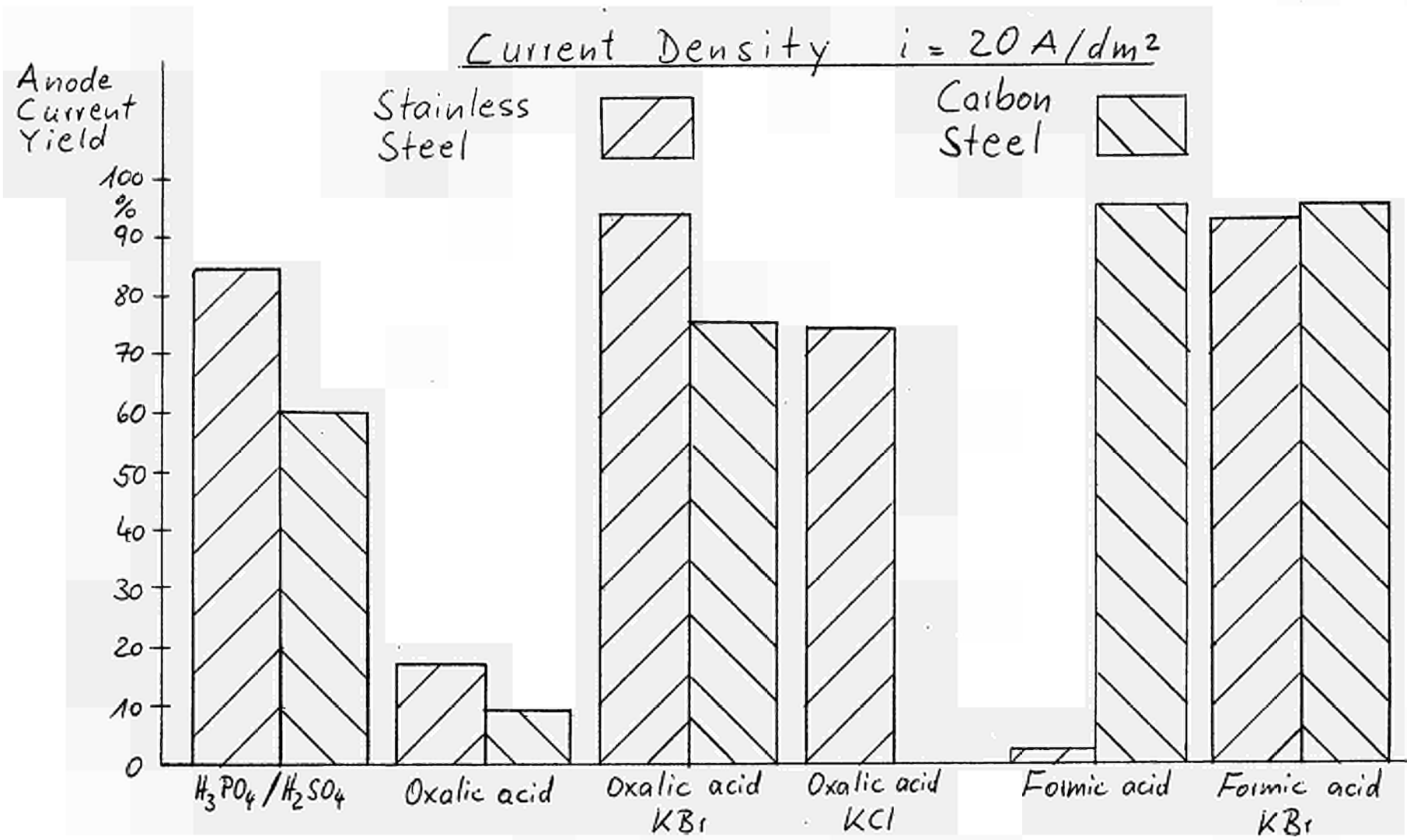


Figure 3. Comparison of electrolytes

2.5. Optimization of Filtering Systems for Various Concrete Decontamination Techniques

Contractor: Salzgitter AG, Salzgitter, Germany
Contract N°: FIID-0005
Working Period: January 1985 - June 1987
Project Leader: W. Ebeling

A. Objectives and Scope

The effectiveness of mechanical and thermal methods for the decontamination of concrete surfaces has already been demonstrated. However, the collection and conditioning of the important amount of generated dust, aerosols and toxic gases needs further development.

As concerns the filtration during thermal decontamination, multi-stage storing filters, as currently used in the nuclear industry, have shown adequate efficiency, but their limited storage capacity precludes an economic operation. Concerning the effectiveness of filtration systems for mechanical decontamination, no extensive investigations have been undertaken, so far.

The aim of this research programme is to investigate various filter systems, such as storing filters, regenerative mechanical filters, electrostatic filters, concerning their separation efficiency, their storage capacity and service life, including an analysis of the amount and size distribution of dust available at each filtering stage. The experiments will use dust generated by the above decontamination methods on non-radioactive concrete samples.

Based on existing data on radioactive concrete surfaces, a theoretical assessment on possible radioactivity inventories in the investigated filter systems will be made, with a view to their optimization for real applications.

B. Work Programme

- B.1. Modification and adaptation of the existing test facility for air filtering systems.
- B.2. Acquisition of components for testing and concrete samples.
- B.3. Selection and mounting of various air filter systems.
- B.4. Implementation of various thermal and mechanical concrete decontamination procedures (flame spalling, grinding, chipping hammer, scarifier).
- B.5. Measurement of airborne dust and aerosols by various methods.
- B.6. Analysis of the measurement records and evaluation of the tested filters with respect to separation efficiency, retention capacity, radioactivity and costs.

C. Progress of Work and Obtained Results

Summary

During the decontamination of the surfaces of the building by stripping the concrete layer by layer, dust particles and aerosols occur which must be separated as thoroughly as possible.

In order to check the varying amounts of dust released during the different stripping processes, preliminary tests were carried out on inactive concrete test pieces. The size of the grains was analysed in the extraction section behind the test cabin, and the dust loads determined gravimetrically.

During the purely mechanical extraction process (particularly when using chisel and spike hammers) we determined relatively small amounts of dust released, which limits the load for the operating personnel.

The subsequent mechanical cleaning of the flame-scarfed surfaces leads, however, to very high dust loads in the crude gas in the extraction section, so that a higher degree of filtering must be expected.

The initial tests with a regenerative type of filter having the suspended matter class "S" revealed that separation degrees of over 99% could be achieved with all stripping processes. Whenever a given pressure drop is reached across the filter, an automatic cleaning process takes place by means of compressed air. The filter was loaded and cleaned 10 times without detracting from the constant separating capacity.

1. Preliminary Tests (B.5.)

In order to check the varying amounts of dust released using the different stripping processes, preliminary tests were carried out on inactive concrete test pieces. The size of the grains was analysed in the extraction section behind the test cabin, and the dust loads determined gravimetrically.

The following tools were used for the tests:

- chisel hammer
- spike hammer
- grinding device
- flame-scarfing jet torch and wire brush for cleaning the flame-scarfed surfaces afterwards.

The suction volume flow during the tests was 3000 m³ /h.

1.1 Dust Loading of Raw Gas

The results of the gravimetric measurements are summarized in Table I. The dust load values indicated in the table are average values of several measurements.

If chisel and spike hammers are used, relatively large pieces of concrete are removed from the surface, and only a minimum amount of fine dust is released. In the case of grinding, however, it is essentially dust that is produced due to the high rotational speed of the grinding wheel. Accordingly the dust load in the suction conduit was 5 to 6 times higher than in the case of the other two, purely mechanical stripping methods.

The tests with flame scarfing jet torches and wire brush for subsequent cleaning of the flame-scarfed surface confirmed the results obtained during the previous research projects. The amount of dust released is so high that during the flame scarfing and especially during the subsequent cleaning of the flame-scarfed surface, extensive requirements of filtration and of protection for the operating personnel have to be met.

1.2 Particle Size Analyses

The particle size analyses, which were carried out with an Aerodynamic Particle Sizer APS 33, revealed very similar results for the chisel hammer and

spike hammer tests, as was to be expected, cf. Figs. 1 and 2. In both cases, the curve of the particle number concentration is relatively flat. The main fraction of the particles lies within the aerodynamic diameter range of 0.6 to 3 μm .

According to expectations, the grinding method produces a shifting of the particle size distribution towards smaller diameters (Fig. 3). The main fraction of the particles lies between 0.7 and 1.5 μm .

Flame scarfing produces a clearly defined maximum between 0.8 and 1 μm (Fig. 4), whereas brushing yields a similar particle size distribution as with grinding (due to the high rotational speed), as was to be expected, cf. Figs. 3 and 5.

The accumulated particle mass distributions, however, show in all cases that these particles play a subordinate role regarding their mass. The relatively few particles with an aerodynamic diameter of over 4 μm account for a total mass proportion of over 90%.

2. Filtering Tests (B.3. and B.4.)

After completion of the preliminary tests, regenerative filters were incorporated into the suction conduit. During the tests, the dust loads were determined both upstream and downstream of the filtering plant, and particle size analyses were carried out.

2.1 Filtering Plant

The dedusting system includes two filter elements of the suspended matter classification "S" arranged behind each other and having a cartridge form. The second filter element serves as a safety filter so that in case the main filter is damaged, no contaminated air will enter the ambient air.

As soon as a given amount of dust has been deposited on the main filter, which causes a pressure drop that can be selected freely beforehand, the cleaning system is cut in automatically. In order to clean the filter, purified compressed air is used which should also be free of water and oil. The separated dust drops through a dust funnel with butterfly valve into a dust collecting tank.

2.2 Results of Measurements

With all concrete stripping methods tried, the degrees of separation achieved with the filtering plant were appreciably higher than 99%. As was to be expected, the particle size distribution maxima in the clean gas are shifted towards smaller particle diameters, cf. Figs. 1 to 5.

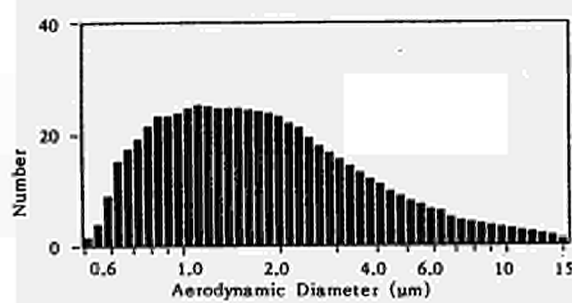
The pressure drop across the filter rises in relation to the amount of dust received. In an unloaded condition, the pressure drop across the filter is 510 Pa at a volume flow of 3000 m^3/h . The filter manufacturer recommends to start cleaning at a pressure drop equal to 980 Pa. This limit value was set at the switchboard of the plant. When this value was reached, the cleaning phase was started automatically. After the filter had been cleaned, the pressure drop was in the order of 680 Pa. The filter was loaded 10 times and cleaned automatically at a constant degree of separation.

With the particle size distributions occurring when stripping concrete, and at raw gas dust contents of approx. 50 mg/m^3 , the manufacturer guarantees that the regenerative filter will have a service life of 2000 hours.

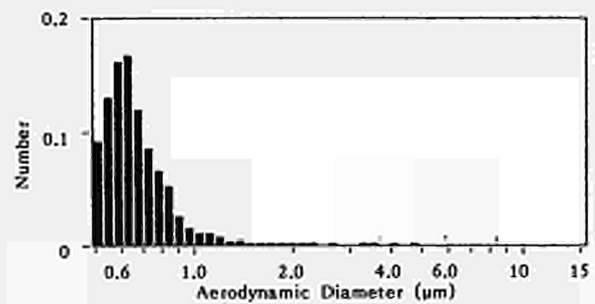
Table 1

Raw gas dust loads in the suction duct with different stripping methods

Stripping Method	Dust Loading in mg/m ³
Chisel Hammer	26,75
Spike Hammer	36,71
Grinding Device	164,5
Flame-Scarfig	110,8
Wire Brush	851,7

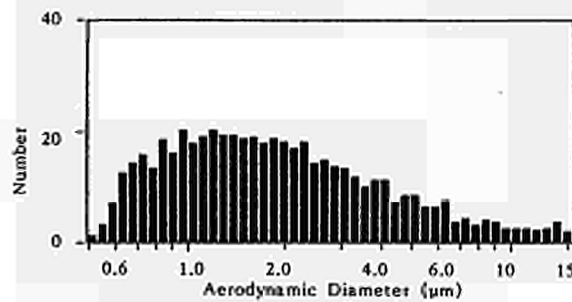


a) Raw Gas

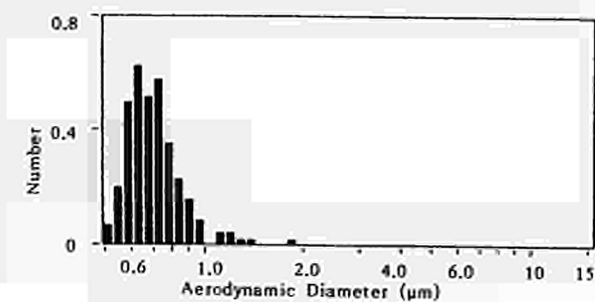


b) Clean Gas

Fig. 1 Number concentration vs. particle size when using the chisel hammer

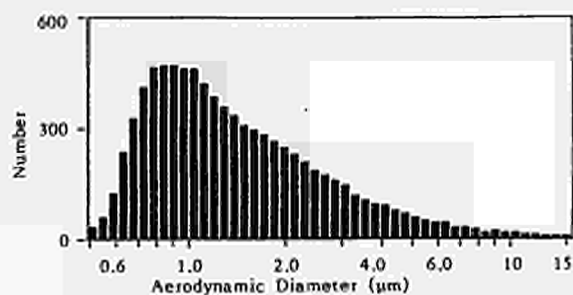


a) Raw Gas

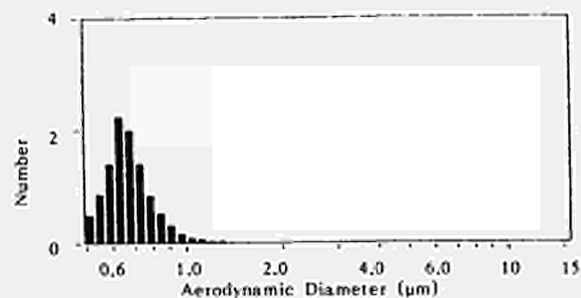


b) Clean Gas

Fig. 2 Number concentration vs. particle size when using the spike hammer

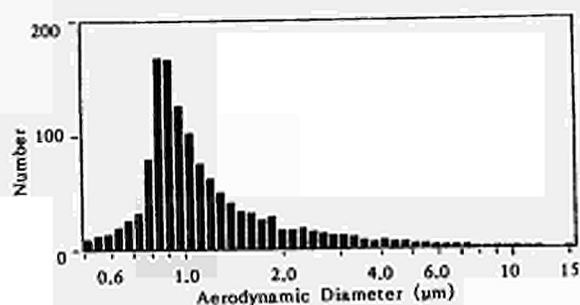


a) Raw Gas

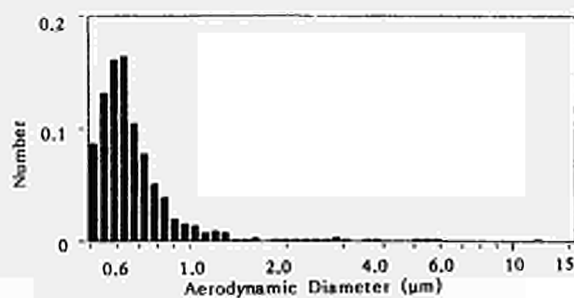


b) Clean Gas

Fig. 3 Number concentration vs. particle size when using the grinding device

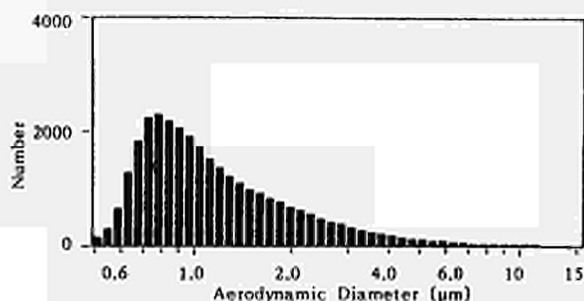


a) Raw Gas

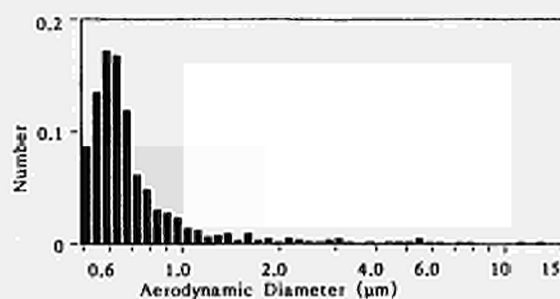


b) Clean Gas

Fig. 4 Number concentration vs. particle size when flame-scarfing



a) Raw Gas



b) Clean Gas

Fig. 5 Number concentration vs. particle size when using the wire brush

2.6. Economic Comparison of Decontamination and Direct Melting with a View to Recycling Scrap

Contractor: Gesellschaft für Nuklear-Service mbH, Essen, Germany
Contract N°: FIID-0029
Working Period: January 1985 - June 1986
Project Leader: K. G. Janberg

A. Objectives and Scope

The decommissioning of nuclear facilities either requires the final disposal of large quantities of contaminated scrap metal or the decontamination to a degree which allows its further use in nuclear or other areas.

Decontamination technology is well developed and in most cases based on the application of highly corrosive agents or electrochemical processes. Recently, direct melting has been added to these procedures as it allows for the separation of Cs and Sr from the base material. However, the volatile contamination agents have to be retained by appropriate filter systems.

The objective of this work is to carry out an economic study of decontamination, direct melting and super-compaction, with a view to recycling of scrap, in order to establish a state-of-the-art cost structure for the decommissioning of nuclear installations. This economic comparison is based on actual clean-up or decommissioning work executed by the contractor under industrial conditions.

This study takes into account the nuclear installations in Germany.

B. Work Programme

B.1. Review studies

B.1.1. Inventory of contaminated metal scrap until 1994.

B.1.2. Review of existing decontamination methods.

B.1.3. Review of licensing conditions for recycling of decontaminated metal scrap.

B.2. Assessment of the investment and running cost of the three following procedures:

- decontamination of scrap metal followed by melting and release,
- direct melting of scrap metal, followed by release,
- super-compaction followed by disposal as radioactive waste.

C. Progress of Work and Obtained Results

Summary

During this period, the inventory of contaminated scrap, the review of existing decontamination methods and licensing conditions were completed and the cost of procedures indicated in B.2. was assessed.

1. Compilation and review of basic information (B.1.)

- The normal scrap metal arisings lie between 20 and 40 tons per reactor and year. In the Germany this mass is complemented by the ones resulting from backfitting (BWR power stations) and decommissioning (KRB A, Otto Hahn, KWL, KKN, VAK, FR2, MERLIN etc.) and eventually in the Nineties also WAK.

So, the total annual scrap mass from normal operation until 1994 will lie between 4,000 and 8,000 t complemented by approximately 10-15,000 from decommissioning. It would be higher if safe long-term enclosure would not be the reference concept for most reactor buildings.

- For the chemical decontamination, there is no upper limit for its application, while electropolishing is more suitable for the lower range, say $< 100 - 200 \text{ Bq/cm}^2$ for the time being. Direct melting is licensed so far only for scrap having less than 74 Bq/g.
- In the FRG, recycling in the industrial area is limited to an activity level of 3.7 Bq/g and/or 0.37 Bq/cm^2 as far as radiation is concerned. If this application-limitation is not feasible, then the activity levels have to be lowered by a factor of ten, making control measurement rather difficult. This is the reason why recycling in form of throw-away waste casks becomes more and more interesting with the increase in the mass of scrap metal.

2. Assessment of the investment and running costs of the three following procedures (B.2.):

- Decontamination by chemical procedures, here, the cost range varies, of course, depending on the original contamination, geometry and total quantity in a range of 2.50 to 9 DM/kg.
- Direct melting, as it is applicable only under 74 Bq/kg, it cannot be compared over the whole range of contamination, but it can be said that its cost-range lies between 3 and 9 DM/kg including cutting, transport and measurement. So, there is no clear and definitive cost advantage compared to decontamination under the existing licensing conditions.
- Super-compaction and direct disposal, this is the only solution for extremely mixed scrap wastes of complicated geometry and unknown material composition. The cost ranges between 2.5 and 5.5 DM/kg. This procedure is to be ruled out for large components of homogeneous constitution.

3. PROJECT N°3:
DISMANTLING TECHNIQUES

A. Objective

The objective of this project is the development of the special techniques needed for dismantling the large steel components (e.g. reactor pressure vessel) and reinforced-concrete structures (e.g. reactor shielding) of redundant nuclear installations, account being taken of the particular requirements due to radioactivity.

B. Research performed under the 1979-83 programme

The following techniques have been tested and developed:

- thermal techniques such as plasma-arc and oxygen cutting and cutting by laser beam;
- mechanical techniques such as sawing;
- explosive techniques for the dismantling of concrete structures.

C. 1984-88 programme

The dismantling techniques needing further development should be chosen account being taken of the results of the first programme. Particular emphasis will be laid on the minimisation of secondary waste and contamination, and of occupational radiation exposure.

The necessary equipment for the remote operation of dismantling and other decommissioning techniques will be an important new aspect for investigation under the proposed programme.

D. Programme implementation

At the end of 1985, seven research contracts were at the stage of execution, and eight contracts were at the stage of negotiation.

3.1. Ventilation and Filtration Techniques for Thermal Cutting Operations

Contractor: United Kingdom Atomic Energy Authority, Windscale
Nuclear Laboratories, United Kingdom
Contract N°: FIID-0006
Working Period: October 1984 - September 1986
Project Leader: J.R. Wakefield

A. Objectives and Scope

The dismantling of nuclear plant calls for the segregation of different materials and combinations of materials. These are largely mild steels, carbon steels and stainless steels. A thermal segregation process has advantages in that it is less sensitive to material thickness and type and is more easily controlled by remote means. Its disadvantage is that it generates high concentrations of sub-micron aerosols which cause rapid plugging of absolute filters. To extend the life of these filters and to reduce the volume of secondary waste, some form of pre-filtration is necessary.

The object of this work programme is to: categorize aerosols produced by a range of thermal cutting processes; identify suitable pre-filtration devices; test them against cutting aerosol challenge; recommend a suitable filtration system which minimises secondary waste production and the man-Rem dose to operators. This work will initially be carried out in a purpose-made rig and will continue to a full scale mock-up of the Windscale AGR plant.

Co-operation with CEA Saclay (contract N° FIID-0007) will take place over the work period and will take the form of information exchange and the interchange of apparatus and personnel.

B. Work Programme

- B.1. Literature survey for identification of former work and of alternative techniques.
- B.2. General investigation into aerosol behaviour for various cutting techniques.
- B.3. Construction of a filtering rig and detailed study of various filtration systems.
- B.4. Assessment of various tested filter systems for their appropriateness in decommissioning applications with active aerosols.
- B.5. Execution of full size ventilation trials including aerosol deposition in ductings and plate out on the decommissioning machine.

C. Progress of work and obtained results

Summary

The literature survey revealed considerable documentation on aerosol production from plasma torches and arc saws, largely under water, but none at all from oxygen powder cutting. It was therefore necessary to embark on a series of ranging runs using this method in order to determine the design of the experiments. These early indications were encouraging in that it was apparent that visibility of the workpiece was not to be a problem (only 1% of kerf is airborne) and that the main occupation lay with plate-out and filtration of activated steel aerosols. This arose from the discovery that a large proportion of the aerosols produced were submicron in size (40% of total) and so would be expected to lead to rapid blocking of the standard HEPA type filters.

A number of parameters of the cutting process were selected as being likely to influence the production rate of submicron aerosols. These included powder injection flow, cutting speed, cutting oxygen flow, size of cutting flame, and type of materials being cut. The latter also included composites typical of those present in the Windscale AGR plant. The results obtained show that powder flow rate and cutting speed have the most influence and that it is possible to determine values which produce the minimum aerosol production. Furthermore, this optimised production rate is one or two orders of magnitude greater than that obtained with zero powder flow. Hence there is a strong incentive to devise methods of oxygen cutting that do not require powder injection.

These experiments also demonstrated that the electrostatic precipitator is a suitable prefiltration device but that additional development work on monitoring its efficiency and the safe removal of active dust is required.

All the above work was carried out in a small rig with a ventilation flow of $5000 \text{ m}^3 \text{ hr}^{-1}$. A considerably larger full scale reactor simulatant was constructed during the period of this review and is about to be commissioned. This will be capable of reproducing the intended ventilation flow in the Windscale AGR of $15,000 \text{ m}^3 \text{ hr}^{-1}$ and will enable the earlier results to be corroborated.

Two meetings have been held in the UK with CEA Saclay and a work plan agreed. The prefiltration module developed at Saclay will be installed in the cutting/filtration rig and presented with the range of aerosols generated as in the experiments described in 2. and 3 below. Cross comparison with earlier results will be made from experiments carried out in both facilities.

Progress and results

1. Literature survey (B.1.)

The production of aerosols from cutting metal by the application of heat depends upon the torch characteristics and the material to be cut. Studies /1/,/2/,/3/ have been made in which these variables have been evaluated. Particle size distributions were found to be in the range $0.2 \mu\text{m}$ to $0.3 \mu\text{m}$ for cutting stainless steel by plasma and oxyacetylene torch /2/. Aerosols production rates were also determined for a number of processes ranging from reciprocating saw to plasma torch /2/. Values in close agreement to these were also obtained for the oxypropane torch /3/. It has been concluded that a standard HEPA type filter will block

long before a radiation limit is reached for the types of activated steels encountered in nuclear plant /4/. Methods of improving the filter burden before blockage occurs have been suggested /5/ and followed by manufacturer /6/. In addition a form of prefiltration device will be necessary to prolong the life of the HEPA filters.

2. Investigations on aerosol behaviour (B.2.)

The work to date has concentrated on the oxypropane cutting systems with and without the addition of iron/aluminium powder. This was carried out in the rig depicted in Figure 1. The flame cutting torch and the vertical material frame are housed in a 120 m³ chamber with viewing panels along one side. Clean air enters at one end and after passing over the cutting zone is collected in ducting via a hood. The ducting is equipped with portholes for the insertion of Anderson impactors and Millipore samplers and also 150 isokinetic sampling and traversing take-offs. The prefiltration equipment (in this case an electrostatic precipitator), HEPA filters and the extract fan are mounted in series and exhaust to atmosphere.

The parameters investigated were derived from an initial list of eleven and comprise:

- Powder injection flow.
- Cutting speed.
- Cutting oxygen flow.
- Flame size (via flame oxygen flow).

These were then related in turn to different steels viz:

- Mild steel.
- Carbon steel.
- Stainless steel.
- Composites of the above.

For each material and set of conditions the following principle parameters were measured within the duct.

- Duct air velocity.
- Velocity profile.
- Particle size distribution.
- Aerosol generation rate.
- Filter differential pressure.
- Chemical analysis of particles (for some runs).

The results obtained for each run were plotted as aerosol generation, both total and submicron, against each of the selected parameters. An example of two such plots is given in Figure 2. These represent aerosol production per length of cut against cutting rate and powder flow rate. These were found to be the main parameters which govern aerosol production. These results were for mild steel of 80 mm thickness. Table 1 summarises the values obtained for this and other materials including simulants of those found in the Windscale AGR plant, for optimised cutting speeds and powder flows.

Table I

Aerosol production rate against cutting speed and powder flow
for various materials

Material	Cutting speed mm. s^{-1}	Powder flow rate g. min^{-1}	Aerosol production rate g. m^{-1} of cut
Mild steel	2.5	100	20
Stainless steel	2.5	100	60
Hotbox (mild on stainless steel)	5.0	80	23
Hoxbox (stainless on mild steel)	2.5	140	40
Pressure Vessel	1.5	140	30

This shows that the stainless steel sample produced considerably more aerosol than mild steel. The simulated hot box section produced results differing by a factor x 2 depending from which side it was cut. The aerosol production rate fell by a further order of magnitude in the absence of powder injection so it is recommended that this method should be used only when essential. In the case of the Windscale AGR plant this would be when penetrations need to be made in thick mild steel plate, when the material geometry is variable, when it is difficult to maintain a regular stand off distance and when cutting composites of materials such as the hot box and the reactor pressure vessel.

3. Preliminary trials on an electrostatic precipitator (B.3.)

Some preliminary trials have been carried out on an electrostatic precipitator on a prefilter. A particle spectrometer was used to measure concentrations upstream and downstream of the precipitator. In runs when mild steel and composite materials were cut using both iron and iron/aluminium powders it was shown that an efficiency of better than 90% was obtained for removal of particles of less than $0.5 \mu\text{m}$. Table II gives the detailed results.

Table II

Penetration of electrostatic precipitator by submicron particles from flame cutting operations

Particle size range μm	% Penetration	
	Mild steel	Composites
0.12 - 0.15	2.4	0.5
0.15 - 0.18	2.7	2.2
0.18 - 0.20	3.3	4.0
0.20 - 0.30	5.3	2.0
0.30 - 0.40	8.4	Not available
0.40 - 0.50	8.7	Not available

4. Assessment of filter systems (B.4.)

Methods of estimating the efficiency of an electrostatic precipitator by measuring the EHT voltage at the plates in under development. This will enable the precipitator to be operated for a longer period before cleaning is necessary. Also a method of remotely removing active dust from the plates is being studied. It has been shown that simple vibration of the plates is insufficient to remove all the burden but if the efficiency can be reliably measured then less than 100% removal can be tolerated.

5. Construction and commissioning of a filtration rig (B.5.)

The full sized cutting and ventilation/filtration rig has been constructed and is in the course of being commissioned. It is illustrated in Figure 3. The specimen to be cut can be mounted horizontally on vertically and the torch is deployed by an industrial robot. A full flow ventilation system of $15,000 \text{ m}^3 \text{ hr}^{-1}$ has been installed together with HEPA filters and provision for a variety of prefiltration devices. Viewing is by closed circuit TV from the control room.

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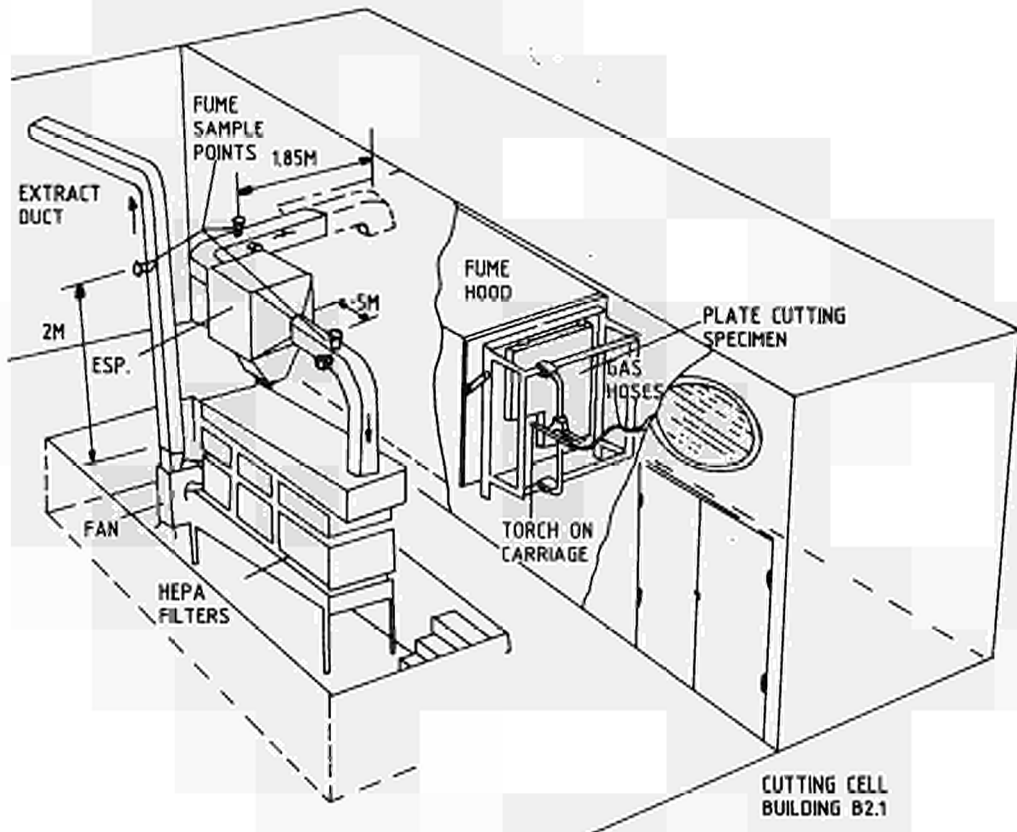


FIGURE 1 FLAME CUTTING AND FUME CHARACTERISATION RIG.

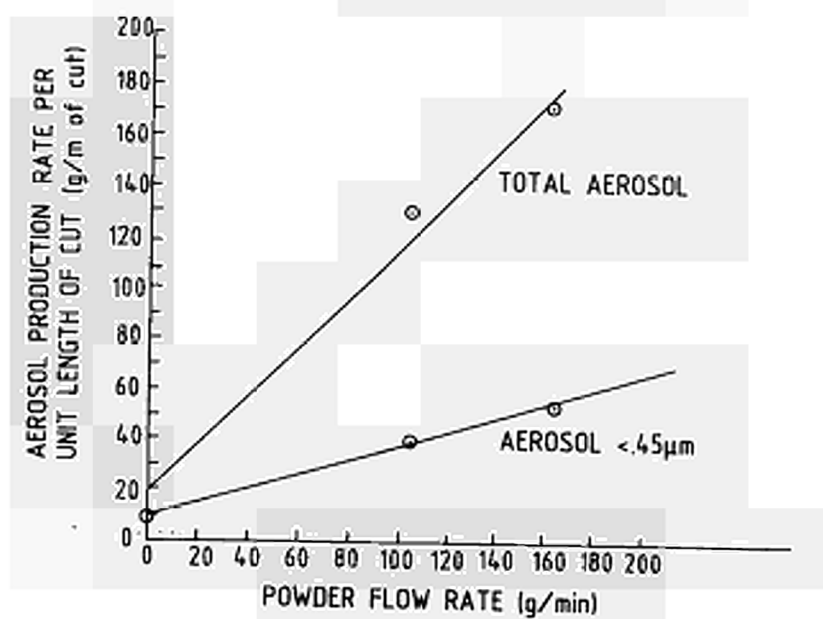
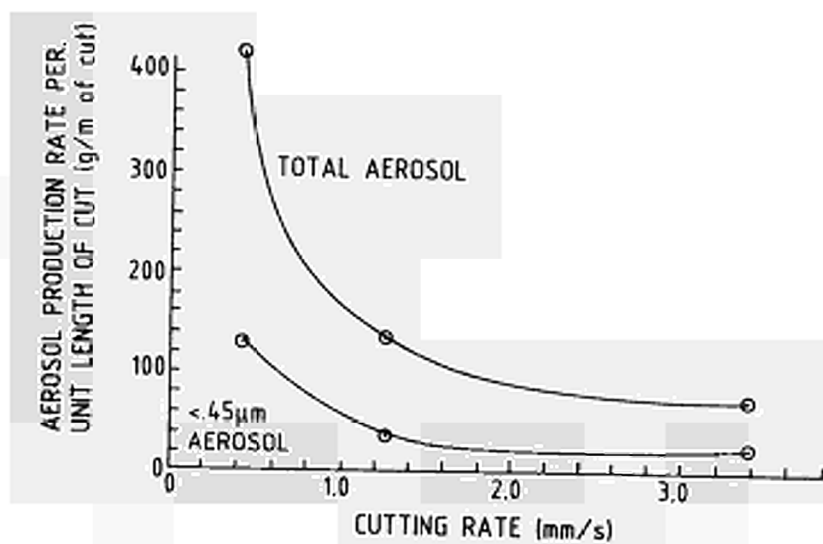
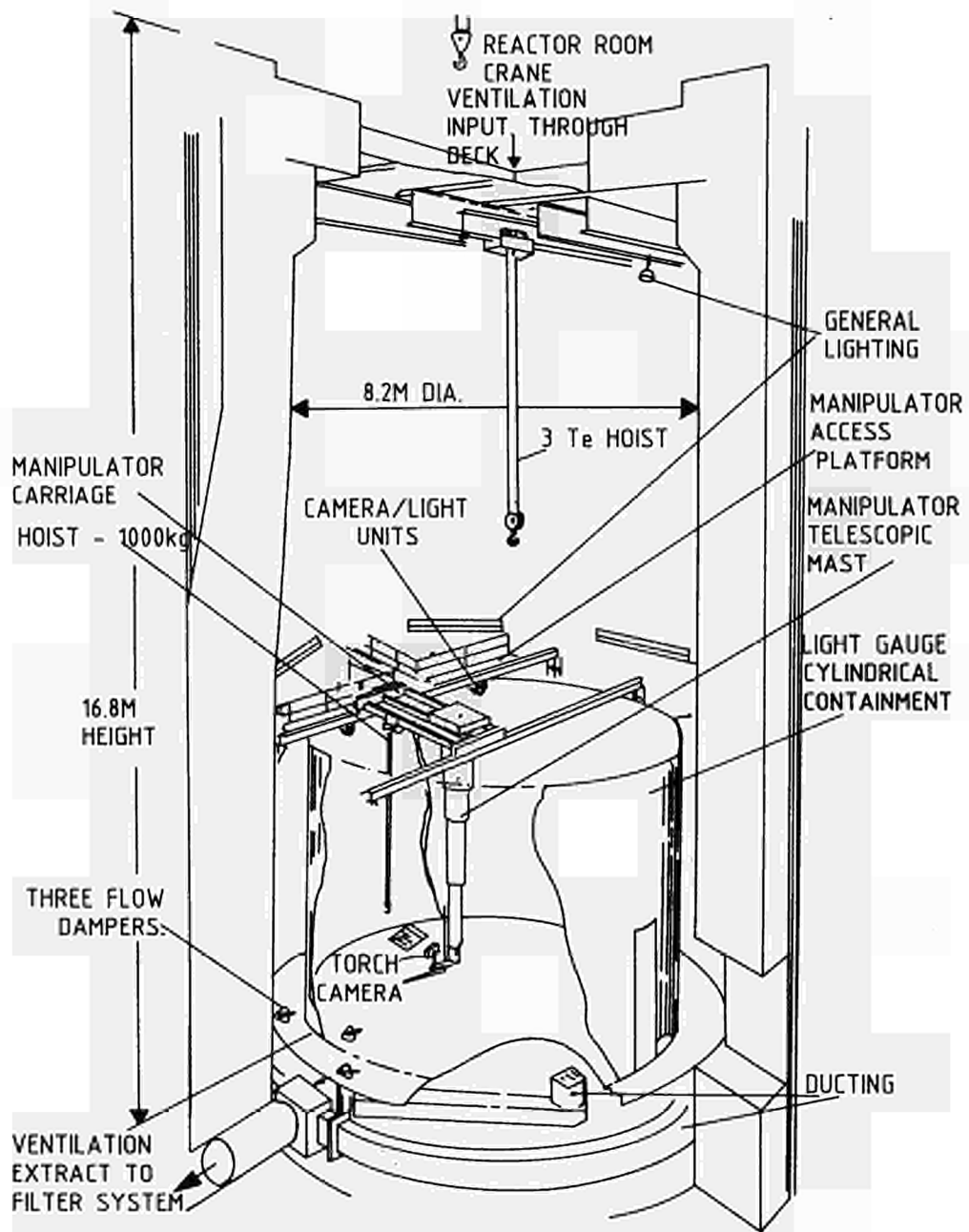


FIGURE 2 MILD STEEL 80mm THICK.



HERO DEVELOPMENT FACILITY.

FIGURE 3 FULL SIZED VENTILATION AND FILTRATION RIG.

3.2. Prefiltering Devices for Gaseous Effluents from Dismantling Operations

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FIID-0007
Working Period: January 1985 - December 1987
Project Leader: M. Pourprix

A. Objectives and Scope

Dismantling processes produce emissions of aerosols which can disseminate contamination in the cell where the cutting operation takes place, and in the ventilation ducts up to the HEPA filters, the last barrier before releases into the environment. Cutting processes, and mainly thermal ones, cause rapid plugging of HEPA filters because of the high concentrations of ultrafine particles produced. To increase the life of HEPA filters and thus to reduce the amount of solid wastes, an efficient cleanable prefiltering device is necessary.

The object of this work is to: categorize the aerosols produced by various cutting techniques, identify the possible captation and prefiltration devices, select them in a reduced-size mockup, evaluate the selected ones on an experimental rig and then use them on an actual dismantling site.

This survey will be done in co-operation with UKAEA Windscale (contract n° FIID-0006 UK).

B. Work Programme

B.1. Collection of data on aerosols and filters associated to various metal cutting techniques and complementary experimental studies on ultrafine particles.

B.2. Design and testing of various aerosol captation devices at the aerosol generating source.

B.3. Design, testing and final selection of various pre-filtration devices in a down-scaled test section.

B.4. Evaluation of a selected prefiltering system in a full scale test section with real cutting effluents.

B.5. Final assessment of selected captation and prefiltration devices by application to radioactive aerosol sources in a dismantling facility.

C. Progress of work and obtained results

Summary

The choice of prefiltration and captation devices require aerosol characterization. A literature survey and our experiments showed that :

- thermal methods like oxy-acetylene cutting torch and plasma torch generate more important rates, mainly in the range of submicronic particles, than mechanical methods like various saws or grinder,
- particles production rate and size are dependent not only on the cutting tool but also on many other parameters like for instance material nature and cutting environment.

To choose a cleaning device, we use a down scaled test mockup with a flow of 3 m³/h and an optimized running. Some prefilters calibrations were made on this mockup with monodisperse aerosols but because co-operative UKAEA experiment parameters, we decide to define a full scale test rig at once. This rig will include a cyclone convertible into electrocyclone, an electrostatic precipitator and a bags filter.

Progress and results

1. Collection of data on aerosols and filters associated to various metal cutting techniques and complementary experimental studies on ultrafine particles. (B.1.)

A literature survey and personal experiments on the secondary emissions produced by mechanical and thermal techniques on metallic structures have been carried out. Mechanical cutting tools evaluated include disc saw, reciprocating saw, band saw, chop saw and grinding wheels. Thermal cutting techniques include oxy-acetylene torch and plasma torch.

We can notice that :

- the data are not numerous
- the thermal cutting techniques produce higher aerosols rates by unit of time and even by unit of cut length (in spite of their rapidity) than mechanical ones. For example, by the cut of 2 inch schedule 40 type 304L stainless steel, aerosol production rate range from 50 mg/cut for the reciprocating saw to about 1000 mg/cut for plasma torch (fig. 1)
- all devices create aerosols in the respirable size range (less than 10 µm aerodynamic diameter). Our experiments and literature survey (table I, fig. 2 and 3) show that the mass median aerodynamic diameter range from 1 to 10 µm for mechanical cutting tools (reciprocating saw, band saw, grinder) and from 0.1 to 0.5 µm for thermal ones (oxy-acetylene torch and plasma torch)
- the aerosol production rate and distribution in size are dependent upon :
 - . the cut plate nature (production is higher for example for stainless steel alone than for carbon steel alone), its cleanness degree (with or without grease for example), its thickness and its impurities
 - . the cutting tool of course and for a same cutting tool upon its power, its speed and eventually upon the choice of the gases
 - . the cutting environment (underwater plasma arc cutting produce a lower aerosol rate at least 30 times than air plasma cutting and the mass median aerodynamic diameter is smaller), the underwater depth (fig. 4 and 5) and evidently upon the sampling point (we point out, for example, that distribution in size are different at the source, near the operator's nose and in the ventilation duct).

A comparison between three tools (grinder, oxy-acetylene torch and plasma torch) on a dismantling site (photos 1 and 2) to cut carbon steel pipes showed what is the most appropriate for this work and that keeping a suction effect in the tube is an efficient captation at the source especially for submicronic particles (fig. 6).

All our experiments emphasized the interest to collect the aerosol at the generating source and to insert prefiltration devices upstream the HEPA filters which do not have particularly a high efficiency for thermal cutting generating aerosol (HEPA filter efficiency is minimum for about 0.1 μm).

2. Design, testing and selection of various pre-filtration devices (B.3.)

First, we intend to use a down-scaled test mockup with a flow of 3 m^3/h and three down-scaled prefilters have been calibrated with monodisperse aerosols (working step B.3) but in our co-operative experiment with UKAEA, it would lead to very long time cutting.

So we designed a full scale test rig at once. This rig will include a cyclone convertible into electrocyclone, an electrostatic precipitator, a bags filter, HEPA filters, isokinetic samplings, a diaphragm and a fan (working step B.4).

The first two prefilters have recently been qualified in the frame of other uses.

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Table I (extracted from / 1 /)

Aerosol particle size distribution and cutting time for metal cutting tools used on 2" schedule 40 type 304L stainless steel pipe

Tool	Range of MMAD ^a (μm)	Geometric standard deviation	Cutting Time (min)	
			Range	Mean
Reciprocating saw	bimodal	N.A.	1.8 - 4.5	3.1
Band saw	1 - 5	2.3	0.9 - 1.9	1.5
Side arm grinder	bimodal	N.A.	N.A.	N.A.
Chop saw	1.5 - 9.0	4.6	1.1 - 1.7	1.5
Oxy-acetylene torch	0.1 - 0.3	2.3	3.0 - 4.2	3.6
Plasma torch	0.2 - 0.3	3.7	0.5 - 0.7	0.6

^aMass Median Aerodynamic Diameter by cascade impaction
N.A = not applicable

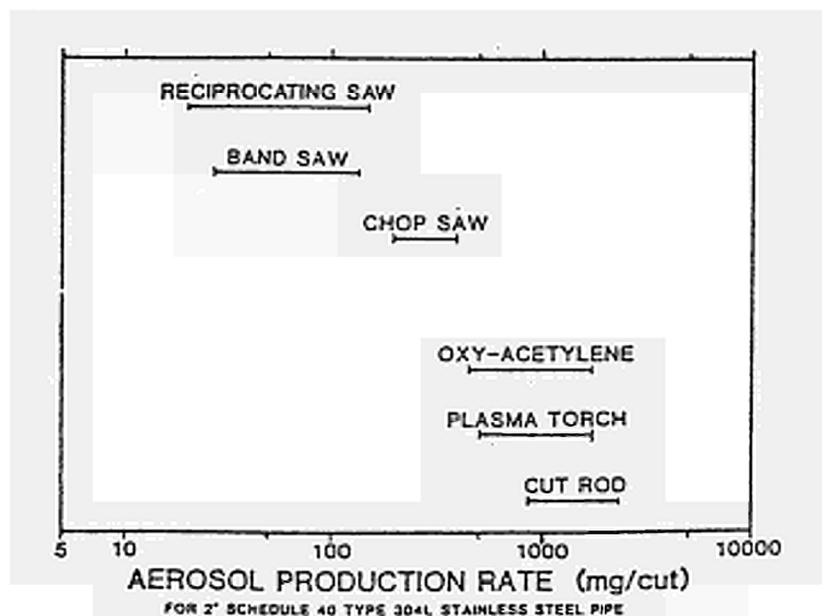


Fig. 1. Aerosol production rates in milligrams per cut with the various tools used on 2-inch Schedule 40, Type 304L stainless steel pipe (extracted from / 1 /)

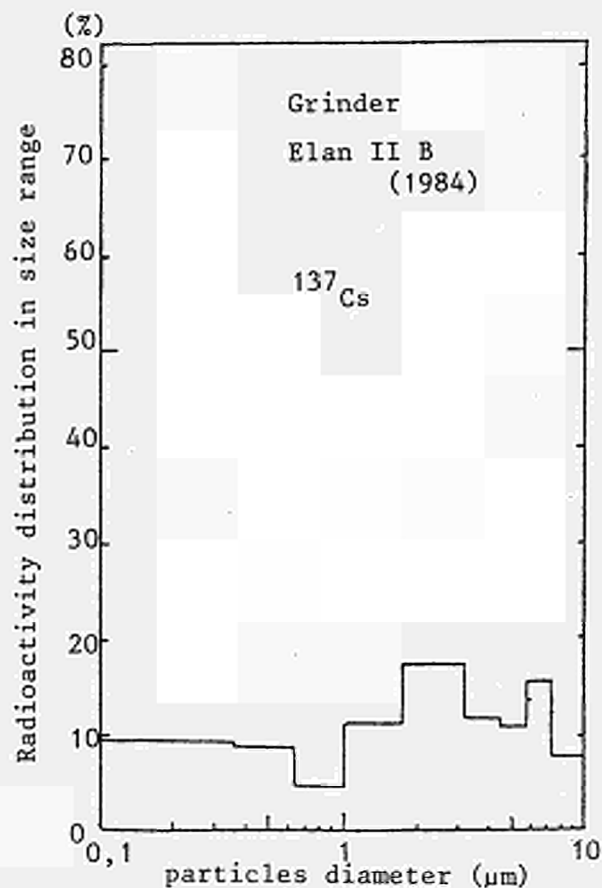


Fig. 2. Radioactivity distribution in particle size range in cutting stainless steel using grinder (extracted from / 2 /).

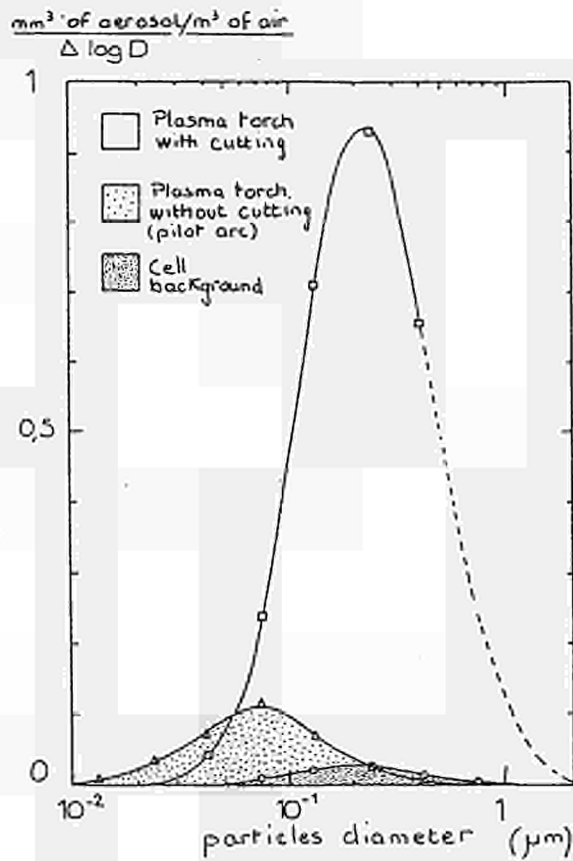


Fig. 3. Aerosol volume per air volume distribution produced by plasma torch with and without cutting in air used on stainless steel plate (extracted from / 2 /)

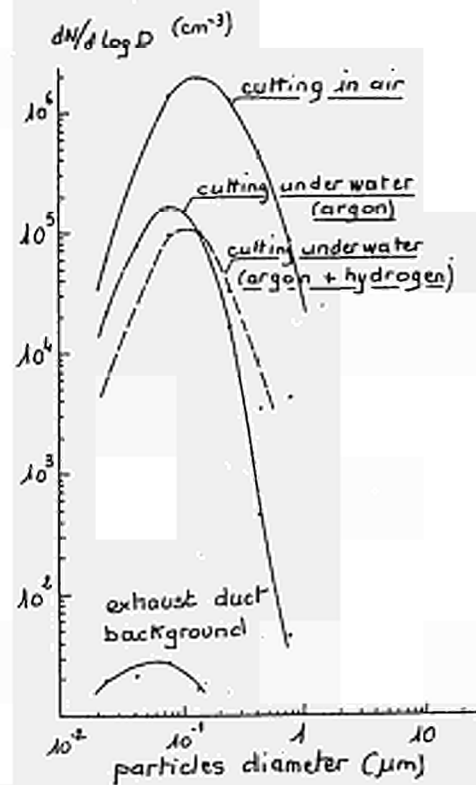


Fig. 4. Numeric particle concentration distribution with plasma torch cutting in air and underwater used on stainless steel plate (extracted from / 3 /)

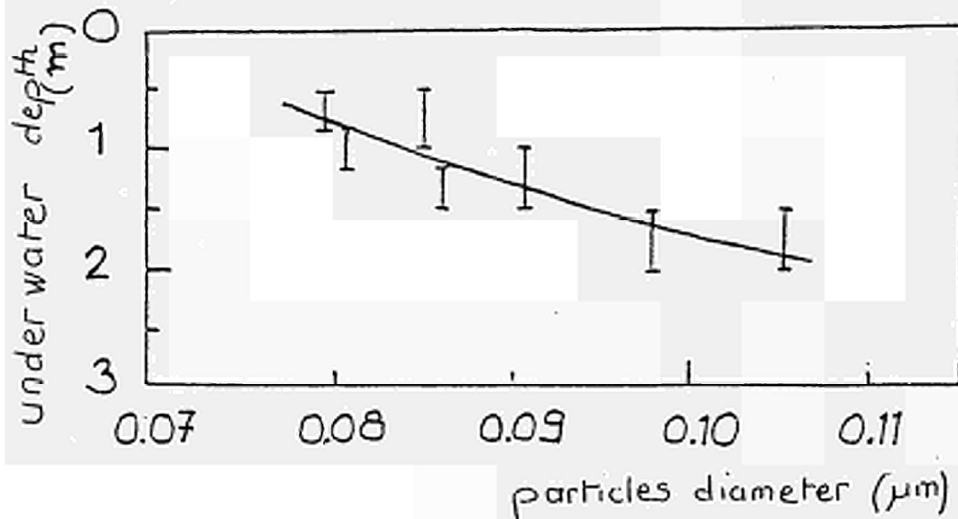


Fig. 5. Particles median diameter variation versus underwater cutting depth (extracted from / 3 /)

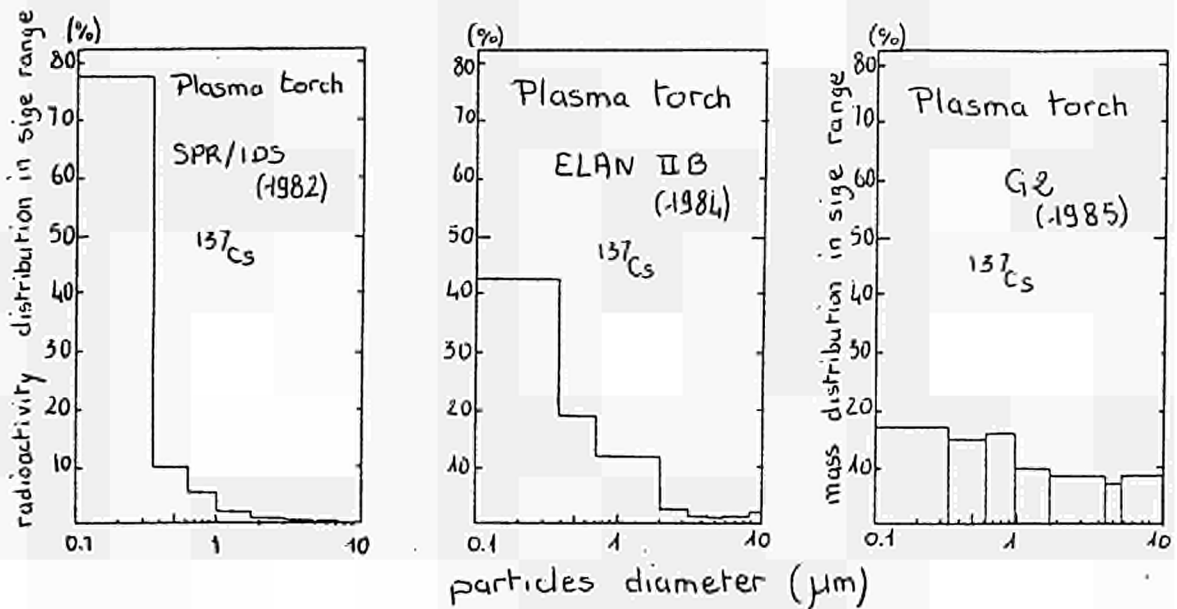


Fig. 6. Aerosol mass (or radioactivity) distribution with plasma torch cutting in air in three different cells. In the third case (fig. on the right), a suction effect was maintained in the cut tube. (extracted from / 4 /)



Photo 1 - Carbon steel tube cutting by oxy-acetylene torch (extracted from / 4 /)

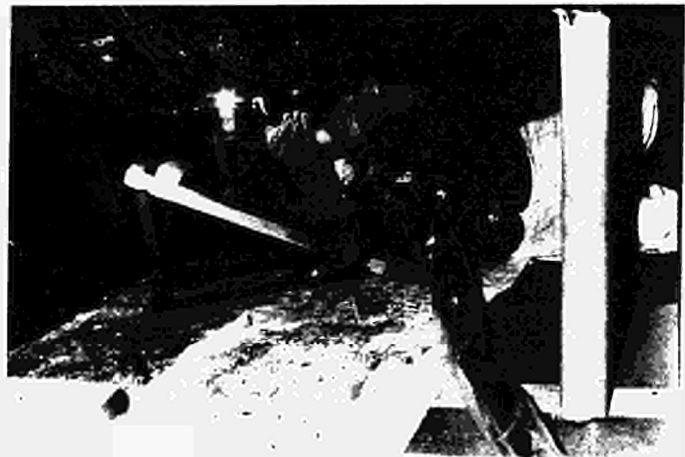


Photo 2 - Carbon steel tube cutting by plasma torch (extracted from / 4 /)

3.3. Dross and Ultrafine Particulate Formation in Underwater Plasma-arc Cutting

Contractor: Heriot-Watt University, Edinburgh, United Kingdom
Contract N°: FI1D-0008
Working Period: January 1985 - March 1987
Project Leader: B. Waldie

A. Objectives and Scope

Underwater plasma-arc cutting is a useful technique for dismantling but produces dross and ultrafine fume particles which must be collected. The overall project aim is to improve understanding of the factors governing formation rates and characteristics of dross and ultrafine fume particles so that these by-products can be better controlled during dismantling.

The research is predominantly experimental, with supporting theoretical work on fluid dynamics of dross behaviour and on formation and behaviour of the ultrafine fume particles. Metal samples to be cut are non-active, the aim being to characterise the basic mechanisms which should be valid for active and non-active metals. Cutting is done in a hyperbaric chamber with simulated water depth up to 10 metres. The former vessel allows the influence of pressure alone to be studied, and the latter, the combined effects of pressure and a water column. Part of the programme involves the development of techniques for collecting and characterising dispersed dross and ultrafine particulates.

B. Work Programme

- B.1. Updated literature review and analyse of data on secondary waste (dross, ultrafine fume particles) generated during underwater plasma-arc cutting of steel.
- B.2. Design and construction of a dross collection system, appropriate for underwater cutting.
- B.3. Design and construction of a collection device for ultrafine fume particles appropriate to underwater cutting.
- B.4. Development of TV and/or photographic techniques for underwater monitoring of the behaviour of cutting waste.
- B.5. Tests on cutting of non-active stainless steel samples in hyperbaric flooded test chambers, with monitoring of dross and ultrafine fume characteristics under various cutting parameters (cutting vertically upwards).
- B.6. Idem B.5., with cutting vertically downwards.
- B.7. Idem B.5., with cutting in horizontal position.
- B.8. Design and construction of a test vessel providing a 10m water depth and monitoring/sampling devices for cutting waste.
- B.9. Cutting tests in the facility developed in B.8. with cutting parameters selected in B.5. to B.7.
- B.10. Analysis of the surface layer material behaviour by trace and compound workpiece techniques.
- B.11. Conclusive assessment of obtained results.

C. Progress of Work and Obtained Results

Summary

An experimental facility has been set up for studying dross and ultrafine fume particle formation in underwater plasma cutting. Cutting is carried out in a flooded 0.66m diameter hyperbaric vessel. Some characteristics of dispersed dross have been determined for horizontal cutting. Formation rates of finer particles suspended in the water and the ultrafine fume particle content of the effluent gas have been measured for low immersion depths. An initial theoretical model for transfer of particles from gas bubbles to water has been developed. Equipment to allow studies at immersion depths to 10 metres is under construction.

Progress and Results

1. Literature Review (B.1.)

Previous research on underwater plasma cutting most relevant to this project is described in five reports - three from Europe/1,4,5/ and two from Japan/2,3/. These five reports deal in particular with fume characteristics and dispersed dross/1,2,3/ as well as other aspects of the process. Topics to be studied here, such as the effect of pressure and water depth on fume and dross formation, the distribution of dross and the transfer of fume particles from gas to water were not described.

2. Experimental Equipment and Techniques (B.2./B.3.)

A 0.66m diameter flooded hyperbaric chamber has been set up for cutting studies B5, B6 and B7. This is shown diagrammatically in Figure 1. The torch moves and the workpiece is stationary. Visual recording of cutting phenomena has been done by still and cine-photography and T.V. through a viewing port (B.4), the former two being preferred.

Ultrafine fume particles in gases leaving the water are collected continuously by filtration (B.3). Sampling of fume particles in the gas below water has not been feasible. Dross particles down to 10 μm are collected (B.2) along the length of a cut as in Figure 1. Fine suspended particles are collected by filtration of the batch of water after a run.

For studies of the effect of water depth, a second rig involving a 0.6m diameter x 12m high cylindrical vessel (Figure 2) has been designed and is under construction (B.8). When completed this will allow sampling of particles in gas at different depths to follow transfer of particles from gas to water.

3. Dross and Suspended Particles in Water (B.5)

Dispersed dross particles from horizontal cutting (B.5) of 304 stainless steel have been characterized as regards yield and particle size distribution. Examples of the form of the particle size distribution are given in Figure 3. The particle size distribution, at least for one thickness (12mm), does not vary markedly with current and cut speed. Present sizes are generally intermediate between those from two previous studies/1,3/. Conditions promoting dross attachment are still under study.

Suspended particles, of much smaller size than the above dispersed dross, represent around 0.2% wt. of total metal removed in cutting experiments to date. Microscopic examination suggests that these comprise both fume particles and very small spheroids formed from

molten dross.

Further work on dross and suspended particle characteristics is to be done under items B.5, B.6 and B.7.

4. Ultrafine Fume Particles

Research on this aspect has included measurement of ultrafine fume particle concentrations by filtration of effluent gas and initial development of a theoretical model of the transfer of such particles from rising gas bubbles to the water. Variables in the model include particle size, bubble diameter and water depth. Cutting so far has been in the horizontal (B.7). Particle content of the effluent gas from cutting with 150mm immersion depth was then equivalent to 0.01% wt. of total metal removed. The influence of water depth is to be studied in the column rig (B.9) and results compared with model predictions.

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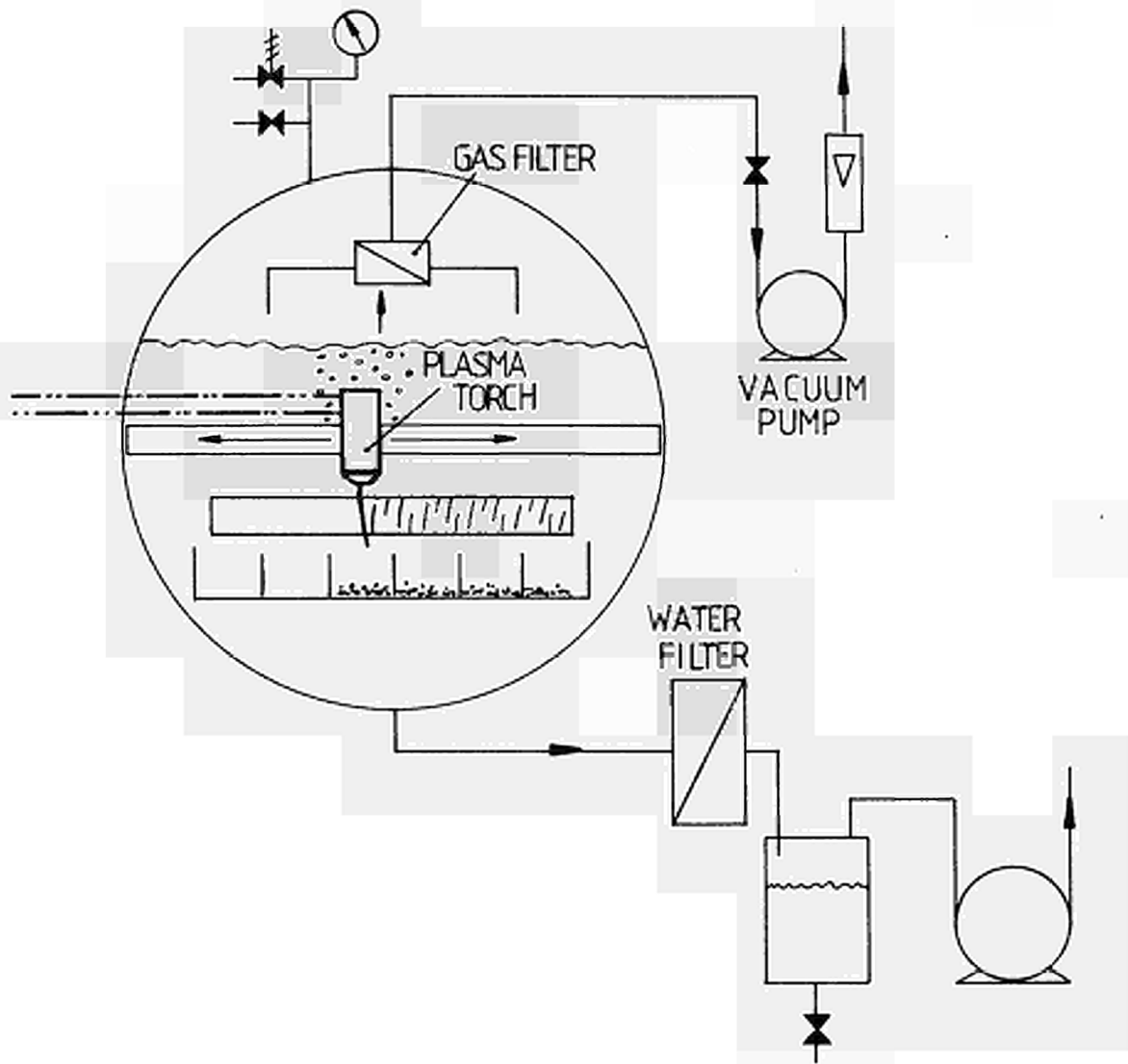


Fig. 1. Schematic of underwater plasma-arc cutting in hyperbaric chamber.

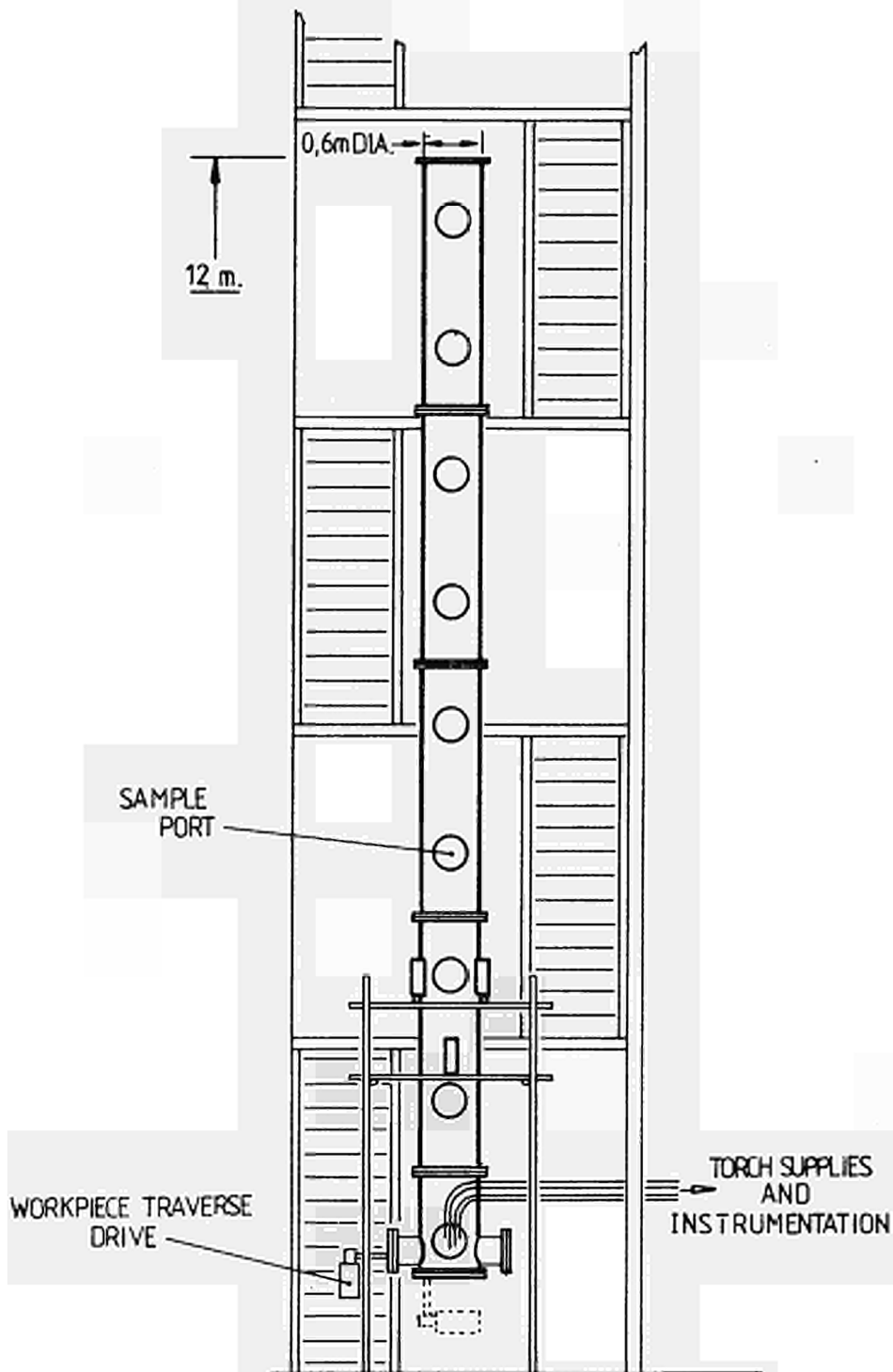


Fig. 2. Schematic of proposed rig for cutting in water depths to 10m.

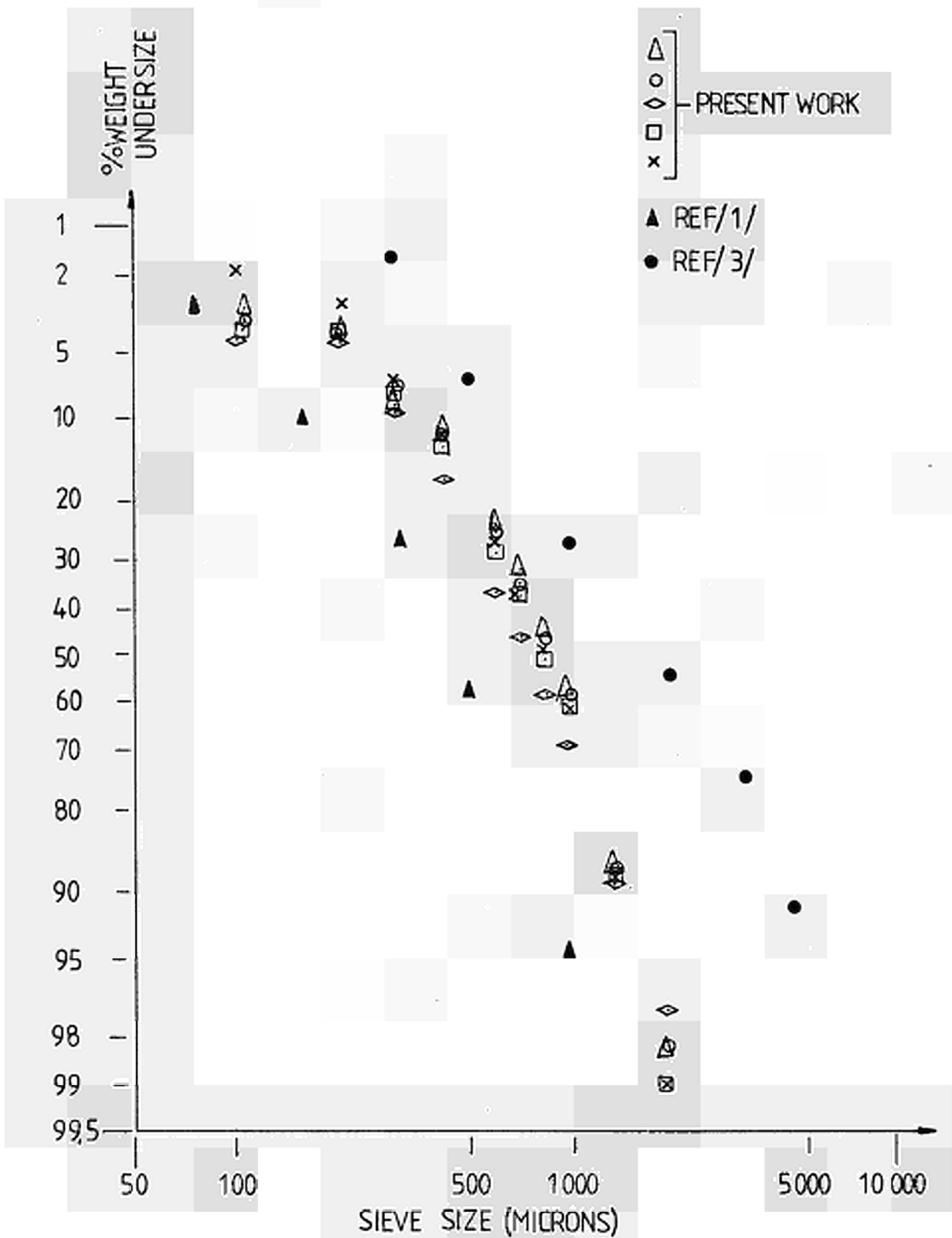


Fig. 3. Particle size distribution of dispersed dross from underwater plasma-arc cutting of stainless steel.

3.4. In-situ Arc-saw Cutting of Heat Exchanger Tubes and of Pipes from the Inside

Contractor: Field Automation, Paris, France
Contract N°: FIID-0009
Working Period: January 1985 - December 1986
Project Leader: P. Thomé

A. Objectives and Scope

The principle of underwater metal cutting by electric arc saw presents some similarities with the arc gouging process and electrode arc cutting. Besides its numerous other advantages as high precision work and small production of cutting waste, this method is especially appropriate for telemanipulation by robots; particularly because of small induced vibrations and cutting forces involved in this process, and by the possibility to use small dimension cutting discs allowing for high accessibility to complex areas.

The present work is mainly aimed at an adaptation of this procedure to in-situ cutting by robots, especially inside of tubes and pipes, with a special objective to dismantling steam generators and other heat exchangers in nuclear installations.

This development comprises design studies for apparatus to be adapted on special crawler or robot arms, laboratory studies of the cutting parameters, miniaturization of cutting heads for their introduction into small diameter ducts.

B. Work Programme

- B.1. Development of methods and tools for internal arc-saw cutting of steam generator (S.G.) tubes (internal diameter of about 19 mm).
 - B.1.1. Design, construction and testing of a miniaturized cutting tool for PWR steam generator channel head.
 - B.1.2. Development and construction of a laboratory testing bench to check process characteristics by external and internal cutting.
 - B.1.3. Design and fabrication of a complete device for internal cutting, based on test results obtained under B.1.1. and complying with in-situ working limitations.
 - B.1.4. Execution of performance tests on representative Inconel tubes (under water).
 - B.1.5. Final assessment on realistic full-scale samples supplied by EdF.
- B.2. Development of methods and tools for internal arc-saw cutting of pipes (internal diameter about 200 mm).
 - B.2.1. Design of a suitable crawler for hoisting the cutting tool.
 - B.2.2. Fabrication of an appropriate cutting tool and laboratory tests to determine the cutting parameters in accordance to the pipe diameter and thickness and to the tool working limitations.
 - B.2.3. Testing of the cutting tool.

C. Progress of work and obtained results

Summary

During the reporting period, the research work has covered following parts of the work programme:

- Study and fabrication of a simple testing bench for a preliminary evaluation of the mechanical and electrical parameters suiting the process, for cutting Inconel and stainless steel tubes (B.1.1., B.1.2.);
- Designing of an "in-situ" arc-saw tool for steam generators (B.1.3.);
- Designing and building of a new general-purpose laboratory bench with accurate movements and appropriate instrumentation;
- Designing of a self-propelled equipment (crawler with arc-saw) for pipes (B.2.1., B.2.2.);
- Starting of cutting parameter studies (B.1.2.).

The internal cutting head for small tubes requires more extensive work than originally contemplated. Main difficulties arise from the miniaturization. A new cutting head for small tubes has been designed.

The laboratory tests confirm the expected performances of the method: high cutting velocity, no-distorsion of the work piece, negligible cutting forces. More than hundred cuts have been performed during this period with external and internal heads.

As concerns the arc-saw crawler, a modular equipment has been designed for pipes with internal diameters ranging from 245 to 335 mm, and a 20mm thickness. This self-propelled crawler should be suitable for travelling in any position with a mean curvature radius above 980mm.

Progress and Results

1. Study and fabrication of a preliminary testing bench for cutting S.G. tubes (B.1.1., B.1.2.)

This simple bench was used for a very first evaluation of the basic parameters:

During the tests, the rotating disc electrode is immersed in a tank, and comes on a tube which is manually positioned. The position of the work-piece allows either internal or external tests. The adjustments of the cutting depth and the cutting speed are manual, the disc electrode is driven by a pneumatic motor.

The tests were made on Inconel 600 tubes ϕ 22,22mm, 1,27mm thick. The cuts were made under water on the external surface of the tubes in order to visualize the results.

The disc has a power supply delivering 100 Amps under 20 Volts, with regulation by thyristors and 50Hz frequency. The pneumatic motordrive can go up to 18000rpm, the electrical supply being made through a slip-ring allowing a maximum speed around 5000rpm.

During the various tests, the following observations were made:

- The water turns rapidly opaque by the presence of dross. A continuous flood of water is necessary for visual observation.
- The wearing of the disc is noticeable, mainly because of sticking due to short-circuiting when the disc contacts the tube; this results from the poor conditions of experimentation (manual adjustments of the workpiece).
- The rotative speed of the disc must be as high as possible (between 3000 and 4000rpm), if the depth penetration and the cutting speed are to be modulated.

- The arc-saw cutting seems to be significantly faster than other cutting techniques. The generator's intensity curve draw peaks at values between 20 and 70Amps.

2. Study and fabrication of a general-purpose testing bench

This development is an extension of the facility described in chapter 1. It is intended for investigations with high precision, of various cutting parameters as rotation velocity of the disc, wear of the disc, as well as optimizing of electrical parameters.

The bench is mainly composed of a water tank and 2 mobile tables for holding the sample and the working tool. Various cutting heads and electrodes can be used on this bench for cutting tubes, pipes and plates.

3. Design of an internal cutting tool for steam generator tubes (B.1.3.)

During the reporting period, only the design of the device for internal cutting of steam generator tubes was completed. This design should have the following characteristics:

- a) mechanical:
 - rotating speed of the disc (about 3000rpm)
 - rotating speed of the head holding the disc (advance: about 10rpm)
 - adjustment of the depth of the cut (penetration)
- b) electrical:
 - amp-values
 - driving motors
- c) miscellaneous:
 - water circuit, dross removal.

The transmission "spear" comprises three coaxial tubes:

- a) the central tube for the rotation of the disc, the power supply and the cooling water,
- b) an intermediate insulated tube for the depth of the cut,
- c) the external tube for the advance of the tool.

The disc electrode is held by a spring secured on the axial tube, which provides for the eccentrication (depth) of the disc.

The depth and the rotating speed of the tooling head is motorized by a single motor, in such a way that for each turn of the head (advance), the depth (eccentrication of the head) is increased by 0.3mm. The maximum eccentrication is 2.4mm corresponding to 8 turns of the head.

4. Design of an "in-situ" arc-saw crawler for pipes with a diameter in the range of 250 to 350mm (B.2.1.)

The design should meet various conditions, such as: incorporated driving device (automotor), capability to ascend and move inside curved ducts, blockage of the working tool during its operation, allowing for use in pipes with different internal diameters and curvature ratios. The crawler is composed by various elements connected by articulations: a front trolley bearing the working tool, a driving element ("actionneur") assuring a step-wise progression inside the duct, and an end trolley for guidance. This crawler is designed to work under water. The final design meets following specifications:

- piping:
 - . pipe curvature radius: 980mm minimum
 - . pipe thickness: 20mm maximum
 - . pipe internal diameter: 245/335mm

- crawler:
 - . self-moving crawler (push-pull type)
 - . displacement speed: 400mm/min
 - . climbing capability for vertical pipes
- arc-saw:
 - . electrode diameter: 200mm
 - . amps: 600A; 1500A with water-cooled cables
 - . rotating speed: up to 3000 rpm
- controlled parameters:
 - . electrode rotating speed
 - . kerf penetration speed (in-feed)
 - . traverse feed.

3.5. Electrochemical Technique for the Segmenting of Activated Steel Components

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen,
Germany
Contract N°: FIID-0010
Working Period: January 1985 - June 1986
Project Leader: W. Stang

A. Objectives and Scope

Electrochemical decontamination has a great importance during the decommissioning works at KRB-A. By this method a thin metal surface layer is removed due to a galvanic process in an electrolytic solution. Using the same principle it is also possible to remove material locally (ECM-technique).

Many advantages of this method indicate that it could be used for disassembling activated components during decommissioning of nuclear power plants. In order to investigate its applicability, experiments with non-active materials from a reactor pressure vessel are carried out.

In the research programme it will be established:

- which cathodes are most suitable for high cutting velocities,
- which amount of sludge (waste) is produced in the electrolyte.

The work in this contract will assess whether electrochemical cutting of activated parts of the KRB-A reactor pressure vessel is a technically useful, low-cost and low radioactive dose procedure.

The experiments are carried out in an existing test facility of AEG-Elotherm in Remscheid.

B. Work Programme

B.1. Modification of an existing test facility for the testing of static and dynamic cathodes.

B.2. Implementation of parametric studies and of the main test programme on various non-active representative steel plates.

B.3. Evaluation of the obtained results and elaboration of recommendations for a possible application to radioactive components.

C. Progress of work and obtained results

Summary

The investigations in the field of the electrochemical cutting techniques were carried out with 8 inactive samples equivalent to the material of the reactor pressure vessel. Three difficult operating methods were tested.

The static method led to an insufficient result because of considerable passivation of basic metal.

When using a cathode with a lateral effective edge the interval method worked without problems.

The continuous method was sensitive against changes of the operating parameters. It was found that the interval procedure is the best suited EC-method for separating active steel components.

Progress and results

1. Present status of the test facility (B.1.)

The facility for the electrochemical cutting of typical materials of the reactor pressure vessel consists of a cathode that is either static or can be moved stepwise or continuously towards the material. The space between the material and the cathode is filled with a salt solution that circulates in a closed circuit. Under the influence of a direct current the material dissolves. The produced hydroxides are separated from the liquid by sedimentation or by centrifugation and compacted in filter presses. A rectifier supplies the required current. The trial tooling (see Fig. 1) consists of a cathode carrier mounted at a brass top plate. The brass electrode fitted at the cathode carrier is coated with a plastic layer. The cathode guide made out of fiber-glass reinforced epoxy resin includes the electrolyte inlets and outlets.

The lower part of the trial tooling is shaped as a frame which is mounted at a basic plate. The frame which is bolted to the cathode carrier keeps the component to be cut.

2. Execution of parametric tests (B.2.)

Eight inactive workpieces with the dimensions 210x107x143 mm were used for the investigations. The workpieces consisted of the basic material 22 NiMoCr 37 with a 6 - 7 mm thick welded surface of material 1.4556 (stainless steel).

The following ECM operating methods were tested:

- static tests
- interval tests
- dynamic tests

The following parameters were varied:

- voltage
- amperage
- concentration of the electrolyte
- kind of cathode
- cutting velocity

3. Implementation of the main test programme (B.3.)

At first the series of static tests were carried out. For this purpose a cathode was fixed in a distance of 0,3 mm from the workpiece.

Then a NaNO_3 -solution was pumped continuously through the work gap cathode - workpiece. After a voltage was fed to cathode and workpiece a certain reaction time ran down. It was established that the welded surface material could be removed by the static method practically without any problems. When the boundary layer basic material - welded surface was reached, considerable passivations of workpiece and cathode occurred so that the process broke down. Therefore the static material removal by ECM is not a suitable process for separating active steel components.

During the second test series the interval method was applied. In this case the cathode was moved towards the workpiece by 0,2 mm every time after a certain reaction time (removal time) had run down. This process was carried out via an automatic program.

When using cathodes with lateral effective edge the process ran down without any problem.

When using a cathode without lateral effective edge the process broke down because of short circuits. The interval method is a conceivable process for separating active steel components, and this method should be taken into consideration because of the results achieved with inactive steel components.

The third test series covered the dynamic process. In this case the cathode was continuously moved towards the workpiece.

It appeared that this process behaved sensitively against changes of electrolyte pressure and voltage. Also this process is conceivable for separating active steel components.

To sum up, the interval process is the best suited EC-method for separating active steel components, since this method offers the best process stability.

The separation of steel components results in a sludge quantity weighing 1.9 times of the weight of the removed material quantity, when dry.

When a filter press is used, the wet sludge weighs 9,5 times of the weight of the removed material quantity.

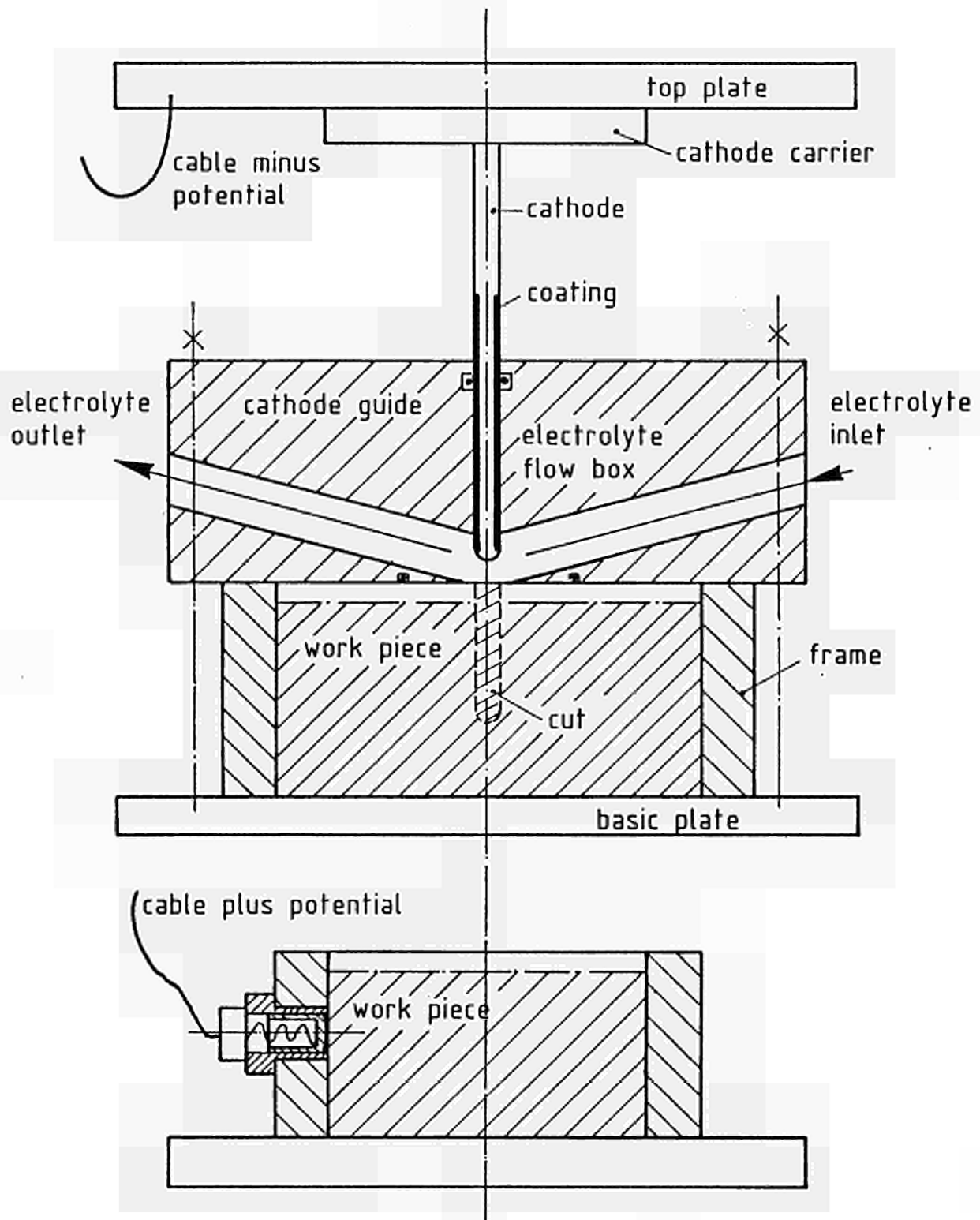


Fig. 1. Trial tooling for ECM cutting technique.

3.6. Explosive Techniques for the Dismantling of Biological Shield Structures

Contractor: Battelle-Institut e.V., Frankfurt, Germany
Contract N°: FIID-0011
Working Period: April 1985 - September 1987
Project Leader: H.U. Freund

A. Objectives and Scope

In the decommissioning of reactor systems, the removal of heavy reinforced or prestressed concrete structures, in which large quantities of concrete and steel have become activated during reactor operation, is considered as a major problem.

To study methods for the safe demolition of such structures, without release of radioactive materials, various techniques are being considered for the cutting of concrete in which a high level of control could be imposed. In the foregoing CEC research programme Taylor Woodrow Construction (TWC), under consideration of one approach, undertook a programme of tests which demonstrated the feasibility of using explosives for such controlled cutting. During the same period the Battelle-Institut e.V. (BF) also demonstrated the feasibility of an approach using "line charges" as opposed to the "point charges" used by TWC.

The present research work aims at complementing, improving and optimising the foregoing work. Extensive investigations will be executed on the adjustment of blasting parameters, material and structural effects, drilling techniques, particle distribution and on procedures for remote handling. Work is carried out jointly - based on a common and complementary work programme - with TWC (contract N° FIID-0012).

B. Work Programme

- B.1. Adjustment of blasting parameters considering separation efficiency and fragment size.
 - B.1.1. Effect of initiation mode - sequential or simultaneous firing - (BF).
 - B.1.2. Effect of charge type and tamping (BF).
 - B.1.3. Effect of charge distribution - hexagonal and parallel line arrays - (BF).
- B.2. Study of the effects of geometrical shape, of the pressure of a steel liner and of reinforcement.
 - B.2.1. Effect of the geometrical shape of the structure and of the presence of a liner (TWC).
 - B.2.2. Effect of the reinforcement array (BF).
- B.3. Drilling and boring of charging holes.
 - B.3.1. Assessment of boring by shaped charges (TWC).
 - B.3.2. Assessment of mechanical drilling (BF).
- B.4. Study of the structural response of the test body and filters to blasting in closed containment experiments.
 - B.4.1. Response of the test body and of its foundation (BF).
 - B.4.2. Study of the blast wave pressure distribution (BF).
 - B.4.3. Effect of blast on air filters (BF).
 - B.4.4. Theoretical assessment and modelling of blast effects (BF + TWC).
- B.5. Investigation of generated dust during blasting.
 - B.5.1. Assessment of particle size distribution of produced rubble as a function of charge burial depth (TWC).
 - B.5.2. Effect of a spray system on mass and size distribution (BF).
- B.6. Final assessment and evaluation of results, including desk studies on procedures for remote handling (BF + TWC).

C. Progress of work and obtained results

Summary

Work on the contract started only in October 1985 after final clearing of legal contract arrangements. It concentrated on the topic of work package B.1 "Adjustment of blasting parameters".

As methods for the practical realization various initiation modes have been considered. Commercially available explosive charges have been investigated on their applicability with regard for bore hole dimensions and tamping.

Progress and results

1. Effect of initiation mode (B.1.1.)

Due to increased safety requirements a mixed electrical/pyrotechnic initiation has been chosen to be best suited as compared to electrical circuitry initiation used in standard demolition blasting (see Fig. 1) /1/ and /2/.

The initiation techniques considered and their characteristic properties are shown in Table I.

2. Effect of charge type and tamping (B.1.2.)

Prior to the first run of experiments a careful design of the charge/bore hole geometry and the tamping is being made. Depending on commercial availability and handling safety criteria various setup modes may be applicable. In Fig. 2 a selection of 3 setups which have been successfully applied in preceding experiments /3/ are shown. Two from these will be incorporated into the first blasting runs.

References

- /1/ THUM, W., Sprengtechnik im Steinbruch- und Baubetrieb, Bauverlag Wiesbaden und Berlin (1978)
- /2/ Handbuch Sprengtechnik, Autorenkollektiv VEB, Deutscher Verlag für Grundstoffindustrie Leipzig (1975)
- /3/ FREUND, H.U. et.al., Battelle-Report BIEV-R-65.036-4 to BMFT Project 02 S 7031 (1983)

TABLE I

Initiation techniques

specification	amount of explosive	timing characteristics	remarks
1) electrical detonator + booster charge	< 1g/bore-hole	multiple simultaneous and/or delayed firing in the range between zero and several seconds	standard blasting initiation technique
2) pyrotechnic cord	~ 10g/m, < 1g/bore-hole	multiple simultaneous firing	insensitive to electrical failure and transient pickup, careful geometrical layout required
3) Norabel shock tube + booster charge	< 0,1g/m < 1g/bore-hole	multiple simultaneous firing and/or delayed firing, delay adjustable by length of cord between zero and several milliseconds	insensitive to electrical failure and transient pickup, simple non-destructive layout of cord array

Note: In present experiments, a combination of 1) and 3) will be employed.

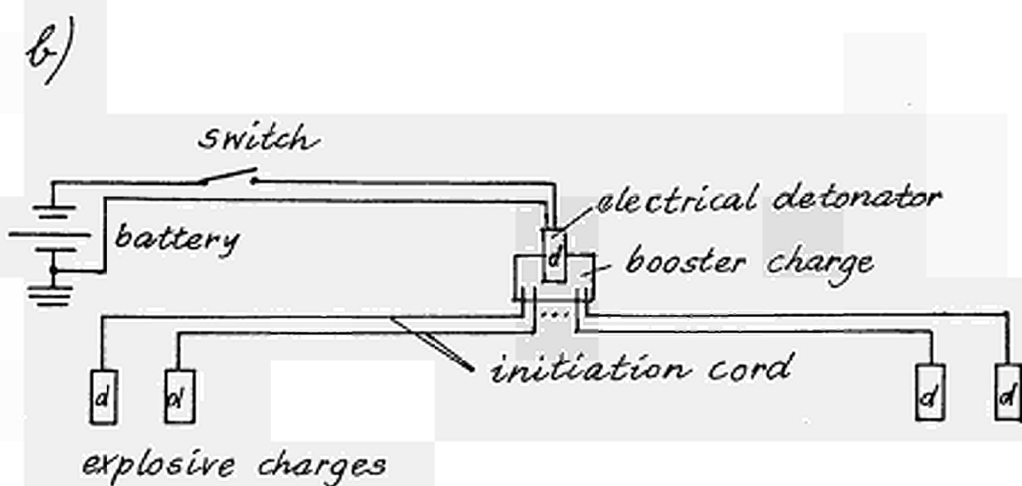
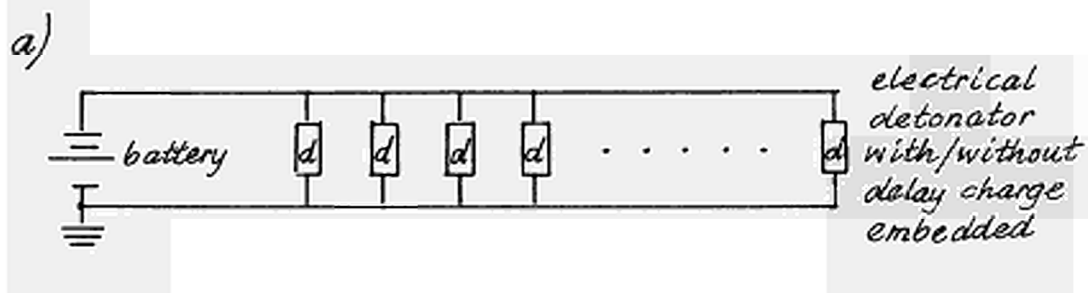


Fig. 1: Initiation circuits:
 a) standard electrical circuitry
 b) mixed electrical/pyrotechnic initiation

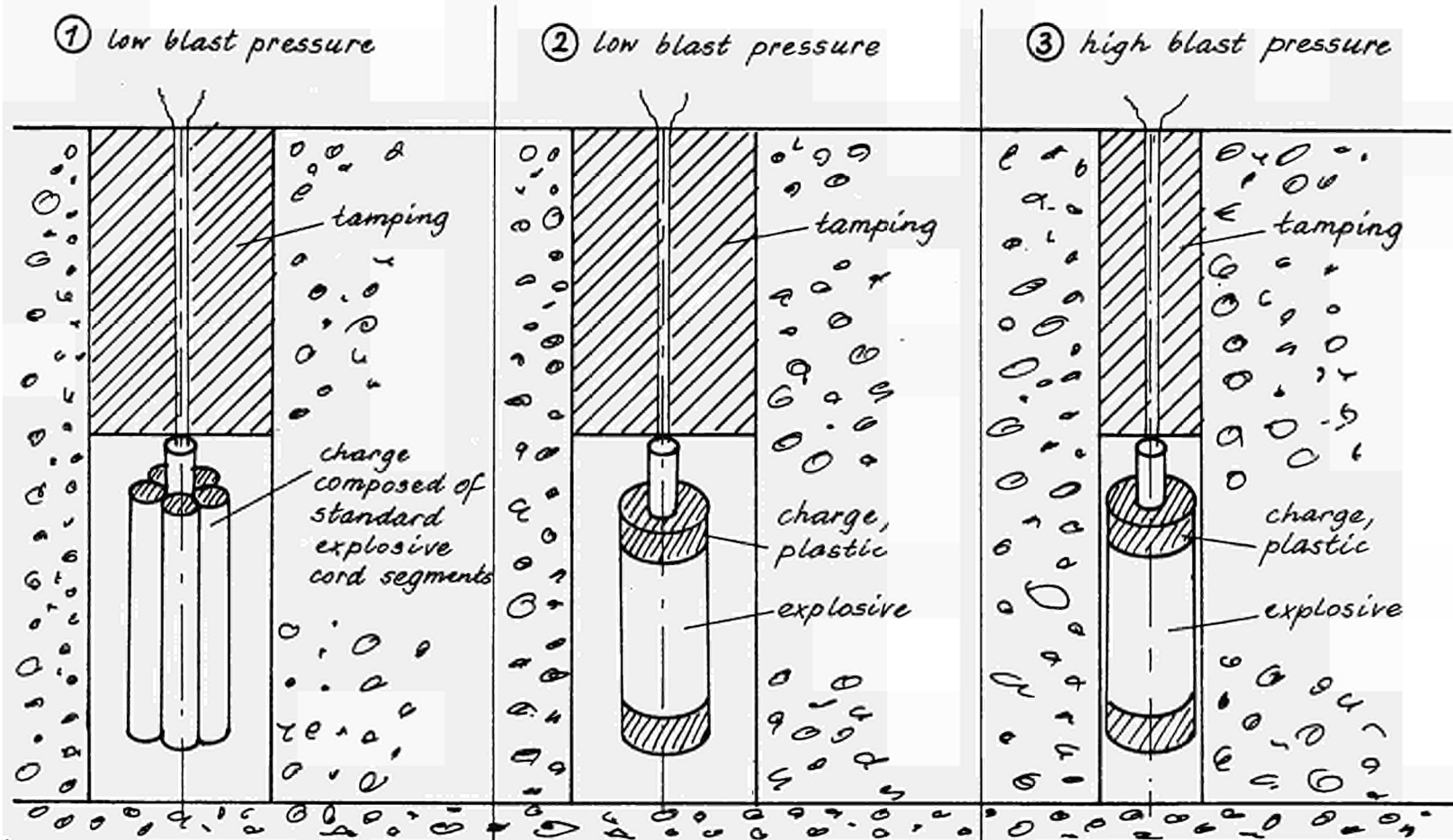


Fig. 2: Modes of charging and tamping considered for the experiments

3.7. Prototype System for Remote Laser Cutting of Radioactive Structures

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FIID-0013
Working Period: November 1984 - April 1988
Project Leader: J. Geoffroy

A. Objectives and Scope

The advantage of cutting by laser beam consists mainly in very small induced cutting forces and in producing only small amounts of cutting waste. The principal aim of the present research is the development and construction of a prototypical laser cutting device for metal structures, which may be contaminated or activated. The system will be designed for remote operation.

An existing 3-5 kW industrial laser will be adapted for transportability and tightness in nuclear environment. The laser transport system will consist in an articulated arm for transmitting the laser beam to a remote cutting location. The arm, operating with 5 degrees of freedom in a polar coordinates system, will be capable of entering an active area through an orifice of a diameter of only 250 mm. Each articulation will be equipped with an electrical D.C. motor enabling positioning by remote command. The actual trajectory of the cutting head will be defined by practical testing.

For commissioning of the developed prototype, a series of cutting tests on typical, but non-radioactive structures as hot cells, pipework, waste containers etc. will be executed, including measurements of generated aerosols and slag.

B. Work Programme

- B.1. Design, construction and functional testing of a robot arm including remote control and command, and tests on the handling of the arising laser cutting waste.
- B.2. Adaptation and coupling of the robot arm to an available laser cutting device.
- B.3. Commissioning and demonstration tests of the complete facility, including laser cutting of various non-radioactive stainless steel components with handling of the arising cutting waste.

C. Progress of work and obtained results

Summary

During the first 1985 semester, the main functions of the dismantling laser robot (ROLD) were defined in the nuclear environments (inside reactor, hot cells). The laser robot's specifications and the basic principles were defined during the 2nd 1985 semester and the conceptual studies were made.

The basic robot configuration is 5 axis freedom (rotation). The definition of the components of the joints (motors, speed reduced, bearing and encoder) has been made.

The computer control has been chosen. It is the computer controlling industrial robot system manufactured by COMMERCY. Furthermore, some tests of cutting of stainless steel were made for evaluating the aerosol products.

Progress and results (B.1.)

The study of the design of a dismantling laser robot (ROLD) was realized with the specifications defined during the first 1985 semester. Its essential function is the cutting of metallic structures in active zone (reactor core, hot cells).

In all the cases, the CO₂ power laser will be installed outside the active zone. An optical guide (crossing cell, telescopic tube, etc...) will join the laser to the robot itself (fig.2), this robot will have a polar geometry with 5 axis of rotation with a developed length of about one meter. It will have to cross a circular hole of about 0,2 m diameter. The computer control will assume the control of the trajectory (teaching and back repeat), as well as the system parameters (speed, power, security, etc..). The optical transmission will be made with 8 mirrors plans at 45° incident following the given principles by the figure n°3. The use of 8 mirrors (instead of the six theoretical necessary) allows to have the coaxial inside and outside beams. This disposition will allow the trajectories command.

The choice of the motors, speed reducer, encoder, has been made in function of the mechanical couples, speed accelerations, previous precisions and also in function of the small available space to place these components in the robot structure. This choice has been done in successive steps and finally allows to respect the crossing contrainte by an hole of 0,2 mm diameter (figure 3). The estimate total weight for the robot is of about 22 kg. For the computer control, we have chosen an industrial manufactured cabinet which command principle is very close to the one we use in our laboratory. This is the "control unit Commercy" using a Gixi computer XI.16. Two micro-processors cards Intel 8086 are used, one of them mainly for the trajectory calcul and the other for the remote control.

In another way, some tests were made concerning the formation of aerosols during one cutting metal operation. We used stainless steel structures of plates of 2 mm thick, which simulate the structures thickness that can be seen in cells (pipes, tubes).

This work was realized with the help of CEA/SACLAY specialists (Service de Protection des Installations Nucléaires, Section d'Etudes Industrielles de Protection).

Table 1 gives the results (the given masses being an average on several tests) 30% of aerosols are produced near the front face of the structures and 70% near the back face. An analysis of the granulometric distribution shows that 60% of the produced aerosols have a diameter lower than 0,35 μm (figure 1).

Table I

Aerosols analysis on stainless steel plates
(2 mm thick)

Test	P watt	V m/min	Pressure bars	Weight in mg/m		
				ma	md	ma/md
1	120	0,2	6	83	450	0,18
			4	77	435	0,17
2	420	1,2	6	120	1287	0,09
			4	85	1236	0,07
3	960	3	6	427	4744	0,09
			4	299	3127	0,09

ma aerosols mass produced by meter of cutting
md sediments mass produced by meter of cutting

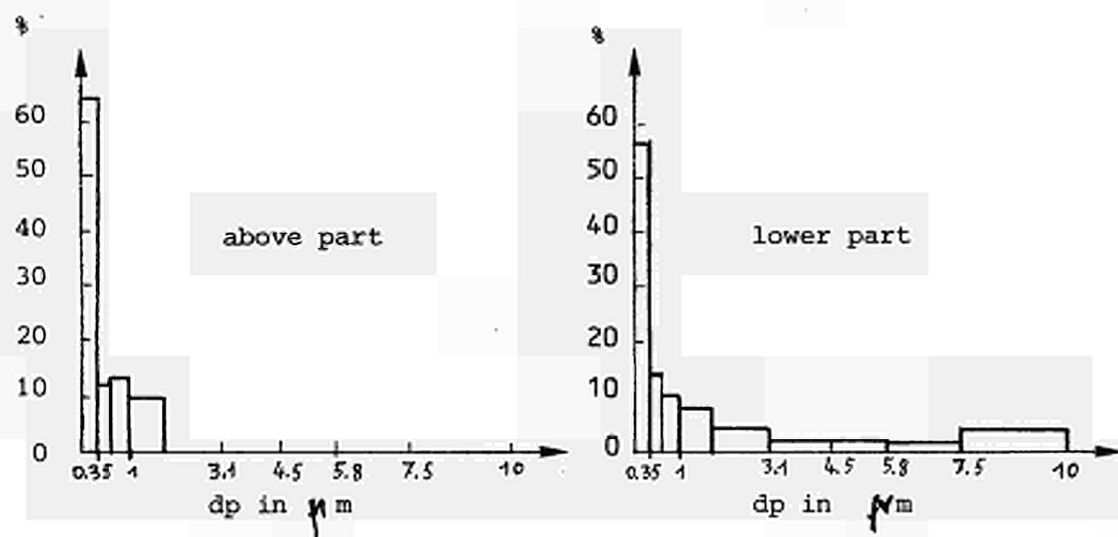


Fig. 1 : Particle size distribution
(Test at 960 W)

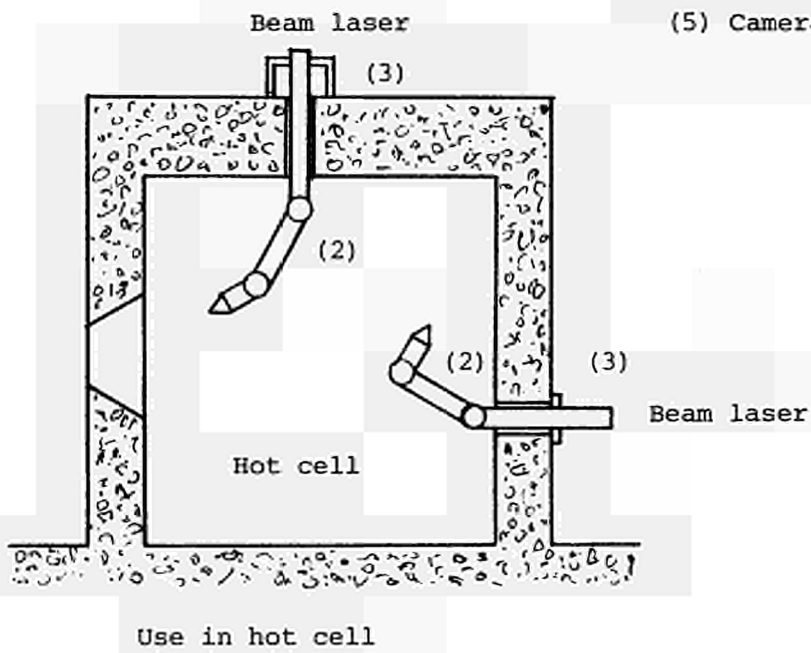
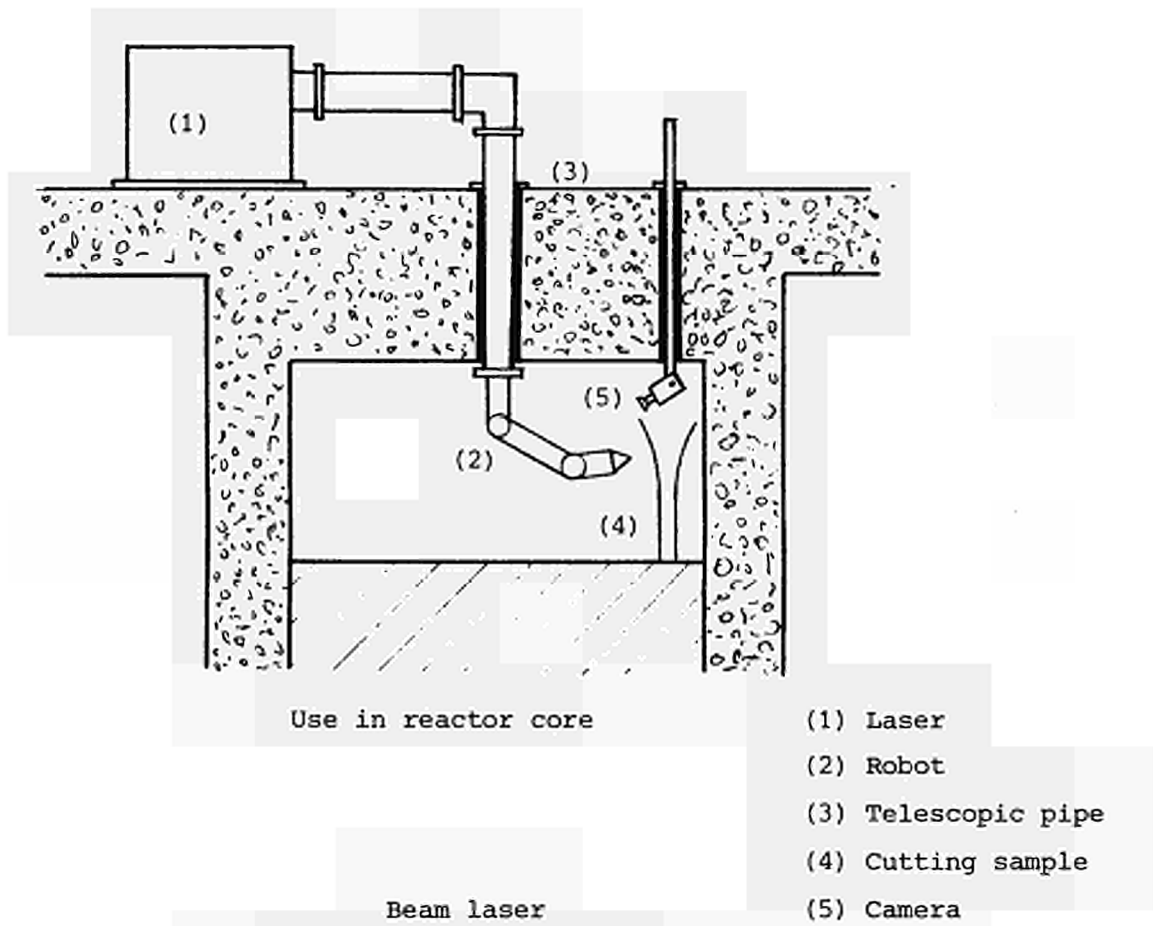


Fig. 2 : Examples for the utilisation of the laser robot

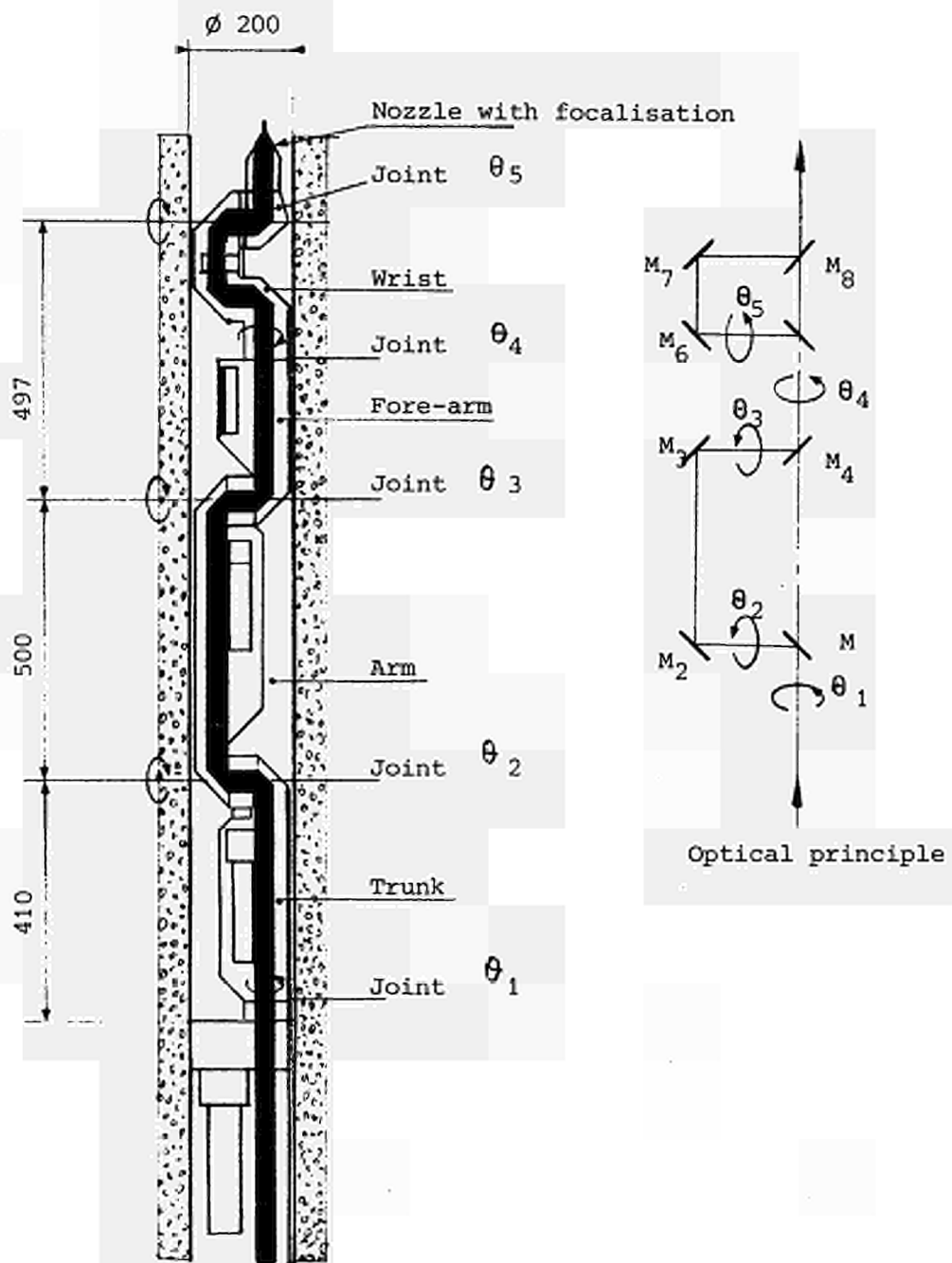


Fig.3 : Rold conception

4. PROJECT N°4:

TREATMENT OF SPECIFIC WASTE MATERIALS: STEEL, CONCRETE AND GRAPHITE

A. Objective

In the dismantling of nuclear installations large amounts of radioactive steel, concrete and - in the case of gas-cooled reactors - graphite will arise. This waste must be suitably conditioned for disposal.

B. Research performed under the 1979-83 programme

The following research work has been performed:

- experiments on the melting of radioactive steel scrap including investigation of the possibility of decontaminating the melt;
- development and assessment of techniques for coating metal and concrete parts in order to immobilise the radioactivity;
- comparative assessment of various modes of treatment and disposal of radioactive graphite.

C. 1984-88 programme

Melting of radioactive steel should be further investigated, on the one hand as a method for immobilising contamination and reducing the volume of waste, and on the other hand as first step towards the possible recycling of the steel.

The work on coating techniques should be continued with a view to the integration of this treatment into appropriate overall waste management and disposal schemes.

Treatment techniques for graphite waste should be developed for at least one management mode, to be selected with due regard to the results of the assessment performed under the 1979-83 programme.

The treatment of plutonium-contaminated steel and concrete from the dismantling of fuel-cycle facilities is a new aspect to be investigated under the 1984-88 programme.

In all these investigations, due attention will be paid to the necessity of adapting the treatment techniques to the final destination of the waste.

D. Programme implementation

At the end of 1985, four research contracts were at the stage of execution, and three contracts were at the stage of negotiation.

4.1. Research into the Melting/Refining of Contaminated Steel Scrap Arising in the Dismantling of Nuclear Installations

Contractor: British Steel Corporation
Contract N°: FIID-0015
Working Period: January 1985 - September 1987
Project Leader: C.R. Gomer

A. Objectives and Scope

This is a research into the melting and refining of contaminated steel scrap arising in the dismantling of nuclear installations. The general aim of the research is to optimize the management of these metal wastes so as with minimum radiological impact to immobilise the various radioactivities in metal and secondary products of minimum volume for storage. Alternatively from some starting contamination or activation level to be determined, to recycle the metal product either for unlimited release or for specific shield or storage containers for more highly radioactive materials. The first research programme 1979-83 yielded a considerable body of knowledge, with radioactivity behaviour in several types of melting recognized. The present work is a continuation study with these and other furnace systems and with examination of behaviour of some smaller presence radioactivities. Radiological safety factors and updated cost benefit for recycling and disposal will also be evaluated.

B. Work Programme

- B.1. Tests on the 5t electric arc furnace with appropriate nuclear scrap and simulated contamination.
- B.2. Investigation of caesium retention in 10t induction furnaces using normal acid slag and low level radiotracer.
- B.3. Melting of activated/contaminated steel waste in a 6t experimental Basic Oxygen Furnace (BOS) in order to examine the Co-retention in slag when Co is present as surface contaminant.
- B.4. Pre-furnace assessment of the contamination of steel waste by monitoring.
- B.5. Investigation of the slag/metal chemistry to identify specific radionuclides (Nb-94, Ni-63, Sr-90, Sb-124, Eu-154 and Am-241).
- B.6. Investigation of the transfer of radioactivities to furnace and refractories with particular view to the concentration effect of the nuclides.
- B.7. Evaluation of radiation exposure (individual and collective) of involved persons and of radioactive emissions to the environment for long-term operation; cost/benefit optimization for re-cycling and disposal based on results obtained.

C. Progress of Work and Obtained Results

Summary

Work has mainly been under programme item B.1 with some required measurements coming under the B.4 and B.7 headings. Thus continuation work on radiocaesium behaviour has been done in electric arc furnace steel melting. This was a radioactivity balance in the Grangetown 5 t furnace with an acidic refractories lining which allowed the use of an acidic slag. Caesium-134 as caesium hydroxide contamination on a steel plate assembly (to resemble say heat exchanger reentrant surface) was substantially retained in slag for at least 30 minutes after plate addition to the melt which is sufficient time for slag to be removed and thus in effect to decontaminate the melt metal. The cost and effort of adapting the furnace for this work are relatively low and thus provides flexibility in decision on the radiological management of caesium radioactivities with the versatile steelmaking capability of the electric arc furnace.

A second test has been made on samples proving the radiological safe reclaim in both process and product, of dismantled pipework from No. 1 Magnox reactor at the Oldbury Power Station. Contamination mainly Co-60 on this pipework requires it otherwise to be low level radioactive waste with attendant inefficient and high cost eventual disposal. The main amount, i.e., 6 t of the bypass CO₂ coolant circuitry has been cut into suitable lengths and is an ideal subject for a 150 t production electric arc furnace reclaim demonstration. The impasse in regard to acceptance by the workforce in production melting however remains.

Also included in this year's work has been a first assessment of the possible benefits in radioactive metal waste management of a new type of versatile electric arc furnace using a plasma arc instead of the usual carbon electrode arc.

Progress and Results

1. Caesium Radioactivity Balance Work

1.1 Introduction

A radioactivity balance melt was needed to determine whether values for radiocaesium retention in slag determined in earlier work on induction furnaces using acidic slags, could be obtained on an electric arc furnace.

1.2 Furnace Preparation

The electric arc furnace is normally worked with a basic slag and basic refractories which is not favourable to caesium retention.

The following work was therefore done on the BSC Grangetown 5 t electric arc furnace (see Fig. 1).

- (a) The normal monolithic hearth was removed and replaced by the more acidic ramming material.
- (b) The basic tarred magnesia walls of the furnace were sprayed with a silica based compound (Hotshot) to give a 5 cm thick cover to the original refractory to a height well above the slag line on the furnace. This is the area where the slag cover, on the molten metal pool washes against the furnace wall and is the area of greatest wear and refractory/slag reaction.

This is an inexpensive method to temporarily transform a basic refractory arc furnace lining to an acidic lining. The furnace can readily be changed back to its more usual role of basic lining/basic slag operation.

1.3 Preparation of Radioisotope Addition

Earlier balance work carried out with a 250 kg melt in a 500 kg induction furnace had studied the retention in slag of caesium hydroxide added as contamination on steel plate. Almost 100% of the added caesium-134 was retained in the slag until after the furnace was tapped.

For the present balance melt two plates were contaminated with a total of 15 μ Ci (0.56 MBq) of caesium-134 as caesium hydroxide. The plate preparation was identical to that for previous radioactivity balance melts, the plates being contaminated by caesium hydroxide droplets. Only one side of each plate was contaminated and the two plates were bolted together with approximately 15 mm separation. This allowed the plates to be added to the furnace without the risk of contamination transfer and simulated internally contaminated material such as Magnox reactor power station bypass pipe work.

1.4 Melt Details

Two tonne of plate fragment scrap charge was melted in the 5 t capacity electric arc furnace; 24 kg of sand plus 12 kg of lime addition produced a total slag cover. The contaminated scrap was added to the molten metal pool via the furnace door and to ensure the caesium had maximum opportunity to react with the slag the scrap was held under the slag until completely melted.

1.5 Results and Discussion

There was no pick up of radiocaesium in any metal sample. All slag samples taken during the trial were analysed to obtain the $\text{SiO}_2:\text{CaO}$ ratio as a measure of the acidity of the slag, a mean value of 3.1:1 is recorded on graph Fig. 2 as are the values 1.3:1 and 0.41:1 for two earlier radiocaesium balance melts in this furnace but with unmodified lining. In these two melts caesium was substantially lost almost immediately in one case and at a fast rate from early in the melt and on down to zero in the other. Figure 2 showing specific activity of slagtime compares the caesium loss rate for the three $\text{SiO}_2:\text{CaO}$ ratios. Since the initial amounts of radioactivity divided by the slag masses were closely similar this graph compares also the whole loss of caesium radioactivity for the three melts.

The immediate substantial caesium loss for the $\text{SiO}_2:\text{CaO}$ 1.3:1 ratio melt is thought due to direct volatilisation when the caesium contaminated plate assembly was seen in this case to float in the slag layer. The loss rate for the remainder of caesium transferred to slag falls into the order of the $\text{SiO}_2:\text{CaO}$ ratios. Thus:-

Remainder Radiocaesium at 30 minutes		
Approx. 80%	SiO_2/CaO	3.1:1
Approx. 20%	SiO_2/CaO	1.3:1
Approx. 9%	SiO_2/CaO	0.41:1

2. Magnox Coolant Duct Pipework Test Melting

2.1 Introduction

Bypass duct from the CO₂ coolant circuits from CEGB Oldbury Nos. 1 and 2 Magnox reactors 2 x 6 t, i.e., 12 t total in all, has now been offered for reclaim melting. Pilot melting showed the material from No. 2 reactor to be satisfactory for large furnace dilution melting, but work force opposition prevented this and the material was disposed of to the UK Drigg site.

It was found that the material from reactor No. 2 required X15 dilution to reach 0.37 Bq g⁻¹ and it was considered of value to determine this factor in the present case and also determine from the overall measured radioactivity in the melt the accuracy of the premelt monitoring assessment. This assessment was done by reactor health physics staff - one of whom attended the trial melt - when the bypass pipework was removed and cut to furnace charging lengths. The inside surface of the pipework was spray painted to fix the contamination and two small representative pieces, Fig. 3, were sent to BSC for pilot melting. These were further monitored by BSC before charging the total 62 kg to the cold furnace with overcover charge of 3400 kg of plate fragment scrap. Realistic large steelmaking practice was observed by oxygen lance blow into the furnace - which at the same time causes considerable fume discharge into the melt shop.

2.2 Results

Results from this work are summarised for premelt monitoring assessment in Table I and for melt and melt auxiliary measurements in Table II.

From the actual dilution factor of X50 and the radioactivity value of 0.02 Bq g⁻¹ attained it is seen that a dilution factor of only X3 is needed to reduce this No. 1 reactor pipework to the UK DoE/BSC registration reclaim value of 0.37 Bq g⁻¹. Also the amount of radioactivity on the pipework was one tenth only of the monitoring estimates.

3. Radiological Safety

Furnace personnel quartz fibre and film badge dosimetry showed only background radiation exposure. Also all personal air samplers (1μ particle retention) showed no airborne radioactivity pick up during any of the trial work. This applied also to the respirator filters worn by the oxygen lance operatives. Table II shows the measured value of environmental air to be at only 0.1% of the DAC value.

4. Plasma Arc Furnace Potential in Nuclear Scrap Melting

Consideration has been given to this furnace (which is still under development and for which an experimental unit exists within BSC) for radioactive steel melting. Thus it has very good operating enclosure, there is less violent arc action promoting better slag cover during melting and very good steelmaking control of slag reactions. It may well be better suited to movement from site to site requiring no tilting mechanism the metal being discharged through the furnace bottom hearth. Radiological and operational factors including comparative costs are being examined.

TABLE I HEALTH PHYSICS ASSESSMENT OF PIPEWORK PRIOR TO MELTING

	Piece 1	Piece 2
Mass, (kg)	37	25
Max. surface dose rate, mR/h (μ Sv/h)	0.07 (0.7)	0.05 (0.05)
Internal surface fixed contamination, μ Ci/cm ² (Bq/cm ²)	2×10^{-3} (74)	2×10^{-3} (74)
Estimated total radioactivity, μ Ci (kBq)	14 (518)	10 (370)
Principal Radionuclides	Cobalt-60, possibly S-35, trace caesium-137	Cobalt-60, possibly S-35, trace caesium-137

TABLE II COMPARISON OF MEASURED RESULTS WITH RELEVANT LIMITS

Sample Type	Measured Value	DAC Bq/m ³ . (Cs-137)	ALI Bq . (Cs-137)	Dept. of Environment Reclaim Registration Limit Bq/g
Environmental Air	1.62 Bq/m ³	2×10^3	6×10^6	-
Extraction Duct Air	3.0 Bq/m ³	Not Applicable	Not Applicable	-
Slag (mass 100 kg)	0.2 Bq/g	-	-	0.37
Metal (mass 34 t)	0.02 Bq/g	-	-	0.37
Extractor Dust Cake (mass 34 kg)	<0.01 Bq/g	-	-	0.37
Total Activity	83 kBq	-	-	-

DAC - (Derived Air Concentrations) are derived limits intended to control exposure over prolonged periods of exposure of up to one year.

ALI - (Annual Limits on Intake) are derived limits which if achieved would result in a maximum permissible whole body radiation dose of 5 rem/year from internal radiation.

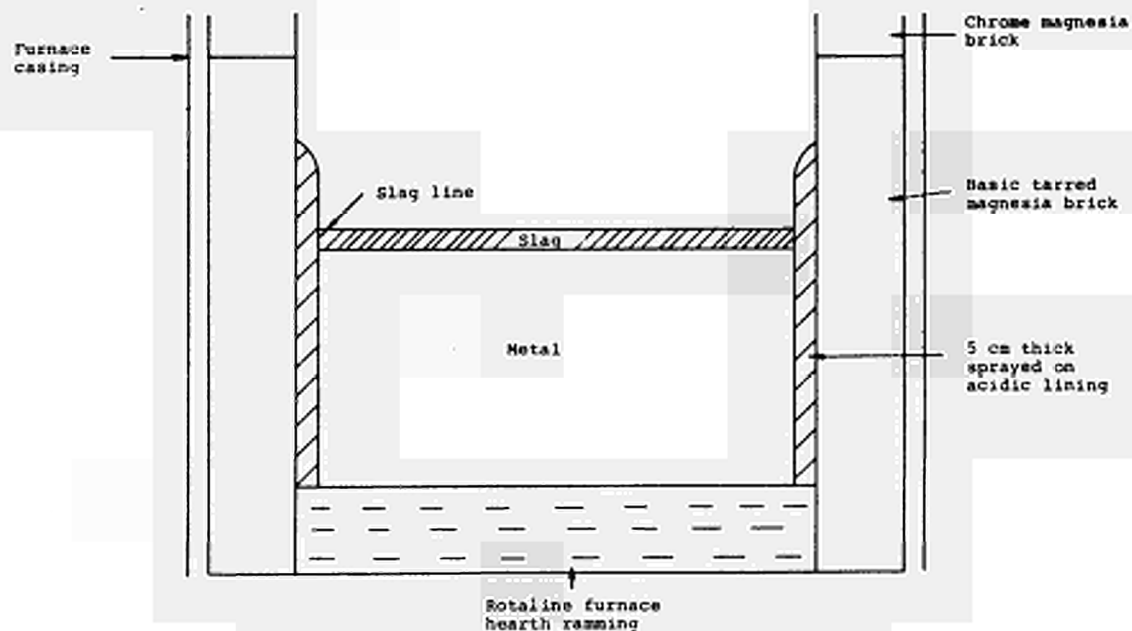


Figure 1. Furnace refractory arrangements

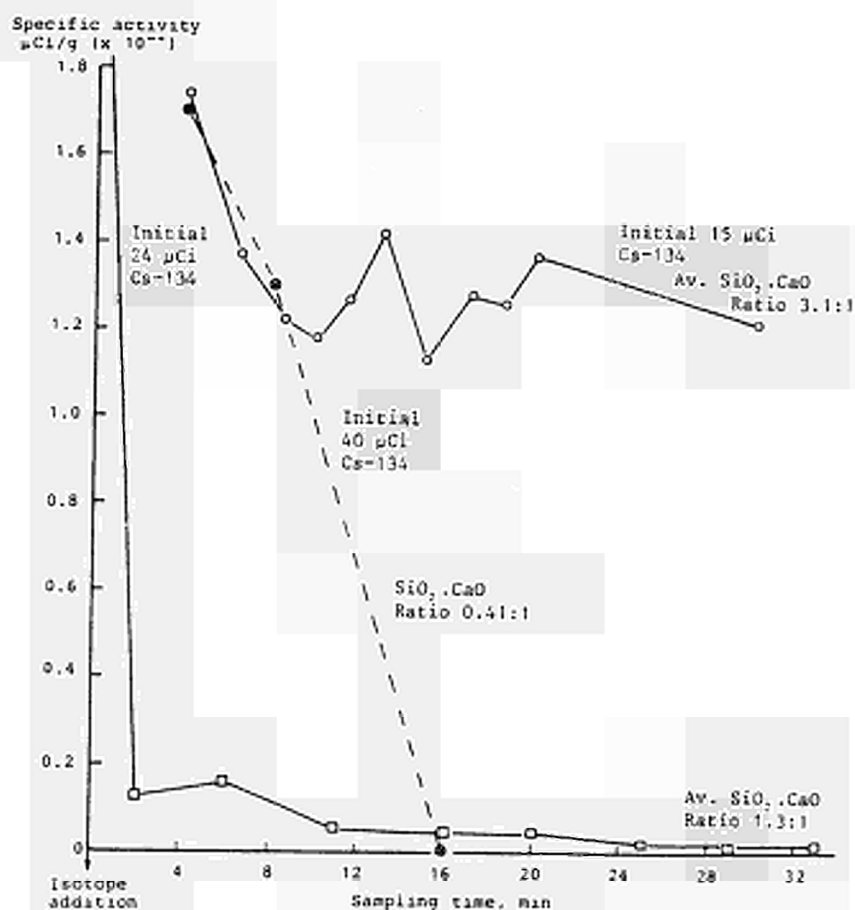
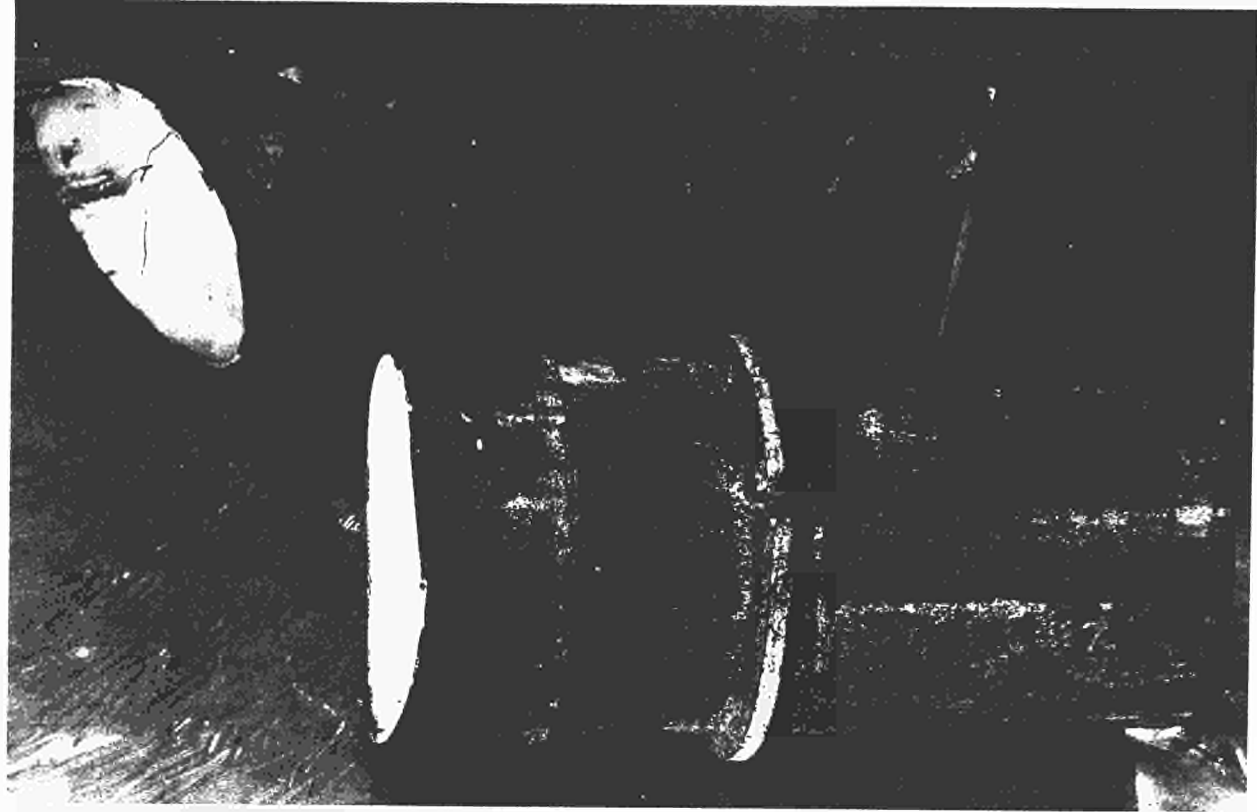


Fig. 2: Comparison of caesium-134 retention with slag acidity



TWO SECTIONS OF BYPASS CIRCUIT PIPEWORK MELTED AT GRANGETOWN FIG. 3

4.2. Melting of Radioactive Metal Scrap from Nuclear Installations

Contractor: Siempelkamp Giesserei, Krefeld, Germany
Contract N°: FI1D-0016
Working Period: November 1984 - December 1986
Project Leader: M. Hasselhoff

A. Objectives and Scope

This research work is based on the results and experience carried out at Siempelkamp in the frame of the first five-year programme (1979-1983). The preceding research work proved that it is possible to melt down contaminated scrap by means of a modified industrial furnace device in compliance with the legal limits and regulations.

The main targets of this research work therefore aims at the reaction of radionuclides during the melting procedure with regard to various material qualities and the harmless recycling of melted-down metal parts coming from refurbishing and decommissioning of nuclear installations.

B. Work Programme

- B.1. Planning and design of the melt device taking into account an existing furnace.
- B.2. Construction of the needed melt device components.
- B.3. Meltwork using as scrap contaminated carbon steel, stainless steel and its mixture.
- B.4. Evaluation of melt results.
- B.5. Technical, economical and radiological consequences.
- B.6. Extrapolation to large commercial reactors (1300 MWe) and comparison with alternative modes with view to economical and environmental aspects.

C. Progress of work and obtained results

Summary

In the course of the present research programme melts were carried out with larger amounts of low contaminated steel waste.

The melts gave further information about the behaviour of single radionuclides during the melting of different steel qualities as well as on the continuous operation of the device. For proving the harmless recycling of the melted-down material, suitable castings have been produced.

Progress and results

1. Planning, design and construction of the melt device (B.1. and B.2.)

From November 1, 1984 until December 31, 1985 basically the following works were carried out: first of all an appropriate melt device based on the experience of the preceding research programme was designed and built, consisting of the following components: melting furnace, charging device and filter system.

The melting furnace is a modified industrial induction furnace with a capacity of approximately 20 Mg of liquid iron. The charging device with a transfer canal system guarantees the maintenance of low pressure in the total device during charging the melting furnace. The filter system consists of a cyclone filter, a tube filter, and a high efficiency submicron particulate airfilter step.

2. Melting of contaminated steel waste coming from refurbishing nuclear installations (carbon and stainless steel) (B.3.)

- Melts carried out:

- . in the first series of tests approximately 35 Mg contaminated mixed scrap was melted down. Due to the high chromium contents of the cast iron melt the utilization was reduced to production of shielding plates, weight approx. 700 kg/piece; dimensions 1000 x 1000 x 100 mm;
- . in the second series of tests contaminated carbon steel with a mass of approx. 30 Mg was melted down. In this series of tests the reproduction of the radiological results could be proved. This steel quality could be used for castings of high quality such as type-A and type-B container;
- . in the third series of tests austenitic stainless steel was used. Out of approx. 34 Mg. material 90 bars were produced for a later recycling. The radiological results basically attest the results concerning the reaction of the single radionuclides already found in the other series of tests. The relatively high percentage of slag (+/- 5%) is remarkable compared with the other series of tests.

- Results:

The results of the series of tests basically confirm these of the preceding research programme. Of the essential radio-nuclides Co-60 and Cs-137 more than 90% of Co-60 is to be found in the metal and more than 80% of Cs-137 in the slag.

Corresponding to the various materials tendentious differences could be found. The ferritic scrap with a magnetite layer gives the activity in a bigger amount to the slag than austenitic scrap, where the activity is bound by diffusion in the metal matrix.

The last melts gave information about the behaviour of other relevant nuclides during the melting process. Helpful for the detection of these nuclides was that the specific activity - due to the small amount - in the slag and filter dust increased, and the homogeneous distribution of the nuclides on the melt allows a long-time measure-

ment by taking one sample from the melt.

. Silver 110 m

Some experiments showed a small amount of silver. The behaviour of silver is very similar to cobalt. This means that 90% of Ag remains in the metal, only 10% goes to the slag and filter system.

. Cerium 144

Cerium 144 was once found in the slag and the furnace liner. The behaviour is similar to Cs-137, due to the higher point of vaporization it cannot escape to the filter system.

. Europium 154

Two of the melting experiments showed a small Europium 154 activity. Similar to the behaviour of Cerium it could only be detected in the slag.

. Manganese 54

Manganese was only found in the slag and the filter dust. It was not possible to prove manganese in the melt - possibly because of the small concentration.

. Zinc 65

Zinc with the low vaporization temperature leaves the metal and partly the slag and is found in the filter system.

3. Further procedure

Two further test melting campaigns are planned for 1986 to give information on activity exchange between melting and slag with respect to binding the adhering activity in the slag. At present test planning for a first series of tests is carried out. Discussions to obtain suitable waste material have already been engaged.

4.3. Separation by Transport in the Vapour Phase of Stainless Steel Constituents

Contractor: CEA-DMG, Grenoble, France
Contract N°: FIID-0017
Working Period: January 1985 - December 1986
Project Leader: G. Tanis

A. Objectives and Scope

A few decades after shut down of a nuclear power plant the stainless steel covering the inside of the pressure vessel is radioactive only due to cobalt 60. The separation of this element from the other constituent elements of the steel would drastically reduce the amount of radioactive waste and would allow the recycling of inactive elements, i.e. most of the steel.

To date no technique is known to lead to an efficient separation of cobalt at reasonable cost and without involving, as intermediate steps, an increase of the amount of waste. The present research consists in a feasibility evaluation of separating Co from stainless steel using vapour phase transport, offering two advantages:

- no additional amount of waste, even for a transient step,
- repeatability allowing separation rates to be reached.

It has never been applied to alloys as stainless steels and the conditions of application have to be assessed by theoretical and experimental research that should end with a first estimate of the feasibility of the suggested process.

B. Work Programme

- B.1. Preliminary work on the vapour phase transport including thermodynamic modelling of the process, set up of computer programs, collection of data and estimate of missing data.
- B.2. Conditioning of the metal to be treated and selection by calculation of appropriate transport gases.
- B.3. Experimental verification of the separation effect of the selected gases on stainless steels.
- B.4. Feasibility evaluation on the most appropriate situation and economical evaluation of the procedure in case of industrial application.

C. Progress of work and obtained results

Summary

Thermodynamic situations of vapour phase transport of the constituents of stainless steels were carried out and lead to the choice of iodine as carrier gas. The calculated separation rates showed that several successive treatments are needed to achieve the elimination of cobalt in Fe, Ni or Cr. Experiments on conditioning of primary material by plasma melting enabled the obtention of particles sizes that seem suitable for the vapour transport process.

Progress and results

1. Preliminary work on the vapour phase transport including thermodynamic modelling of the process, set up of computer programs and collection of data. (B.1.)

The preliminary work of carrying out a thermodynamic simulation of the suggested process was completed in 1985. It consists of:

- the definition of standard conditions for the thermodynamic modeling of the vapour phase transport, conditions under which transport capacities and partition coefficients could be compared. The practical thermodynamic parameters were also defined: temperature and rate of iodization.
- the setting up of several existing computer programs and their adaptation to the present case.
- the establishment of a data set by gathering data from thermodynamic data bank, from literature or by estimate of missing data from classical thermodynamic models.

2. Conditioning of the metal to be treated and selection by calculation of appropriate transport gases. (B.2.)

Various conditioning techniques were reviewed and a process leading to small metal particles was chosen because of its simplicity and its adaptation to the treatment of pieces. This process consists of a plasma surface melting of the steel with pulverization of metal droplets and solidification of these droplets into small particles with a surface/volume ratio suitable for the gazeification.

A thermodynamic evaluation of transport capacities of chlorine and iodine, associated or not with hydrogen, led to the choice of iodine as carrier gas. Thermodynamic and diffusion properties of the various phases involved enable an operating temperature range of 1300 - 1600 K to be established. Hydrogen slightly affects the transport capacity of iodine with respect to the metallic elements. This effect could be used to enhance separation but was not studied in detail.

A thermodynamic simulation of transport in the presence of alloys was carried out. It showed that the separations could be done in several steps, i.e. successive transports. The separation of Fe and Ni are quite easy but the separation of Cr from Co is problematic. An improvement of the separation coefficient will be sought in this last case.

This section is completed at 90 %, only a few calculations will be needed depending on the results of the experimental program.

3. Experimental verification of the separation effect of the selected gases on stainless steels (B.3.)

Laboratory experiments are now in progress, in order to check the agreement with thermodynamic calculations.

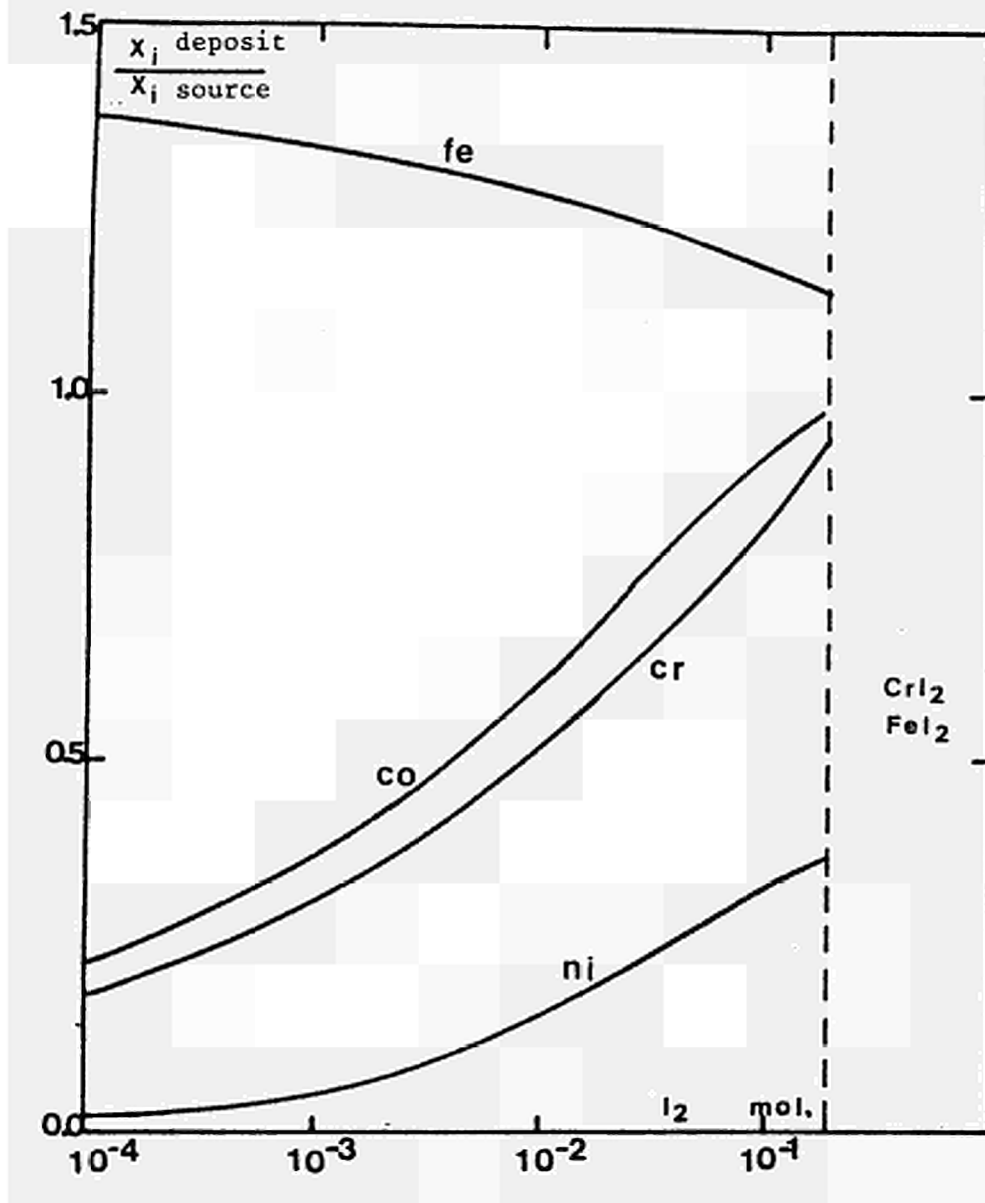


Fig. 1. The transport capacities of a carrier gas with respect to metals could be defined as the amount of mole of element transported when a given volume (10 litres) of gas, initially at equilibrium with the element and given amounts of carrier gas (I_2 ou $I_2 + H_2$) has its temperature raised by $100^\circ K$. The results calculated for Co and Fe are reported in the figures to show the difference in behavior of Co and Fe, responsible for the separation effect.

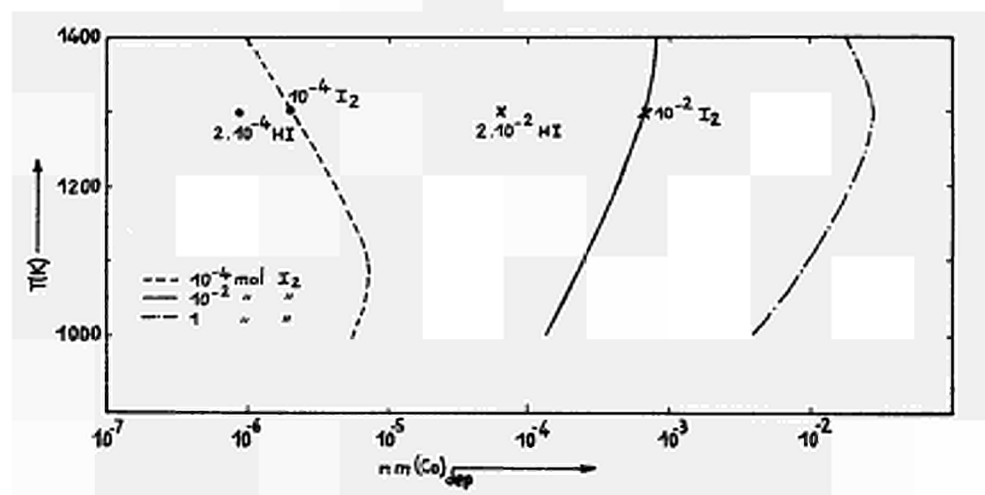
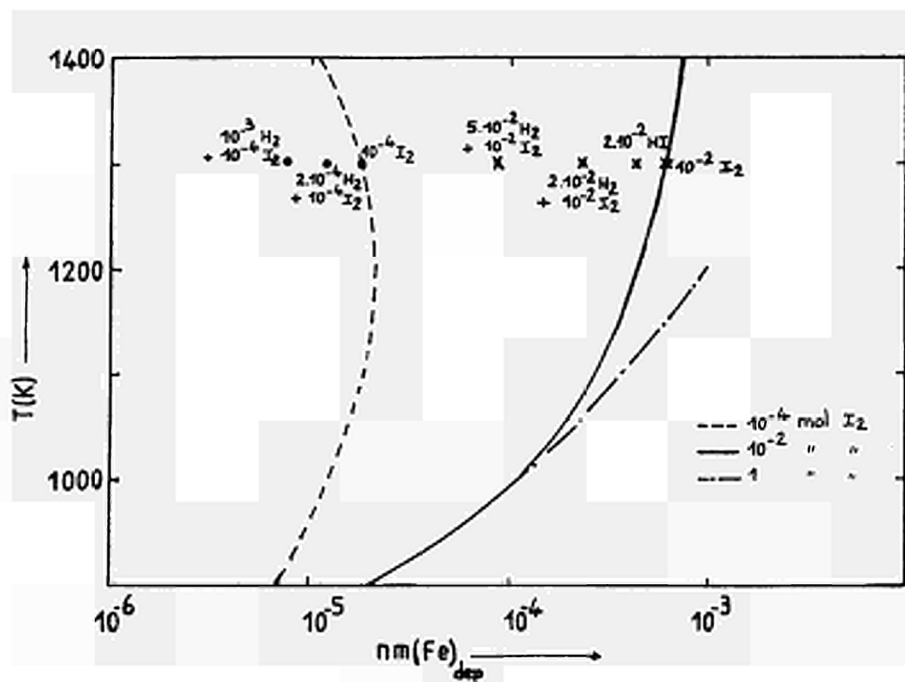


Fig. 2. The separation coefficients for the constituents of alloys are defined as the ratio of the content of a given element in an alloy deposited from the gas phase to the content of this element remaining present in the initial source alloy which has been treated. The results show that Fe and Ni separation is easy when Co and Cr are transported almost identically.

4.4. Immobilization of Contamination by Coating with Polymers on Large-Size Wastes with View to Storage

Contractor: Commissariat à l'Energie Atomique, Grenoble, France
Contract N°: FI1D-0018
Working Period: January 1985 - December 1988
Project Leader: Ch. de Tassigny

A. Objectives and Scope

Characteristics of polymers are convenient for producing coatings with good properties of durability and mechanical resistance. The study concerns the development of thick coatings with polymers on metallic pieces or on concrete. Indeed, an important thickness allows the lowering of diffusion of radionuclides and protects directly the surface of contaminated pieces without complementary process of cutting or embedding. New possibilities are then found in the field of handling, transport and storage of large size wastes issuing from dismantling of nuclear plants.

The aim of the programme is to demonstrate the possibility of applying this type of coating on real contaminated pieces coming from dismantling. Attention will be given to controls of representative radionuclides diffusion, mechanical and temperature resistance.

B. Work Programme

- B.1. Feasibility study of a procedure suitable for coating large components.
- B.2. Mechanical, thermal and radio-diffusion optimization of polymer coating with particular respect to geometry, surface conditions and nature (metal, concrete ...) of the components.
- B.3. Study of a mobile projection apparatus suitable to be adapted for application in the nuclear area and able for projection thick coatings on large components as given in B.2.
- B.4. Preparation of a projection area at pilot plant scale to demonstrate the feasibility of the procedure on components > 1m and on large dimension low level waste.
- B.5. Application of the procedure within the frame of a dismantling project in France or another EC country.

C. Progress of work and obtained results

Summary

The work performed during 1985 consisted mainly in a study of long-term diffusion of Co-60 and Cs-137 through polyurethan coatings, the establishment of quality controls required for process approbation by storage authorities, and the definition of a mobile spray unit suitable for nuclear environment.

Items B.1., B.2. and B.3. of the programme started during the middle of the year, later than foreseen in the initial planning. B.4. started earlier.

Progress and results

1. Feasibility study of a procedure suitable for coating large components (B.1.)

- Polymeric formulation

Several polyurethen have been examined; the composition of the polyol and of the hardener was regarded in connection with toxic regulation for industrial applications (some hardeners content aromatic components). The time of polymerisation is also an important parameter for obtaining a thick coating during a short time.

- Mechanical and physical characteristics

Preliminary results were obtained for an industrial polyurethan (URE-FLEX): /1/ and /2/. A new study started to define the following parameters:

- . resistance to fire and control of gas evolution,
- . stability with temperature action,
- . resistance against climatic ageing: UV, thermal cycles, damp, water,
- . resistance against mechanical aggressions: impact, erosion, stamping, tearing.

Samples were made with industrial projection unit. Study of these samples is in progress.

2. Optimisation of polymeric formulation (B.2.)

Tests for measuring the diffusion of the main radionuclides (Co-60 and Cs-137) in low level wastes show (fig. 1 and 2) that:

- during the first 200 days the diffusion is very low for this kind of polyurethan UREFLEX,
- an acceleration of diffusion occurs after 200 days,
- the results are better with one diffusion cell.

These tests will be continued.

3. Study on a mobile spray unit (B.3.)

A feasibility study concerning the coating of low level wastes issuing from the dismantling of a nuclear reactor (G1 at Marcoule) has shown the utility of a mobile unit able to coat in-situ contaminated components. This mobile unit has to be made with biological screen and ventilation adapted to the two following purposes: filtration of high efficacy for nuclear contamination and trapping of paint particles.

A two-modules apparatus has been chosen for treating pieces of various dimensions (fig. 3).

4. Feasibility study (B.4.)

The feasibility of this process is being demonstrated on a simplified scheme suitable for nuclear environment, i.e.:

- biological shield (concrete),
- industrial painting projector purchased with teleoperation,

- filtration system.

Application on real radioactive large-size waste from dismantling G1 is foreseen during 1986.

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/1/ de TASSIGNY, Ch., report EUR 9666 (1985)

/2/ de TASSIGNY, Ch., DAVIES L1., BRAMBILLA, G., Conference on the decommissioning of nuclear power plants, 22-24 May 1984, Luxembourg.

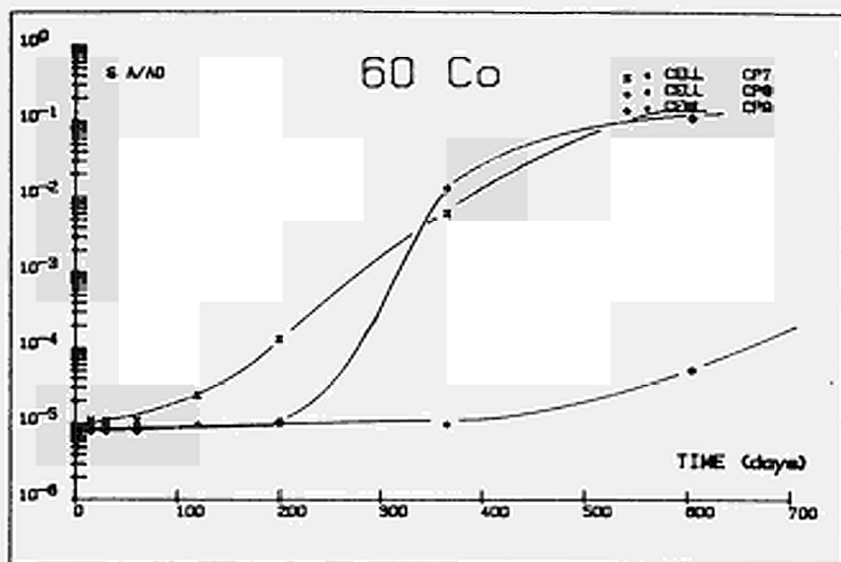


Fig. 1. Diffusion of ^{60}Co in polyurethan "UREFLEX".

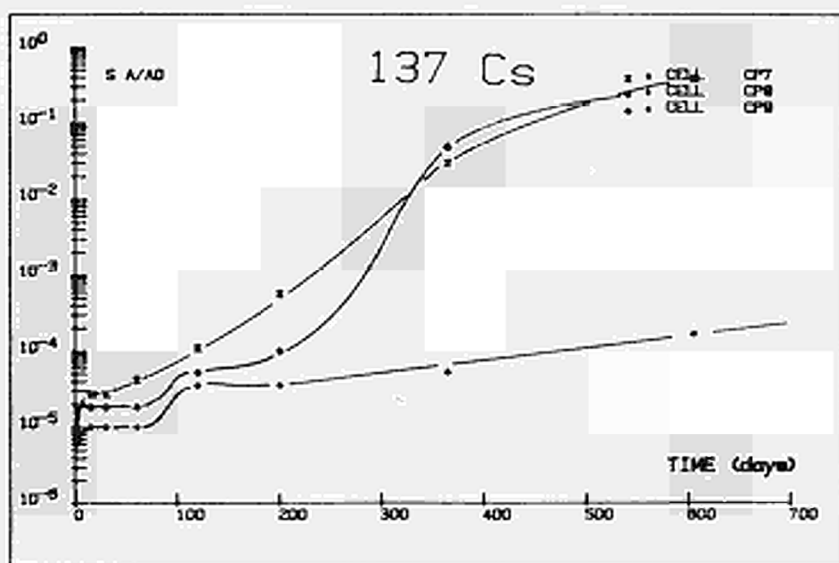


Fig. 2. Diffusion of ^{137}Cs in polyurethan "UREFLEX".

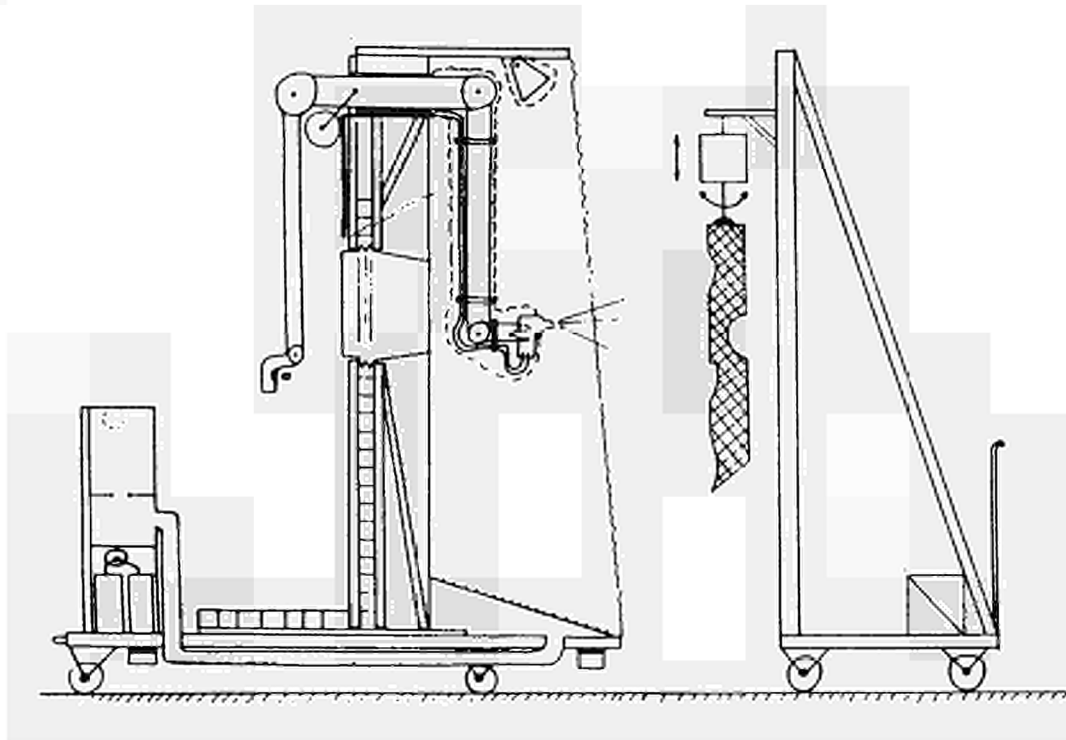


Fig. 3. Project of a coating mobile unit.

5. PROJECT N°5:

LARGE CONTAINERS FOR RADIOACTIVE WASTE PRODUCED IN THE DISMANTLING OF NUCLEAR INSTALLATIONS

A. Objective

Radioactive waste resulting from the dismantling of major reactor components must be transported in larger units than those at present used for other types of radioactive waste, in order to reduce the amount of cutting required and, consequently, the radiation exposure of personnel and the costs of the decommissioning.

B. Research performed under the 1979-83 programme

A system study has been performed, which made it possible to define the types of large transport and/or disposal container needed for bulky radioactive waste resulting from the dismantling of nuclear power plants.

C. 1984-88 programme

In the light of the results of the above-mentioned system study, large transport and/or disposal containers should be developed. The performances of the waste/matrix/container system under conditions representative of envisaged waste repositories should be studied. The control methods for verifying the suitability of the containers for land storage, sea dumping, transport, etc., according to the specific technical requirements for these different utilisations, will be considered.

D. Programme implementation

At the end of 1985, three research contracts were at the stage of negotiation.

6. PROJECT N°6:

ESTIMATION OF THE QUANTITIES OF RADIOACTIVE WASTE ARISING FROM DECOMMISSIONING OF NUCLEAR INSTALLATIONS IN THE COMMUNITY

A. Objective

The low-level radioactive waste produced in the dismantling of nuclear installations will ultimately constitute a substantial part of the overall volume of radioactive waste generated by nuclear industry. The objective of this project is to estimate the quantities of various categories of radioactive waste that will arise from the decommissioning of nuclear installations in the Community. This involves the definition of reference strategies for decommissioning and is therefore to be regarded as a long-term task.

B. Research performed under the 1979-83 programme

The following research work has been performed:

- analysis of concrete samples from various nuclear power plants in order to determine the composition and extension of long-lived radionuclides in shielding structures;
- analysis of steel samples in order to determine the composition of long-lived radionuclides in reactor components;
- preparation of a methodology for evaluating the radiological consequences of the management of very low level waste produced in the dismantling of nuclear power plants;
- review of the measuring techniques required for the purpose of deciding whether or not material from the dismantling of nuclear power plants is radioactive.

C. 1984-88 programme

Research should be performed in the following main areas:

- improved estimate of the quantities of radioactive waste arising from the decommissioning of typical nuclear installations, account being taken of the results of the first five-year programme (in particular Projects N°2 and N°6);
- study of strategies for the decommissioning of nuclear installations and for the management of the radioactive waste arising therefrom, account being taken of the waste disposal facilities existing or being developed in various member countries;
- characterisation of the radioactivity associated with components and structures of nuclear installations, with emphasis on long-lived radionuclides (analyses complementary to those performed under the first five-year programme); in-situ measurement techniques for the localisation and identification of radionuclides, including the case of mixtures of alpha, beta and gamma emitters;
- residual activity levels below which activated and/or contaminated parts could be re-used and corresponding measurement methods.

D. Programme implementation

At the end of 1985, three research contracts were at the stage of execution, and five contracts were at the stage of negotiation.

6.1. The Assessment of Low-Level Contamination from Gamma-Emitting Radionuclides

Contractor: Imperial College Reactor Centre, Silwood Park, UK
Contract N°: FI1D-0019
Working Period: October 1984 - December 1987
Project Leader: P.W. Gray

A. Objectives and Scope

The objective of this research programme is to evaluate a new analytical technique that should improve the precision of the inferences that can be made about radionuclide activity from area measurements of small-area spectral peaks obtained using multi-channel spectrometry.

These improvements are based on the application of Bayesian peak fitting, a method of peak fitting that allows the information contained in a spectrum to be used more fully than is possible with the method of gross counting, which is currently employed. It follows that activity estimates and confidence intervals for activity should be more precisely defined, and the resources required to obtain a specified detection limit should be reduced.

An assessment of the extent of this improvement, and of whether this improvement is sufficient to warrant using the slightly more complicated Bayesian approach, is the main objective of this research programme.

B. Work Programme

- B.1. Equipment procurement, installation, acceptance testing and planning.
- B.2. Collection of sample spectra and the assessment of spectral instability.
- B.3. Development of Bayesian peak fitting and the construction of Bayesian prior densities.
- B.4. Spectral simulation and peak fitting for different values of peak area, background level, and other relevant parameters.
- B.5. Construction of a hypothesis test that the peak area is zero, and the determination of its properties.
- B.6. Construction of an estimator for peak area, and the determination of its properties.
- B.7. Construction of a confidence interval estimator for peak area, and the determination of its properties.
- B.8. Generalisation of the hypothesis test to several radionuclides.

C. Progress of work and obtained results

Summary

Selection, installation and acceptance testing of a spectrum stabilizer and a microcomputer have been completed. Facilities for transferring spectral data from the spectrometry system to the microcomputer have been provided.

The standard Bayesian approach to peak fitting has been modified so that the dependence of peak area estimates on the characteristics of the spectrometry system employed can, to a large extent, be treated analytically rather than numerically. As a result, a more thorough simulation programme is possible than was originally envisaged, and the results of the simulation programme can be applied to any spectrometry system for which a Bayesian prior density can be constructed.

A general statistical framework has been established for the construction of Bayesian prior densities, and this framework has been applied to provide an exact solution for the Bayesian prior density associated with a stable spectrometry system.

Progress and results

1. Introduction

The techniques that are currently used to make inferences about radionuclide activity from area measurements of small-area spectral peaks are based on gross counting, and do not make use of the information that is available regarding permissible peak shapes; however, a technique that employs peak fitting allows this information to be taken into account. With a peak fitting technique, a parametric class of functions is fitted to the spectral peak; peak shape is described by a response parameter vector, which reflects the characteristics of the spectrometry system employed, and by a sample parameter vector, which reflects the characteristics of the sample being measured.

By calibrating the spectrometry system, a Bayesian prior density can be constructed for the response parameter vector; this density function describes the uncertainty in the response parameter vector arising from measurement uncertainties in calibration, and from the instability of the spectrometry system due to temperature drifts in the electronics. Conventional peak fitting can be adapted to make use of the information contained in this Bayesian prior density.

For the Bayesian peak fitting technique, the likelihood function of a small-area peak is multiplied by the Bayesian prior density of the response parameter vector and normalized to form a posterior Bayesian density; the maximum of this posterior Bayesian density is obtained, and used to estimate the sample and response parameter vectors of the small-area peak. The Bayesian prior density acts as a penalty function during the maximization procedure, improving the convergence properties of the algorithm employed. Furthermore, as the response parameter vector is well defined following system calibration, the dispersion of the posterior Bayesian density is somewhat less than that of the likelihood function (suitably normalized), and hence the uncertainty associated with each sample parameter, and with the peak area in particular, is reduced.

2. Decoupled Bayesian peak fitting (B.3.)

Due to the mathematical complexity of the likelihood function, the distributions of the statistics required to make inferences about the sample activity must be determined by simulation. Unfortunately, the posterior Bayesian density depends not only on the sample and response parameters associated with the small-area peak, but also on the auxiliary

parameters defining the shape of the Bayesian prior density. For most applications, these auxiliary parameters correspond to the elements of the covariance matrix of the response parameter vector; and the values of these auxiliary parameters will depend on the care with which the spectrometry system has been calibrated, and on fluctuations in the peak centroid due to gain and zero level instability. The dependence of the posterior Bayesian density on the auxiliary parameters adds considerably to the complexity of the simulation programme, and limits the range of spectrometry systems to which the results of the simulation programme can be applied.

A modification to the standard Bayesian approach has been developed that decouples the auxiliary parameters from the sample and response parameters, so that simulation with respect to the auxiliary parameters is not required. Instead of maximizing the posterior Bayesian density with respect to the sample and response parameters, only the likelihood function is maximized; however, the likelihood function is maximized with respect to the sample parameters alone, with the response parameter vector held fixed at a value equal to the mean of the Bayesian prior density. The probability density function of the peak area, obtained by simulation, will depend on the sample parameter vector, the response parameter vector, and the mean of the Bayesian prior density. This density can be regarded as a conditional density with respect to the response parameter vector; by randomizing this density with respect to the Bayesian prior density, the response parameter vector is removed and the auxiliary parameter vector is introduced, so that the randomized density depends on the sample parameter vector, the auxiliary parameter vector, and the mean of the Bayesian prior density. The efficiency of this approach is less than that of the standard Bayesian approach, but the loss is negligible given the high precision with which the spectrometry system can be calibrated.

Hence, since the auxiliary parameters of the Bayesian prior density are no longer involved in the simulation programme, a more thorough simulation programme is possible, and the results of the simulation programme can be applied to any spectrometry system for which a Bayesian prior density has been constructed.

3. A Bayesian prior density for a sourced peak and a stable spectrometry system (B.4.)

A Bayesian prior density has been constructed for a spectrometry system for which instability can be neglected - the uncertainty in the response parameter vector is largely due to measurement uncertainties arising during system calibration - and for which the peak of interest is "sourced" - a strong source producing the spectral peak that is to be measured is available.

A large spectral peak is obtained during calibration of the spectrometry system, and an estimate of the response parameter vector associated with this peak is obtained by maximum likelihood estimation. This process is repeated N times, so that the sampling distribution of the response parameter vector is determined empirically, with the sample mean, \bar{X} , and the sample covariance matrix, $A/(N-1)$, given by

$$\bar{X} = \frac{1}{N} \sum_{i=1}^N X_i \quad \text{and} \quad A = \frac{1}{N-1} \sum_{i=1}^N (X_i - \bar{X})(X_i - \bar{X})^T$$

respectively. It is assumed that the maximum likelihood estimator has a multivariate normal distribution, with mean μ and covariance matrix Σ .

The Bayesian approach can be applied not only to the estimation of the sample parameters associated with a small-area peak, but also to the calibration of the spectrometry system. The posterior Bayesian density obtained from the calibration of the spectrometry system becomes the Bayesian prior density used for estimating the sample parameters of a small-area peak.

The Bayesian approach may be applied to the calibration of the spectrometry system as follows. The joint probability density function of \bar{X} and A is regarded as the joint likelihood function of μ and Σ . An uninformative prior density must be chosen for μ and Σ , since no information exists about these parameters before calibration. By generalizing the uninformative prior densities most commonly chosen for these parameters in the univariate case, a joint prior density of the form

$$|\Sigma|^{-p/2}$$

is advocated, where p is selected to maintain consistency between Bayesian and frequentist inferences based on the same statistic. It has been shown that for an m dimensional response parameter vector, the joint posterior Bayesian density of μ and Σ is the product of a multivariate normal distribution and an inverse Wishart distribution:

$$n_m(\mu|\bar{X}, \Sigma/N) w_m^{-1}(\Sigma|A, N+p-1),$$

where

$$n_m(\mu|\bar{X}, \Sigma/N) = (2\pi)^{-m/2} |\Sigma/N|^{-1/2} e^{-(1/2)(\mu-\bar{X})^T(\Sigma/N)^{-1}(\mu-\bar{X})}$$

and

$$w_m^{-1}(\Sigma|A, N+p-1) = \frac{|A|^{(N+p-m-2)/2} e^{-(1/2)\text{tr}(\Sigma^{-1}A)}}{|\Sigma|^{(N+p-1)/2} 2^{m(N+p-m-2)/2} \Gamma_m[(N+p-m-2)/2]}$$

By integrating the joint density with respect to Σ , it can be shown that the marginal density of μ has a multivariate elliptical t-distribution:

$$\frac{\Gamma[(m+r)/2]}{(r\pi)^{m/2} \Gamma(r/2) |\Psi|^{1/2} [1+(1/r)(\mu-\bar{X})^T\Psi^{-1}(\mu-\bar{X})]^{(m+r)/2}}$$

where r equals $N+p-2m-1$ and Ψ equals $A/[N(N+p-2m-1)]$. The value of p can be shown to be $m+1$ by comparing the distribution of a certain function of μ with Hotelling's T-square statistic.

4. Spectrum stabilization and data transfer (B.1. and B.2.)

A Nuclear Data 595 digital spectrum stabilizer for use with a Nuclear Data 6600/6700 spectrometry system has been purchased to determine the range of values for the auxiliary parameters of the Bayesian prior density that are typical of commercially available spectrometry equipment. (B.1.)

Installation and acceptance testing of the stabilizer have been completed. A communications programme to transfer selected regions of interest within a spectrum to the mainframe computer has been written and tested (B.2.).

6.2. Development of Methods to Establish Curie Content of Radioactive Waste from Decommissioning Projects

Contractor: UKAEA, Windscale, UK
Contract N°: FIID-0020
Working Period: December 1984 - September 1987
Project Leader: F.G. Brightman

A. Objectives and Scope

A review is required of the impurity concentrations and the resultant long-lived radioactivities, in materials to be consigned to low and medium active disposal facilities. Sampling methods are to be developed which are applied along with analysis methods currently available, to demonstrate sufficiently detailed knowledge of beta, X-ray and gamma radioactivities from waste.

Development of calculation methods, and demonstration of their validity for assay of radioactivities in waste material in several geometries, is required as part of a decommissioning demonstration project.

The final objective of the programme is to provide an easily used and acceptable method of assay which will have wide application.

B. Work Programme

- B.1. Analysis of Co, Ni, Nb and low-level trace impurities in representative WAGR material samples.
- B.2. Development of suitable sampling methods.
- B.3. Review of present analysis data.
- B.4. Design and test of the final sampling/analysis scheme.
- B.5. Supply of samples.
- B.6. Design test of codes for curie assay.
- B.7. Tests using source array of Co-60 simulating tube, plate or mixed waste geometries.
- B.8. Revision of the codes using tests results.

C. Progress of work and obtained results

Summary

A review has shown that isotopes of cobalt, niobium, silver and europium are the principal gamma-emitters of importance. Initial determinations on WAGR steels and archive steel samples have demonstrated the concentration ranges of these impurities. The most important, cobalt, was present at 140-350 µg/g in mild, and 300-1500 µg/g in stainless steel. Sampling and analysis techniques are being explored further.

Encouraging progress has been made with a surface procedure for active steel, using abrasive paper followed by direct X-ray Fluorescence Spectroscopy. Further evaluation of neutron activation analysis for silver and europium measurements suggests that these may be difficult to determine if the original samples are already strongly active.

Code development was begun, with the objective of estimating the activity within a concrete waste box from measurements of the dose rate outside the box. A start was made on the WESTD code development, allied to the RANKERN main shielding code. Predictions of source activity were made with 25 percent accuracy. The general work progress status is as follows:

- . B.1., B.2. and B.5. are progressing normally,
- . B.4. and B.6. are delayed,
- . B.3. is completed.

Progress and results

1. Analytical development (B.1., B.2., B.3.)

Analytical research in support of Curie Assay can be divided into three main areas, (a) analysis of archived steel samples of relevance to WAGR, (b) analysis of WAGR samples and (c) development of remote sampling techniques suitable for use on activated steels. In collaboration with the Central Electricity Generating Board, we have shown that cobalt, niobium, silver and europium are likely to present the principal radiological hazard at the time of reactor decommissioning and disposal.

- Archived samples

A small number of archived steels of similar specification to those used in WAGR have been obtained from various sources and analysed for cobalt and niobium. In addition, previous analyses of steel by Harwell have been reviewed, and data for cobalt, niobium, silver and europium collated. These data show cobalt concentrations in mild steel to lie in the narrow range 140 - 350 µg/g, whereas stainless steels cover the range 300 - 1500 µg/g. Niobium levels are less than 100 µg/g except in niobium stabilised steels (e.g. Type 347 used for some thermocouple sheathing) where levels of 6000 - 8000 µg/g (0.6 - 0.8 percent) have been found. Silver shows the largest variation with levels between < 10 and 400 ng/g. Europium levels are < 200 - 400 pg/g. This work has highlighted the potential problem of sample inhomogeneity both in terms of batch to batch variation of same specification steel, and in terms of variation within a single casting. Silver appears the worst of the four trace elements in this respect.

- WAGR components

Analysis of WAGR components has been restricted to stored, spare components (e.g. loop-tubes) which are believed to be representative of those in the reactor. It has proven difficult to obtain WAGR pressure vessel samples because of the discovery of large quantities of asbestos lagging. This material is being removed, which will enable access to the required samples in 1986. The analysis of those components obtained fits the pattern outlined above for the archived samples. For silver however, in one high pressure loop tube sample, a concentration of 4000 ng/g was observed.

- Remote Sampling of Active Samples

Evaluation of a sampling procedure using abrasive paper followed by direct X-ray fluorescence measurement has continued. The silicon carbide paper used previously has been replaced with diamond paper. This change has reduced trace element levels in blanks, and now allows silicon to be measured in the steel. Silicon can be present at up to percent levels and its measurement may assist in the identification of steel type.

The computer programme used has been modified to measure the 16 most common elements in steels (which include both cobalt and niobium). As the sample adhering to a single piece of abrasive paper cannot be weighed, it is necessary to determine all the major and minor elements and express the concentration of the element of interest as a fraction of the sum total of amounts of major and minor elements.

For a series of Certified Reference Materials, the "rubbing" sampling technique is giving data in excellent agreement with the certificate for some elements, including niobium and cobalt, but is giving poor agreement for others. Work continues to investigate this discrepancy. Some typical results are given below for elements which consistently give good data:

	STAINLESS STEELS		MILD STEELS	
	NBS C1153	NBS 1170a	NBS 1167	NBS 1168
% Cr found	16.9	17.4	0.60	0.79
expected	16.69	17.42	0.54	0.69
% Ni found	8.90	9.00	1.09	2.14
expected	8.77	8.89	1.03	1.99
% Mo found	0.26	0.28	0.18	0.16
expected	0.24	0.25	0.20	0.19
% Nb found	0.46	-	NOT DETECTED	
expected	0.50	-	0.006	0.022
% Co found	0.12	0.09	0.15	0.03
expected	0.127	0.096	0.16	0.03

2. Sampling/Analysis scheme (B.4., B.5.)

Samples were supplied at intervals in 1985 for the analyses reported above. Investigations were made of the reasons for engineering difficulties (for example, the presence of asbestos lagging) and other problems which tended to delay simple preparation. The use of an optical spectrometer, such as the trolley-mounted, computer-assisted device used in steel manufacturing, was considered. However, towards the end of the year, it became clear that the analytical performance for Cobalt-59 determination would not be good enough. A 20 percent accuracy at 150 ppm of Co-59 is the target. In December 1985 investigation began for possible use of a laser microanalyser, as developed by the UK Central Electricity Generating Board for its gas-cooled reactors. Engineering design of a sampling facility was also in progress for use during Windscale Advanced Gas-Cooled Reactor decommissioning (1981-1995).

3. Code Tests in an experimental facility (B.6., B.7.)

Code development continued during 1985 but arrangements for an experimental test of code predictions were delayed into 1986. However, this delay is not expected to affect the overall work programme.

The objective of the code development is to estimate the activity within a box containing active waste from measurements of dose rate outside the box. The first version of the code assumed that the source activity would be represented by a polynomial expression with ten unknown parameters. Ten measurements of dose rate were supplied and the solution was obtained using standard computer routines. In tests the method was satisfactory in some cases but for others, negative source activities were predicted in some regions of the box. The method was therefore discarded.

The second version of the code assumed the source activity was not a continuous function of position. The box volume was assumed to be divided by planes parallel to the three axes into a number of sub-volumes. It was further assumed that the source was made up from a uniform source over the whole box plus a peak source in one of the sub-volumes. It was assumed that six dose rates around the box are provided. The attenuation between each dose point and each sub-volume was determined. Then, for each sub-volume in turn pairs of dose points were considered in turn to provide an estimate of the whole volume and sub-volume sources. Any negative results led to the rejection of the pair. At the end of this process the results were analysed to decide which sub-volume gave the best match to the dose rate input.

Two tests have been performed on the second version of the code. In the first there was a uniform source of activity throughout the volume. An established code (RANKERN) provided the dose rates at specified positions. The new code (named WESTD) calculated the activity from the dose rates. The original source activity was 3.3E10/s and WESTD gave a value of 2.7E10/s, a 20 percent difference. The second test had a source near one corner of the box, and no source throughout the rest of the box. The code WESTD predicted the source to within 20 percent again.

The two tests therefore gave predictions of good accuracy, but further tests are needed.

6.3. Systems for Contamination Measurements on Curved Surfaces

Contractor: Reaktorwartungsdienst und Apparatebau GmbH, Jülich,
Germany
Contract N°: FIID-0021
Working Period: July 1984 - January 1987
Project Leader: St. Pawlytsch

A. Objectives and Scope

Large quantities of low-level radioactive waste is produced during refurbishing, maintenance and dismantling of nuclear installations, which could be re-used or recycled. In order to fulfil authority regulations, precise and safe measurements methods should be used, even on curved surfaces (e.g. inside tubes and pipes).

The objective of this research is the development and testing of a detector system for measurement of very low-level radioactivity, even near background level, suitable for irregularly-shaped surfaces like inside small diameter tubes.

B. Work Programme

- B.1. Development of a basic electronic equipment, suitable for the existing various prototype round and flat detectors with integrated gas supply and analogic part; testing with prototype detectors in the laboratory and under real conditions (KRB-A, Gundremmingen); development of further detectors to complete the range.
- B.2. Development of an optimised stationary and portable digitally working unit with background subtraction; development of semi-automated or automated measurement systems for irregular surfaces and improvement at laboratory scale.

C. Progress of work and obtained results

Summary

In 1984, the firm Reaktorwartungsdienst und Apparatebau GmbH in Jülich set itself the task of developing proportional counters which could also detect surface contamination on irregularly-shaped surfaces. Eighteen months later, Proportional Counters were available in series production in round or flat form. It was frequently demonstrated that the measurements are extremely well suited for the detection of surface contaminations in inaccessible positions.

In the 2nd semester of 1985, work was initiated with respect to an automatic or semi-automatic measurement rig designed to carry out clearance measurements on decontaminated condensor tubes. The particular emphasis of the development work in this period laid on the completion of the hand-held device as a prototype for series production. However, this prototype has not been completed yet, because of unexpected difficulties with the integrated gas supply. Parallel to this work, the calibration trials were continued, the leakage problems on the detector solved, the first range of detectors produced and the flat counter further perfected. Success was achieved in the development of a very flat counter which will have a maximum height of only 20 mm and an automatic device for seal testing.

At the end of 1985, extensive measurement programmes were initiated in the nuclear power stations Brunsbüttel, Gundremmingen and Krümmel.

Progress and results

1. Development and testing of a stationary system with analog-electronic part for measurement of contamination on irregular surfaces and in inaccessible places (B.I.)

In various trials in the nuclear power stations of Gundremmingen, Brunsbüttel and the experimental reactor in Jülich, it was clearly established in May/June 1985 that the measurements were suitable for the detection of surface contamination even in inaccessible places. For the first time, direct contamination measurement could be carried out in the interior of scaffolding pipes or other piping.

The detectors operate with a special cable, also newly developed, via a stationary measurement rig and an interconnected adapter. This special cable contains not only the electrical conductors but also the tubes for the gas supply. For this cable, these detectors and this adapter, a special plug connection was developed. From April until July 1985, these plug connections were submitted to continuous testing. In September both could be accepted as appropriate for series production.

In 1986, the following detectors will be available:

- Round PZ 15 R
for contamination measurements in tubes with an interior diameter of more than 22 mm,
- Round PZ 30 R
for contamination measurements in tubes with an interior diameter of more than 35 mm,
- Round PZ 50 R
for contamination measurements in tubes with an interior diameter of more than 57 mm,
- Flat with Side Window
for contamination measurements in channels and U-profiles with an interior width of more than 34 mm,

- Flat without Side Window
for contamination measurements of smooth surfaces.

All detectors are constructed for hand use, consisting of special proportional flow counters with integrated input discriminator. The discriminator and the pre-amplifier are incorporated into the hand-grip. Only because of this arrangement the new contamination counters can be applied as a measurement system (interchangeable).

Larger scale measurement trials started at the end of 1985 after developing reasonable calibration compounds with the corresponding reference geometries. After several attempts, success was attained in May 1985 in testing appropriate standard compounds. For these tests, the nuclides Sr-90, Am-241 and Co-60 were used as reference sources. Their suitability was established in various laboratory experiments and also in on-site measurements. The reference activities are 37 Bq or 370 Bq. In exceptional circumstances as, for example, in plateau measurement, activities of 3700 Bq or 37000 Bq may be used.

Parallel to the development of the proportional counters and the necessary electronics for hand use, appropriate manufacturing aids were worked out for a later series production. In this respect, the winding device, the automatic seal tester and the so-called detector test device should particularly be mentioned.

The detector test device was developed for automatic plateau measurements. It was possible in April/May 1985 to present a completed test device. At the end of 1985, all plateau measurements were carried out automatically by means of this device.

Towards the end of 1985, the construction of a rig for the automatic or semi-automatic clearance measurement of condenser tubes was initiated.

7. PROJECT N°7:
INFLUENCE OF PLANT DESIGN FEATURES ON DECOMMISSIONING

A. Objective

The objective of this project is to identify and develop reasonable improvements in the design of nuclear installations with a view to decommissioning.

B. Research performed under the 1979-83 programme

Activities on the following subjects are in progress:

- control of the cobalt content of reactor steels and testing of cobalt free materials to substitute cobalt alloys;
- surface coatings to protect concrete against contamination;
- reactor shielding design features that facilitate dismantling;
- documentation system for deferred decommissioning;
- review and catalogue of design features facilitating decommissioning.

C. 1984-88 programme

Some of the subjects studied under the 1979-83 programme are expected to need continued development under the 1984-88 programme. In addition, design features of certain fuel-cycle installations (e.g. reprocessing plants) should be examined with a view to decommissioning.

D. Programme implementation

At the end of 1985, one research contract was at the stage of execution, and four contracts were at the stage of negotiation.

7.1. Decontamination and Remote Dismantling Tests in the ITREC Reprocessing Pilot Plant

Contractor: ENEA/Trisaia Energy Research Centre, Policoro, Italy
Contract N°: FIID-0022
Working Period: July 1985 - June 1988
Project Leader: T. Candelieri

A. Objectives and Scope

The ITREC plant was originally conceived and built as an integrated unit for reprocessing and refabrication of fuel elements. Fuel elements containing uranium and thorium are processed without separation of the fission products. Moreover, the processed material contains Th-228, a strong gamma emitter. The refabrication is, therefore, carried out in a cell fitted with adequate shielding, using remote-operated equipment and techniques. All equipment belonging to the main chemical process is installed in modular units, which provide for remote-controlled removal after appropriate decontamination of the individual unit (rack) for maintenance and modification of equipment (Rack Removal System). This system allows the remote transfer of process equipment from the hot cell to the decontamination cell and its decontamination to levels low enough to permit safe access for the workers of maintenance operations.

The ITREC plant has been operated under hot conditions from 1975 to 1979.

The scope of this research is to evaluate the advantages of the Rack Removal System in the dismantling of reprocessing installations. The objective of this work is to verify experimentally the possibility of the decontamination of any particular module and the capability of the remote dismantling of components installed in the mobile rack. In particular, the main objective is to develop remotely operated equipment for the dismantling of centrifugal contactors.

B. Work Programme

- B.1. Design and construction of cutting equipment for dismantling the centrifugal contactors of Rack 6 bis in the ITREC plant.
- B.2. External and internal decontamination of Rack 6 or 6 bis, with a first operation in the hot cell, followed by complete cleaning in the decontamination cell.
- B.3. Testing of dismantling by remote cutting of the centrifugal contactors with the highest contamination.
- B.4. Design and construction of a storage container for the conditioned dismantled centrifugal contactors.

C. Progress of Work and Obtained Results

Summary

The feasibility study of the dismantling device has been developed on the basis of previous experience carried out in the ITREC plant for decontamination and maintenance works. Maintenance and post-accident experience is being used in planning and the forthcoming decommissioning project in order to reduce work exposure and costs.

The dismantling device will set up in the decontamination cell, where it will operate, with the help of two manipulators and using video camera, to handle fixtures to support pipes during cutting and subsequent removal and transport of the centrifugal contactor. After transfer of the major radioactive source (centrifugal contactor), the access into the decontamination cell will be allowed for intervention. These operations include the centrifugal contactor replacement on rack 6 bis and the welding pipe. The "operative unit" of the original project is shown in Figure 1; appropriate modifications will be brought on the basis of cutting tests done with different laboratory mechanical cutting methods.

Progress and Results (B.1.)

A dismantling device (basic project) has been completed in six months according to the 1st step of the programme. The conceptual design has been developed on the basis of the general philosophy of the ITREC pilot plant which enables remote maintenance operations on equipment. The "Rack Removal System" provides the remote-controlled removal of the single equipment's rack from the hot cell (chemical process) to the decontamination cell. The present project is adequate to accomplish remotely operations for removal of the single centrifugal contactor from the rack 6 bis.

The dismantling device consists of:

- carrying structure
- operative unit.

The whole device has been developed considering the characteristics of the facilities available on the ITREC plant as well in the decontamination cell and in the rack 6 bis itself. In order to facilitate working conditions, some improvements have been made on the rack 6 bis and others will be necessary on the decontamination cell to adapt it for this purpose.

Some cutting techniques have been tested in order to optimise the cutting tools for pipe.

The feasibility study of the dismantling device and cutting pipe tests has been completed; research to develop remotely operated cutter-unit is under way in the ENEA Laboratory in order to adapt multiple saw-blade or commercially available shears.

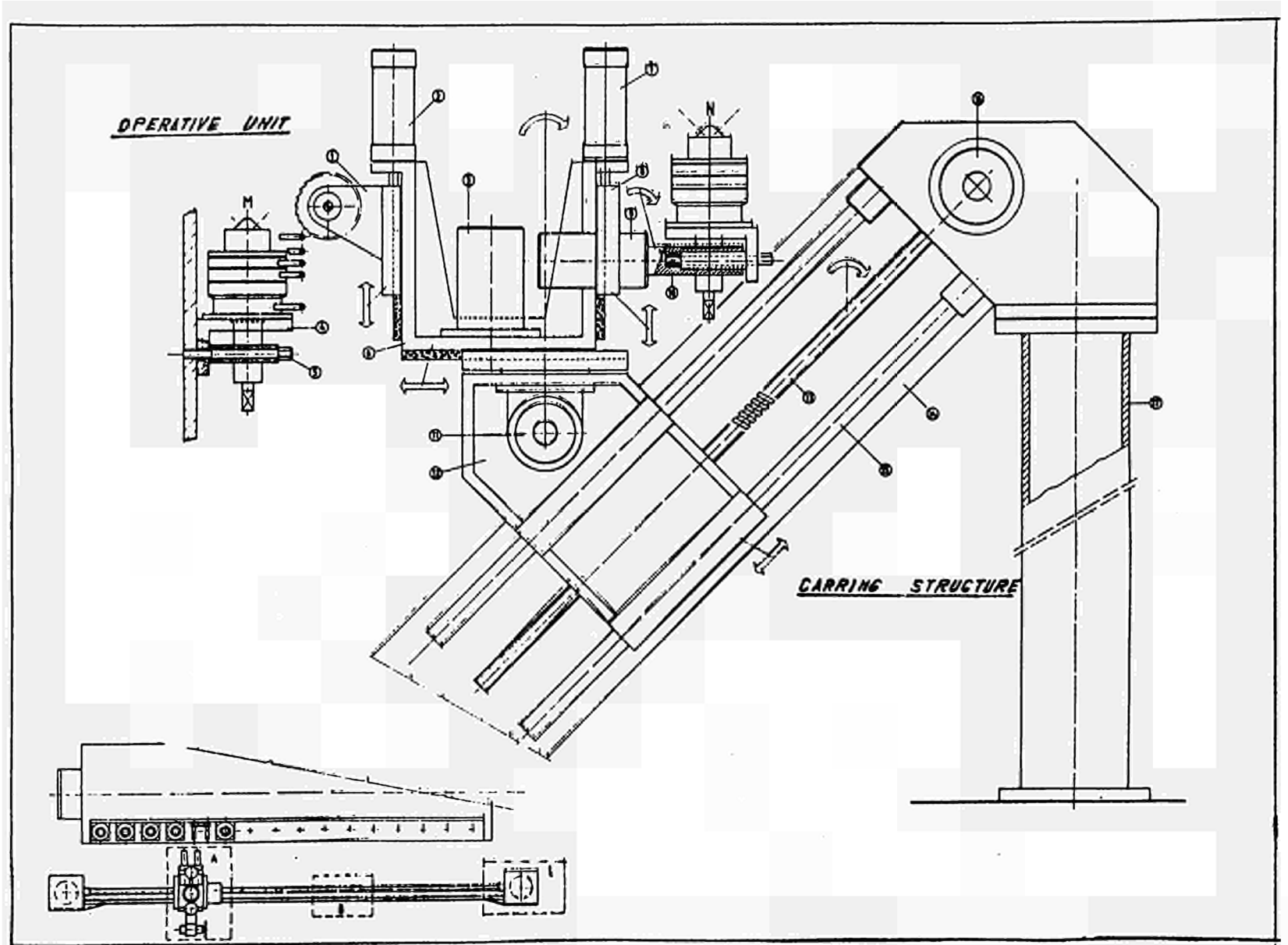


Figure 1. View of dismantling device

8. SECTION C:

TESTING OF NEW TECHNIQUES UNDER REAL CONDITIONS

In the course of the progressive development of new techniques, ever greater importance will attach to the testing of these techniques under representative conditions, in particular the presence of radioactivity. Industrial decommissioning operations undertaken in Member States would offer valuable opportunities for such testing. Because of the importance of this subject, it has been added to the 1984-88 programme as a separate section.

At the end of 1985, six research contracts were at the stage of execution, and six contracts were at the stage of negotiation.

8.1. Dismantling and Decontamination of a Feedwater Preheater Tube Bundle of Garigliano BWR

Contractor: Ente Nazionale per l'Energia Elettrica, Rome, Italy
Contract N°: FIID-0023
Working Period: November 1984 - October 1987
Project Leader: A. Bertini

A. Objectives and Scope

The decontamination for decommissioning purposes has not yet been applied extensively for the total cleaning of large components.

In the frame of heat-exchanger decontamination, only soft chemicals have been applied on large scale and unrestricted release levels have never been obtained. Many problems are connected with tube bundles which have very large surfaces and which are contaminated both inside and outside.

The scope of the present work is to demonstrate the feasibility of dismantling and decontamination of a large component coming from a first generation BWR (Garigliano). Experience gained in other plants will be taken into account, in the sense that the decontamination of the shell, and probably of the tube-sheet, may be carried out by electrochemical way. This study will be mainly focused on the decontamination of tube bundles.

Moreover, the estimation of the amount and the composition of secondary wastes produced is an aim of the work. Finally, the importance of decontamination techniques in decommissioning and, in particular, for the unrestricted release of turbine house building parts, of systems and components, will be evaluated.

The study will result in the assessment and qualification of an effective and economic technique for the decontamination of large and complex components with a production of secondary wastes in limited quantities.

B. Work Programme

- B.1. Preliminary evaluation of the characteristics of the selected feed-heater, including operating data, with respect to water chemistry and radioactivity inventory.
- B.2. Determination of the radioactivity inventory of the feedheater including measurements on scrap samples.
- B.3. Laboratory investigations on the ultrasonic and chemical procedures on representative samples, including tests on an appropriate treatment of the spent decontaminant.
- B.4. Definition and selection of the most suitable procedure for the determination of the residual activity inventory.
- B.5. Design and construction of an appropriate decontamination facility.
- B.6. Dismantling and decontamination of the feedheater, treatment of the spent decontaminant, conditioning of the secondary wastes and determination of the residual activity inventory.
- B.7. Evaluation of obtained results and final assessment for potential application to components of full-size BWR plants.

C. Progress of work and obtained results

Summary

The preliminary investigations which led to the choice of the feed water preheater to be dismantled and decontaminated are reported.

The present situation of the BWR Garigliano nuclear power plant (with details of the history of its operation), the water chemistry of the cooling cycle, the radiometric features of the steam-condensate-feed water cycle, the design and construction features of the four feed water preheaters are described in particular.

Any characteristics were insufficient to recommend a preheater above the others. It was therefore decided to conduct activities on preheater N° 4 since it is the one which has operated at the highest temperature and it is situated at floor level.

The metallographic and chemical characterizations of a tube exchange sample of the preheater N° 4 were performed. The thicknesses of the oxide layers measured on the tube sample were: 1-8 μm on the inside surface and 2-10 μm on the outside surface. The presence of the species SiO_2 , $\alpha\text{Fe}_2\text{O}_3$ and αFe was detected on both the inside and the outside surfaces by X-ray diffraction.

The preliminary tests with the ultrasound and chemical decontamination process were reported too. The results show that generally the combined process is more effective than the sum of the single contributions.

Progress and results

1. Introduction

The Garigliano nuclear power plant is equipped with a boiling water forced circulating system reactor, dual cycle, with a turbo-nuclear unit of 160 MWe. The plant went into commercial operation in April, 1964, was operated by ENEL and produced $12.5 \cdot 10^9$ kWh with a utilization factor equal to 68%, until August, 1978, when it was closed down because of a breakdown in a secondary steam generator /1/. In March, 1982, the plant was definitely declared out of commission.

The radioactivity contained in the turbine building due primarily to Co-60 and Cs-137, is a very small part of the total inventory of radioactivity in the plant; nevertheless, this is spread over very wide surfaces with specific levels of contamination which, on the inner surfaces of the systems, vary from 3 to $6 \cdot 10^3$ Bq/cm² and, on the outer surfaces, from 0.04 to 4 Bq/cm².

It is advisable to undertake massive decontamination operations in order to rid the turbine building of contamination, with the aim of reaching the limit of surface radioactivity which will allow the material to be released unconditionally. These operations, on account of the variety of components and materials used and the difference in type and level of contamination, require several different techniques of decontamination /2/.

2. Operational data (B.1.)

The water chemistry of the condensate-feed cycle can be divided into two distinct periods, separated by the replacement of the feed water preheaters in 1968 /3/. This replacement was made necessary principally by problems of core load losses, caused by the orifices of the fuel components being blocked due to a high rate of deposit. Before 1968, the N° 1 and 2 preheaters had the tube-bundles in Cupro-Nickel and N° 3 and 4 in Monel; afterwards preheaters with the tube-bundles in AISI 304 L stainless steel were installed.

The differences in the chemistry of the water are displayed above all in the concentration of the corrosion products, which decreased notably after the above-mentioned period. In particular the concentration of insoluble iron, in the form of Fe_2O_3 and $Fe(OH)_3$, to vary, remaining on an average around 15 ppb while the concentrations of Cu and Ni are considerably reduced.

After evaluating all the information about the preheaters it was decided to conduct the experiments on preheater N° 4 for the following reasons:

- it was the last one before entry into the secondary steam generators and is therefore the one where the feedwater operated at the highest temperature, the contaminated oxide should have formed in a worse condition (more compact and tenacious) with respect to that in the other preheaters;
 - it is situated at level +10.00 (floor level) and therefore it has more space to move and the insulation has been already partly stripped.
- Some views of the preheater are shown in Figure 1 and 2.

3. Measurement of total contamination (B.2.)

For measurement of the total surface contamination (smearable and fixed) samples were taken by removing the oxide layers and deposits (crud) from known surface of measurement. The samples were drawn both directly on site, by means of appropriate tools chosen to remove the whole of the contamination on the surface under examination, and after removal on site of radiologically representative samples and subsequent removal in the laboratory, using appropriate mechanical and/or chemical methods. Measurement of the samples was carried out by means of Ge-Li gamma spectrometry in known geometry, after appropriate treatment of the samples.

The results of the measurement of total surface specific radioactivity are reported in Table I. In all the samples, the composition in percentage of the various radionuclides shows that Co-60 predominates, with percentages around 90. For the feed water preheater N° 4 the total radioactivity is estimated in 10^3 MBq.

4. Metallographic and chemical characterization of oxides (B.3.)

The morphological characterization of the contaminated oxides and deposits which cover the surfaces of the exchange tubes has been performed by Scanning Electron Microscope.

The outside surface of the tubes appears to be covered by regular iron oxide crystals with size 0.2-3 μm . The inner surface appears to be similar but the crystals are not regular and are more dispersed. The surface microanalysis by EDS showed, apart Fe, the presence of Si and Cr at low concentration.

In order to evaluate the thickness of the oxide layer a tube sample was encapsulated, prepared and polished for optical microscopy observations. The oxide layer thickness does not appear regular but many spots are present. Inside the tube, the thickness ranges from 1 to 8 μm , while outside it ranges from 2 to 10 μm . Some observations by SEM with Cross section microanalysis by EDS on the outside surfaces show the presence of Cu in the oxide.

For evaluating the chemical elements and the crystallographic species present in the oxide layers, analysis by X-ray fluorescence (XRF) and X-ray diffraction (XRD) were done. The results show the presence of SiO_2 , $\alpha\text{Fe}_2\text{O}_3$ and αFe on both the inside and the outside surfaces of the preheater tube samples.

5. Laboratory tests on the ultrasonic and chemical process (B.3.)

The decontamination process chosen for the tube bundle of the preheater is the application of ultrasounds in a chemical bath of inorganic acids. Strong mineral chemicals such as HF, HNO_3 , HCl and their mixtures, have been selected taking into account the results of previous investigations on the chemical decontamination of AISI 304 Garigliano materials (from primary recirculation loop) /4/.

Some preliminary tests on the ultrasound and chemical process were performed in order to have a better basis for the future tests on the tube bundle materials. The batch tests were carried out in two common little ultrasound tanks, the test specimens were small flat pieces and/or pieces of little tubes.

The preliminary test materials were: aluminium, oxidized (not contaminated) carbon steel, non-oxidized AISI 304 stainless steels and contaminated AISI 304 stainless steel from 24" primary pipe of the Garigliano power plant.

The test chemical solutions were: NaOH 0.8% wt (for aluminium tests), HCl 4% vol. and HF 0.35% vol. + HNO_3 5% vol.. The concentrations of HCl and of HF- HNO_3 mixture were chosen because they were not effective on contaminated samples during a chemical treatment (without ultrasounds) Of course many tests were performed in demineralized water in order to have reference values.

The tests were started at room temperature; no degassing and no solution stirring were performed before and during the tests.

The effectiveness of the process on non-active specimens were evaluated on the base of weight less measurements. In the tests with contaminated specimens the decontamination effectiveness was defined as the ratio between the initial and final Co-60 radioactivity counted by a Na-I detector in a shielded box.

The preliminary analysis of the radioactivity data, in terms of DF, show the following relationship:

$$DF = DF_U \cdot DF_C \cdot DF_S$$

where: DF, is the total decontamination factor with the ultrasound and chemical process; DF_U , is the decontamination factor for the process with ultrasounds without chemicals; DF_C is the decontamination factor for the process with chemicals without ultrasounds; DF_S , is adjunctive DF due to the synergic effect of ultrasounds and chemicals. DF_S is a complex function of many parameters such as kind and concentration of chemicals, time, material, type of surface and contamination and so on.

In the cases investigated it appears that DF is greater for the $HF+HNO_3$ solution than for the HCl solution.

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Table I - Total contamination in preheater N° 4

Surface	Sample	Sampling area (cm ²)	Crud (mg)	Contamination (Bq/cm ²)	Co-60 (%)	Cs-137 (%)
outside (steam)	outside exchange-tube	50	161	3	88	12
inside (feedwater)	inside exchange-tube	33	153	60	100	-
shell (inside)	man way	100	814	6	99	1

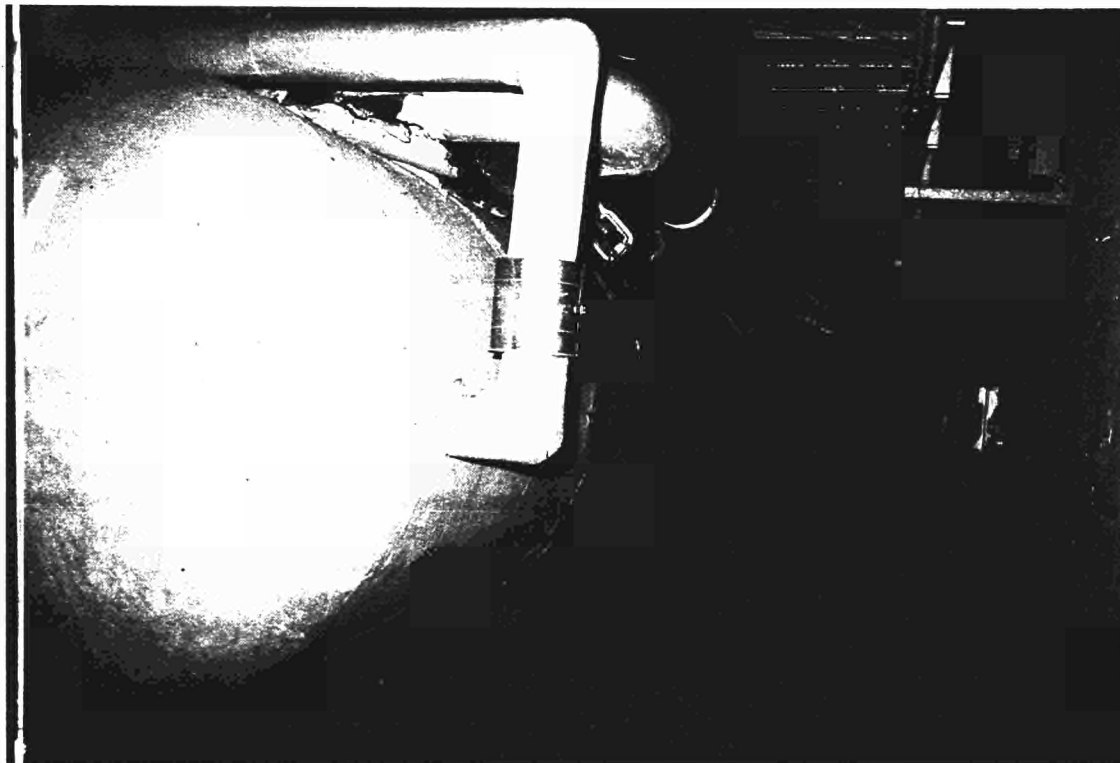


Figure 1 - Views of preheater N° 4.

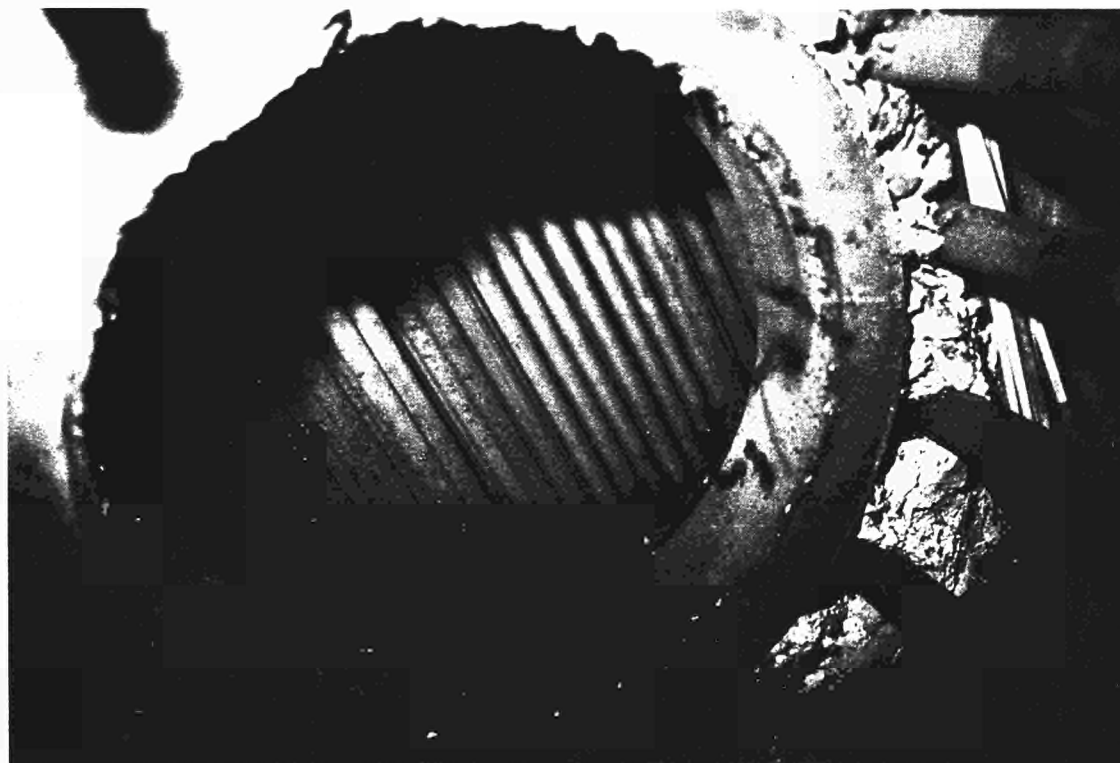


Figure 2 - Views of the man-way with the sampling area.

8.2. Conditioning, Transport and Dismantling of Very Large Plutonium Glove-Boxes

Contractor: Belgonucléaire, Dessel, Belgium
Contract N°: FIID-0024
Working Period: July 1984 - June 1986
Project Leader: J. Draulans

A. Objectives and Scope

The decommissioning of standard-sized plutonium glove-boxes has been performed in several countries since several years. However, the dismantling of very large alpha-radiating units has yet to be demonstrated.

Plutonium research laboratories and mixed-oxide fuel fabrication plants have to be partially dismantled in the near future. During these dismantling tasks, severe problems will arise with the decommissioning of huge glove-boxes containing very large and heavy equipment. These units have to be conditioned and transported to an ad-hoc installation for dismantling and final disposal. The techniques used until now for the conditioning and the transport of small units are not applicable in this field. Indeed, new techniques have to be developed for assuring at any time the leak-tightness of such units up to the moment of their dismantling.

The aim of the research is to develop concepts needed and to execute and demonstrate decommissioning operations on five large glove-boxes of the Dessel mixed-oxide fuel fabrication plant. These operations include conditioning, transportation on public roads to an external dismantling cell, dismantling and assessment of applied techniques.

B. Work Programme

- B.1. Conditioning for allowing safe transportation of five large plutonium glove-boxes, formerly used for mixed-oxide fuel fabrication .
- B.2. Preparation and safe and leak-tight transportation of five large glove-boxes to a special dismantling installation.
- B.3. Adaptation for air-tight introduction of the glove-boxes into the dismantling cell, execution of the dismantling by a selected appropriate procedure and final assessment of the applied techniques with recommendations for further applications.

C. Progress of work and obtained results

Summary :

5 large glove boxes have been removed from the BELGONUCLEAIRE Mox plant and transferred directly to the SCK/CEN dismantling plant. 2 glove boxes have been completely dismantled. Their parts are stored in final disposal drums, filled with concrete. The 3 others are in a temporary storage awaiting final dismantling.

BELGONUCLEAIRE has chosen a dismantling in a specialized institute, so special authorizations were required for the transportation of these glove boxes over the public road. At the same time the problems of removing the glove boxes out of the plant building, their transportation and their introduction into the dismantling cell were threatened.

Most of the glove boxes have been rotated over 90°. A special technique was used : a solid metal frame was built around every glove box to be rotated. The heavy equipment inside was then fixed to this outer frame to avoid any force transmission of the equipment on the glove box frame; furthermore a special glove box rotating system was built and tested for the glove box rotating operation.

After careful cleaning, 4 heavily contaminated glove boxes have been rotated without any major problem and without any contamination release out of the glove box. All 5 have been packed and transported without major problems.

The major problem encountered during the dismantling of the first glove box was the strong increase of the plutonium concentration in the air of the dismantling room, requiring supplementary actions for fixing the contamination on the glove box inner walls and the equipment walls.

Progress and Results

1. Conditioning for allowing safe transportation of five large Pu glove boxes, formerly used for mixed oxide fuel fabrication. (B.1.)

First the glove boxes have been separated from each other and from the surrounding ones by removing the equipment installed between 2 glove boxes in plastic tunnels, reinforced by means of wirescreens. Some difficulties have been encountered during the removing of the rather heavy equipment (up to 100 à 200 kg).

Once isolated, each glove box has been carefully cleaned, dry and wet. Due to the high plutonium presence, up to 194 g Pu after cleaning, strong limits were imposed on the quantity of water allowed in the glove box. Furthermore up to 63 operators have been cleaning during an average of 6,6 h/operator on one glove-box due to the high dose rates. Information concerning the glove boxes and the cleaning operation is given in tabel I.

The smaller equipment, difficult to stow was then removed and the remaining parts carefully stowed : in some cases by welding metal strips between different parts. In any case care has been taken not to fix the inner equipment on the upstanding walls or the roof of the glove box.

All surfaces inside the glove box were then covered with a contamination fixing product and all glove and lockports closed by means of an inner cover.

Each glove box to be rotated was then surrounded by a strong metal frame called "Stiffening frame, onto which the heavy equipment inside the glove box was fixed by means of special built hooks passing through glove or lockports. The location of these hooks was such that the heavy equipment was completely supported by the outer frame and the 20 mm thick

glove box bottom in order to avoid any supplementary load on the glove box frame during the lifting and rotating .

All empty voids were then filled with a packing material.

The stiffened glove box was then rotated in a special built frame for lifting and rotating. Figure 1 shows both the stiffened glove box and the rotating frame. 4 of the 5 glove boxes had to be rotated. No mayor problem was encountered during these operations, carried out without contamination release.

The glove box in its transportation position was then installed on the wooden packing bottom, covered by 2 plastic layers. Each of the plastic layers was then used as the bottom of a complete plastic bag surrounding the glove box. Only an air breathing filter was connected through the plastic bags to the atmosphere. A wooden transportation case was then installed around the packed glove box. Lifting chains were connected through the case and the plastic bags on the stiffening frame.

Information concerning these tasks for glove box A13 is grouped in table II.

2. The transportation of the glove boxes.(B.2.)

Procedures for transportation on the public roads were established during details discussions, competent authority approvals were obtained on the basis of these procedures.

No major problems have been encountered during the truck loading and unloading operations and the transportation itself.

3. Adaptation for air tight introduction of the glove boxes into the dismantling cell, execution of the dismantling by a selected appropriate procedure and final assessment of the applied techniques with recommendations for further applications. (B.3.)

The outer dimensions of some stiffened glove boxes are such that the whole unit could not enter into the entrance lock of the alpha area. Therefore a lock chamber, called "reception centre" had to be constructed in front of the entrance lock, into which a part of the stiffening frame could be dismantled so that the glove box could be introduced into the alpha area through the entrance lock.

The "reception centre" was made from dismountable wooden panels covered with polyethylene sheet. Access to the reception centre could if necessary be done through a safety lock. Continuous alpha counting, depression meter and a turnable camera were installed. All the operators and supervisors involved in the preliminary dismantling of the stiffening frame were protected by means of frogmensuits.

First the 2 plastic bags around the glove box have been removed, no contamination was found on the plastic. The work proceeded by the dismounting of a part of the stiffening frame. Only the side parts of the stiffening frame had to be dismantled, a width reduction of about 12 cm being sufficient, so the equipment inside the glove box could still be suspended onto the stiffening frame. The removed stiffening parts were not contaminated. The glove box is then introduced through the entrance lock of the alpha room into the latter. The dismantling was performed by a crew of 5 operators. 31 drums, with a capacity of 200 l, were filled with dismantled parts and bitume. The main problem encountered during the dismantling was the high contamination level in the alpha area requiring successive cleaning and contamination fixing operations.

TABLE I

Some data concerning the glove boxes and their cleaning operation.

Glove box number		A01	A10	A11	A12	A13
Glove box name		Dosing PuO ₂ UO ₂ and scrap	Outlet Calc. Furnace	Scrap Transport	Scrap Homo- genization	Scrap Milling
Glove box	dimensions in m (lxwxh)	3.2x1.0x(2.0+2.9)	2.7x1.0x3.5	1.5x1.0x3.5	1.5x1.0x3.4	2.4x1.0x4.1
	volume in m ³	6 + 1	9.3	5.2	5.0	10.0
	window surface in m ²	17.9	17.0	16.2	15.7	26.5
	number of gloves	62	65	30	43	63
Average/highest gamma dose rate (in mR/h) contact window <u>before</u> cleaning		80,5/180	76,25/200	17,16/21,5	82,1/150	27,9/41,3
Average/highest gamma dose rate (in mR/h) contact window <u>after</u> cleaning		24,6/37	82,25/113	7,97/9,4	54,85/67	9,87/13,7
<u>During cleaning :</u>						
- number of operators involved		55	63	19	47	60
- required working time in hr.		297	416	80	263	248
Quantity of Pu in g						
- present after cleaning		194	179	85	103	78
- removed during cleaning op.		653,9	282,3	80,6	169,1	244,3
Quantity of solid waste pro- duced in l.		932	876	1072	932	1020

TABLE II

Decommissioning of glove box A13 - Grouped data.

	NUMBER OF OPERATORS INVOLVED	REQUIRED WORKING TIME IN HOURS	TOTAL GAMMA DOSE PEN- DOSIMETER mR	TOTAL GAMMA DOSE FINGER- BADGE IN mR	AV. GAMMA BODYDOSE AS RECEIV. BY 4 MAIN OPER. IN mR	QUAN. OF ALPHA CONTAM. WASTE (vol. in l)				WEIGHT IN KG OF COND. WASTE	QUANTITY OF SUSPECT WASTE IN l	
						in 28 l drums		in 200 l drums			in 28 l drums	
						N° of drums	Total vol.	N° of drums	Total vol.		N° of drums	Total vol.
Glove box disconnecting operation	2,-	72,-	166,-	n.a.	n.a.							
Glove box cleaning operation	60,-	248,-	1527,-	6990,-	n.a.	15	420	3	600		-	-
Glove box displac. prep. + displac.	n.a.	567,4	n.a.**	n.a.	706,5							
de-mounting of com- frames	n.a.	621,-	n.a.	n.a.	268,5	4	112	none	none		11	308
mis- lifting + sion- rotation	n.a.	80,-	n.a.	n.a.	8,-							
ing packing	n.a.	217,-	n.a.	n.a.	22,5							
Glove box dismantling + destruction and conditioning	5	1540,-				-	-	31	6200		-	-
TOTALS		3345,4				19	532	34	6800	10.000	11	308
Conclusions	Ratio	$\frac{\text{Effective Volume}^*}{\text{Glove box Volume}} = \frac{2,48}{10} = 0,248$				Ratio $\frac{\text{Volume Final Dispos. Drums}}{\text{Volume of Glove Box}} = \frac{6,2}{10} = 0,62$						
	Ratio	$\frac{\text{Weight of condit. Final Disposal drums}}{\text{Weight of glove box + stiffening frame}} = \frac{10.000}{4.500} = 2,22$				Ratio $\frac{\text{Vol. of Final Disposal Drums}}{\text{Effective volume}} = \frac{6,20}{2,48} = 2,5$						
* Effective volume : real volume of equipment and structural components ** n.a. : not available												

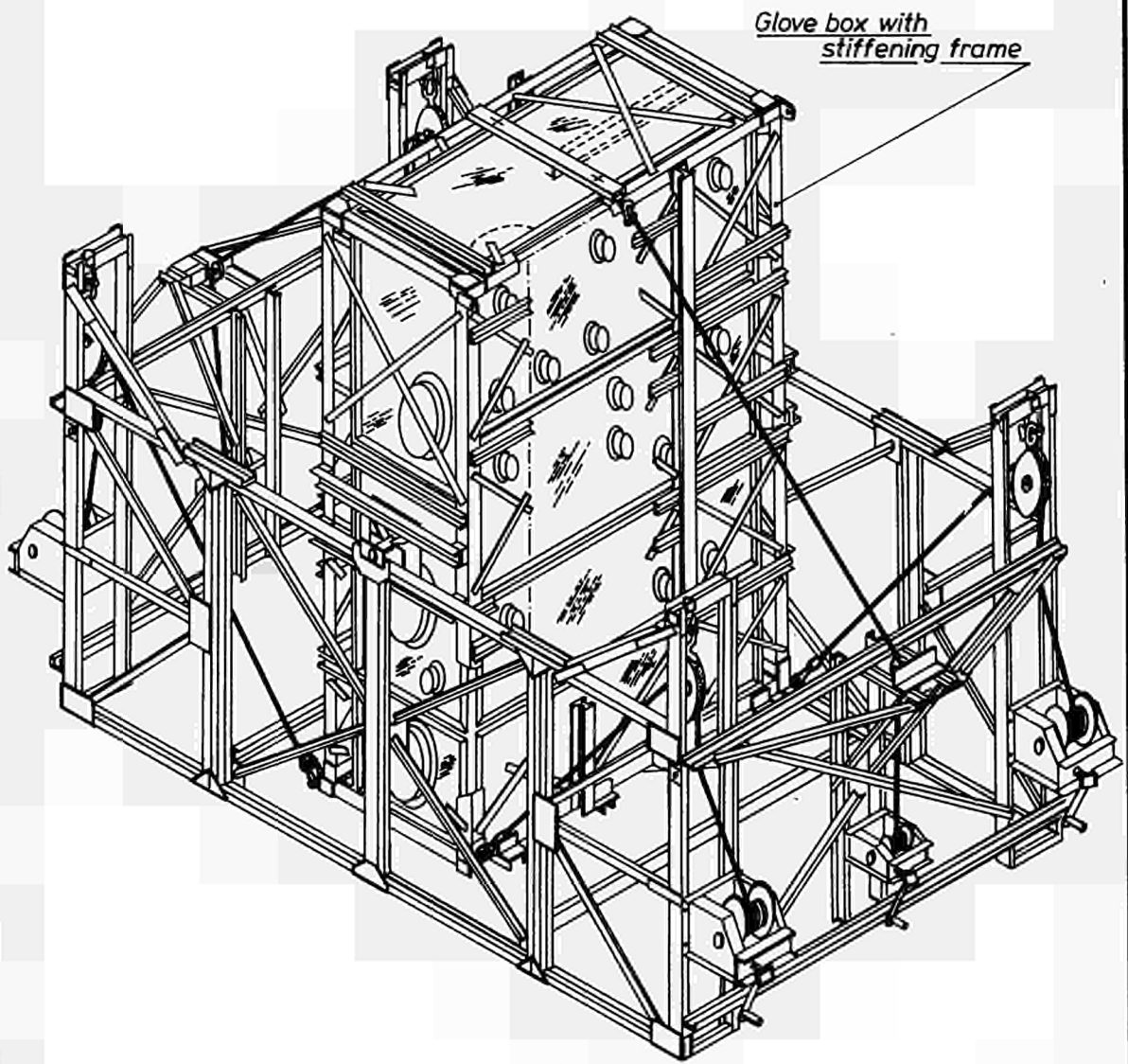


Fig. 1. Frame for lifting and rotating of glove-boxes with glove-box.

8.3. Large-Scale Application of Segmenting and Decontamination Techniques

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen,
Germany
Contract N°: FIID-0025
Working Period: January 1985 - December 1988
Project Leader: W. Stang

A. Objectives and Scope

As one of the first nuclear power plants of Germany, Gundremmingen unit A has been operating from 1966 to 1977 until an incident occurred with subsequential damage to the plant. In 1980 it was decided to decommission the plant. While for the reactor and auxiliary buildings a concept of safe enclosure was issued, some selected systems in the turbine house were dismantled and decontaminated. The aim was to reduce the radioactive waste volume as much as possible and to reclaim usable materials.

This research work is aimed at the development and optimisation of dismantling and decontamination techniques as well as measurement methods for residual activity, appropriate for a large-scale application (300 Mg). Economics and health-physic considerations are main criteria in this research.

B. Work Programme

- B.1. Selection and large-scale application of techniques for the cutting of components from the turbine house.
 - B.1.1. Classification of components for dismantling.
 - B.1.2. Laboratory tests of various cutting techniques with subsequent selection for the most appropriate application.
 - B.1.3. Large-scale application of selected cutting techniques on various components.
- B.2. Selection and large-scale application of techniques for the decontamination of components from the turbine house.
 - B.2.1. Classification of components for decontamination.
 - B.2.2. Laboratory tests of various decontamination techniques with subsequent selection for the most appropriate application.
 - B.2.3. Large-scale application of selected decontamination techniques on various components.
 - B.2.4. Reassessment of existing procedures to facilitate unrestricted release, based on melting of metallic scrap.
- B.3. Detailed studies on electrochemical decontamination.
 - B.3.1. Control and optimized use of electrolytes.
 - B.3.2. Development of continuous regeneration procedures for electrolytes.
 - B.3.3. Development of continuous regeneration procedures for acids.
 - B.3.4. Investigations for optimal conditioning of secondary waste arising from electrolytes and acids.
- B.4. Optimization of methods for the determination of the residual activity.
 - B.4.1. Classification of components for residual activity measurements.
 - B.4.2. Testing of various measuring techniques with subsequent selection.
 - B.4.3. Large scale application of selected methods for residual activity measurements on various components.

C. Progress of work and obtained results

Summary

In the current year 1985 the following large scale activities were carried out:

- dismantlement of the main condenser
- dismantlement of the turbine shielding
- removal of insulation

Small scale experiments and tests were initiated in

- cutting techniques
- decontamination techniques
- survey methods for unrestricted release of materials.

Progress and results

1. Selection and large scale application of techniques for cutting of components from the turbine house (B.1.)

In the field of metal cutting techniques three classical methods were investigated

- sawing
- torch cutting
- grinding

The suitability for large scale use was evaluated under technical, economical and radiological aspects.

In general one can say:

Sawing is the most preferable method because of minimal release of aerosols and acceptable efficiency if large stationary machines are in use.

Torch cutting has the advantage of more flexibility in case of adverse circumstances like difficult accessibility, curved cuts, thick walls, large diameters etc.

Grinding is recommended only for very low contaminated non-ferrous metals and stainless steel with thin walls. In this case it has been found that the release of aerosols is extremely high and cuts can be done by hand only on tubes or small bars.

The disadvantage of thermal cutting by torch or grinding machine is the requirement of continuous aerosol filtering and mask protection of the working personnel.

Fig. 1 shows the aerosol concentration obtained by tests.

Fig. 2 indicates the relations between the average degree of contamination and the aerosol concentration for each of the cutting techniques.

Based on the experiences of these tests the best suited segmentation method was selected for the dismantling of the main condenser.

The procedure of the dismantling was as follows:

- cooling inlet and outlet covers were opened
- about 17.000 condenser tubes were drilled, capped and removed
- support plates were removed by torch cutting.

2. Selection and large scale application of techniques for decontamination of components from the turbine house (B.2.)

Several decontamination procedures were investigated with respect to their large scale applicability.

A combined decontamination method of wire brushing prior to electropolishing was tested. Some pipes of different systems of the turbine house were treated by this way. It was found that merely brushing was not sufficient to fall below release value. Pre-brushing results in an about 35 % reduction of effort required for electropolishing.

The exact treatment times for the combined decontamination method as well as for electropolishing are given in table I and table II. A further combined method consisted in ultrasonic cleaning and pickling. These investigations with different cleaning liquids such like water, sulfuric acid and detergent solution, provided no sufficient results. The reason was the tight adhesion of the layers.

Investigations were also made on the basis of laboratory tests whether the condenser pipes can be subjected to a reutilization after melt-down. Partially brass pipes were melted with a slag-forming constituent. Some melting tests were made with pre-heating. The best results were obtained without use of a slag-forming constituent and without pre-heating. The activity flow chart for this version is represented in Fig. 3. A decontamination-factor of about 2 was achieved. Other melting tests with pre-decontaminated condenser pipes will be carried out.

3. Detailed studies on electrochemical decontamination (B.3.)

During the use of electrolyte solutions for electrochemical decontamination an acid concentration of about 40 % (by weight) proved very advantageous because of

- higher capacity of dissolved iron
- favourable starting conditions for regeneration.

The facility designed for regeneration of exhausted electrolyte, consisting of a reaction vessel, an oil heater, an evaporator, several storage vessels and a pump stand, was installed and after acceptance by the TÜV (Technical Supervisory Association) operated.

A procedure was developed for the conditioning of secondary waste arisen during the regeneration of electrolytes; the conversion of waste into an unreactive form with a high density will favour final storage.

4. Survey methods for determination of the residual activity (B.4.)

After the dismantling of about 30 tons of insulation wool a measuring arrangement for a liquid scintillation measurement of the bags filled with insulation wool was designed. Furthermore a micro-processed contamination monitor for fast determination of the residual activity of large components was installed.

Table I

Treatment times for the combined decontamination method of wire brushing prior to electropolishing

Diameter (mm)	Treatment time for dp-samples (min)	Treatment time for sp-samples (min)	Treatment time for ko-samples (min)
DN 350	146	206	146
DN 400	147	207	147
DN 450	147	207	147
DN 606	165	225	165
DN 800	178	238	178
DN 1000	193	253	193

dp: attacked by steam
sp: attacked by feed water
ko: attacked by condensate

Table II

Electropolishing times without pretreatment

Diameter	Electropolishing time for dp-samples (min)	Electropolishing time for sp-samples (min)	Electropolishing time for ko-samples (min)
all	370	250	220

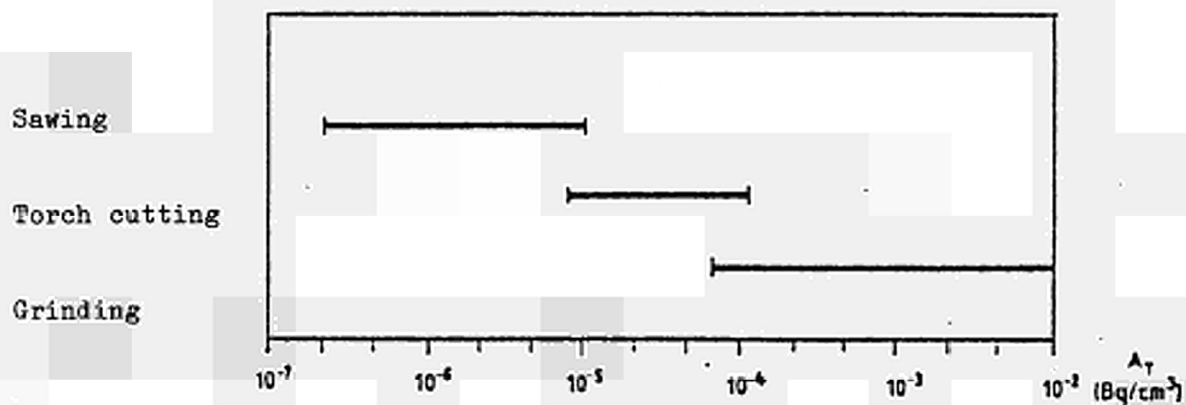


Fig. 1. Aerosol concentration (A_T) for tested metal cutting techniques

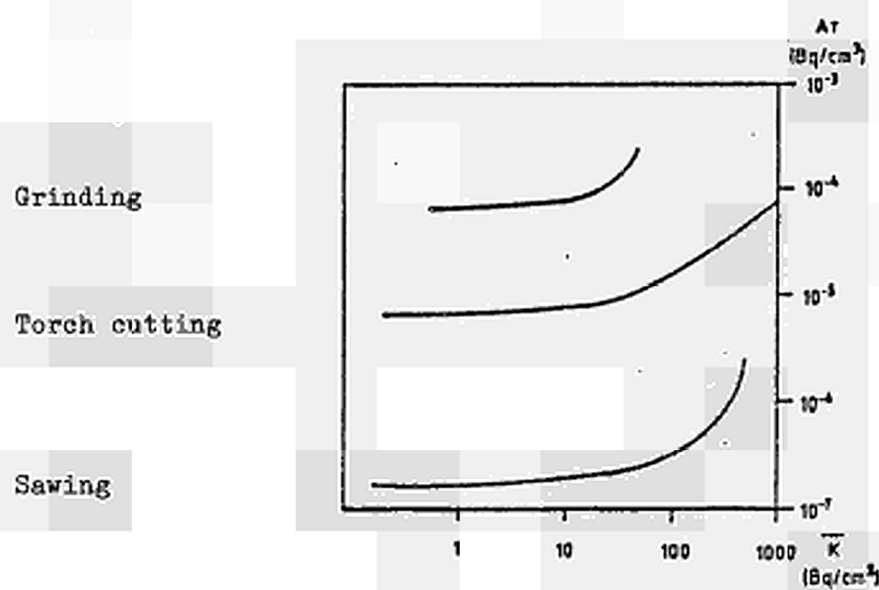


Fig. 2. Relation between average contamination \bar{K} and aerosol concentration A_T

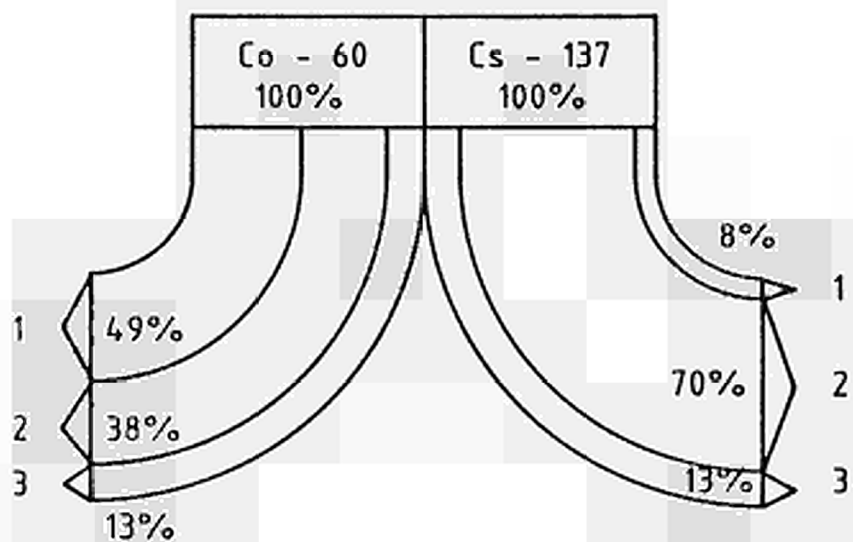


Fig. 3. Activity flow chart after melt-down of condenser pipes

1. Residual activity in the melt
2. Residual activity in the slag and crucible
3. Residual activity in the filters

8.4. Development of Techniques to Dispose of the Windscale AGR Heat Exchangers

Contractor: United Kingdom Atomic Energy Authority, Windscale
Nuclear Laboratories, United Kingdom
Contract N°: FIID-0026
Working Period: January 1985 - June 1987
Project Leader: J.R. Wakefield

A. Objectives and Scope

The objective of this research work is to dismantle two heat exchangers of the Windscale Advanced Gas-cooled Reactor. This will demonstrate that such plant can be decontaminated and dismantled for disposal without environmental hazard and without exceeding the prescribed radiation dose limits to operatives. The purpose of decontamination is to enable hands-on methods of dismantling to be used and avoid expensive and time-consuming remote operations. A further objective is to establish the nature of corrosion and contamination within the heat exchanger gas-side in order to provide data for future decommissioning of similar plant. Prior to this work, it was known that the radiation levels on the outside of the heat exchangers exhibited a peculiar distribution, but there was insufficient knowledge of the detail to predict the nature of the contamination. However, by removing a limited number of samples, having regard to the doses incurred by the operatives, it is expected to be able to identify the contaminants and recommend methods of removal. It is intended to select two such methods for use on the two heat exchangers and engineer them to minimise the quantities of liquid reagents and secondary waste to be handled. A complete costing and dose inventory will be maintained as the operations proceed.

Other organisations involved in the fulfilment of this contract are: UKAEA Winfrith (characterization of contamination and selection of decontamination method), BNFL Sellafield (provision of road and rail transport, waste disposal), other contractors (designing, manufacturing and operation of the decontamination plant).

B. Work Programme

- B.1. Characterization of surface corrosion and contamination on extracted samples of the heat exchangers, and selection of an effective contaminant allowing for an acceptable disposal of the secondary waste.
- B.2. Design, manufacture and installation of a decontamination plant.
- B.3. Decontamination of the heat exchangers and conditioning of the secondary waste.
- B.4. Dismantling of the boilers and conditioning of the scrap for final disposal.

C. Progress Of Work And Obtained Results

Summary

Heat exchanger tube samples were removed and sent to UKAEA Establishment, Winfrith. These were examined and the depth of oxide on the gas-side (outer) surface was measured. Evaporator and economiser tubes were found to have a 10 μm of magnetite and the superheater tubes deposit was in the form of a 150 μm thick reaction zone having a distinct chromium profile ranging from 1% at the parent metal to 12% at the surface. The predominant gamma emitters were confirmed as Cs^{137} and Co^{60} . The values measured were such that a decontamination factor of 30 at the point of highest activity would be required to give a radiation level of 0.1 mSv hr^{-1} at working distance or 0.5 mSv hr^{-1} at contact.

The recommended method of achieving this decontamination factor is by a fill, soak and drain process using 1.0 M hydrochloric acid + 0.0025M citric acid.

The liquid effluent, which amounts to 160,000 litres per fill, requires neutralisation before being pumped to the British Nuclear Fuels effluent disposal plant. A Functional Specification has been prepared in draft and commercial organisations will be invited to submit proposals for the supply and operation of a suitable plant. A detailed design exercise is in progress on the provision of a pipe line to carry this effluent to the BNFL plant. Also consideration is being given to developing a second method of decontamination, possibly using acid refluxing, for use on the second heat exchanger.

Progress And Results

1. Assessment Of The Problems - Determination Of Chemistry (B.1.)

Heat exchanger tube specimens were removed and sent for analysis to the UKAEA Establishment, Winfrith. The evaporator and economiser samples were found to have a surface layer of magnetite 10 μm thick, the outer 8 μm , being loose. The superheater tube deposit was in the form of a 150 μm thick reaction zone having a distinct chromium profile ranging from 1% at the metal to 12% at the surface. The principle contaminants were revealed to be Cs^{137} and Co^{60} . Mean Cs^{137} levels on the economiser and the tops of the other tube banks were (in k Bq cm^{-2}) 10 on the superheater, and 1.0 on the other tubes. Corresponding Co^{60} levels were 0.15 and 0.06 giving Cs:Co ratios of 67:1 and 17:1. The specimens from the bottom of the evaporator bank gave mean Cs and Co levels around 25 and 0.5 k Bq cm^{-2} respectively.

Ring sections of tube were tested with hydrochloric acid and nitric acid solutions with citric acid at 20°C and 50°C. It was found that the solutions ceased to act effectively after about 90 minutes and that more than two applications did not produce an appreciably higher Decontamination Factor. At ambient temperatures (only practicable for the full scale plant), DF's between 18 and 85 were obtained for Cs^{137} using 0.5M hydrochloric acid + 0.0025M citric acid. Under similar conditions, 0.5M nitric acid + 0.0025M citric acid gave DF's of 5 to 30. It follows that to obtain the required DF of 30 at 20°C, a solution of 1M hydrochloric acid + 0.0025M citric acid is required and this is the recommended method. This should reduce maximum radiation levels to 0.1 mSvhr^{-1} at working distance (0.5 mSv hr^{-1} at contact) to give operator maximum dose limits 0.1 rem/day. It is likely that these dose levels will be much less in practice.

2. Design, Manufacture And Installation Of Decontamination Plant (B.2.)

A functional specification for decontaminating a heat exchanger has been prepared in draft. This will be followed by an engineering plan which takes into account the restraints imposed by the plant and associated services. This plan will be offered to the commercial organisations who have expressed interest in the project and they will be invited to tender for the work. The proposal for an acid reflux system is still on the table but funds for its development to a viable process are being sought.

Advanced discussions with British Nuclear Fuels have confirmed that the hydrochloric acid/citric acid effluent, suitably neutralised, is acceptable for disposal via their plant. It will be necessary to design and install a suitable pipeline to carry the active effluent from the WAGR site to the BNFL plant. The resolution and acceptance of this proposal has occupied considerable time and effort and although not forming part of this contract, has delayed the next phase by six months.

The heat exchangers have been prepared for the commencement of the decontamination process by removal of the external asbestos insulation. External pipework has been removed and temporary seals have been fitted to the penetrations. Access ladders and platforms have been cleared away and the floating support bearings have been replaced by solid pads. The concentric gas ducts have been severed and removed on one heat exchanger and work will continue on the remainder. All bioshields have been penetrated for the installation of pipework for decontamination and ventilation equipment.

8.5. Pilot Decommissioning of a Mixed-oxide Fuel Fabrication Facility

Contractor: British Nuclear Fuel plc, Sellafield, United Kingdom
Contract N°: FIID-0027
Working Period: July 1984 - June 1988
Project Leader: A.P. Colquhoun

A. Objectives and Scope

The objectives of the project are to pilot the development of technology for the decommissioning of facilities used in the fabrication of mixed-oxide fuels. Based on existing experience the aim is to establish the procedures which are the most cost-effective overall under the specific constraints on the disposal of wastes arising and on the radiation exposure of personnel.

The development programme is integrated within the decommissioning of the Co-precipitation Plant which was used to produce mixed-oxide powder for the fabrication of fast reactor fuel. The Plant is on two floors and occupies a total floor area of some 320 m² within which are housed 14 glove boxes, 2 furnaces, 5 tanks, a scrubber vessel, ventilation ducting and pipework.

The techniques to be tested are those which meet the specific constraints and for which previous research has indicated the potential for large-scale application. Decontamination, dismantling and packaging of the wastes are the operations involved while the radiation, contamination and ingestion hazards impose restrictions on the methods of working. It is towards the most effective overall procedure that the techniques will be concerted.

Included in the aspects of this development are the minimizing of the amount of alpha-contaminated waste material, the minimizing of the radiation exposure to personnel, the identification of the best means of in situ decontamination, and the most suitable means of measuring in-situ the alpha contamination. Finally, a comparison of costs and radiation exposure from alternative techniques in the real application of decommissioning will be made.

B. Work Programme

- B.1. Detailed planning for most appropriate decontamination and dismantling, including technical specifications and safety assessments for proposed methods and plant modifications and submissions for company and regulatory approvals.
- B.2. Execution of modifications, testing of equipment, rehearsing of proposed procedures on plant simulations, followed by in-situ decontamination.
- B.3. Rehearsing of dismantling and packaging procedures on plant simulations, followed by in-situ dismantling and packaging.

C. Progress of work and obtained results

Summary

This report describes work carried out in support of the development aspects of the project to decommission the mixed oxide fuel fabrication facility. Work was initiated on the main project in September 1984 but was interrupted between April and September 1985 before the Contract was signed in November 1985. Most of the earlier work was aimed at defining the total Decommissioning Project of which the integral development aspects form the major part of the contract. Progress on these development aspects is largely determined by the timing of the phases of the main project. All the work reported forms part of step B1 as defined in chapter B. A scope of work document for the project was produced and used to formulate Outline Specifications for Engineering Modifications needed to facilitate the work. Work was initiated on various Specifications in support of the newer techniques provisionally identified for evaluation. A Specification for Freon (Arklone) decontamination equipment has been produced and others drafted for Reusable Modular Containments, a Portable Pressurised Suit Shower and a small scale Electrolytic decontamination facility.

Progress and results

1. Scope of Work document

This document was produced first in order to clearly identify the work involved and more particularly the following.

- a) The extent of the decontamination and dismantling work requirement.
- b) The newer techniques to be applied and evaluated.
- c) The engineering modifications necessary to facilitate work implementation.
- d) The preliminary flow chart for the work and its end points.

Fact finding visits were made to other UK organisations to view appropriate projects and techniques, to discuss relevant issues and to compile and collate necessary data to supplement literature searches. Discussions were held with staff at the United Kingdom Atomic Energy Establishments at Harwell and Winfrith and at the Ministry of Defence Establishment at Aldermaston all of whom are doing work of relevance to the project. This confirmed the correctness of the work approach in the Scope and led to more detailed studies to produce Outline Specifications suitable for design work which will lead to cost estimation.

2. Outline Specifications for Mechanical, Electrical and Civil Modifications

The main modification areas identified for the existing plant to engineer the following requirements were:

- a) The rerouting of pipework and the provision of additional systems for the addition, use and disposal of decontaminant liquids.
- b) The provision of additional lifting equipment.
- c) The creation of transport routes for glove boxes out of the working areas into an area suitably converted for use as a waste packaging station for despatch to storage and treatment facilities.

- d) The recovery and restoration of the decommissioned area for reuse.

3. Development Aspects

One conclusion from the data collection phase was that although some techniques being considered are sufficiently developed to require only adaptation to meet Site specific constraints, others need significant pre-testing before a final decision on their use in the project can be taken. To facilitate this process further priority consideration was given to the areas of the Scope covering Freon (Arklone) decontamination, Reusable Modular Containments and Plutonium Monitoring.

4. Freon Decontamination

The use of this and other non-aqueous solvents has been extensively studied by others and this previous work has established a potential for the removal of loose gross contamination typically found in powder handling operations. In the plant to be decommissioned the radiation levels from the residual powder could contribute significant dose uptake to operators until reduced. The use of Freon to flush away for recovery the powder may be a way of rapidly reducing radiation fields prior to operator intervention for more intensive decontamination or dismantling. An Outline Specification for equipment development and supply was prepared and discussions have been held with a number of potential sub-contractors. The proving of the feasibility of the concept, the definition of key operating parameters, and the safety assessment and approval for use of the system are seen as requiring resolution. Identified problems to be addressed are those of solvent loss into ventilation systems and filtration and recovery of the particulate material.

5. Reusable Modular Containments

This concept utilises reusable structures as contamination barriers during operations requiring the breaking of primary containments eg removal of glove box panels. Their use in conjunction with protective and strippable coatings for contamination control is a potentially attractive alternative to existing ad hoc methods for such in-situ work stations. A Supply Specification has been drafted based on a system in use at the UKAEA's Winfrith Establishment but optimised for the planned operations in terms of sizes and modular panel configurations. An accompanying Supply Specification has been drafted for Pressurised Suit Portable Shower Facility. After cost estimation an evaluation will be made against existing systems before finalising the proposed extent of their initial test usage.

6. Plutonium Monitoring

Work was initiated aimed at producing a Specification for any additional capability needed to supplement instruments currently routinely used in production and waste management operations and to provide an integrated monitoring system. Ideally this will be able to do the following.

- a) Establish the initial plutonium inventory and profile before decontamination.

- b) Monitor changes in contamination levels as decontamination proceeds.
- c) Assay plutonium on arisings from dismantling being sent to further treatment facilities.
- d) Measure residual contamination levels to meet disposal criteria for shallow land burial.

7. Other Development Work Aspects Initiated

A Supply Specification is being prepared for a nitric acid based Electrolytic Decontamination System. It is envisaged that an existing glove box, which presently has an acid cleaning bath for plant components, will be modified to house appropriate equipment. This will evaluate the cost effectiveness of the final cleaning of dismantled components to shallow land burial levels.

8.6. Testing of New Techniques in Decommissioning of a Fuel (U, Th) Fabrication Plant

Contractor: NUKEM GmbH, Hanau, Germany
Contract N°: FIID-0028
Working Period: July 1985 - December 1988
Project Leader: H.D. Fricke

A. Objectives and Scope

This research work is aimed at the assessment of new procedures in the framework of the decommissioning of a plant for the production of Material Test Reactor (MTR) and Thorium High Temperature Reactor (THTR) fuel elements.

Important issues in this work are the preparation of detailed uranium and thorium contamination distribution maps in walls and floors, the execution of various dismantling and decontamination operations under health physics control, the large-scale treatment of arising primary waste and the minimization of secondary waste. The work will be concluded with an assessment of gained experience, with possible recommendations for future work on similar facilities.

B. Work Programme

- B.1. Preparation of a map of the distribution of the contamination within different parts of the fuel fabrication plant.
- B.2. Determination, by analyses of representative samples, of the penetration depth of uranium and thorium in various parts of the facility.
- B.3. Controlled decontamination and dismantling of the internal components and of all auxiliary equipment of the plant.
- B.4. Assessment of appropriate conditions for the removal of contamination from the walls of the facility and its implementation after acceptance by the Regulatory Bodies.
- B.5. Decontamination of the floors and their removal.
- B.6. Testing of new decontamination procedures for less accessible parts.
- B.7. Determination of the residual activity and possible further dismantling of less accessible parts.
- B.8. Conditioning and assessment of the residual activity of metal waste for re-use by melting.
- B.9. Large-scale decontamination of the demolition rubble based on existing laboratory-scale methods.
- B.10 Minimizing of the secondary waste from decontaminants.
- B.11 Testing of a NUKEM procedure for container sealing.
- B.12 Evaluation of obtained results.

C. Progress of work and obtained results

Summary

During the working period activities were started, according to the working steps B1-3 and B 10 of the work programme. For all contaminated areas within the fuel fabrication plant NUKEM-A a contamination map was prepared, including surface dose rates of walls, floors and equipment and proposals for decontamination. At some selected places sampling of wall and floor materials was started for determination of penetration depths of U and Th. As one example for a contaminated facility the dismantling of a pyrolysis reactor has been performed, which shall be further decontaminated. Various decontamination agents were tested with regard to interference with NUKEM's waste water treatment system. Activity levels for walls, floors and conventional equipment were found to be in the area $< 2 \text{ Bq/cm}^2$ until 6 Bq/cm^2 , at fabrication facilities until 60 Bq/cm^2 , these values include non removable and removable activity.

Progress and results

1. Preparation of a contamination map (B.1.)

The NUKEM-A-facility is in some areas slightly contaminated with U- and Th-materials, caused by 25 years production of high temperature reactor- und material test reactor fuel. Within the program the activity levels of all contaminated areas were listed to a contamination map, which includes measured contamination levels for walls, floors, conventional equipment (tables, chairs etc.) and technical facilities. Additionally first surface decontamination experiments were performed with commercial available decontamination agents.

Contamination level measurements are performed with a commercial available Geiger-Müller counter (Kontamat, FAG-Kugelfischer Company) and by a standardized wipe test.

The summarized results from more than 500 measurements are:

- contamination of walls, floors : < 2 to 6 Bq/cm^2
- contamination of conventional equipment: < 2 to 6 Bq/cm^2
- contamination of process facilities : < 2 to 60 Bq/cm^2

A first attempt of a classification has been performed, which shows possibilities of decontamination to achieve values below acceptable activity limits and U-recycling. Depending on contamination levels, materials and surface conditions (corrosion, roughness, porosity) following qualitative results are obtained:

- easy decontamination (wiping, simple cleaning)
metal surfaces, ceramics, PVC and terrazzo (floors) $\leq 0,1 \text{ Bq/cm}^2$
- chemical and physical decontamination (acid flushing, scraping, sand blasting e.g.)
metalls, ceramics, polymers, acryl glass, walls, floors, concrete
- residual materials (capable for U-recovery by incineration, melting, dissolution and extraction)
burnable trash, metal scrap, terrazzo, concrete

At the time the contamination map will be completed and computerized for easy actualization.

2. Penetration depth of Uranium and Thorium (B.2.)

According to the program, measurements of penetration depths of Uranium and Thorium into wall and floor materials will be performed by core drilling, cutting cores in thin layers (mm), activity measurements or dissolution, liquid sampling and drying for α -counting and spectrometry. In the working period a suitable core drilling equipment was selec-

ted and laboratory equipment has been installed, including a diamond saw for cutting and an IR-dryer for pretreatment of specimen for α -counting.

3. Dismantling of components (B.3.)

As one example for a contaminated facility dismantling of a pyrolysis reactor was started. The reactor encounters following parts:

- outer body with filling pipe and mixing elements (1000 mm \emptyset , ~ 3000 mm height)
- inner body (780 mm \emptyset) with mixing elements
- reactor lid

Materials in contact with pyrolyzed residues have been made of Inconel 600.

The interior wall of the outer body and the interior body are contaminated with a slag layer (1-2 mm), which contains depleted Uranium and Thorium. This layer shall be completely removed by grinding and/or sand blasting. The decontamination work will start in the first quarter of 1986, results will be obtained by activity measurements at the surfaces and removed materials.

4. Minimizing of secondary wastes (B.10.)

Four decontamination agents, commonly used for cleaning of contaminated areas, have been tested with regard to their interference with the precipitation and flocculation process, used at NUKEM for waste water treatment.

Experiments were performed by treatment of 1 m³ batches of cleaning effluents, doted with Uranium and Thorium, with precipitation agents, investigation of sedimentation velocity of the precipitate and measurement of the activity in the supernate. For each agent it was possible to meet activity levels in the supernate below fixed release limits for NUKEM-A. No interference of the sedimentation, compared with waste waters without cleaning agents, occurred at the use of two decontamination agents, which will be used for further work.

ANNEX I

LIST OF PUBLICATIONS RELATING TO THE RESULTS OF THE 1979-83 PROGRAMME ON THE DECOMMISSIONING OF NUCLEAR POWER PLANTS

A. Annual Progress Reports

"The Community's Research and Development Programme on Decommissioning of Nuclear Power Plants - First Annual Progress Report (year 1980)", EUR 7440, 1981.

"The Community's Research and Development Programme on Decommissioning of Nuclear Power Plants - Second Annual Progress Report (year 1981)", EUR 8343, 1983.

"The Community's Research and Development Programme on Decommissioning of Nuclear Power Plants - Third Annual Progress Report (year 1982)", EUR 8963, 1984.

"The Community's Research and Development Programme on Decommissioning of Nuclear Power Plants - Fourth Annual Progress Report (year 1983)", EUR 9677, 1985.

B. 1984 European Conference

Schaller, K.H., Huber, B. (ed). Decommissioning of nuclear power plants - Proceedings of a European Conference held in Luxembourg, 22-24 May 1984. Graham & Trotman Ltd, London. EUR 9474 EN.

C. Final Contract Reports

Boothby, R.M., William, T.M. (1983). The control of cobalt content in reactor grade steels. European Appl. Res. Rept., Nucl. Sci. Technol. Vol. 5, No. 2, Harwood Academic Publishers. EUR 8655.

Lörcher, G., Piel, W. (1983). Dekontamination von Komponenten stillgelegter Kernkraftwerke für die freie Beseitigung. EUR 8704.

Kloj, G., Tittel, G. (1984). Thermische und mechanische Trennverfahren für Beton und Stahl. EUR 8633.

Harbecke, W. et al. (1984). Die Aktivierung des biologischen Schields im stillgelegten Kernkraftwerk Lingen. EUR 8801.

Verral, S., Fitzpatrick, J. (1985). Design concepts to minimize the activation of the biological shield of light-water reactors. EUR 8804.

Eickelpasch, W. et al. (1984). Die Aktivierung des biologischen Schields im stillgelegten Kernkraftwerk Gundremmingen Block A. EUR 8950.

- Verry, P., Lecoffre, Y. (1984). Décontamination de surfaces par érosion de cavitation. EUR 8956.
- Allibert, M.; Delabbaye, F. (1984). Extraction du cobalt des aciers inoxydables. EUR 8966.
- Ebeling, W. et al. (1984). Dekontamination von Betonoberflächen durch Flammstrahlen. EUR 8969.
- Boullitrop, D., Rouet, D. (1984). Etude de la décontamination au moyen de supports gélifiés. EUR 9102.
- Peselli, M. (1984). Individuazione quantitativa delle impurezze del contenitore a pressione del reattore del Garigliano. EUR 9167.
- Avanzini, P.G. et al. (1984). Valutazione delle caratteristiche di progetto che facilitano lo smantellamento delle centrali nucleari PWR. EUR 9191.
- Regan, J.D. et al. (1984). Design features facilitating the decommissioning of Advanced Gas-cooled Reactors. EUR 9207.
- May, S., Piccot, D. (1984). Détermination analytique d'éléments traces dans des échantillons de bétons utilisés dans les réacteurs nucléaires de la Communauté européenne. EUR 9208.
- White, I.F. et al. (1984). Assessment of management modes for graphite from reactor decommissioning. EUR 9232.
- Goddard, A.I.H. et al. (1984). Trace element assessment of low-alloy and stainless steels with reference to gamma activity. EUR 9264.
- Bregani, F. et al. (1984). Chemical decontamination for decommissioning purposes. EUR 9303.
- Glock, H.J., et al. (1984). Dokumentationssystem für den Abbau von Kernkraftwerken. EUR 9343.
- Ahlfänger, W. (1984). Zusammensetzung von Kontaminationsschichten und Wirksamkeit der Dekontamination. EUR 9352.
- Brambilla, G. et al. (1984). Vernici per la fissazione della contaminazione superficiale dei materiali. EUR 9358.
- Paton, A.A. et al. (1984). Civil engineering design for decommissioning of nuclear installations. Graham & Trotman Ltd, London. EUR 9399.
- Bittner, A. et al. (1985). Konzepte zur Minimierung der Aktivierung des biologischen Schilfs. EUR 9442.
- Brambilla, G., Beaulardi, L. (1985). Rivestimenti rimovibili per la protezione di superfici in calcestruzzo dalla contaminazione. EUR 9463.

- Arndt, K.-D. et al. (1984). Thermisches Trennen von plattierten Komponenten des Primärkreises von Kernkraftwerken. EUR 9479.
- Rawlings, G.W. (1985). Development of large diamond-tipped saws and their application to cutting large radioactive reinforced concrete structures. EUR 9499.
- Borody, I.I. et al. (1985). Treatment of active concrete waste arising from dismantling of nuclear facilities. EUR 9568.
- Bernard, A., Gerland, H. (1985). Recherche et caractérisation de revêtements pour la protection des structures en béton. EUR 9595.
- De Tassigny, C. (1985). Immobilisation de la contamination sur des métaux par revêtement de polymères. EUR 9666.
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- Antoine, P. (1985). Intégrité à long terme des bâtiments et des systèmes. EUR 9928.
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- Hasselhoff, H., Seidler, M. (1985). Anlage zum Einschmelzen von radioaktiven metallischen Abfällen. EUR 10021.
- Chavand, J. et al. (1985). Découpage de composants métalliques par fissuration intergranulaire. EUR 10037.
- Gauchon, J.P. et al. (1986). Décontamination par des méthodes chimiques, électrochimiques et au jet d'eau. EUR 10043.
- Smith, G.M. et al. (1985). Methodology for evaluating radiological consequences of the management of very low-level solid waste arising from decommissioning of nuclear power plants. EUR 10058
- Gomer, C.R., Lambley, J.T. (1985). Melting of contaminated steel scrap arising in the dismantling of nuclear power plants. EUR 10188.
- Da Costa, L. et al. (1985). Review of systems for remotely controlled decommissioning operations. EUR 10197.
- Price, M.S.T., Lafontaine, I. (1985). System of large transport containers for waste from dismantling light water and gas-cooled nuclear reactors. EUR 10232.

ANNEX II

MEMBERS OF THE ADVISORY COMMITTEE ON PROGRAMME MANAGEMENT
IN THE FIELD OF THE DECOMMISSIONING OF NUCLEAR POWER PLANTS (1)

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(1) See also footnote to Annex III.

(2) From 30.4.1984.

(3) Till 30.4.1984.

ANNEX III

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-
- (1) This Committee was established by the Council Decision of 29 June 1984 dealing with structures and procedures for the management and coordination of Community research, development and demonstration activities (OJ N° L 177, 4.7. 1984, p. 177). By the same Council Decision, the Advisory Committee on Programme Management in the field of the decommissioning of nuclear power plants was dissolved. However, the latter Committee continued to carry out its task until the new Advisory Committee effectively started functioning, i.e. end of 1984.

European Communities — Commission

EUR 10740 — The Community's research and development programme on decommissioning of nuclear installations. First annual progress report (year 1985)

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Luxembourg: Office for Official Publications of the European Communities

1986 — VI, 140 pp. — 21.0 × 29.7 cm

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This is the first annual progress report of the European Community's programme (1984-88) of research on decommissioning of nuclear installations. It shows the status of the programme on 31 December 1985.

This first progress report, covering the period of putting the programme into action, describes the work to be carried out under the 27 research contracts concluded, as well as initial work performed and first results obtained.

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