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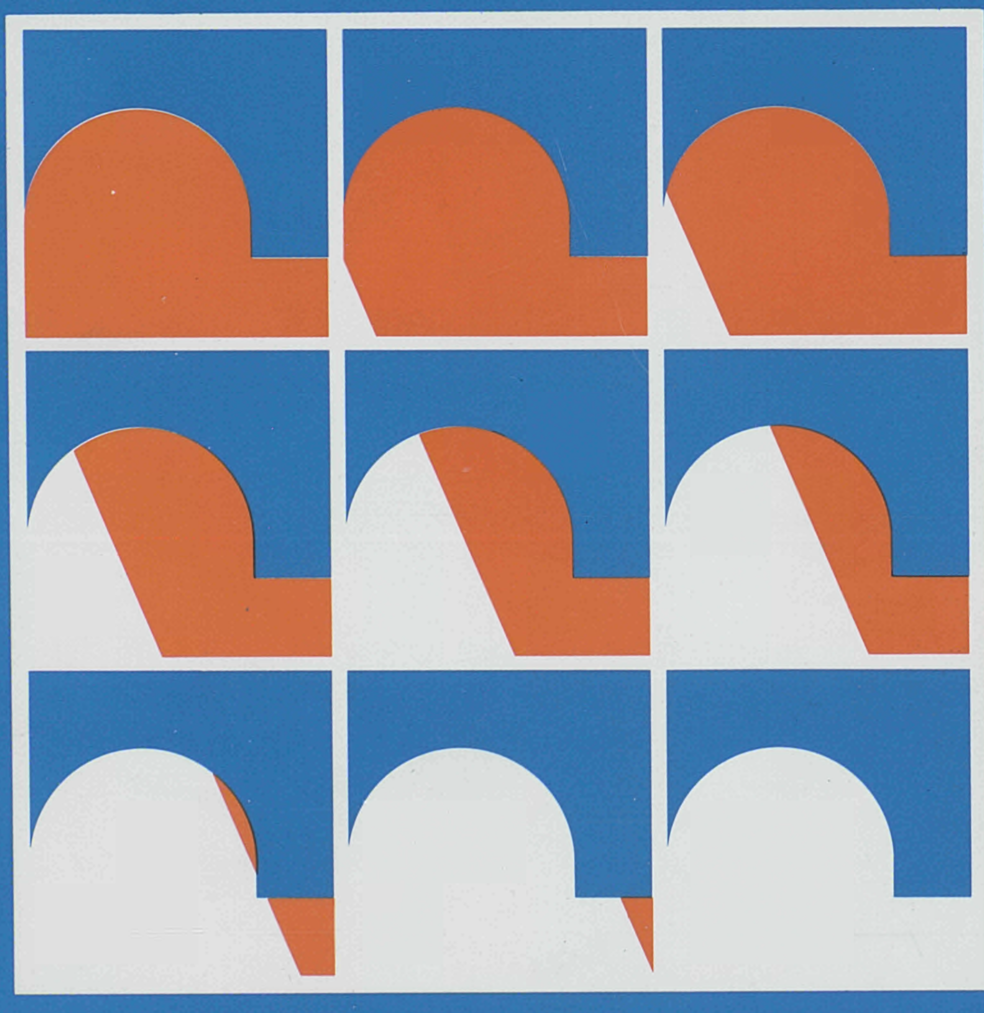
Commission of the European Communities

nuclear science and technology

The Community's research and development programme on decommissioning of nuclear installations (1989-93)

Annual progress report 1992



Report

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The Community's research and development programme on decommissioning of nuclear installations (1989-93)

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FOREWORD

The following third Annual Progress Report summarises the activities of the European Communities R&D Programme on Decommissioning of Nuclear Installations for the year 1992. (Annual progress reports 1990 and 1991, see ref. /1/ and /2/).

This programme was adopted by the EC Council in March 1989 /3/ to find "effective solutions which are capable of ensuring the safety and protection of both mankind and its environment against the potential hazards in decommissioning".

As a large number of older nuclear facilities will be taken out of service in the next ten years, the public, the industry and national regulations are becoming increasingly concerned over the occupational doses, environmental hazards and the costs which could be incurred in the decommissioning of such plants. The European Community, well aware of these concerns, has since 1978 operated and financed research programmes in this field.

The 1989-1993 programme concerns the following areas:

A. Research and development projects concerning the following subjects:

Area N° 1: Long-term integrity of building and systems;
Area N° 2: Decontamination for decommissioning purposes;
Area N° 3: Dismantling techniques;
Area N° 4: Treatment of specific waste materials: steel, concrete and graphite;
Area N° 5: Qualification and adaptation of remote-controlled semi-autonomous manipulator systems;
Area N° 6: Estimation of the quantities of radioactive wastes arising from the decommissioning of nuclear installations in the Community.

B. Identification of guiding principles relating to:

- the design and operation of nuclear installations with a view to simplifying their subsequent decommissioning,
- the decommissioning operations with a view to making occupational radiation exposure as low as reasonably achievable,
- the technical elements of a Community policy in this field.

C. Testing of new techniques in practice:

- pilot projects,
- alternative tests,
- staff secondment.

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 for this five-year programme amounts to 33.8 million ECU.

Work on the four pilot dismantling projects was successful. Also during 1992, Part B "Identification of guiding principles" has achieved satisfactory progress and the common action to collect data relevant to cost, occupational doses, working time and waste arisings is now a fully operational part of the programme. Both activities are covered in this report.

I am most grateful to the contractors who have produced most of the substance of this report and who, this year, provided a particular effort to make its timely publication possible.

For its compilation and editing I wish to thank my colleagues, Messrs R Bisci, B Huber, K Pflugrad, E Skupinski and R Wampach. As by the time of the publication of this report, Mr Huber and Mr Skupinski will no longer be among the team managing the Communities' R&D on nuclear decommissioning, the outstanding merits, the commitment and the untiring efforts of these two "founders" of the programme should also be acknowledged in this report.

R SIMON
Head of the Programme

References

/1/ "The Community's research and development programme on decommissioning of nuclear installations (1989-1993)". Annual progress report 1990. EUR 14227.

/2/ "The Community's research and development programme on decommissioning of nuclear installations (1989-1993)". Annual progress report 1991. EUR 14498.

/3/ Council Decision of 14 March 1989 adopting a research and technological development programme for the European Atomic Energy Community in the field of the decommissioning of nuclear installations. OJ No. L 98, 11.04.1989, p. 33.

/4/ Commission Communication concerning the research programme on the decommissioning of nuclear installations (1989 to 1993). Call for research proposals. OJ No. C 24, 31.01.1991, p. 8

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LIST OF ABBREVIATIONS - CONTRACTORS' NAMES AND ADDRESSES

AEA-Culch.	Atomic Energy Authority Technology Culcheth, Wigshaw Lane, UK-Cheshire WA3 4NE
AEA-Culh.	Atomic Energy Authority Technology Culham, UK-Abingdon, Oxfordshire OX14 3DB
AEA-Harw.	Atomic Energy Authority Technology Harwell, UK-Oxfordshire OX11 0RA
AEA-Wind.	Atomic Energy Authority Technology Windscale, UK-Seascale, Cumbria CA20 1PF
AEA-Winf.	Atomic Energy Authority Technology Winfrith, UK-Dorchester, Dorset DT2 8DH
ARC	Arc Kinetics Ltd, 38, The Fairway, UK-Daventry, Northamptonshire NN1 4NW
BAI	Benelux Analytic Instruments, Vaartdijk 22, B-1800 Vilvoorde
BE	Battelle Europe - Battelle-Institut e.V. Frankfurt, Am Römerhof 35, D-60486 Frankfurt/Main
BNF	British Nuclear Fuels plc, Sellafield Works B403, UK-Seascale, Cumbria CA20 1PG
BS	Brenk Systemplanung, Heinrichsallee 38, D-52062 Aachen
Bureau A+	Bureau A+, Godsweerdersingel 87, NL-6041 GK Roermond
CEA-Cad.	Commissariat à l'Energie atomique, Centre de Cadarache, B.P. N° 1, F-13108 St. Paul-lez-Durance
CEA-FAR	Commissariat à l'Energie atomique, Centre de Fontenay-aux-Roses, 60 Avenue du Général Leclerc, B.P. N° 6, F-92265 Fontenay-aux-Roses
CEA-Sac.	Commissariat à l'Energie atomique, Centre de Saclay, F-91191 Gif/Yvette
CEA-Valrhô	Commissariat à l'Energie atomique, Centre de la Vallée du Rhône, B.P. N° 171, F-30205 Bagnols/Cèze Cedex
CIEMAT	Centro de Investigaciones Energéticas Medioambientales y Tecnológicas, Avenida Complutense 22, E-28040 Madrid
COGEMA	Cie Générale des Matières nucléaires, B.P. 270, F-50107 Cherbourg
COMEX	Comex Nucléaire, 36 boulevard des Océans, F-13275 Marseille
DLR	Deutsche Forschungsanstalt für Luft- und Raumfahrt e.V., Pfaffenwaldring 38-40, D-70569 Stuttgart
ENEA	Ente per le Nuove Technologie, l'Energia e l'Ambiente, Viale Regina Margherita 125, I-00198 Roma
ENEL	Ente Nazionale per l'Energia Elettrica, Via R. Rubattino 54, I-20134 Milano
ENRESA	Empresa Nacional de Residuos Radioactivos S.A., Calle Emilio Vargas 7, E-28043 Madrid
ENSA	Equipos Nucleares S.A., Plaza del Marqués de Salamanca, E-28043 Madrid
EPC	S.A. d'Explosifs & Produits chimiques, rue de la Dynamite, F-13310 St-Martin de Crau
EWN	Energiewerke Nord GmbH, Greifswald, D-17509 Lubmin
FHGF	Fachhochschule Giessen-Friedberg, Wiesenstraße 14, D-35390 Giessen
Framatome	Framatome, Tour Fiat Cedex 16, F-92084 Paris-la-Défense
Goodwin	Goodwin Air Plasma Ltd, Kernan Drive, UK-Loughborough, Leiston LE11 0JF

IND	International Nuclear Decommissioning (BNF), Sellafield, UK-Seascale, Cumbria CA20 1PG
KA	Kraftanlagen Aktiengesellschaft, Im Breitspiel 7, D-69126 Heidelberg
KEMA	N.V. Keuring van Elektrotechnische Materialen, Utrechtseweg 310, NL-6812 ET Arnhem
KfK	Kernforschungszentrum Karlsruhe, D-76344 Eggenstein-Leopoldshafen
KKWR	Kernkraftwerk Rheinsberg, D-16831 Rheinsberg
KRB	Kernkraftwerk RWE-Bayernwerk GmbH, Dr.-August-Weckesser-Straße, D-89355 Gundremmingen
LAINSA	Limpiezas y Acondicionamientos Industriales S.A., El Payeter 13, E-46008 Valencia
NIS	NIS Ingenieurgesellschaft mbH, Donaustraße 23, D-63452 Hanau
NNC	National Nuclear Corporation Ltd, Booths Hall, Chelford Rd, UK-Knutsford, Cheshire WA16 8QZ
Noell	Noell GmbH-Nuklear Service, Alfred-Nobel-Straße 20, D-97080 Würzburg
NRPB	National Radiological Protection Board, Chilton, UK-Didcot, Oxfordshire OX11 0RQ
Radia	Radiacontrôle, Route de Lyon 44, F-38000 Grenoble
RNL	Risø National Laboratory, P.O. Box 49, DK-4000 Roskilde
RWE	Rheinisch-Westfälisches-Elektrizitätswerk AG, Kruppstraße 5, D-45128 Essen
RWTHA	Rheinisch-Westfälische Technische Hochschule Aachen, Reutershagweg 4, D-52074 Aachen
SCK/CEN	Studiecentrum voor Kernenergie/Centre d'Etudes de l'Energie Nucléaire, Boeretang 200, B-2400 Mol
SG	Siempelkamp Giesserei GmbH & Co, Siempelkampstraße 45, D-47803 Krefeld
Siemens-KWU	Siemens AG, Bereich Energieerzeugung KWU, Hammerbacherstraße 12-14, D-91058 Erlangen
Siemens-BEW	Siemens AG Brennelementewerk, Rodenbacher Chaussee 6, D-63457 Hanau
SSP	Stangenberg, Schnellenbach und Partner GmbH, Viktoriastraße 47, D-44787 Bochum
Taywood	Taylor Woodrow Engineering Ltd., Ruislip Road 345, UK-Southall UB1 2QX
TNO	Netherlands Organization for Applied Scientific Research, P.O.Box 155, NL-2600 AD Delft
TÜV-Bay.	Technischer Überwachungsverein Bayern e.V., Westendstraße 199, D-80686 München
TÜV-SWD	Technischer Überwachungsverein Südwestdeutschland e.V., Dudenstraße 28, D-68167 Mannheim
TWI	The Welding Institute, Abington Hall, UK-Abington, Cambridgeshire CB1 6AL
UDA	Universidad de Alicante, Carretera de San Vincente del Raspeig s/n, E-03099 Alicante
UH-IW	Universität Hannover, Institut für Werkstoffkunde, Appelstraße 11A, D-30167 Hannover
VAK	Versuchsatomkraftwerk Kahl GmbH, Postfach 6, D-63796 Kahl/Main

SECTION A: RESEARCH AND DEVELOPMENT PROJECTS

1. AREA No. 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 area is to determine the measures required for maintaining shut-down plants in a safe condition and to assess the radiological consequences of costs.

B. Research performed under the previous programmes (1979-1988)

The research work has been focused on the following main subjects:
- inspection of selected nuclear power plants and examination of materials as they exist therein, in order to determine the mode and pace of degradation;
- methodology studies of the measures necessary for maintaining plants in safe condition and for keeping the necessary ancillary equipment operable.

C. 1989-1993 Programme

Research in this area should be pursued with a constant moderate effort, enlarging the data base and exploiting the growing experience, in order to establish confidence in long-term forecasts. This involves in particular:
- collection of additional experimental data, eg repetition of past examinations after a time interval of about five years, in order to determine the rate of degradation and derive or check forecasting rules;
- comparison of confinement methods applied at specific shutdown nuclear installations in Member States;
- assessment of the merits of the Safe Storage option in the decommissioning of nuclear installations other than reactors.

D. Programme implementation

At the end of 1992, one research contract relating to Area No. 1 was at the stage of execution.

1.1. EXAMINATION AND LONG-TERM ASSESSMENT OF NUCLEAR POWER STRUCTURES

Contractor:	TEL, SSP, NNC	
Contract No.:	FI2D-0048	
Work Period:	April 1991 - June 1993	
Project Manager:	D C POCOCK, TEL	
	Phone: 44/81/575 47 23	Fax: 44/81/575 40 44

A. OBJECTIVE AND SCOPE

This work programme describes two separate activities. Taywood Engineering Ltd (TEL) and Stangenberg, Schnellenbach and Partner (SSP) will collaborate to perform the examination and long-term assessment of nuclear power structures, and National Nuclear Corporation Ltd (NNC) will separately assess the risk of rapid stress corrosion cracking of carbon/low alloy steel and intergranular attack of stainless steel components.

The first activity (TEL/SSP) will be directed towards substantiating and redefining predictive models of the mode and pace of deterioration of nuclear power plant structures. The planned work will comprise re-visiting of sites previously examined, obtaining additional experimental data and making further assessment. Furthermore, efforts will be made to include investigation at stations not previously visited within EC member countries. All results will be incorporated into a coherent data base and proposals for a planned inspection and maintenance system will be produced. There is also a need to identify means of monitoring and assessing the ongoing state of tendons and components of buildings and structures which provide protection to nuclear plant. Theoretical research will be conducted into the long-term behaviour of the prestressed concrete pressure vessel (PCPV) at THTR Schmehausen as an example. The results will be used to develop a monitoring programme for the PCPV. The planned work is a complement to contracts FI1D-0030, FI1D-0031 and report EUR 12758.

The second activity (NNC) will be directed towards prediction of the levels of nitric acid which will form in nuclear power plants being decommissioned and assessment of the consequences of such levels with respect to corrosion/degradation effects which could adversely influence subsequent component removal.

B. WORK PROGRAMME

B.1. TEL Work programme

- B.1.1. Re-inspection of Nuclear Power Plants
- B.1.2. Extension to further Nuclear Power Plants
- B.1.3. Compilation of Systematic Data Base
- B.1.4. Planned Maintenance System Development
- B.1.5. Prestressing Options Study in collaboration with SSP
- B.1.6. Monitoring Requirements for Prestressed Concrete.

B.2. SSP Work programme

- B.2.1. Development of a planned maintenance system, including long-term behaviour of materials and structured components
- B.2.2. Study of parameters of a prestressed concrete vessel with regard to partial destressing
- B.2.3. Monitoring requirements for a prestressed concrete pressure vessel and recommendations

B.3. <u>NNC Work programme</u>

B.3.1. Design review
B.3.2. Prediction of nitric acid concentration
B.3.3. Literature survey
B.3.4. Corrosion assessment
B.3.5. Environmental control
B.3.6. Further work.

C. Progress of Work and Obtained Results

Summary of Main Issues

Inspections have been planned and carried out at six nuclear power stations (Berkeley, Bradwell, Trawsfynnyd, Wylfa, Hunterston and Dungeness). Data has been gathered from both the site inspections and subsequent laboratory tests to assess the time taken to initiate reinforcement corrosion. The results from the inspections are currently being analysed. A review has been conducted of possible non-destructive testing techniques for monitoring the condition of prestressing tendons.

Investigations of the long-term behaviour of prestressed concrete pressure vessels during the post-operational state were undertaken. Data was compiled to establish the boundary conditions for the analysis and the measured and calculated creep strains were compared.

Decommissioning strategies being employed at Berkeley, Hunterston 'A' and Latina Magnox power stations have been reviewed and NNC's attention has focused on corrosion aspects and the long-term integrity of structures.

A similar policy of environmental and corrosion monitoring of the primary circuit is now proposed by Scottish Nuclear for Hunterston 'A' to that which has been adopted by Nuclear Electric at Berkeley. An in-vessel atmosphere of dry air has been maintained during the following defuelling at both stations, and monitoring of the air composition and steel corrosion will continue through Stage 1, at least, of the "care and protective maintenance" period of decommissioning.

Although it has been predicted that significant levels of nitric acid form when moist air is subjected to gamma radiation, no measurements have as yet been made at either site to confirm this.

Unfortunatley, we have so far been unsuccessful in co-ordinating our investigation with decommissioning work at Latina.

Progress and Results

1. Inspection of Nuclear Power Plants (B.1.1, B.1.2.)

An inspection strategy was established and carried out during the inspection of reinforced concrete structures six nuclear power stations (Berkley, Bradwell, Trawsfynnyd, Wylfa, Hunterston and Dungeness). The aim of the inspections and associated laboratory testing is to yield the data required by predictive models used to assess the time taken to initiate reinforcement corrosion, where corrosion is caused by either chloride ions or carbonation of the concrete. These tasks are now substantially completed and are providing data for input to subsequent tasks.

2. Compilation of Systematic Database (B.1.3.)

Results from inspections undertaken in the preceding CEC programme and this current programme have been collated into a tabular form suitable for inputting into a database. The format of the database is currently being investigated.

3. Prestressing Options Study in Collaboration with SSP (B.1.5.)

A structural analysis has been carried out to determine the state of stress in a PCPV at the end of its service life and after the removal of all prestressing tendons. The PCPV at Wylfa was selected for this study. Analyses were carried out at four load states;

a) End of service life.

b) As (a) but without reactor gas pressure.

c) As (b) but with concrete temperatures at ambient.

d) As (c) but without prestressing loads.

Particular attention was given to assessing the effects of temperature-related creep in the concrete. The analysis was carried out on an axisymmetric basis using the PAFEC finite element system. The results from the analysis showed that removal of prestress at the end of 40 years of operation is likely to generate tensile stresses adjacent to the inner liner. The magnitude of the stress is probably within the limits that can be carried by the concrete. It would be beneficial to allow the PCPV to cool over a long period of time before following a planned destressing procedure.

4. Monitoring Requirements for Prestressed Concrete (B.1.6.)

An initial investigation into the capability of impact echo and integrated optical fibre test techniques has been carried out. This investigation has been followed by a wide ranging desk study into other NDT techniques for assessing the condition of prestressing tendons. Both the initial investigation and the subsequent review indicated that at present, there is no technique currently available capable of detecting the presence of small flaws at the concrete steel interface in prestressed concrete structures.

5. Study of Parameters of a Prestressed Concrete Vessel with regard to Destressing (B.2.2)

The investigations of the long-term behaviour of prestressed concrete pressure vessels during the post-operational state have been continued.

An extensive literature evaluation has been carried out on the creep of concrete for PCPV. The design creep law for the concrete of the reference vessel of the THTR nuclear power plant Hamm-Uentrop has been compared with other creep functions, which in addition consider the dependence on the age of concrete, when loads are applied. On this basis the influence of the age at loading has been introduced into the THTR creep law.

Using this modified creep function, mechanical analyses of an axisymmetric model of the THTR-PCPV have been made, which enclose a post-operational period of nearly 100 years. The concrete stresses under full prestressing, 50% prestressing of all tendons and full destressing have been calculated and assessed. Likewise the strain behaviour over such a period has been analysed.

6. Corrosion assessment (B.3.4.1)

It is required to demonstrate that pressure vessel integrity will be maintained throughout the decommissioning period. During Stage 1 of the "care and protective maintenance" phase, specimens of irradiated and oxidised material will be monitored at regular intervals to establish rates of corrosion.

Stress corrosion specimens will be included in the corrosion monitoring programme.

7. Corrosion specimens (B.3.4.2)

A sample canister containing corrosion specimens has been retrieved from the lower dome below the core at Berkely. The activity levels are such that the canister must be handled inside 'caves' and information on the depths of corrosion is awaited.

Specimens are also being recovered from the top dome region above the core where activity levels will have been higher. Additional pre-oxidised specimens are being procured for use in the corrosion monitoring programme. 'C' ring specimens for assessing stress corrosion cracking are also being manufactured and these will be strategically positioned around the primary circuit. Most specimens will be located in Reactor 1 and possibly a few also in Reactor 2.

Attempts are also being made to analyse a specimen selected from the graphite monitoring samples (which had been exposed in the reactor over its full operating life) for evidence of nitrate. This could provide valuable information with respect to the ability of the graphite moderator to absorb any nitric acid formed eg during man access shutdowns, which would have a major influence on the nitric acid levels present in the circuit.

8. Monitoring the corrosion atmosphere (B.3.5.1)

During the "care and maintenance" phase of decommissioning at Berkeley Power Station the in-vessel environment will be monitored via fixed sampling tubes and thermocouples. The tubes will be of different lengths and inserted from the pile cap level where the samples will be analysed. Access from the charge face to below the diagrid is via two specially cleared channels from which the channel gage will have been removed. Additional sampling may be possible using the existing BGD system.

A similar procedure is proposed during the corresponding decommissioning phase at Hunterston A.

Atmosphere monitoring and corrosion monitoring will also be undertaken (a) outside the vessels and (b) around the site.

Information on the microclimate in the lagging will be obtained in order to monitor possible corrosion below any lagging on the outside of plant. Damp areas or thin sections such as the lower gas ducts are areas of particular concern.

9. Formation of nitric acid (B.3.5.2)

Depending on the actual level of activity present it is clear from preliminary basic calculations that substantial quantities of nitric acid may be formed. Where large volumes of air occur the surface area is also large and metal loss because of iron replacement is not expected to be excessive. However, cold pockets of metal may exist which gives rise to condensation and concentration of nitric acid. The detail geometry of components at Berkeley and Hunterston A is under consideration to assess where metal loss by corrosion may be critical to the integrity and safety of the plant.

With regard to stress corrosion cracking (SCC) there is ample evidence that SCC can occur with very low levels of nitric acid vapour ie less than 100 ppm, and concentrations in excess of this level will have prevailed for many thousands of hours. There is therefore scope for loss of plant integrity because of SCC in steel components which incorporate high residual stress such as non-stress relieved welds, bellows, bends etc.

Enhanced corrosion on specimens retrieved from the reactor vessels after defuelling are likely to provide the first evidence of acid attack.

10. Further work (B.3.6)

The work so far reported has included a large speculative content and very little has been produced in the way of measured data and results, even from Berkeley as the lead power station for this decommissioning study.

Results from the first corrosion specimens removed from Berkeley since the "end of generation" are unlikely to be available in time to be included in the technical report. Manufacture of new corrosion specimens is targetted for completion by the end of March 1993, and monitoring of the in-vessel atmosphere will not be in operation before this data.

The Commission is therefore invited to consider extending the terms of the contract to allow this information to be included in the final report, possibly as an appendix.

2. AREA No. 2: DECONTAMINATION FOR DECOMMISSIONING PURPOSES

A. Objective

The objective of this research 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 previous programmes (1979-1988)

The following decontamination techniques have been developed and assessed:
- techniques using aggressive agents in liquid and gel-like form;
- electrochemical techniques using various electrolytes;
- hydromechanical techniques (high-pressure water lance, ultrasound);
- decontamination of concrete surfaces by flame jetting.

C. 1989-1993 Programme

Research in Area N° 2 should be pursued with a reduced effort focused on selected techniques. As a new subject, the use of liquid chemical agents carried by a large volume of air, in the form of foam or fog, should be developed with a view to decontaminating large-volume systems. Thermal techniques for removal of concrete surface layers should be investigated from a more general and fundamental view than in the past.

D. Programme implementation

Six research contracts relating to Area No. 2 were in progress during 1992, from which one was completed in the same year.

2.1. ON-LINE DECONTAMINATION OF COMPLEX COMPONENTS FOR UNRESTRICTED RELEASE, USING ULTRASONIC WAVES IN A FLOWING AGGRESSIVE CHEMICAL AGENT

Contractor:	ENEL, Milano	
Contract No.:	FI2D-0016	
Work Period:	July 1990 - June 1993	
Project Manager:	F BREGANI	
	Phone: 39/2/72 24 30 46	Fax: 39/2/72 24 34 96 (or 39 15)

A. OBJECTIVE AND SCOPE

Previous experiments made by ENEL on small valves, using aggressive chemicals, showed that zones with residual contamination remain inside the components.

The present work aims at solving this problem by enhancing the decontamination effectiveness with the action of focused ultrasonic waves. The main objective of the project is to set up and test in real conditions a new decontamination process based on the simultaneous use of ultrasonic waves and aggressive chemicals, with ultrasonic transducers applied outside the components.

This decontamination process, if its expected performances are confirmed, could become a useful tool in decommissioning activities. It should allow to increase the amount of decontaminable parts without having to spend many man-hours and man-Sv (thus, without dismantling before decontamination).

The project is based on experimental investigations, mainly at laboratory scale but also in plant scale. It is the continuation of work performed by ENEL in the framework of previous EC programmes on decommissioning (contract DE-B-005, report EUR 9303; contract FI1D-0002, report EUR 12878; contract FI1D-0023, report EUR 13255).

B. WORK PROGRAMME

B.1. Evaluation, selection and acquisition of special ultrasonic transducers **to be applied to complex components from outside.**

B.2. Decontamination tests on specimens and components **in the DECO loop.**

B.2.1. Preparation of the DECO loop for testing; selection and characterisation of test specimens and components from Garigliano BWR.

B.2.2. Decontamination tests on contaminated specimens.

B.2.3. Decontamination tests on valves: radioactivity measurements, decontamination factor evaluation and secondary waste assessment.

B.2.4. Data analysis.

B.3. Decontamination and dismantling of a part of a real system of a nuclear power station

B.3.1. Preparation of the system part to be decontaminated.

B.3.2. Initial radioactivity characterisation.

B.3.3. Process design and configuration.

B.3.4. Decontamination.

B.3.5. Dismantling and final radioactivity measurements.

B.3.6. Evaluation of secondary wastes.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The design, calibration and testing of the prototype ultrasound transducer and generator was performed in 1991.

During 1992, laboratory activities on contaminated materials were completed; tests performed in the DECO loop on both stainless steel and carbon steel radioactive components (valves and pipes) have allowed to optimize the process parameters for field tests.

Finally, the experimental configuration and procedures for field tests at the Garigliano BWR were defined.

Progress and results

1. Decontamination tests (B.2.)

1.1. Decontamination tests on contaminated specimens (B.2.2.)

Static decontamination tests on radioactive carbon steel specimens were performed in an ultrasonic tank in order to optimize the decontamination solution composition. Results indicate that, when a HCOOH 15% + HF 3% solution is used, the synergic action of ultrasounds and chemicals determine an almost complete removal of the surface activity (well below 1 Bq/cm^2 without an increase in the base metal corrosion (Figure 1).

1.2. Decontamination tests on small components (B.2.3.)

Decontamination tests in the DECO loop were performed on two small stainless steel valves (1 inch and 2 inches internal diameter) from auxiliary systems of the Garigliano BWR, in a HNO_3 5% + HF 1.5% (by volume) solution with the ultrasound transducers applied on the external surface (Figure 2). Moreover, two tests on carbon steel pipes (one with ultrasounds + chemicals and the other with only chemicals) were performed in a HCOOH 15% + HF 3% solution.

1.3. Data analysis (B.2.4.)

The results on a 2 inches valve were compared with those of a similar valve decontaminated in the previous programme using only chemicals.

The comparison indicates that the synergic action of ultrasounds and chemicals can drastically reduce the activity on components of complex geometry where chemical decontamination gives poor performances (Figure 3).

As far as the tests with the carbon steel pipes are concerned, results indicate that the HCOOH + HF solution can reduce the surface activity well below 1 Bq/cm^2 (Figures 4 and 5).

2. Decontamination and dismantling of a part of a real system of a nuclear power station (B.3.)

2.1. Process design and configuration (B.3.3.)

Two lines containing carbon steel components (labelled system A and system B) were selected for demonstration tests at the Garigliano BWR plant; the experimental configuration and the test procedures were defined according to the results of the decontamination tests discussed previously.

The view and layout of the "system B" which will eventually be used for testing, along with the position of the transducers, are shown in Figure 6. It has a hold-up volume of 124 l and internal contaminated surface of 33 m^2. The main components are:

- 6" piping (total length 7 m) with two elbows and two "T" connections;
- two manual 4" valves;
- one manual 6" valve.

The application of the ultrasonic transducers in selected points (see Figure 6) will allow a direct comparison with the action of chemicals without ultrasounds, with a correct evaluation of the effect of the ultrasounds.

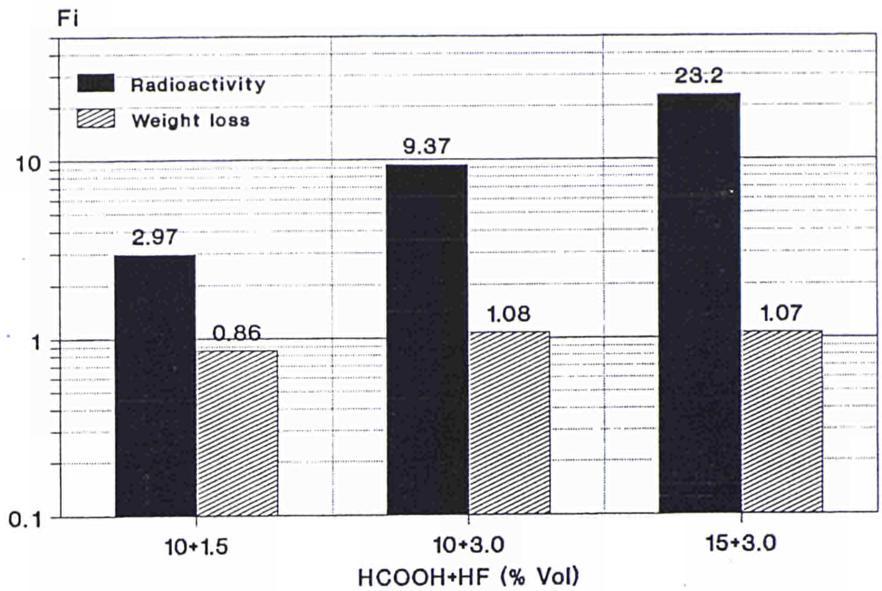


Figure 1 - Comparison of radiometric and weight loss measurements for tests with carbon steel specimens, where $F_i = \dfrac{\text{chemical + ultrasonic effects}}{\text{chemical effect}}$

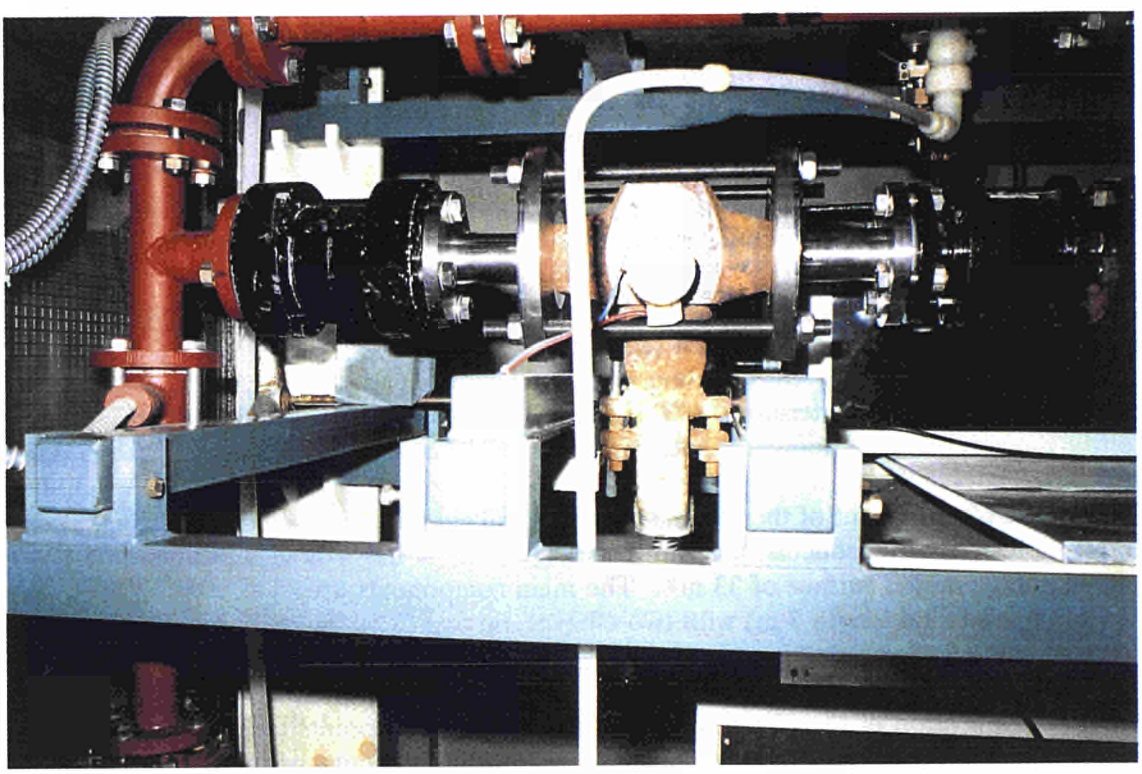


Figure 2 - View of a stainless steel valve mounted on the DECO loop.

Testings conditions		Test in this project		Test in previous project (Ref.EUR 12878)	
Specimen label		SS VALVE V2(*)		SS VALVE N°2 (*)	
Nominal size	(inches)	2		1½	
Initial weight	(g)	16560		18500	
^{60}Co initial activity	(kBq/cm^2)	1.00 - 1.40		0.11	
Decontamination solution HNO_3 + HF	(% volume)	5 + 1.5		5 + 1.5	
Test time	(min)	45	(DYNAMIC)	15	(WASHING)
				430	(1st STEP, DYNAMIC)
				960	(2nd STEP, STATIC)
				65	(3rd STEP, DYNAMIC)
				15	(WASHING)
Flow rate	(m^3/h)	7.0		6.5	
Velocity	(m/sec)	0.9		1.6	
Test temperature	(°C)	25 - 55		20 - 40	
Presence of UT		YES		NO	
UT frequency	(kHz)	21.7	(1 TRANSDUCER)	-	
		24.1	(2 TRANSDUCERS)		
UT power	(W)	300	(PER TRANSDUCER)	-	

(*) SS = Stainless steel

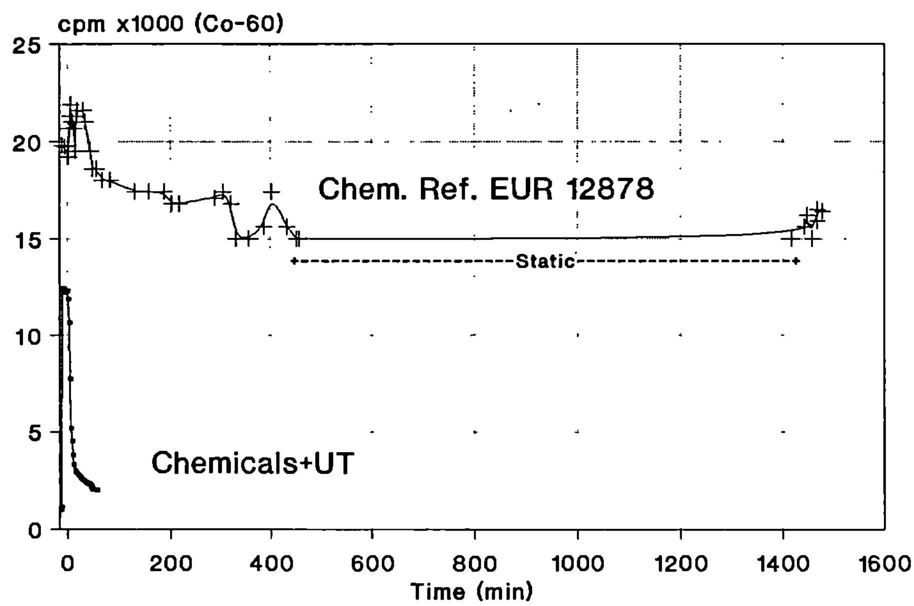


Figure 3 - Comparison of the Co-60 activity trend for tests with stainless steel valves with and without ultrasounds.

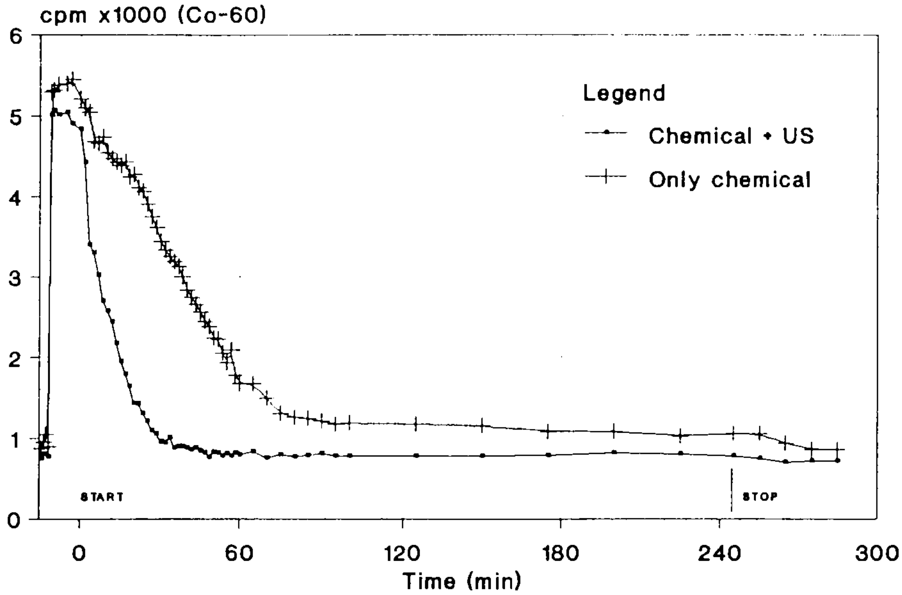
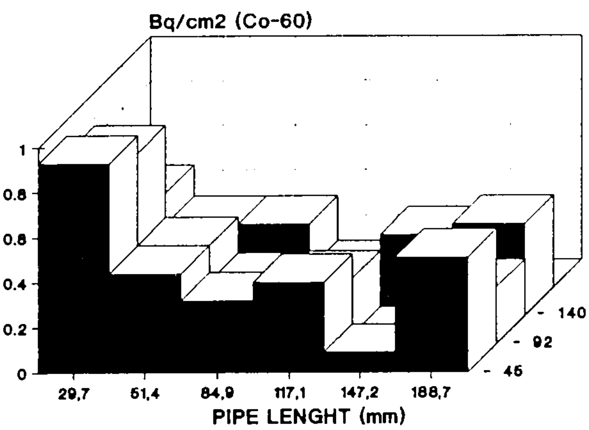


Figure 4 - Result of the decontamination test on the DECO loop with carbon steel pipes. Co-60 radioactivity vs time.



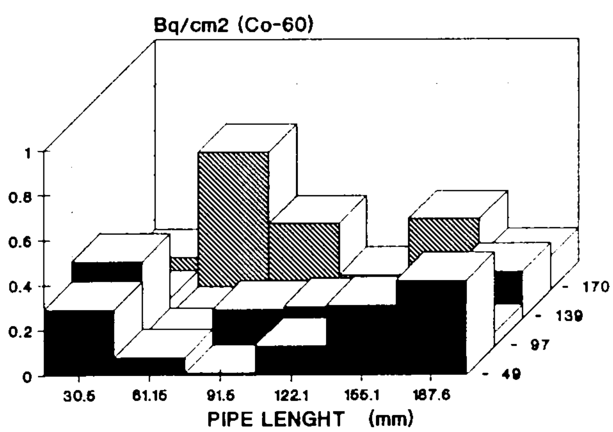
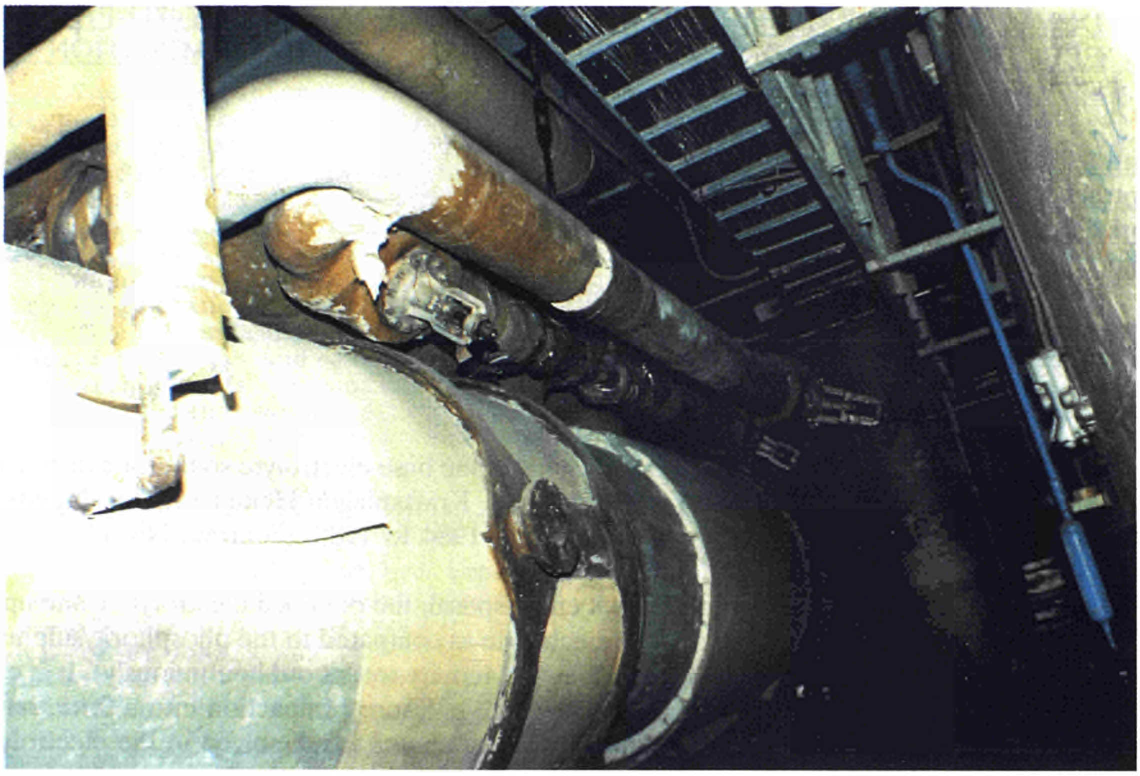


Figure 5 - Residual activity distribution on the carbon steel pipe after the test with only chemicals and chemical + ultrasounds.



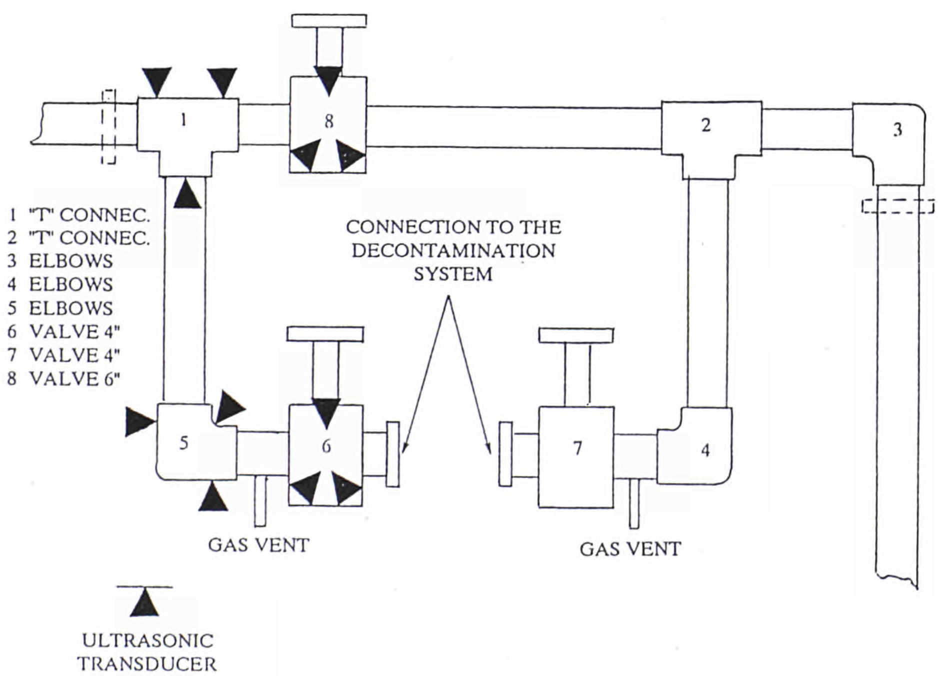


Figure 6 - View and lay-out of the loop B selected for the demonstration tests at the Garigliano BWR plant.

2.2. DEVELOPMENT AND OPTIMISATION OF AN EASY-TO-PROCESS ELECTROLYTE FOR ELECTROCHEMICAL DECONTAMINATION OF STAINLESS STEEL

Contractor:	KA, Heidelberg
Contract No.:	FI2D-0020
Work Period:	July 1990 - June 1992
Project Manager:	A STERINGER
	Phone: 49/6221/39 42 50 Fax: 49/6221/39 47 07

A. OBJECTIVE AND SCOPE

This work aims at optimising an acetyl-acetone base electrolyte so that it can be used for electrochemical decontamination of stainless steels. Kraftanlagen Heidelberg developed the electrolyte under the preceding EC programme from 1984 to 1988, (contract No. FI1D-0004, report EUR 12383).

With regard to waste management and disposal, the obtained electrolyte came up to all expectations. An advantage of the organic electrolyte as compared to the phosphoric/sulphuric acid electrolyte is its long radiological service life (the activity settles out continuously). It is easy to convert the crystalline by-product (sediment) by high-pressure compaction into a form that is suitable for disposal. As only little residues of acetyl-acetonates are dissolved in the electrolyte, it is possible to considerably reduce the electrolyte volume by evaporation.

In tests with radioactive samples of carbon steel, the obtained results concerning removal effects, duration of treatment, surface quality, and decontamination factors, were satisfactory or good. However, pitting was observed in the tests with samples of stainless steel. As a consequence, the surface was not uniformly removed. Parts of the original surface were visible for a long time. This resulted in poor decontamination factors or long treatment times, respectively. In addition, larger volumes of secondary wastes were produced than with a uniformly removed surface. It is therefore required to optimise this electrolyte, if it is to be used for the treatment of stainless steel.

B. WORK PROGRAMME

B.1. Quantitative investigations concerning the dissolution mechanism
B.2. Optimisation of the aqueous electrolyte through replacing the potassium bromide by other conductive salts.
B.3. Investigations into scattering and its effect on abrasion, surface quality and decontamination factor.
B.4. Development of a water-free electrolyte.
B.5. Decontamination tests with contaminated samples.
B.6. Processing of spent electrolyte.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of Main Issues

The dissolution mechanism when using the potassium fluoride (KF) electrolyte and glycol electrolytes was investigated and partly clarified.

The replacement of KBr by KF as conductive salt showed positive results. Satisfactory removal rates and good anode current efficiencies along with smooth surfaces were achieved.

The electrolyte along with KF as conductive salt demonstrated that the throwing behaviour depends on the viscosity and on the rated current density.

When using glycol as solvent along with potassium bromide as conductive salt, no pitting was found on the electrode.

The decontamination tests with the aqueous electrolyte using KF as conductive salt, and also the tests with the organic glycol electrolytes along with KBr as conductive salt showed good results.

Waste management options (recovery of electrolyte and waste treatment) have been investigated and treatment methods recommended.

Progress and Results

1. Quantitative Investigations concerning the Dissolution Mechanism (B.1.)

In the annual progress report 1991, the results achieved so far using an aqueous electrolyte were shown.

The dissolution mechanism postulated earlier for the conductivity salts on CL^- or Br^- basis was verified. The experiments with the KF-electrolyte were carried out using cyclovoltametric methods. The following results were obtained:

Anode reaction:

1. $^{TR}m \rightarrow {}^{TR}m^+ + e^-$ (E)
2. $^{TR}m^+ + F^- \rightarrow {}^{TR}m\text{-}F$ (C)

(C) = Chemical reaction
(E) = Electron transition
^{TR}m = transition metal

These steps took place in the sequence E - C inside the electrochemical double layer.

When diffusing into the electrolyte volume, the anion exchange takes place on the transition metal ion:

3. $^{TR}m\text{-}F + acac^- \leftrightarrow {}^{TR}m\text{-}acac + F^-$ (C)

The equilibrium shown in 3. was the subject of the experiments to determine the life-time of the electrolyte. It will later be shown that an unsatisfactory equilibrium would be detrimental to the application time of the electrolyte.

The mechanism postulated for the dissolution of conductive salts based on KBr or KCl was also confirmed valid for the organic electrolyte using KBr as conductive salt:

Anode reaction:

4. $x^- \rightarrow x^{\cdot} + e^-$ (E) $x^- = Br^-, Cl^-$
5. $x^{\cdot} + {}^{TR}m \rightarrow {}^{TR}m^+ - x^-$ (C)
6. $^{TR}m\text{-}x + acac^- \leftrightarrow {}^{TR}m\text{-}acac + x^-$ (C)

However, consideration must be given to the fact that Fe^{2+} may remain in solution or soluble $Fe(acac)_2$ can be formed.

The following may be recapitulated:

1. In the presence of F^-, the transport reaction (E) is determined by the metal/metal ion transition.
2. In the presence of Br^-, the Br^- is oxidized and dissolves the metal.
3. The metal halogenides are converted to metal acetyl acetonates in a chemical step (C). This is the basis for the entire process.
4. The use of KF in an organic solvent such as glycol showed an obscure behaviour. This electrolyte had no decontamination effect due to the bad removal performance; further tests would only be of theoretical interest.

2. Optimization of the Aqueous Electrolyte by substituting KBr through other Conductivity Salts (B.2.)

The equilibrium described in B.1, and its effect on the life_time of the electrolyte must be investigated with the objective being to minimize the KF-losses through complex formation.

The equilibrium (3) is dependent on the acetyl acetonate concentration.

To be able to increase the acetyl-acetonate concentration, the concentration of isopropanol had also to be increased.

The acetyl-acetone proportion was kept constant in all further experiments.

With this electrolyte the surface quality (pitting?) and removal behaviour and concurrently the fluoride concentration were determined.

A decreasing fluoride concentration can be seen until after approx. 30 min. and a current load of about 9000 Cb a stationary equilibrium is reached. If the electrolysis is interrupted, the F^--concentration rises again as F-ions are released from the complex, finally achieving the initial concentration (see figure 1).

The filtration was carried out to ensure that the regeneration of fluoride resulted from soluble complexes. This is very important for the process as the sediment, which should be fluoride-free, is continuously separated from the solution.

The pH-value or the electrolyte changes from an initial value of 6.5 and approches an asymptotic value of 7.5. In this range no differentiated removal behaviour was observed.

A very important conclusion may be drawn from these experiments; a pH-control in the process is not necessary.

With this composition an electrolyte had been developed with the following properties and specifications:

1. High current yield.
2. High current efficiency and removal rate so that the required short decon tamination times are met.
3. Smooth surfaces.
4. Easy-to-remove residue as waste.
5. High life time of electrolyte.
6. Simple recycle and/or disposal.
7. Low specific waste volume.

3. Decontamination Tests with Contaminated Samples (B.5.)

Test samples doted with ^{59}Fe were decontaminated. The results were reported in the 3rd PR, July - December 1991.

The decontamination of samples from a NPU with varying degrees of contamination was to be demonstrated.

Decontamination of LL-activity material

In a long-term experiment, 5 pieces of piping with an area of 31 cm² each and an average contamination of 11.5 Bq/cm² were decontaminated in 500 ml electrolyte using a current density of 15 A/dm².

After a decontamination time of 6 - 16 minutes and the removal of 20 - 24 µm material, no activity above the lower detectable limit could be found (see figure 2).

The decontamination process shows that:

1. approximately 90 % of the activity arising from contamination is contained in the deposited layer (metal oxides, haematite, magnetite),
2. removing approximately 30 µm of material will be sufficient for decontamination. This, of course, applies only if the layers found in actual case are not thicker than those found in the tested material samples.

The activity in the electrolyte accumulates during decontamination operations. This is very important as far as the life_time of the electrolyte is concerned.

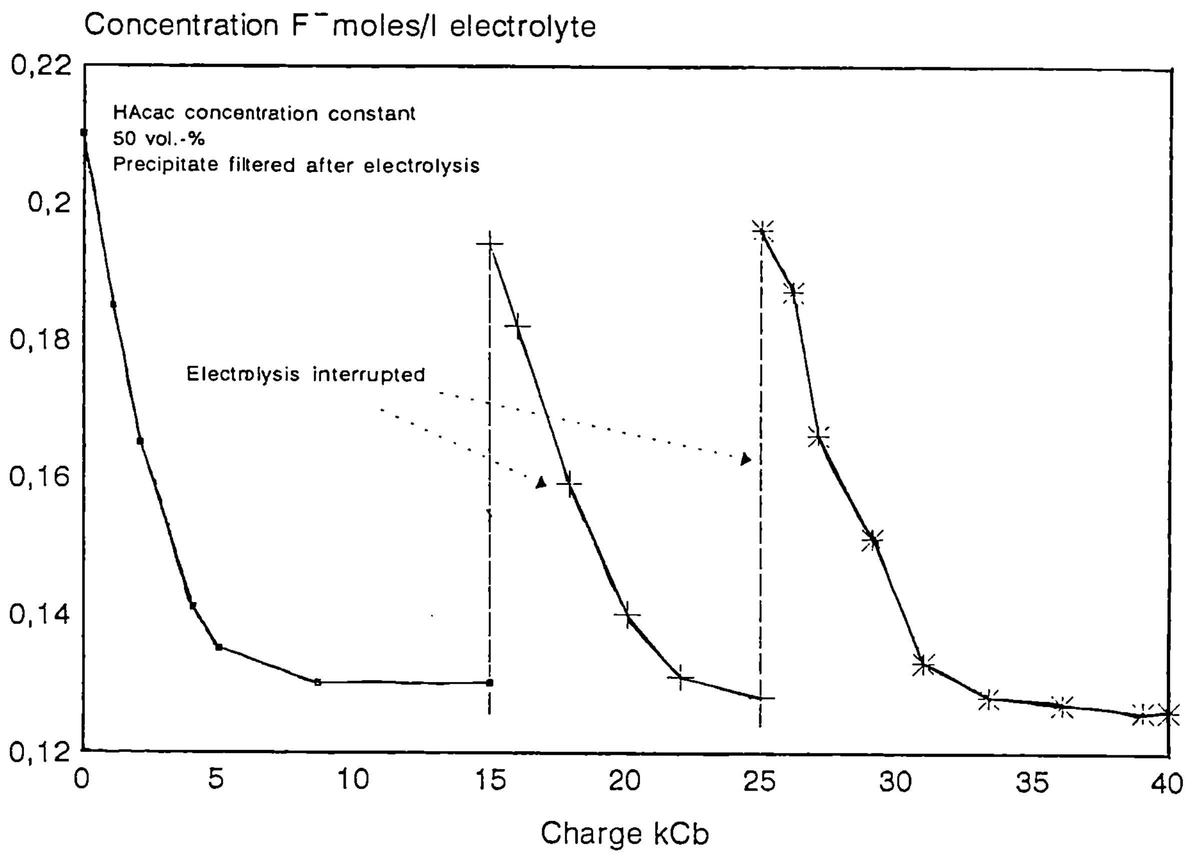


Fig. 1 Fluoride Concentration after interrupted Electrolysis

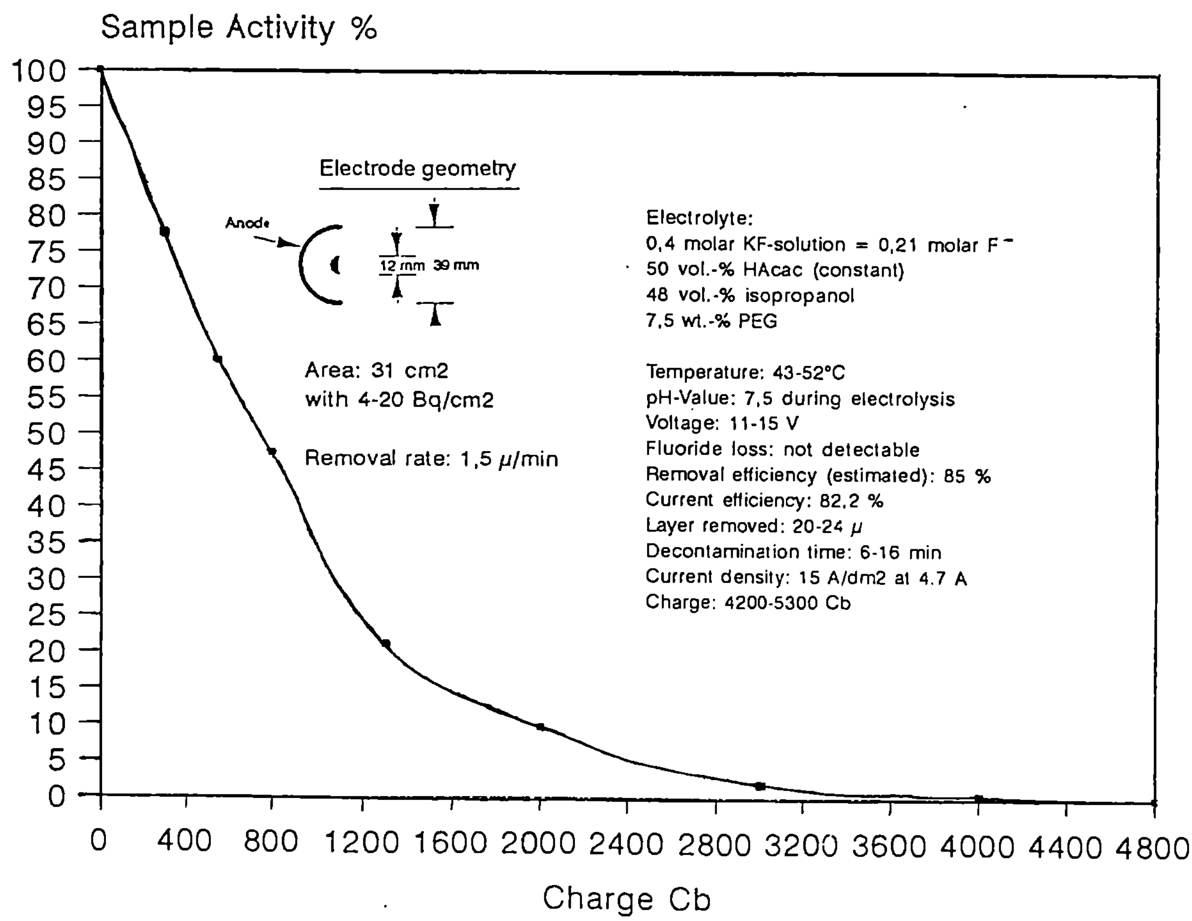


Fig. 2 Activity Loss from LLW-Samples versus Charge

In the case of the electrolyte, a considerable share of the activity is dependent on the ^{60}Co concentration. However, as ^{60}Co is precipitated and therefore is removed from the electrolyte, the activity share of ^{137}Cs must increase. The activity distribution between sediment and electrolyte was therefore studied.

Cooling the electrolyte from 45 °C to 25 °C lessens the solubility product of the metal acetyl acetonates; this further reduces the activity in the electrolyte.

Activity distribution between solution and precipitate

Analysis and measurement show that:

1. the solution (electrolyte after filtration) is enriched with mainly ^{137}Cs. ^{60}Co and traces of ^{241}Am were also registered.
The ^{60}Co activities found are in accordance with the corresponding $Co(acac)_2$ solubility product.
2. the precipitate contains almost only $^{60}Co(acac)_2$ as activity carrier.

The electrochemical life time of the electrolyte is almost indefinite when the concentrations of acetyl-acetone and isopropanol (losses through evaporation) are kept constant.

The electrolyte is regenerated by passing it through an ion exchange unit when the activity level limit is reached; this is essentially caused by ^{137}Cs.

The tests showed that using this electrolyte, up to 4.5E+5 Bq/kg dry ion exchanger could be adsorbed.

The filtered electrolyte had an activity amounting to 180 Bq/l after ion exchange in comparison to 21 kBq/l before treatment; this means that a decontamination factor of approximately 116 had been achieved.

The activity level of the electrolyte lay below the lower level of detection. The same result was realized using ^{59}Fe.

The regenerated electrolyte had a pH-value of 4 and can be used again without modification; during the process, the pH-value rose to 7.5.

4. Waste Disposal and Recycling Possibilities (B.6.)

Disposal of the electrolyte will be necessary when the electrochemical requirements and/or surface quality of the samples to be decontaminated are no longer met.

Suitable treatment routes for conditioning and disposing of the waste electrolyte have therefore been suggested (see fig. 3).

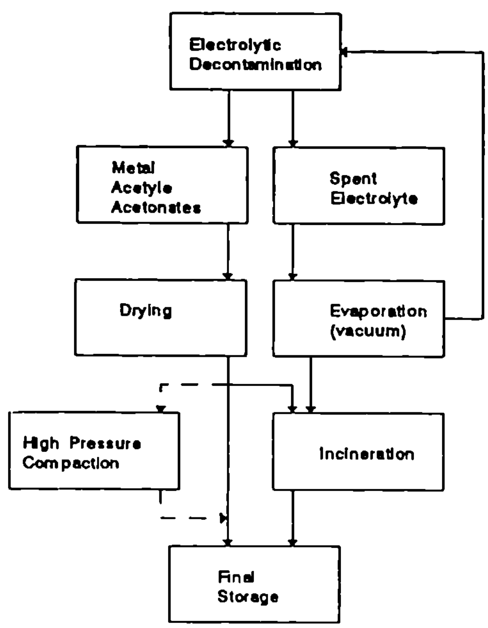


Fig. 3: Waste management options

A deterioration of surface quality may be recognised by optical means or by the decrease in removal rate.

The developed electrolyte showed a very high life time in the tests carried out.

After treating a large amount of steel, i.e. a decontamination campaign has been ended, the electrolyte is easily regenerated and is ready for new service.

Recycling of the aqueous electrolyte

The distillate consists of the components water, isopropanol and acetyl-acetonate. After gas chromatography, KF as conductive salt and solvent is added so that the original composition is again reached.

The residue tends to decompose together with smoke generation at higher temperatures. Because of this, sodium hydroxide is added to the electrolyte before distillation; the metal hydroxide sludge prevents the decomposition of the metal acetyl acetonates up to a temperature of some 170 °C. The amount of sodium hydroxide needed is calculated according to the metal content of the residue.

The sludge is then incinerated.

Disposal of the metal acetyl-acetonates arising from the aqueous electrolyte

During the process, metal acetyl-acetonates are continuously being precipitated. They may easily be separated from the electrolyte either by sedimentation or filtration.

The metal acetyl-acetonate mixture as activity carrier is a mechanical mixture. It can be disposed of after drying according to disposal regulations. The mixture of dried metal acetyl-acetonates is stable up to 140 °C.

The metal salts decompose at 170 °C and with air they burn forming metal oxides.

The specific calorific value of $Fe(acac)_3$ in mixture with $Ni(acac)_2$ and $Cr(acac)_3$ was found to be 23.9 kJ/g.

A possible option for the disposal of metal acetyl-acetonates may be the high pressure compactation of the waste.

Disposal of the organic electrolyte on a glycol basis

Only acetyl-acetone and iso-propanol may be separated from the organic electrolyte by distillation at normal pressure as the temperature should not be higher than 170 °C. The glycol can also be separated when vacuum distillation techniques are used; the organic fraction may be recycled or incinerated.

2.3. MICROWAVE SYSTEM TO SCARIFY CONCRETE SURFACES

Contractor:	ENEA, Casaccia
Contract No.:	FI2D-0024
Work Period:	January 1991 - December 1993
Project Manager:	P CORLETO
	Phone: 39/6/30 48 40 55 Fax: 39/6/30 48 39 51

A. OBJECTIVE AND SCOPE

For the decommissioning of nuclear installations, it may be necessary to scarify masonry or concrete walls, removing at least 20-25 mm of plaster or concrete, in order to eliminate the incorporated contamination. Among the available techniques, the one based on the effect of microwaves upon the water contained in cement is very interesting; the water evaporates and the steam pressure within the pores shatters the cement in small splinters. This method is suitable for remote operation and produces no liquid effluents.

The research project concerns a microwave system consisting in a bell for the scarification and the suction of the splinters and in a support structure for the bell, compatible with the remote handling systems available at the ITREC plant. The system will be manufactured, set up and tested at the ITREC plant in Trisaia on non-radioactive and on radioactive concrete surfaces.

As regards the innovative aspects of the research programme, it is intended to optimise the interconnection between the microwave generators, develop an efficient system to contain and collect the particulate, improve the efficiency of the particulate filtration and, finally, render the whole system flexible and easily operated.

B. WORK PROGRAMME

B.1. Design and construction of a prototype microwave system

B.2. Design and construction of the support structure.

B.3. Trial operations on a non-radioactive concrete wall.

B.4. Testing of the prototype on a radioactively contaminated concrete surface.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The microwave system was designed and manufactured in 1991 (B.1, B.2); it comprises: a microwave device with three 5.5 kW magnetrons generating 2450 MHz waves, a support structure to move the device over the concrete surface to be scarified and a vacuum apparatus to remove the debris produced.

In the course of 1992, tests on non-radioactive concrete surfaces were performed, and the optimum generating parameters of the system were determined. During the tests, several modifications were envisaged to improve the system and it was decided to implement some of them. The relevant design and construction work is now under way; within June 1993, the testing of the upgraded system will take place. The tests on the radioactively contaminated surfaces will be carried out during the second semester of 1993.

Progress and results

1. Trial operations on a non-radioactive concrete wall (B.3.)

The tests on non-contaminated concrete are performed in the Technological Hall at ENEA's Trisaia Research Centre. For the trial operations, the concrete slabs obtained at the Trisaia Centre have been used for the preliminary tests. They are covers of trenches manufactured more than twenty years ago and of unknown characteristics.

Two slabs were expressly produced for the tests; they are described in the 1991 annual report. Two more slabs are at present under construction, similar in size and rebar disposition to the two now available, but differing as to composition; additives have been included to render them similar to the concrete used for Italian power stations.

Test results. About ninety scarification tests were carried out in order to examine the effect the following parameters:

- the total microwave power (6.84 to 17.1 kW)
- the advancement rate of the microwave device (1.68 to 8.4 mm/s)
- the stand-off, distance of the irradiation heads from the concrete surface (15 to 30 mm)
- the number of generators in operation,

have upon the dimensions of the scarified area, the duration of the scarification run, the quantity and dimensions of the debris produced, the number of explosions per run.

The first thirty-one tests were aimed at determining roughly the operational parameters of the apparatus, while the scarification efficiency was only of secondary interest. Scarification with long and uniform furrows was not obtained, except in two cases; but it was possible to establish that a total magnetron power of 17.1 kW, an advancement rate of 1.68 mm/s and a 20 mm stand-off produced the best results.

After the first tests, modifications were executed to the suction system. The cross section and shape of the hose at the connection with the suction bell were changed to allow the flow of larger fragments; a second more external skirt was provided for the suction bell to avoid the debris being propelled outside the bell.

The tests from the thirty-second one onwards were carried out on the two purpose-made slabs. With a total power of 17.1 kW, a stand-off of 20 mm and an advancement rate of 1.68 mm/s, scarification have been obtained in about 10-11 min over areas of 1200-1400 mm x 150 - 200 mm, reaching depths of 15-35 mm. The quantity of debris produced was between 6 and 9 kg. The removal rate achieved varied from 5 to 6 cm^3/s but, in some cases, 12 cm^3/s was reached. The number of explosions per run varied within the 30-70 range. The furrows obtained were continuous and presented well-defined limits (see Figures 1 and 2). No damage or flaws appeared on the concrete structure, notwithstanding the numerous explosions that had occurred.

When the microwave system was run more and once over the same area, a deeper furrow was produced; in one case, a depth of 6 cm was reached. The depth obtained appears to be limited by the rebars that physically impede the penetration of the microwave system.

The rebars after visual examination have not appeared to be impaired by the microwave irradiation. Moreover, the steel bars apparently have no influence on the shattering of the concrete; and neither do the dimensions of the bars. Their dimensions also do not influence the reflected energy; about 25% of the microwave power produced by the generators is reflected by the reinforced concrete independently from the size and mesh of the steel bars present.

The importance of having three microwave generators with adjacent irradiation heads operating simultaneously has been demonstrated in the last tests performed. Actually, with one or with two generators in function, the results obtained are greatly inferior to those with the three generators operating simultaneously at full power.

All the debris produced were transferred to the vacuum system's drums. It was not possible to remove the debris satisfactorily when adjacent furrows overlapped, because the bell's skirt did not come in contact on all sides with the surfaces being scarified and the conditions for suction ceased. Therefore, each furrow was generated at some distance from the adjoining one, and in these cases, the debris were successfully removed. Future design work is planned to have the vacuum system operate properly also with overlapping furrows.

The debris produced are mostly (about 60% w) above 5 mm in diameter and the quantity of fine powder is below 0.1 % w. No trace of dust was noticed in the atmosphere during the tests. Even when the vacuum system was not performing at its best, as was the case when furrows overlapped, there was apparently no dust in the air. With respect to the concrete, no difference was found in the scarification efficiency for the two types of cement (325 and 425) used for the tests; both had been cured for over four months. Instead, a negative effect on the efficiency was noted when, as mentioned, old concrete, albeit of uncertain characteristics, was utilised. Paint did not affect the scarification efficiency, as was shown by covering with epoxy paint one half of one side of the slabs.

The microwave leakage close to the suction bell containing the irradiation heads was measured at various distances, using a Narda instrument with a top-scale reading of 100 MW/cm^2. At 5 cm from the bell, the leakage was below 2 MW/cm^2 in the optimal shielding conditions, that is when the machine was at the start of the run; but during operation the leakage grew, because, as the machine advanced, the furrow produced the movement of the skirt and the rocking of the machine caused the microwaves to escape. Values of about 10 MW/cm^2 were found at about 50 cm; these are above those allowed by regulations; but it must be considered that the equipment is to operate in a radioactive environment without the operators in its vicinity. Anyhow, further work will be necessary to overcome this inconvenience.

In conclusion, it was possible to achieve a flexible, reliable and easy to operate apparatus, which, by means of an appropriate interconnection between the microwave generators, begets continuous scarification of the concrete surfaces and efficient removal of the debris produced.

During the tests, areas have been identified that might be further investigated profitably, like the design of the suction bell to obtain overlapping furrows, the improvement of the system's manoeuvrability and the scarification efficiency as related to the additives in the concrete. To improve the system's manipulation capability, it is envisaged to connect the magnetrons to the irradiation heads by means of flexible wave guides, 6 m long or more; it will thus be responsible to position the magnetrons outside the radioactive area while the irradiation heads connected to the suction bell will be within the hot area, thereby facilitating the maintenance and enhancing the durability of the wave-generating system.

The design and construction of the modifications are at present under way. In March, the upgraded system will be tested on non-radioactive concrete.

2. Testing of the prototype on a radioactively contaminated concrete surface (B.4.)

In June 1993, the tests on the radioactive concrete surfaces will take place in the service corridor above the process cells at the ITREC Plant (pilot reprocessing) at the Trisaia Centre.

Figure 1. Test n. 49 on slab 325

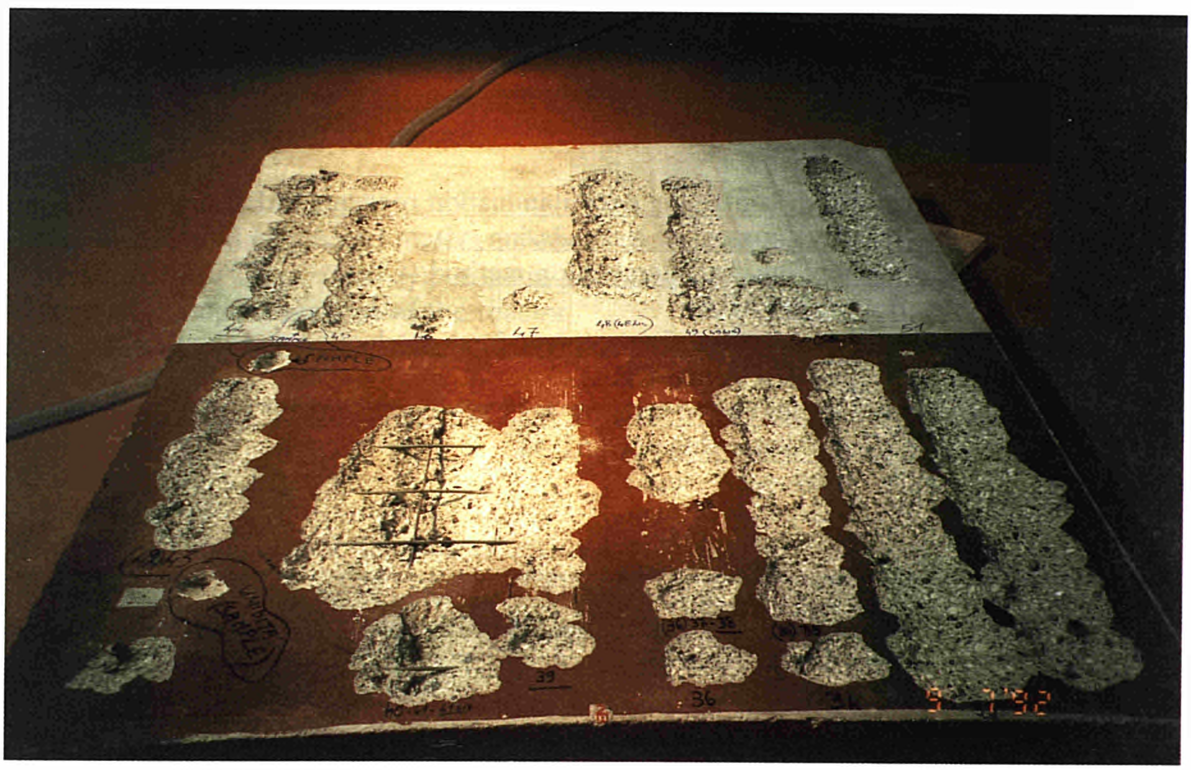


Figure 2. Overall view of slab 325

2.4. DECONTAMINATION OF LARGE-VOLUME NUCLEAR COMPONENTS USING FOAMS

Contractors: CEA-Cad, AEA Winf
Contract No.: FI2D-0035
Work Period: October 1990 - June 1993
Coordinator: J P GAUCHON, CEA-Cad
Phone: 33/42 25 61 93 Fax: 33/42 25 35 45

A. OBJECTIVE AND SCOPE

There are only a few methods for in-situ decontamination of very large components usually in complex forms, such as large valves, reservoirs, heat exchangers, turbines, vessels, boilers.

The foam application processes have the major advantage of using only small quantities of liquid and being able to forcefully penetrate everywhere. Suitable chemical reagents are added to the foam, which acts a dynamic carrier.

In this contract, a technique of permanent foam circulation will be sought, so that decontamination can last for several hours in order to be as effective as possible and to use only a minimum amount of liquid. Decontamination factors of over 100 are expected.

The objectives of the programme are to:

- develop and demonstrate an effective in-situ decontamination technique for large-volume components using chemical foams containing decontamination reagents;
- minimise the volume of secondary wastes produced and demonstrate a treatment and disposal route, e.g. electrolytic processes, wet oxidation.

B. WORK PROGRAMME

B.1. Chemical foam formulation containing decontamination reagents (AEA and CEA)
B.2. Foam production and development of a circulation system (AEA and CEA)
B.3. Small pilot tests to qualify the decontamination method (CEA)
B.4. Secondary wastes treatment (AEA)
B.5. Design, construction and operation of a prototype foam production and circulation rig; non-radioactive demonstration (AEA and CEA)
B.6. Industrial application by radioactive tests on a 25 m^3 contaminated vessel from Winfrith Steam Generating Heavy Water Reactor (AEA)

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

Parametric studies were conducted to finalize a foam formulation with properties (moisture, lifetime, circulation) suited to the decontamination of the Winfrith de-aerator.

Two processes have been developed in order to treat effluent produced by the foam decontamination process.

1. Chemical foam formulation containing decontamination reagents (1.)

The process retained after definition in a liquid medium is applied in two steps at room temperature: a degreasing step in soda medium, followed by a descaling step in sulphuric medium. Both foams are composed of two surfactants and one alcohol.

The surfactants are:

- an amphoteric surfactant: sulphobetain (Amonyl 675 B)
- a non-ionic surfactant: glycoside alkyl-ether (Oramix CG110-60).

The alcohol is a destabilizing agent which enables the control of the humidity and drainage of the foam. In soda medium, 2 Pentanol is used, whereas 4 Methyl 2 Pentanol is required in sulphuric medium.

Degreasing treatment

The aim of this operation is to facilitate the contact between the reagents and the surface to be treated during the descaling sequence. The formulation and operating conditions retained are:

NaOH: 2 mol/l Oramix: 0.4 % Amonyl: 0.3 % 2 Pentanol: 0.3 % (at the start)	Air flow rate: 122 l/h Liquid flow rate: 10.6 l/h Initial bulk factor # 12.5

This foam is particularly well adapted, as it allows a good drainage wall (26%) and transfers 40% of the liquid at the top of the reactor.

Descaling treatment

The surface decontamination is mainly achieved during this sequence. Studies were carried out to optimize the (Oramix/Amonyl) ratio in terms of irrigation of the wall and supply of the reagents at the upper sections of the reactor.

The formulation giving best moisture, drainage and coalescence levels was found to be:

H_2SO_4: 2 mol/l Oramix: 0.8 % Amonyl: 0.3 % 4 Methyl 2 Pentanol: 0.5 % (at the start)	Air flow rate: 180 l/h Liquid flow rate: 13.8 l/h

Figure 1 shows the characteristics of this specific foam.

Efficiency of the decontamination

The formulation was applied to a sample from the Winfrith de-aerator. The results shown in Table I confirm the possibility of reaching decontamination factors close to 100 and residual activities less than 1 Bq/cm^2. It should be noted that in this test, the sample was rinsed being simply left to soak in water.

2. Foam production and development of a circulation system (B.2.)

Tests were carried out to optimize the generation and the circulation of the foam. Parametric studies were conducted to improve the operating conditions of static foam generator.

The best results were obtained with a generator with following characteristics:
- length: # 15 cm
- volume = 12.2 cm^3
- weight of Foraflon: 1.83 g.

Investigations were also made on the circulation of the foam ensuring no pluggage of the experimental loop. This loop was composed of:
- a reactor (capacity: 30 l; height: 1 m)
- a mixer (6 l)
- a foam generator.

3. Small pilot tests to qualify the decontamination method (B.3.)

The assembly of a pilot unit equipped with a 500 l reactor was completed. A capacity of about 2.1 m^3 is to be connected to this pilot unit soon.

4. Secondary wastes treatment (B.4.)

The study and development of a process for the treatment of the waste produced by foam decontamination were carried out. Two processes based on the catalytic oxidation of the organic material in the solution were investigated: the Wetox and the Ag (II) processes (Fig. 2).

Both processes are suitable, as they enable the destruction of 85.90 % and 94.6 % of the organics respectively. These figures may though be enhanced by further improvement.

TABLE I: Decontamination of test specimens from the Winfrith de-aeretor

Test No.	Foam characteristics reagent	Time of contact (hour)	Radiochemical activity (Bq/cm^2)		D.F.
			Initial	Final	
10	NaOH (3 mol/l) Gas vector: air	0	89.16 $\pm$ 5.39	-	
		1	-	81.69 $\pm$ 4.95	1.1
	then H_2SO_4 (2 mol/l) Gas vector: air	0	81.69 $\pm$ 4.95	-	
		2	-	62.07 $\pm$ 3.77	1.3
		4	-	7.37 $\pm$ 0.48	11.1
		6	-	0.83 $\pm$ 0.06	98

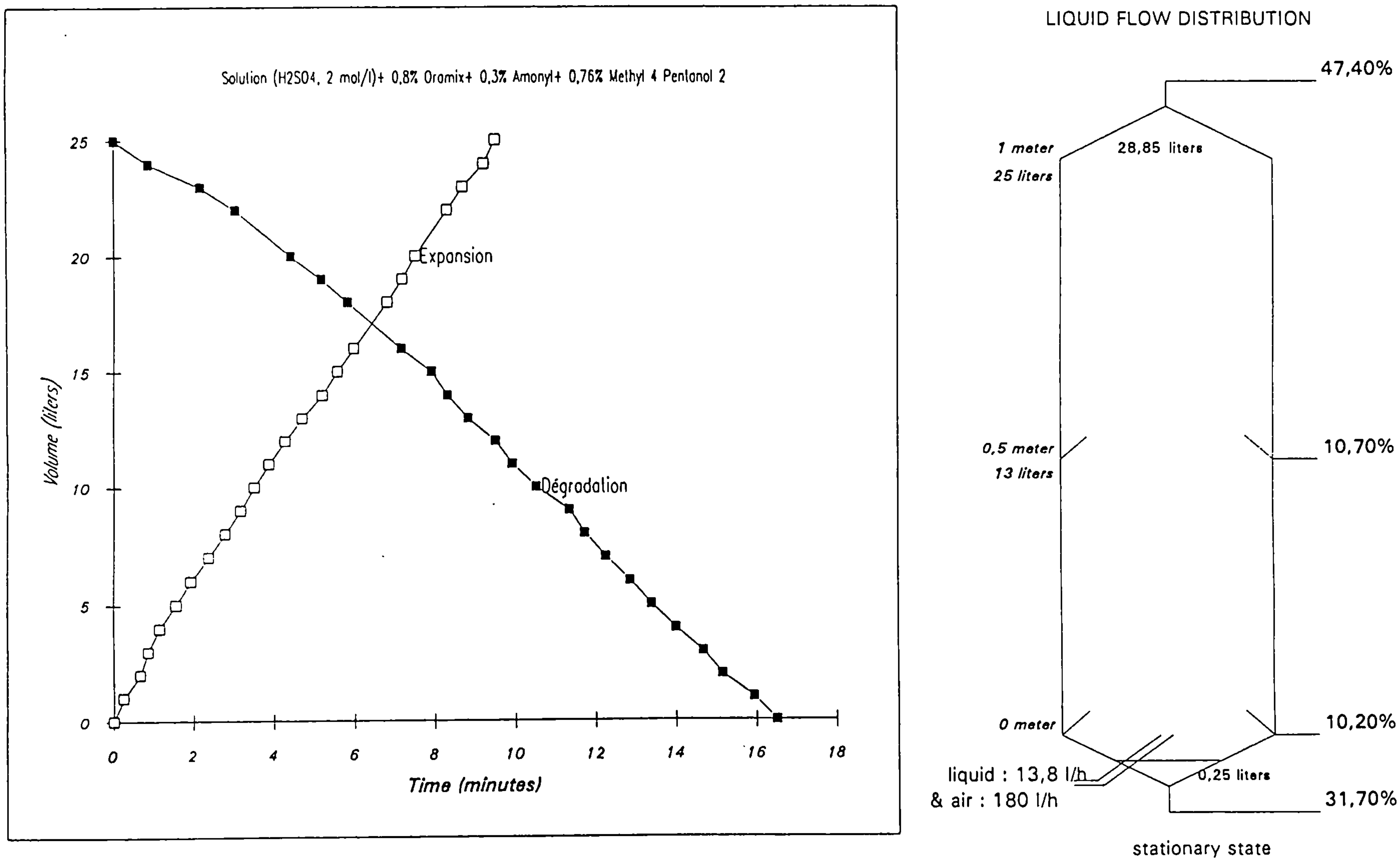


FIG 1: FOAM EXPANSION AND DEGRADATION, PICKLING THIRD STEP

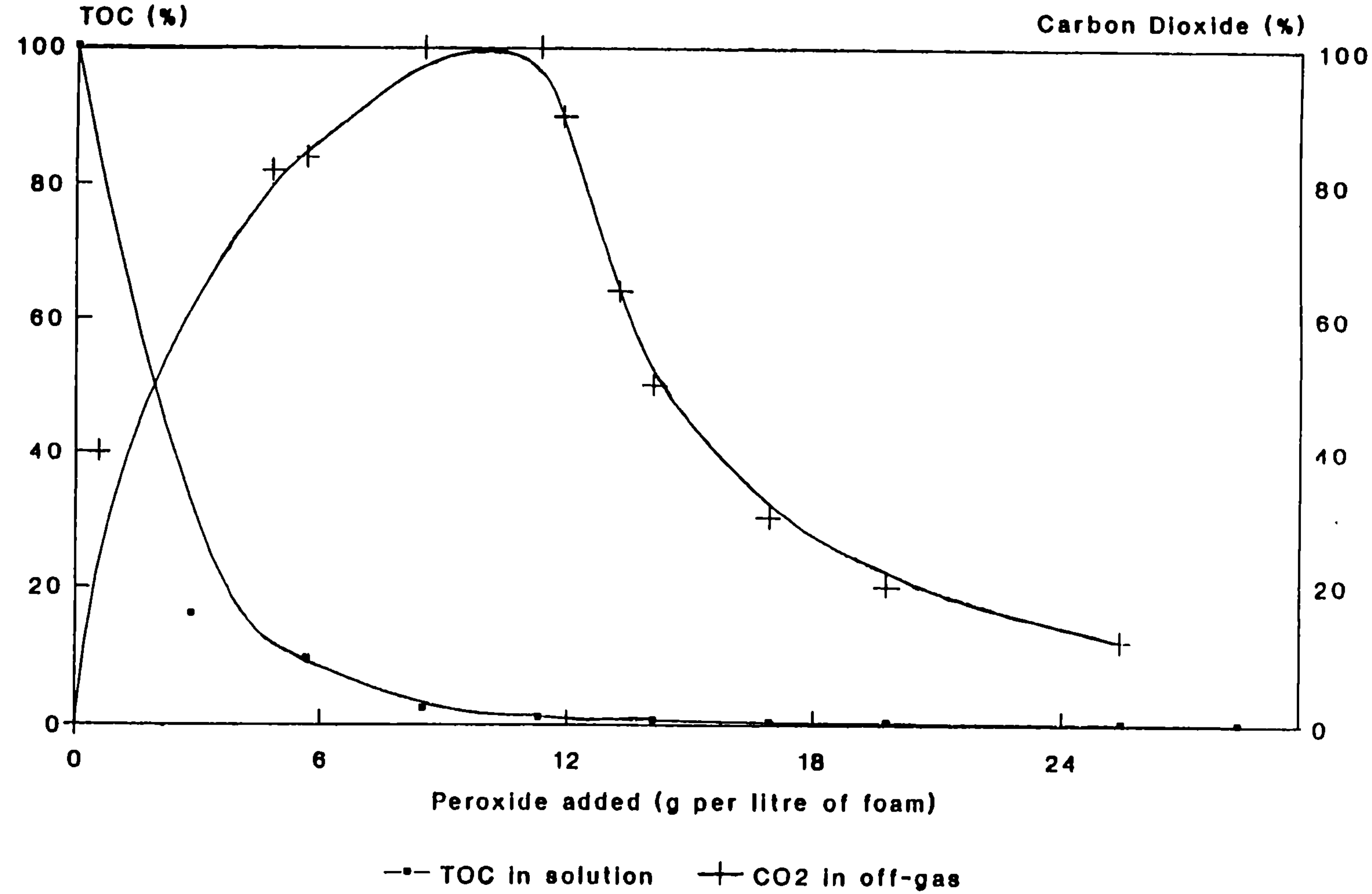


FIG 2 : WETOX PROCESS, CARBON REMOVAL. DECONTAMINATION FOAM

2.5. DECONTAMINATION OF AN EVAPORATOR OF A PILOT REPROCESSING PLANT (EUREX-SALUGGIA) USING A CHEMICAL AGENT DISPERSED AS FOG

Contractors:	ENEA-EUREX, Saluggia
Contract No.:	FI2D-0043
Work Period:	January 1991 - June 1993
Coordinator:	V CALI, ENEA-EUREX
	Phone: 39/11/483 225 Fax: 39/11/483 280

A. OBJECTIVE AND SCOPE

The programme proposes to develop a technique using a chemical agent dispersed as fog for the decommissioning of nuclear installations, finalised towards the decontamination of the thermosyphon evaporator used for the concentration of the end product (enriched uranium and plutonium solutions) of ENEA's pilot reprocessing plant EUREX at Saluggia.

The programme includes:

- a theoretical study of the processes that the inside walls of the evaporator have undergone in contact with the U-Pu solution, in order to obtain valid hypotheses on the behaviour of the contaminants during their attack and removal during decontamination;
- cold and hot laboratory tests to determine the characteristics of a class of chemical compounds (acids, specific complexing agents for actinides) with a great affinity for the contaminants to be removed and capable of operating in an aqueous phase, possibly together with inert carriers such as micelle aggregates;
- cold tests on a mock-up of the evaporator provided with a pneumatic circulation system, in order to optimise the transport and attack procedures of the selected chemical agents;
- a feasibility study of the actual hot tests to be carried out at the plant, in order to test the technology developed during the previous phases;
- hot decontamination tests of the evaporator, if feasible and subject to licensing authorisation.

This research programme aims at obtaining consistent information on a novel approach towards the decontamination of components of nuclear installations: decontamination by means of high affinity chemical reagents, in an aqueous medium for a good surface contact, using the equipment for the circulation and atomising of the solution already existing at the installation.

B. WORK PROGRAMME

B.1. Theoretical studies
B.1.1. Literature review
B.1.2. Theoretical investigation
B.2. Laboratory tests
B.3. Mock-up tests
B.3.1. Design and manufacture of a mock-up
B.3.2. Simulation tests
B.4. Plant tests
B.4.1. Decontamination tests
B.4.2. Analyses
B.5. Evaluation of results

C. Progress of work and results obtained

Summary of main issues

In 1992 the main effort was directed towards experimental tests on cold and hot specimens contaminated with Cerium, Uranium and Plutonium.

The contamination and decontamination apparatuses, envisaged to work in glove-boxes, were designed, assembled and tested.

The errors associated with the adopted experimental procedures were clearly defined.

Plutonium contamination was mainly investigated using various chemical reagents. The results were used to define a foam formulation assuring a good decontamination with a small volume of reagents.

Operational tests are being performed also on a mock-up contaminated with natural Uranium.

Due to the need of a larger number of the tests using Plutonium and to some changes in the programme of EUREX's management, the end of the contract was shifted to 1993.

Progress and results

1. Laboratory tests (B.2.)

Some preliminary tests were run with Cerium and Uranium solutions before using Plutonium.

The experiments were mainly directed at the evaluation of the stripping capacity of the surfactants aqueous solutions under mild conditions (moderately acidic pH, room temperature...) as a preliminary step before the application of active foams. The experimental conditions correspond in fact to those in which the stability of several common foam-forming systems is good. In particular it is known that foaming solutions containing polioxyethylene-type non-ionic surfactants (Brij-35) are not stable at very low and very high pH values, due to the presence of hydrolytic reactions.

1.1. CERIUM TESTS

Contamination. Two different Cerium oxides were prepared from $Ce_2(CO_3)_3$ dissolved in 3.5 M HNO_3 or 7 M H_2SO_4. Cerium oxides were precipitated from these acid solutions with NH_4OH and the hydroxydes were then heated at temperatures from 100 to 300 °C. The samples obtained from nitric acid are more insoluble, indicating a different composition of the oxides.

The results seem to indicate that the quantity of the contaminants at the end of the procedure is substantially reproducible (Relative Standard Deviation-RSD=13%).

Decontamination. The results obtained after treatment with various reagents indicate that TOPO (pH 3.5) seems to be the most active decontaminating agent. However, it has been noted that, due to the very low solubility of the Cerium oxides, the amount of the Cerium extracted is too low.

1.2. URANIUM TESTS

Some test specimens were contaminated with a natural Uranium solution, heating at reflux (starting concentration: 21 g/l; final concentration: 207 g/l).

We can draw the following conclusions:

- the contamination procedure has a good reproducibility (RSD<10%).
- the Decontamination Factor (DF) increases with the concentration and the temperature of the reagent (from A to C in Table I).
- the thermal treatment (100-500 °C) on contaminated specimens has no relevant influence on the DF.

1.3. PLUTONIUM TESTS

1.3.1. APPARATUS

Inside two Glove-Boxes an apparatus for contamination/decontamination studies was installed (Figure 1):

- in GB 1 a heating device is used for the ***contamination*** of specimens in various conditions, using a U/Pu solution (100 g/l). After this cycle, the specimens, washed according to a standard procedure, are transferred to another glove-box with the sealed bag technique.
- in GB 2 the specimens are treated with the chosen ***decontamination*** reagents and then dried and counted with an alpha detector to determine the Plutonium present.

1.3.2. EXPERIMENTAL RESULTS

1.3.2.1. Performance of the experimental apparatus and procedures.

Two typical decontamination experiments are reported in Figure 2. The activity ***versus*** time presents a regular decrease for both systems (HNO_3 2M and Brij-35 0.02M).

The reproducibility of the decontamination is shown on Figure 3, where four independent experiments performed with HNO_3 2M give similar results (the errors, expressed as RSD, are less than 5%).

1.3.2.2. Decontamination results.

The decontamination efficiency of aqueous surfactants solutions varying the concentration (pH 3.5) or varying the pH (concentration: 0.02M)g/l is reported in Figure 4, in comparison with HNO_3 as reference. As expected, the dissolving power increase with the pH, but there is also a clear favourable effect of the concentration (from 0.02 to 0.06 M) which may be attributed to the effect of the surfactants on interfacial tension. The presence of surface-active agents probably allows a better penetration of the solution within the adsorbed layer, although a moderate tendency to form metal complexes from the oxyethylene surfactant groups cannot be excluded.

A second set of experiments was performed using surfactant solutions containing dissolved complexing agents. In this case the results, shown on fig. 5, indicate a different activity of the ligands at a fixed pH value (3.5). After a contact time of ca. 90 min. , the decontamination efficiency observed is in the order:

CUPF < HNO_3 0.1M < 8-HX < TOPO < HNO_3 2M *

After a longer contact time (ca. 450 min.), the decontamination factor obtained working with surfactant solutions becomes higher than that of the HNO_3 0. 1 M, TOPO being always the most active ligand.

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* CUPF : Cupferron
8-H-X : 8-Hydroxyquinoline
TOPO : Trioctilphosphine-oxid

The adopted formulations where then utilised in conjunction with foam-forming systems. The decontamination efficiency seems to be similar. The complete set of the results will be presented in the next Report.

2. Mock-up tests (B.3.)

The experimental tests are underway with Uranium solutions.

3. Discussion of the results.

It was demonstrated that the system model envisaged is able to provide the requested informations on the contamination/decontamination process. The results so far obtained indicate that, utilising the optimised test parameters (time, concentration, pH), a decontamination factor higher than 10 ≈ 20 can be easily achieved.

Table I : Cerium decontamination.

REAGENT	TEMPERATURE [°C]	DF [Decontamination Factor]	RSD [Relative Standard Deviation]
A - HNO_3 2M 20 °C - 2 hours	100	1.2	27
	300	2.0	27
	500	2.0	38
B - HNO_3 2M 100°C - 5 hours	100	76	19
	300	64	23
	500	33	33
C - HNO_3 6M 100°C - 6 hours	100	205	85
	300	410	67
	500	246	60

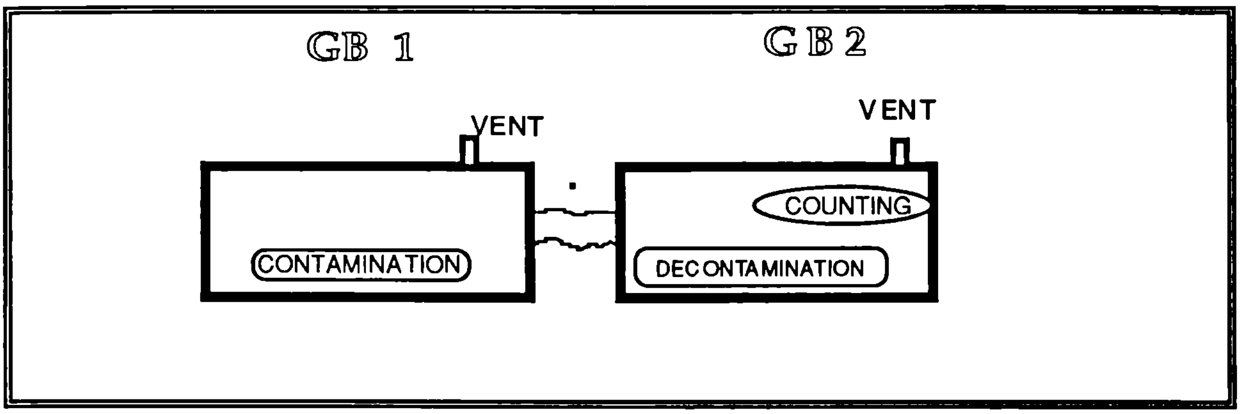


Figure I : Glove-boxes for contamination/ decontamination.

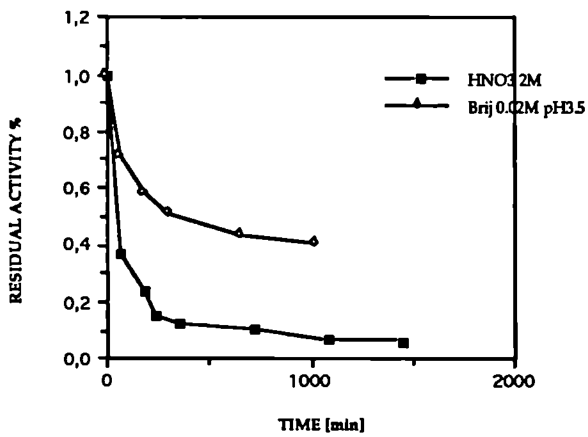


Figure II : A decontamination experiment.

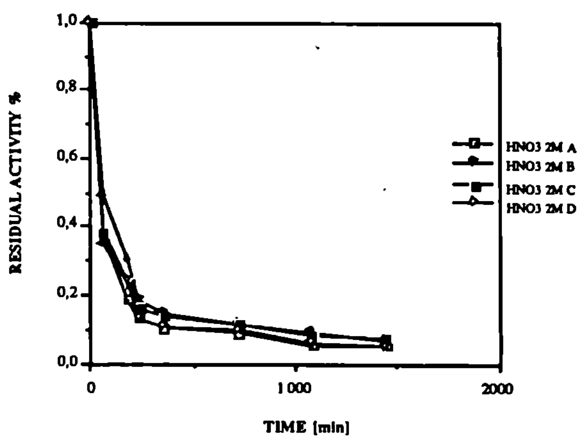


Figure III : Decontamination repeatability

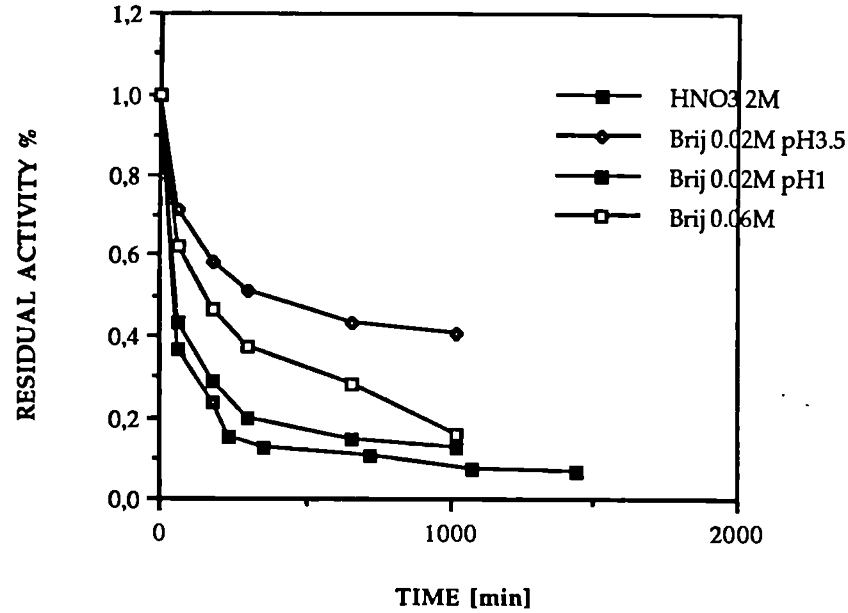


Figure IV : Decontamination : pH and concentration effect

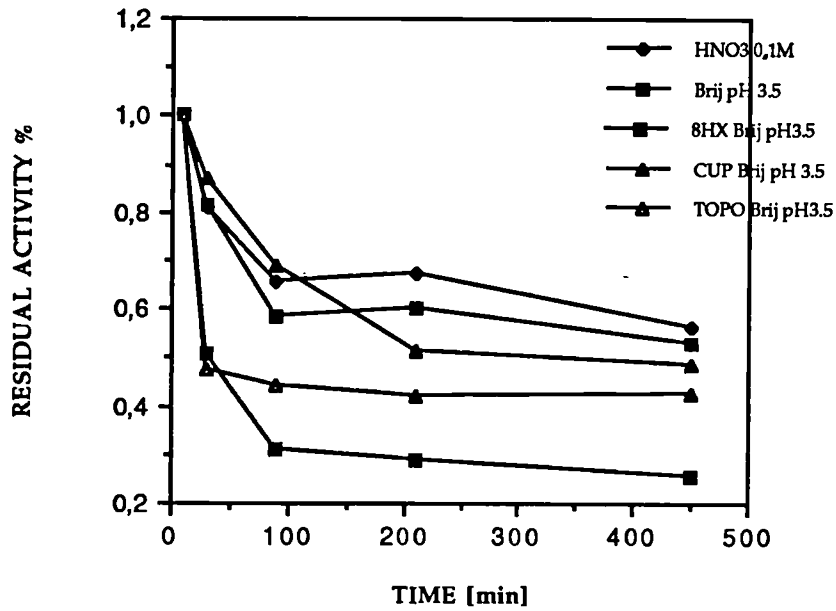


Figure V : Decontamination : various ligands

2.6. DECONTAMINATION TECHNIQUE USING A DISPERSED CHEMICAL AGENT

Contractors: BATTELLE INSTITUT Frankfurt
Contract No.: FI2D-0054
Work Period: September 1991 - March 1993
Coordinator: G POß
Phone: 49/69/79 08 25 84 Fax: 49/69/79 08 86 80

A. OBJECTIVE AND SCOPE

The objective of this research is to develop a technique using a chemical agent dispersed as fog for the decontamination of large size components of nuclear installations. The proposed project investigates the decontamination factors which can be achieved via this method using a lab-scale experimental set up focusing on the decontamination of austenitic steel.

The programme essentially includes:

- construction and testing of the experimental set up;
- adaptation of a droplet size and concentration measuring system;
- decontamination tests with non-active samples to optimise the process parameters;
- decontamination tests with radioactive samples in order to verify the efficiency of this method.

This research programme aims at obtaining consistent information on a new approach towards the decontamination of components of nuclear installations: decontamination by means of high affinity chemical reagents, in an aqueous medium for a good surface contact, using methods already existing in other technical fields.

B. WORK PROGRAMME

B.1. Construction and testing of the experimental set up
B.2. Adaptation of a droplet size and concentration measuring system
B.3. Experiments with non-radioactive samples for the optimisation of the process parameters
B.4. Verification experiments with radioactive samples for the determination of the decontamination factor

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

In 1992, working phases B.2, B.3 and B.4 were performed. Experiments applying the thin film technique to Cr-Ni steel endless belts by means of the apparatus developed in B.1 demonstrated that removal rates of approximately 0.5 μm are obtained for a single etching fluid layer. The maximum film thickness of the aqueous like low viscous agent which can be deposited on sloped surfaces without draining off is about 20 - 30 μm. To reduce the number of etching cycles for the removal of typically 20 μm metal, this would lead to unfeasible repetition rates. To avoid this problem, the viscosity of the etching agent was increased up to a gel-like consistency by adding methylcellulose. Applying the gel as a 1 - 2 mm uniform layer was found to work adequately. Below a critical portion of the additive, the diffusive processes seem to be unaffected. Decontamination experiments with austenitic steel targets originating from a steam generator pipe of the Gundremmingen KRB-A with overall activities in the range of 1000 to 10,000 Bq delivered reductions of at least a factor 2 for a single process cycle.

Progress and results

1. Experiments with non-radioactive samples for the optimization of the process parameters (B.3)

Experiments were performed with the apparatus developed in B.1 etching copperfoils with HNO_3 and compared to bath etching experiments. It was demonstrated that the same amount of etching liquid $\Sigma\Delta V$ results in about the same removal of metal $\Sigma\Delta V_{met}$ within the same time ($\Sigma\Delta t$) independent of the number of volume and time fractions ΔV and Δt. This means one thick acid film $V_o = \Sigma V_{liquid}$ removes in a long time interval $t_o = \Sigma\Delta t_n$ the same amount of metal $V_{met,o} = \Sigma\Delta V_{met}$ as n thin films of acid in successive short time intervals.

These experiments were repeated using a Cr-Ni Steel foil (St 4301) and an etching fluid consisting of a mixture of H_2O, HNO_3 and a fluor salt (Blacit TS from Blasberg Solingen /1/) which is oxidized by the HNO_3 to hydro-fluoric acid HF. Under the assumption of the demonstrated equivalence of bath and multiple thin film etching, experiments with HNO_3/HF at various concentrations were performed to attain an empirical correlation between speed of metal removal and concentration of the two acid components.

With the reference mixture, typically used for industrial Ni-Cr-Steel etching (500 ml H_2O + 500 ml HNO_3 (65%) + 100 g F-salt = 100%), the correlation with concentration (when diluted with H_2O) is shown in Fig. 1, having a wide maximum at 80-60% with a metal removal rate of about 2 μm/h.

When the salt concentration is increased (at the same H_2O/HNO_3 ratio), the etching rate increases almost linearly and reaches a maximum of about 15 μm/h at about 400 g salt per litre liquid (Fig. 2). Smaller and larger HNO_3 concentrations result in a decrease of the etching rate. The behaviour of the thin film etching is quite similar to the Cu/HNO_3 system but the mathematical treatment can be simplified in the almost linear range of Fig. 1 and 2. The measurements were performed at room temperature, because this is the practical condition to be expected. At high temperature (> 50°C), the etching speed increases considerably (more than 30 μm/h) as has been shown in orientation measurements.

2. Droplet size measurements of the atomized etching liquid (B.2)

As described above, the deposited fine dispersed fog results in the expected metal removal rates and makes sense in case very fine layers (approx. 0.1 - 0.5 μm) should be removed. Because the overall thickness of liquid which can be deposited on a surface before rinsing occurs (15-30 μm) primary droplet size (originally envisaged in the 5 μm range) was recognized to be of minor importance, in particular with regard to the frequent cycles necessary to remove thicker layers up to 20 μm. Therefore, droplet size measurements were performed and shortly described in /2/ but not further extended.

3. Reduction of the etching cycles by increasing the viscosity of the etching fluid (B.3.)

To reduce the number of successive aerosol etching cycles for the removal of thicker layers - typically 20 μm layers - experiments were performed with a modified etching fluid. Methylhydroxyethyl-cellulose (Hoechst AG: Tylose MHB 3000p) was found to be an excellent substance to turn the etching fluid (HNO_3/HF) into a gel. The viscosity of the acid was increased by adding 2-4 weight percent Methylcellulose or "Agar" to the standard etching fluid. The result was a "pudding-like" gel that remains chemically and physically stable for a time of at least several hours, which is sufficient. The gel obtained is still suitable for atomization by acid-resistant nozzle atomizers - if necessary supported by electrostatic charge - and deposited in layers up to a few millimeters, without drain off. The etching efficiency of the gel and the total quantity of the waste material are hardly altered.

To remove the gel after the etching process, it was considered necessary to liquefy the agent with a spray of sodium acetate and to skim or suck off the residuals with a scanning suction tube without applying additional rinsing fluid. However, in the course of the experiments, it was found that nitric gases which form during the etching process cause a liquefaction of the gel in the contact layer. During the etching process, this effects leads to a self-induced rinsing of the gel from vertical or sloped surfaces. The liquefied gel film drains off automatically after about 10-15 minutes, depending on the initial viscosity and film thickness. The number of successive etching cycles for the removal of 20 μm can be reduced considerably to 2-3 steps.

The remaining low viscous etching film may contain deposits of fine black particles which result from destroyed methylcellulose after the chemical attack during the etching process. These particles drain off normally with the liquefied etching film (see Fig. 3) unless on horizontal surfaces, where they form a black deposit on the target (see Fig. 4). This film is resistant against rinsing water but can be removed with wipers or erosive sprays, for instance jets of liquid N_2.

3. Verification experiments with radioactive samples for the determination of the determination factor (B.4)

The etching experiments with radioactive samples and the measurements of activity of the targets were performed in a KGB laboratory. To prepare the experiments with the original contaminated samples, the subsequent steps of treatment were tested by means of a Cr-Ni-steel dummy in advance. The samples used for the decontamination experiments were segments of tubes originating from a steam generator pipe of the KRB-A. The target was of nearly cubic form with 5 cm edge length and contaminated though not activated. The initial contaminations of the samples were in the range of 100 to 850 Bq/cm². The experiments were limited to horizontal-positioned targets. The objective was to verify the chemical efficiency on the contaminants which are of complex chemical nature. Before etching, the samples were degreased with acetone. No measurable removal of contaminants was observed on the wiping paper. The standard composition gel was applied to the samples with approximately 0.2 ml/cm², corresponding to a 2 mm film thickness. After 30, 60, 90 and 120 minutes of induction time, the liquefied gel was removed with rinsing water. The residual black deposit resulting from destroyed methylcellulose was removed with wiping paper.

The initial and the final activities were measured separately from respective activities of the collected degreasing, rinsing and wiping test samples. It was found that the complete acid attack with the 2 mm gel film needs less than 30 minutes. No further decontamination effects could be observed afterwards. After one cycle, about 30% of the contaminants could be rinsed off, 30% were contained in the "*black deposit*" which could be easily wiped off mechanically. The residual contamination after one etching cycle is 40%.

The contaminated "*black deposit*" will be considerably reduced on vertical surfaces. The particles suspended in the gel which liquefies with the etching process are transported with the liquid. In case of vertically or sloped positioned targets, water rinsing can be avoided because the thin film residual after draining off of the etching fluid contains only a negligible amount of contaminants and hardly any active acid. These decontamination tests seem to qualify the thick film technique as a kind of pretreatment for a rough dose reduction of heavily contaminated components.

References

/1/ Blasberg Oberflächentechnik, Solingen, Germany; Gebrauchsanweisung für "BLACIT TS-B" Beisalz BN 25610

/2/ E.G. Lierke, G. Poß: *Decontamination technique using a chemical agent dispersed as fog*. Research contract No. FI2D-0054, Second Progress Report, June 1992.

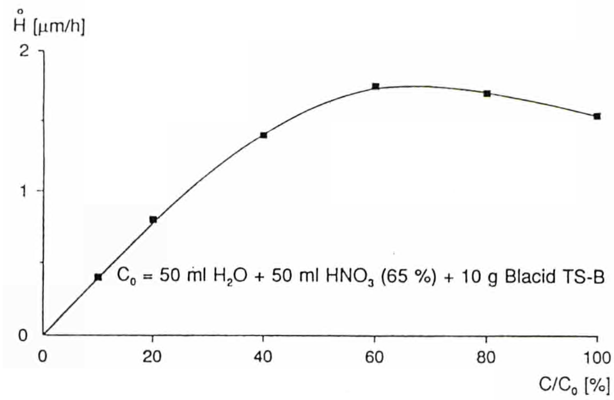


Fig. 1: Bath-Etching of Cr-Ni-Steel (No. 4301) with HNO_3/HF at room temperature

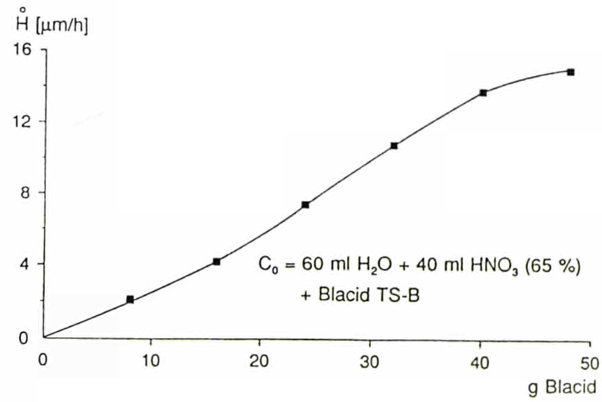


Fig. 2: Bath-Etching of Cr-Ni-Steel (No. 4301) with HNO_3/HF at room temperature

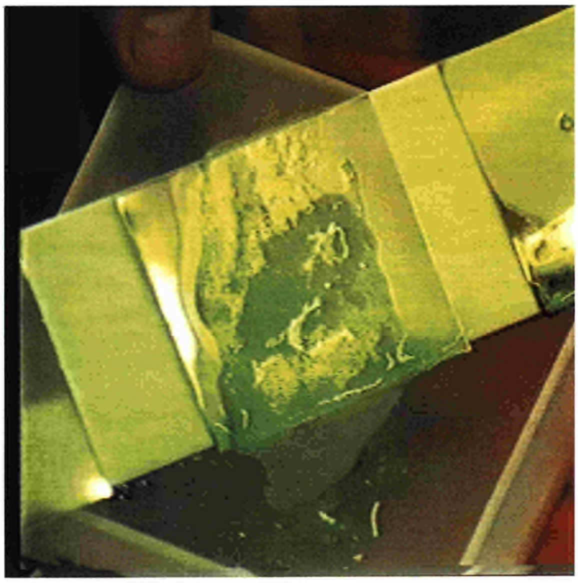


Fig. 3: Liquefied gel draining off

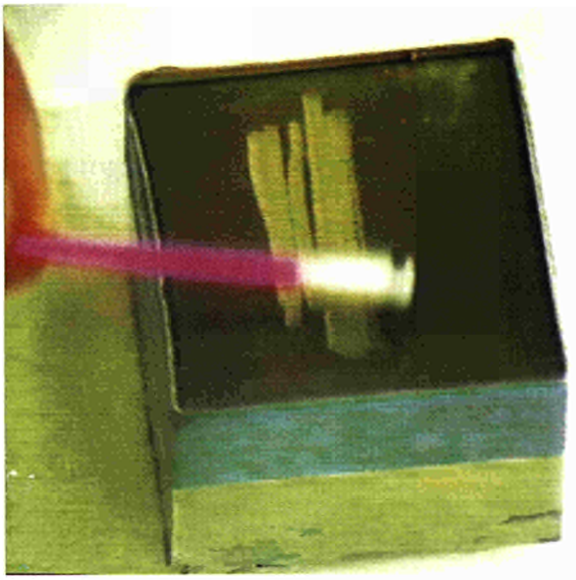


Fig. 4: Black deposit of destroyed methylcellulose

3. AREA No. 3: DISMANTLING TECHNIQUES

A. Objective

The objective of this research 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. Subjects of the research performed under the previous programmes (1979-88)

The following main dismantling techniques were developed and tested:
- thermal techniques such as plasma-arc and oxygen cutting and cutting by laser beam;
- mechanical techniques such as abrasive water jet cutting;
- explosive techniques for the dismantling of concrete structures.

C. Programme 1989 to 1993

Research in this Area should be pursued vigorously with particular respect to the:
- development of the arc-saw technique for cutting thick-walled steel components;
- further development of the electrolytic technique for segmenting thick steel sections;
- comparative assessment of various segmenting techniques with reference to standard cutting tasks;
- full-scale testing of controlled explosive techniques for dismantling of concrete and metal structures.

D. Programme implementation

Eleven research contracts relating to Area No. 3 were in progress in 1992, from which three were completed in the same year.

3.1. EFFECTIVENESS AND LONG-TERM BEHAVIOUR OF CLEANABLE HIGH EFFICIENCY AEROSOL FILTERS

Contractor:	TÜV Bayern	
Contract No.:	FI2D-0007	
Work Period:	October 1990 - June 1992	
Project Manager:	P BOEHM, TÜV Bayern.	
	Phone: 49/89/5190 3165	Fax: 49/89/5190 3191

A. OBJECTIVE AND SCOPE

Because of the high quantity of dust generated by various cutting/dismantling processes, frequent replacement of high-efficiency sub-micron particulate air filters is necessary. If such filters could be cleaned during service, costs for the replacement of the filters, radiation exposures and the amount of secondary waste could be reduced.

The effectiveness in long-term operation (approx. one year) of high-efficiency submicron particle air filters will be investigated in the framework of the dismantling of the Niederaichbach nuclear power station (KKN) in Germany.

A high-efficiency submicron particle air filter system will be exposed to heavy dust generation during the remote-controlled dismantling of KKN primary circuit pressure tubes, and therefore must be dedusted periodically. The dust is radioactively charged (essentially Co-60 and Fe-55). The radioactivity could amount to approx. 1.10^5 Bq/g (pressure tubes and moderator tank) and the dose rate to 0.1 Sv/h. There is at present no experience on the effectiveness and the long-term behaviour of high-efficiency submicron particle air filters that are dedustable during operation.

B. WORK PROGRAMME

B.1. Installation of the filters

B.2. Determination of the main parameters of the clean filter station

B.3. Continuous measurements (pressure pickups, air humidity and temperatures) during cutting of KKN primary cooling circuit (activated cooling channel tubes inside the reactor vessel)

B.4. Final evaluation including radiation exposure of workers, secondary waste arisings, specific costs, effectiveness and long-term behaviour of the filter system.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

As a main result of the investigations and measurements at the Niederaichbach nuclear power plant, it can be stated that the use of automatically cleanable filters (Fig. 1) during the dismantling of radioactive components considerably reduces the quantity of radioactive waste filter elements. Furthermore, less (manual) filter replacements reduce radiation exposure to the staff of the filter plant.

Progress and results (B.3.)

The cutting tools used in 1992 are given on Figures 2a and 2b. Due to this, the average pressure difference of the S-filter station climbed from 1000 Pa to 2000 Pa till mid-October 1992.

The analysis of the measurements shows that an optimal cleaning effect is reached at a pressure difference of about 1,100 Pa at the S-filters. Figure 3 shows the increase of the pressure differences between two cleaning intervals with respect to the cutting tool used. Since the beginning of the remote-control dismantling, 2,800 automatic cleaning intervals of the S-filter have been carried out. Most of the cleaning intervals were stated during work with path grinder and ring saw. At the single use of the ring grinder from 21 September 1992 to 21 October 1992, 660 cleaning intervals took place.

The plastic bags in the dust boxes were changed four times by October 1992. They contained altogether 1,300 kg of radioactive dust (Fig. 4).

As shown on Figure 5, before using cleanable filters at the main filter plant, 602 filters of the first filter stage and 96 filters of the second filter stage had to be replaced. Since the moment cleanable filters were used, only 48 filters of the first stage and 12 filters of the second stage had to be replaced. This corresponds to an estimated minimization of radioactive waste of about 8,500 kg, including the necessary packing of the replaced filters.

Only two filters of the filter plant with cleanable filters had to be changed due to defects.

The contract was extended by 6 months, i.e. to July 31, 1993; due to changes in the KKN decommissioning programme, though, the filter plant will not be used anymore in 1993. The contract is therefore to be concluded by the preparation of the final report.

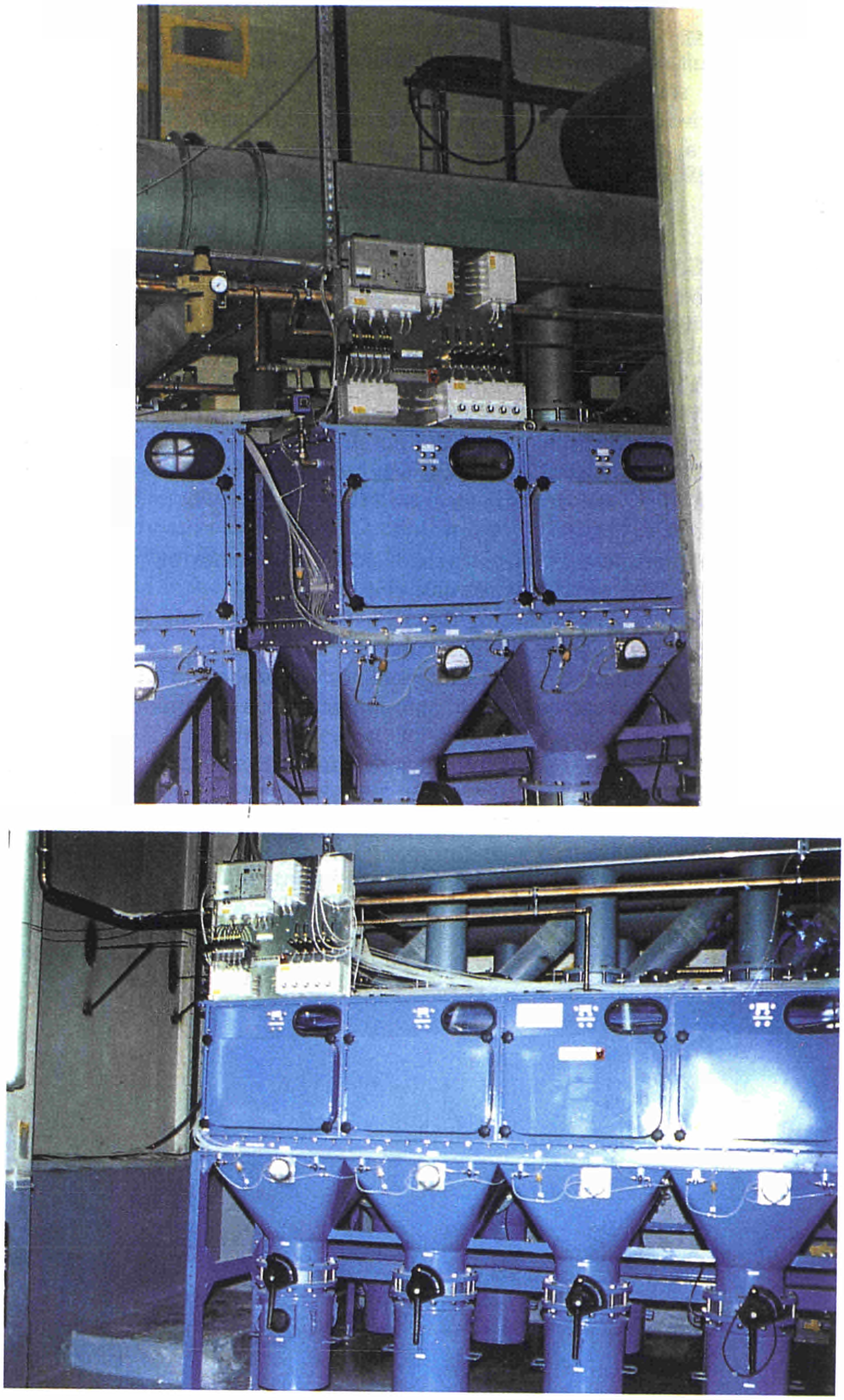


Figure 1: Filter bank housing

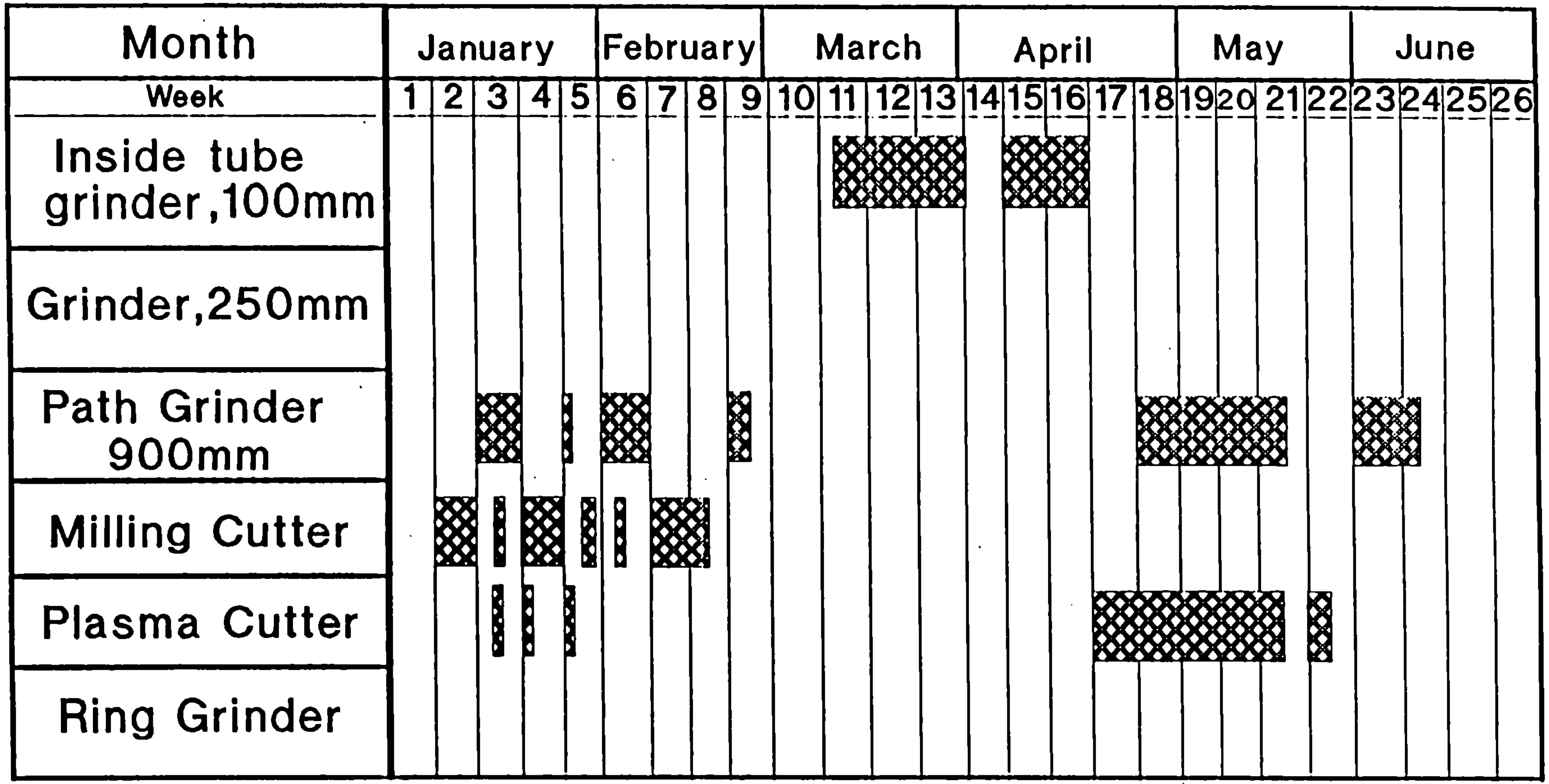


Figure 2a: Timetable for cutting methods 1992

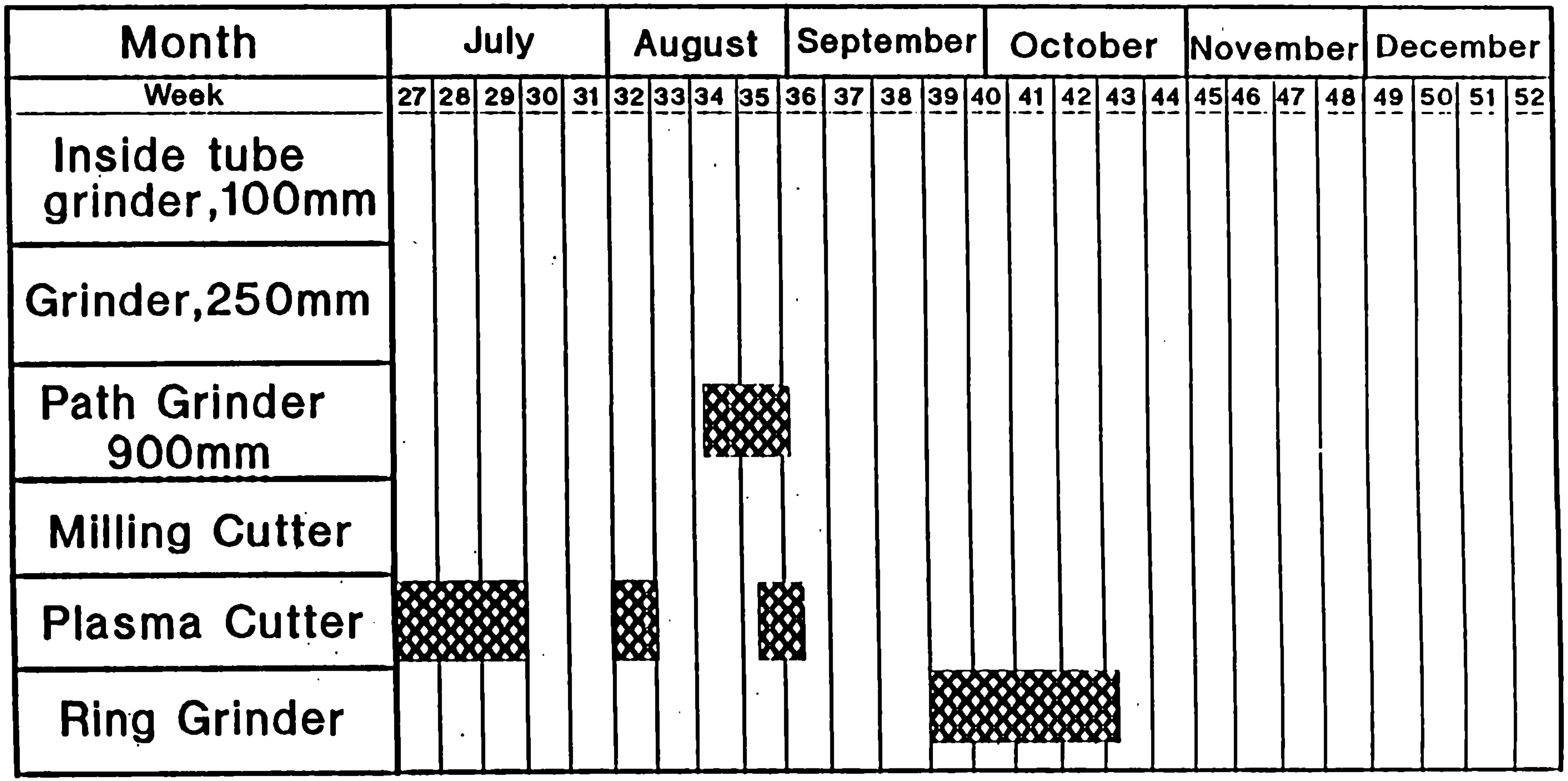


Figure 2b: Timetable for cutting methods 1992

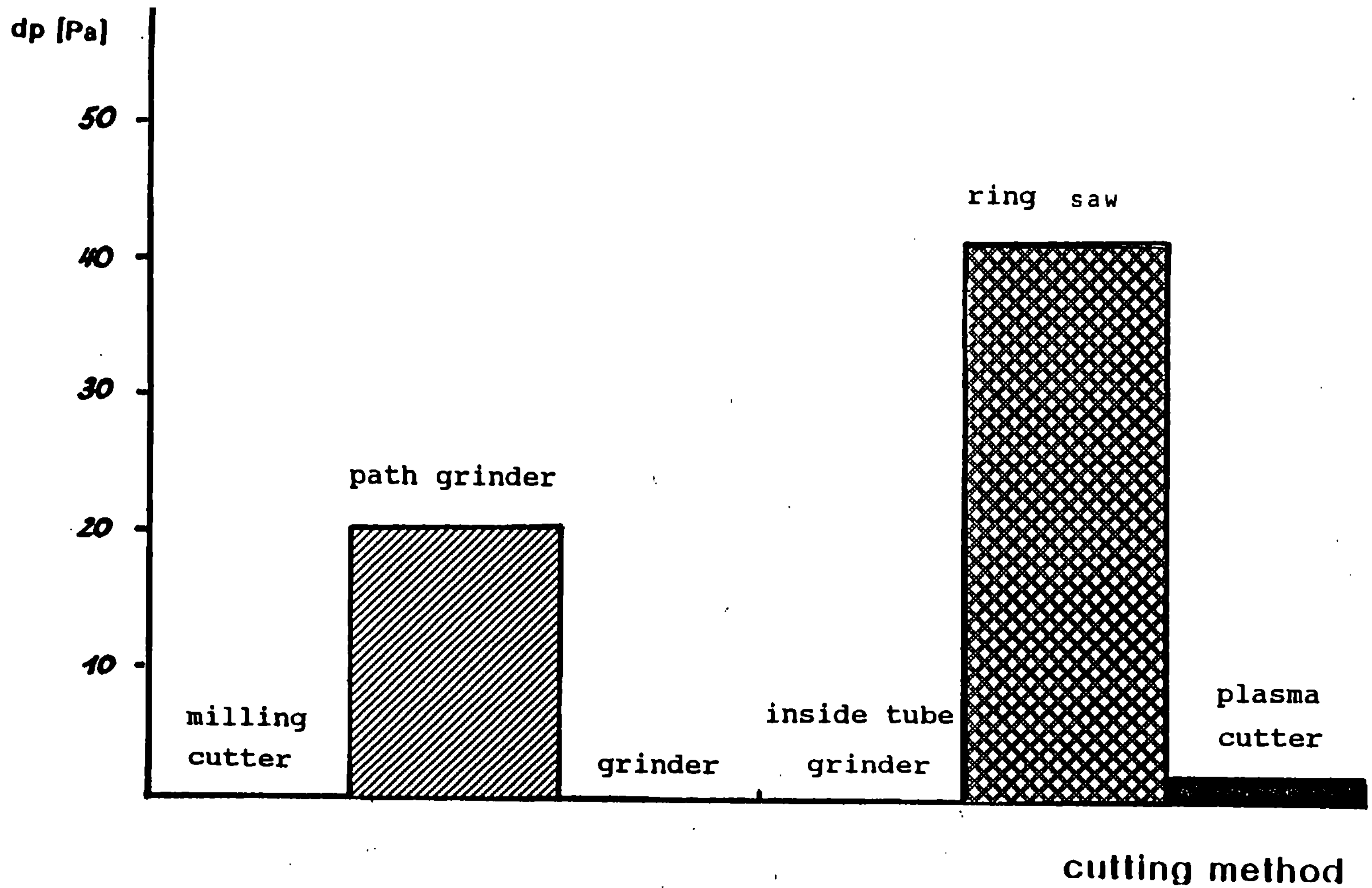


Figure 3: Climb of the filter pressure difference between two cleaning intervals (within 10 min)

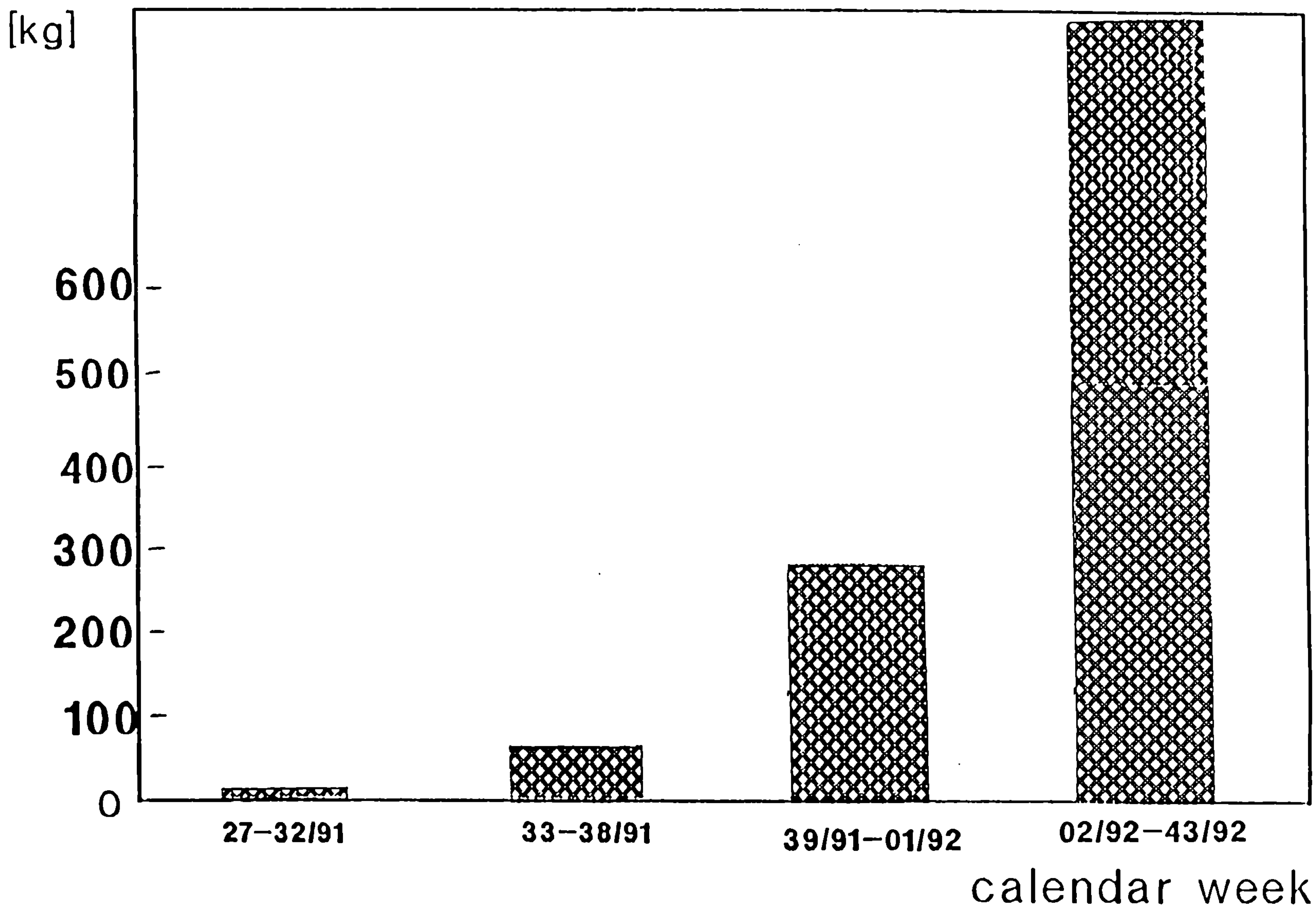


Figure 4 : Dust mass in the plastic bags 1991/92

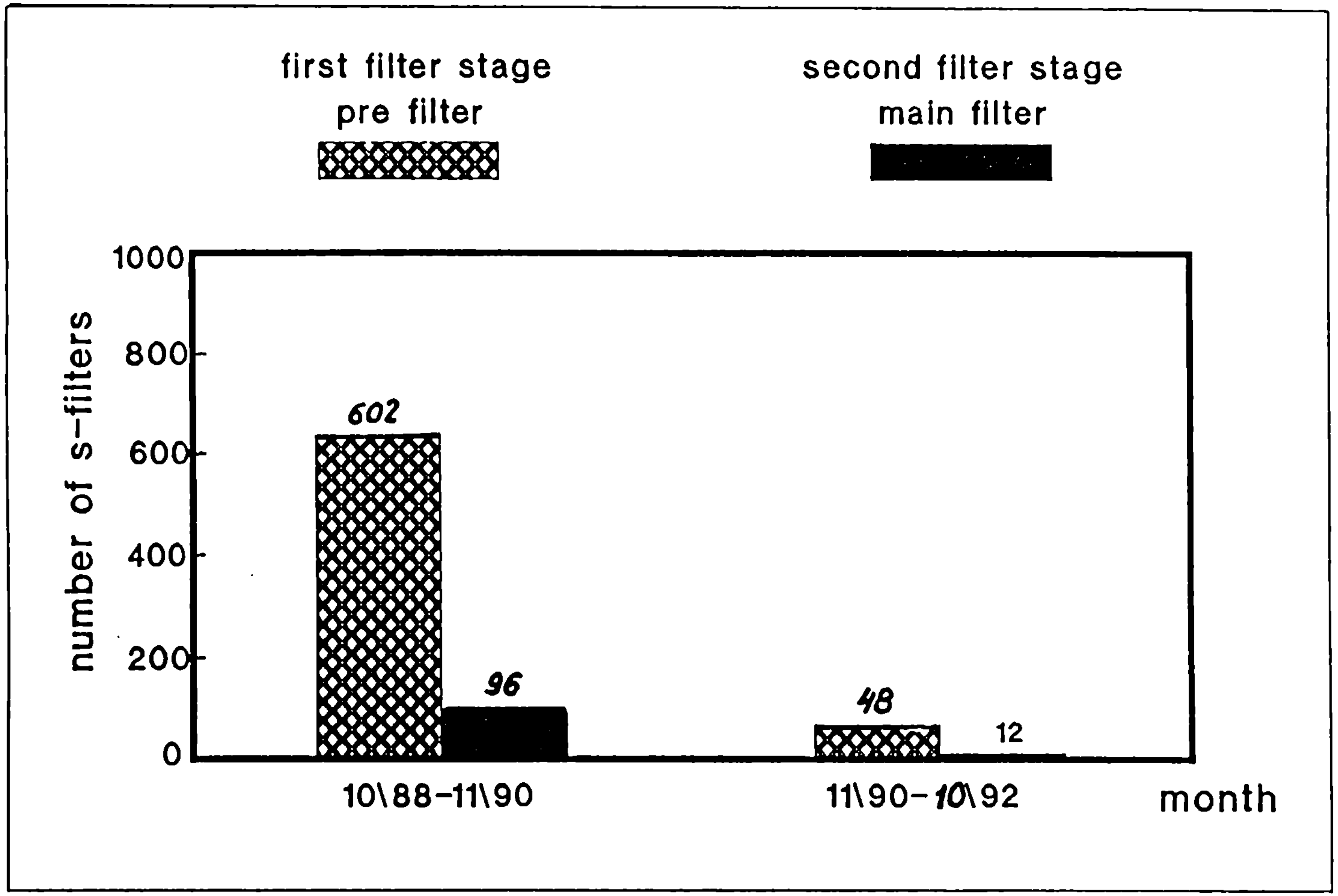


Figure 5 : Timetable of filter replacement of the main filter plant (HFA)

3.2. ABRASIVE WATER JET CUTTING TECHNIQUE FROM THE STAGE OF LABORATORY INTO REAL APPLICATION

Contractors:	UH-IW, CEA-Sac	
Contract No.:	FI2D-0009	
Work Period:	July 1990 - December 1992	
Coordinator:	H LOUIS, UH-IW,	
	Phone: 49/511/762 4320	Fax: 49/511/762 5245

A. OBJECTIVE AND SCOPE

In order to qualify the cutting by abrasive water jets for application in contaminated or activated environment, the cutting techniques developed for laboratory application (CEC contracts FI1D-0069 and FI1D-0067) are to be adapted for remote-controlled application. Secondly, concepts for the handling of the secondary waste are to be developed and proved.

First, the existing abrasive cutting head is to be adapted to remote-controlled work under a water shield up to 15 m, in an inaccessible environment. For this application, methods have to be implemented and proved to control the cutting operation, for instance the state of wear and the cutting results (e.g. depth of the kerf, cutting through). Additionally, parts showing wear are to be remotely replaced so as to allow long-term reproducible operation.

The second step concerns investigations on the secondary waste. Besides a calculation of the composition and amount of secondary waste depending on cutting parameters, strategies will be developed and tested to catch the waste as close as possible at the place of production. Filtration techniques to separate abrasives and cut material from water and air will be adapted from other cutting techniques and will be tested.

All tests will be carried out under non-radioactive conditions, but at real scale in special water basins. The aim of this research work is to set up a tool which is suitable for work under realistic conditions. A control system and the remote replacement of worn parts are further important aims of this research work.

B. WORK PROGRAMME

B.1. Definition of cutting parameters for decommissioning purposes (UH-IW)

B.2. Development of controlling systems for processes parameters and the cutting result (UH-IW)

B.2.1. Preparation of a two-dimensional feeding mechanism for underwater cutting tests.

B.2.2. Development of an on-line controlling system to detect the state of wear inside the cutting head.

B.2.3. Development and adaptation of controlling methods to verify the cutting result during or just after cutting.

B.2.4. Design of a cutting head which includes controlling systems, cutting tests to qualify the sensor systems.

B.3. Development of methods to remotely replace worn parts of the cutting head under water (UH-IW)

B.4. Characterisation and handling of secondary waste

B.4.1. Preparation of test facilities for measuring aerosols and suspended particles when cutting in air and under water (UH-IW).

B.4.2. Measurement and characterisation of the secondary emissions when cutting or kerfing in air or under water (CEA).

B.4.3. Development of methods to lower the spreading out of emissions in air or under water (UH-IW).

B.4.4. Cutting tests to determine the efficiency of measures to lower the emissions and to determine the filtration systems (UH-IW, CEA).

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

The application of abrasive water jets for decommissioning purposes is restricted by the amount of secondary waste as there are water and used abrasive particles. Tests were carried out to quantify the waste (particle size, aerosols) for cutting and kerfing under water and in air. In addition methods were investigated to lower the spreading of the particles during operation (hood) and to treat the waste for recycling and/or handling (hydrocyclone).

For real application it seems to be possible to recycle the abrasives. Therefore in future methods have to be developed to treat the waste to get abrasives with sufficient properties.

Progress and results

1. Preparation of the test facilities to quantify secondary waste (B.4.1.)

To characterise the produced waste tests were carried out in cooperation with IW and CEA. In a hermetically sealed water basin steel and copper samples were cut and kerfed. Sedimented particles, particles suspended in water and aerosols were measured and analysed.

The plates to be cut were either placed under water (for cutting and kerfing under water) or in air. However water is kept in the tank in order to absorb the abrasive water jet.

The experimental device with the associated samplings is schematised on the figure 1.

The ventilation circuit is composed of an inlet HEPA filter (in order to filtrate the air entering into the tank), an integral filter of 160 mm diameter (in order to collect all the aerosols drawn into the exhaust duct), an orifice plate (which has been calibrated and allows to know the flow rate) and a fan. Nozzles are installed in the exhaust duct to allow isokinetic sampling.

The sedimented drosses are collected manually at the bottom of the tank and their size distribution is determined with the use of several sieves.

About 80 samples (suspended particles in water, abrasives, and mainly deposits on filters) have been analysed by ICP (Inert Coupled Plasma) in order to measure the proportion of several elements (Cu, Fe, Mg, Ni, Cr).

2. Characterisation of secondary emissions (B.4.2.)

8 experiments were carried out during the first series and 6 in a second series as indicated in table I.

The total secondary emissions were evaluated for all the experiments.

A) Balance of secondary emissions

Solid emissions

The results of weight analyses are given in table II for both test series.

4.10^{-8} to 7.10^{-5} of the total solid mass is drawn into the exhaust duct.

When the cutting or the kerfing takes place underwater (depth = 100 to 200 mm), the quantity of aerosols is divided by about 200 compared to working in air (comparison between experiments No. 1 and 2 and experiments 5 and 6/7).

The amount of aerosols increase linearly with the amount of used abrasives.

The kerfing produces three to four times more aerosols than the cutting.

Liquid emissions (vapours)

Between 0.07 to 2 % of the used mass of water is drawn into the exhaust duct, mainly in the vapour phase.

The kerfing induces 10 times more water in the exhaust duct than the cutting in air but not under water.

Underwater operation decreases the amount of water in the exhaust duct compared to operation in air (factor of 4 for cutting and 35 for kerfing).

B) Size distribution of abrasives and sedimented dross

The size distributions of abrasives are illustrated by figures 2 and 3.

Figure 2 gives the comparison of cutting and kerfing for application under water. There are no significant differences between underwater operation and operation in air.

Kerfing produces a larger amount of smaller particles than cutting through. This effect is the same for machining steel as well as copper.

There is no important difference between machining copper and steel.

Using different abrasive flow rates (figure 3) has nearly no influence on the size distribution of the used abrasives.

The mass median diameters of the used particles are given in table I.

C) Characterisation of the aerosols in the exhaust duct

The concentration of aerosols in the exhaust duct has varied from 0.04 mg/m^3 to 58.7 mg/m^3 depending on the place of operation (underwater or in air), the nature of operation (cutting of kerfing) and the abrasive flow rate of abrasives (1.8 to 7.2 g/s) when using no hood.

To characterise the size distribution of the aerosols, the mass median aerodynamic diameter (MMAD) is used. The real diameter can be calculated from the MMAD by dividing by the square root of the density of the particles

$$d_{real} = MMAD * \delta^{-0.5}$$

The MMAD is 4 μm (geometric standard deviation = 2.9) for the particles produced by cutting copper or steel in air, the MMAD becomes equal to 6 μm (geometric standard deviation = 2.36) for kerfing copper or steel (1. series of tests).

D) Gas

During the operations under water a hydrogen analyser was used. There is no discernible production of hydrogen (< 0.05 l/min).

3. Methods to lower the spreading out of the emissions (B.4.3.)

The expense of handling the secondary waste is mostly influenced by the spreading out of the abrasives after machining. A reduction of the spreading out causes a decrease of mass of material which has to be treated.

Cutting through

In case of cutting through the abrasive water jet is passing the workpiece producing the cut. A reduction of the spreading out of the used particles can be realised using a catcher system on the rear side of the workpiece but in many cases of decommissioning work the rear side is not accessible (tubes, housing of machines and pumps).

Another possibility to lower the spreading out in case of cutting through is the method of multiple-pass cutting. During a first pass the workpiece is kerfed as deep as possible. The remaining thickness is only a few mm. A second pass is used for cutting through the workpiece. Because of the small remaining thickness the traverse rate can be very high. This fact causes a reduction of spread-out abrasives, in case the reflected particles during kerfing are caught in total by a hood.

To separate the particles from water various techniques can be used. With regard to the reduction of "tertiary" waste such as filters the use of a hydrocyclone for separation seems to be a sufficient solution.

A hydrocyclone which is designed for cleaning applications in nuclear power plants was tested during the reported research work. The results of the tests are given in B.4.4.

Kerfing

The reflected particles can be caught by a hood in case of kerfing. The hood has to be fixed as close as possible to the cutting head and the workpiece to catch as many reflected abrasives as possible. The water inside the hood can be sucked away continuously.

The hood (fig. 4) was adapted in the zone of jet reflection but not covering the cutting head. This geometry allows most of the air bubbles to rise in the water according to the effect of buoyancy. By this effect the bubbles and the reflected particles can be separated by finding a sufficient position for the hood. The flow which can be sucked away from the hood includes only a small amount of air which can be handled by the hydrocyclone.

4. Tests to determine the efficiency of the measures (B.4.4.)

The test equipment which was used was the same as described in B.4.1.

To compare the efficiency of different methods for lowering the spreading out of the particles kerfing tests were carried out under water, the material which was kerfed was copper as given in B.4.1. Data of the tests carried out are given in table I.

A) Size distribution of abrasives and sedimented dross

The mass median diameters of the different tests are given in table I.

The particles of copper are more concentrated in the smaller sieve ranges. The proportion of copper for particles smaller 32 μm is about 12 % compared to 3 - 4 % in overall spectra.

The use of the cartridge filters (type U2-20-Z, 2 μm) (test. nr. 13) lead to problems regarding the circulated water flow rate. Although 10 filters were used parallel the pressure drop after few minutes of filtering was very high. The total capacity of each cartridge filter is between 200 and 2000 g depending on the operating conditions but the suction system was not able to overcome the pressure drop of the wet and dirty filters.

This effect was verified in test nr. 13. For this test the hydrocyclone as well as the cartridge filters were used to filter the basin in total. Starting with a water flow rate of 350 l/min through the filtration system the pressure drop increased very fast. After few minutes the flow was decreased to 90 l/min water.

The particle distribution of the gathered waste is given in fig. 5 for the content of the basin and the cartridge filter. The amount of waste in the hopper was very low (756 g; 1.2 % of the total mass) because of the small flow rate through the hydrocyclone as the result of the pressure drop in the cartridge filters. The efficiency of the whole filtration system was very low. 91.4 % of the particle-mass remained in the basin although the system was operating 2 hours after finishing kerfing.

Tests nr. 15 and 16 point out the effect of the abrasive flow rate on the cutting efficiency. Both tests were carried out with the indirect hood (see B.4.3.), the hydrocyclone for separation but without the cartridge filter system.

The median diameter of the particles which are remaining in the basin is smaller than the particles gathered by the cyclone.

For both tests the mass-% of the particles gathered in the hopper of the hydrocyclone are given related to the total mass of solid waste (figure 6). For bigger particles the efficiency of the cyclone is quite good. For particles bigger than about 100 μm for both tests more than 80% of the mass was filtered by the cyclone. For smaller particles the efficiency decreases. 30 - 40 % of the total mass of solid waste consists out of particles smaller than 100 μm when kerfing copper under water (see figure 2, 3) (assumption: same size distribution of solid waste of test 11 and 15 / 16). With respect to this fact it is necessary to increase the efficiency of the hydrocyclone regarding particles smaller than 100 μm to gather more mass in the filtration system.

Nevertheless the use of a hydrocyclone seems to be a useful technique to separate solid and liquid waste produced during abrasive water jet cutting. For the tests nr. 15 and 16 using an indirect hood 64.1 resp. 69.4 mass-% of the solid waste were separated by the cyclone.

The use of cartridge filters is no sufficient way because of the high amount of smallest particles which clog the filters very fast. Only when using a hydrocyclone which is able to separate a much higher percentage of particle mass even for particles smaller than 100 μm the additional use of a cartridge filter can be a good method to separate the remaining small amount of particles from the water.

B) Characterisation of the aerosols in the exhaust duct

The concentration of aerosols for all the tests of the 2. series is inferior to 0.05 mg/m^3. All tests were kerfing tests under water.

Due to the small collected masses although operation time was 2 hours or more the size distribution of the aerosols cannot be determined in mass.

By chemical analysis the size distribution of particles can be determined for Cu, Fe and Mg. For all experiments the size distribution is bi- or trimodal. One mode is below 0.3 μm and the other between 1 and 4 μm. If a unimodal MMAD (see chapter 2C) can be evaluated (exp. 15 and 16) it is about 2 μm (Cu).

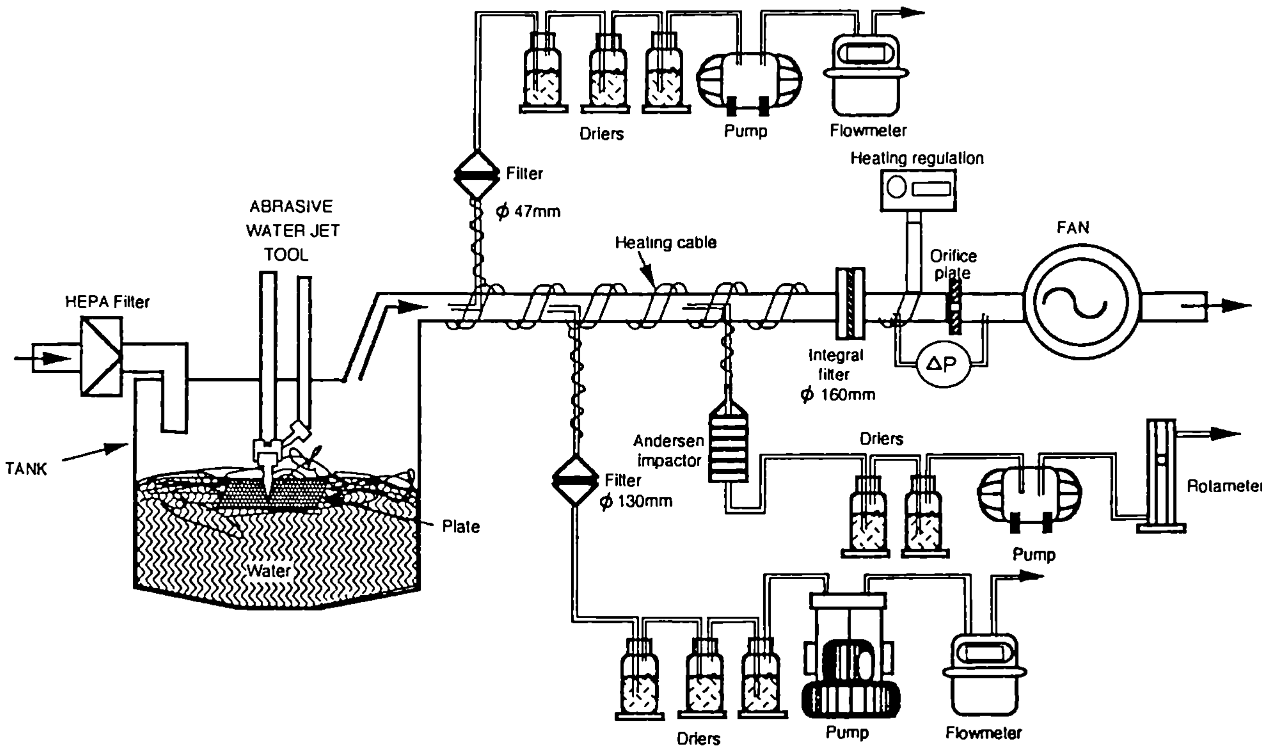


Figure 1 Experimental setup

Table I Main features of the experiments

no.	operation	place	material thickness mm	abrasive mass flow rate g/s	mass median diam.	catch sy- stem	vacuum cleaner
1	cutting	u w	copper 10	7.2		-	-
2	cutting	air	copper 10	6.8		-	-
3	cutting	air	steel 10	6.8		-	-
5	kerfing	air	copper 20	6.3		-	-
6/7	kerfing	u w	copper 20	5.9	120 μm	-	-
8	kerfing	air	steel 20	7.1		-	-
9	cutting	air	copper 10	3.4		-	-
10	cutting	air	copper 10	1.8		-	-
11	kerfing	u w	copper 20	6.1	134 μm	-	-
12	kerfing	u w	copper 20	failed	-	hood	cy.+ca.
13	kerfing	u w	copper 20	6.8	149 μm	basin	cy.+ca.
14	kerfing	u w	copper100	6.6	144 μm	-	-
15	kerfing	u w	copper 20	6.2	136 μm	hood	cyclone
16	kerfing	u w	copper 20	2.05	134 μm	hood	cyclone

(u w = under water; cy. = cyclone; ca. = cartridge filters)

Table II Secondary emissions

Cut No.	Material thickness	Operation place	Cut length mm	Workpiece mass loss $g.m^{-1}$	Used abrasive $g.m^{-1}$	Sedimented dross $g.m^{-1}$	Sedimented dross % TC	Suspended particles $g.m^{-1}$	Suspended particles % TC	Aerosols $g.m^{-1}$	Aerosols % TC	Used water $g.m^{-1}$	Water in exhaust duct $g.m^{-1}$	Water in exhaust duct % of used water	Aerosols $g.m^{-2}$ of cut edge
1	copper 10 mm	cutting underwater	6 528	99.6	3 074.4	3 003.5	97.5	77	2.4	$1.2\ 10^{-4}$	$3.6\ 10^{-6}$	11 688	4.7	0.04	$1.2\ 10^{-2}$
2	copper 10 mm	cutting in air	6 528	110.1	2 914	2 819.2		-		0.027	$9.7\ 10^{-4}$	11 688	19.5	0.17	2.7
3	steel 10 mm	cutting in air	3 672	80.3	3 704	3 464.9		-		0.035	$9.8\ 10^{-4}$	14 788	23.6	0.16	3.5
5	copper 20 mm	kerfing in air	3 672	158.5	2 693	2 674.6		-		0.136	$5\ 10^{-3}$	11 683	194.9	1.7	8.2
6/7	copper 20 mm	kerfing underwater	6 533	152.9	2 532	2 324	98.7	31.5	1.3	$7\ 10^{-4}$	3.10^{-5}	11 679	5.5	0.05	$4.2\ 10^{-2}$
8	steel 20 mm	kerfing in air	3 672	143.5	3 860	3 196.3		-		0.211	$6.5\ 10^{-3}$	14 788	268	1.8	11.7
9	copper 10 mm	cutting in air	6 528	100.3	1 862	1 419.1		-		0.0165	$1.1\ 10^{-3}$	14 813	40.4	0.27	1.65
10	copper 10 mm	cutting in air	6 528	98.0	1 534	1 515.5		-		0.0136	$8.8\ 10^{-4}$	23 284	71.1	0.30	1.36
11	20	Underwater	21 252	144.2	2 612	2 644	99.3	17.7	0.7	1.10^{-4}	4.10^{-6}	11 669	8.1	0.07	6.10^{-3}
12	20	Underwater	2 500	15.6	4 200	NM		NM		$1.8.10^{-4}$		21 600	NS		6.10^{-2}
13	20	Underwater	21 252	152.8	2 922	3 051	97.2	88.2	2.8	$2.2.10^{-6}$	7.10^{-7}	11 669	11.7	0.10	$1.3.10^{-3}$
14	100	Underwater	3 036	729.2	19 779	19 679	99.5	105.2	0.5	$4.2.10^{-4}$	2.10^{-6}	81 686	81.9	0.10	6.10^{-3}
15	20	Underwater	21 252	126.4	2 580	2 312	99.4	13.8	0.6	8.10^{-6}	3.10^{-7}	11 905	15.1	0.13	$4.7.10^{-4}$
16	20	Underwater	21 252	117.7	1 704	1 700	99.3	12.5	0.7	1.10^{-6}	6.10^{-8}	22 586	73.6	0.33	$5.6.10^{-6}$

TC = Total solid mass collected
NM = not measured
NS = not significative

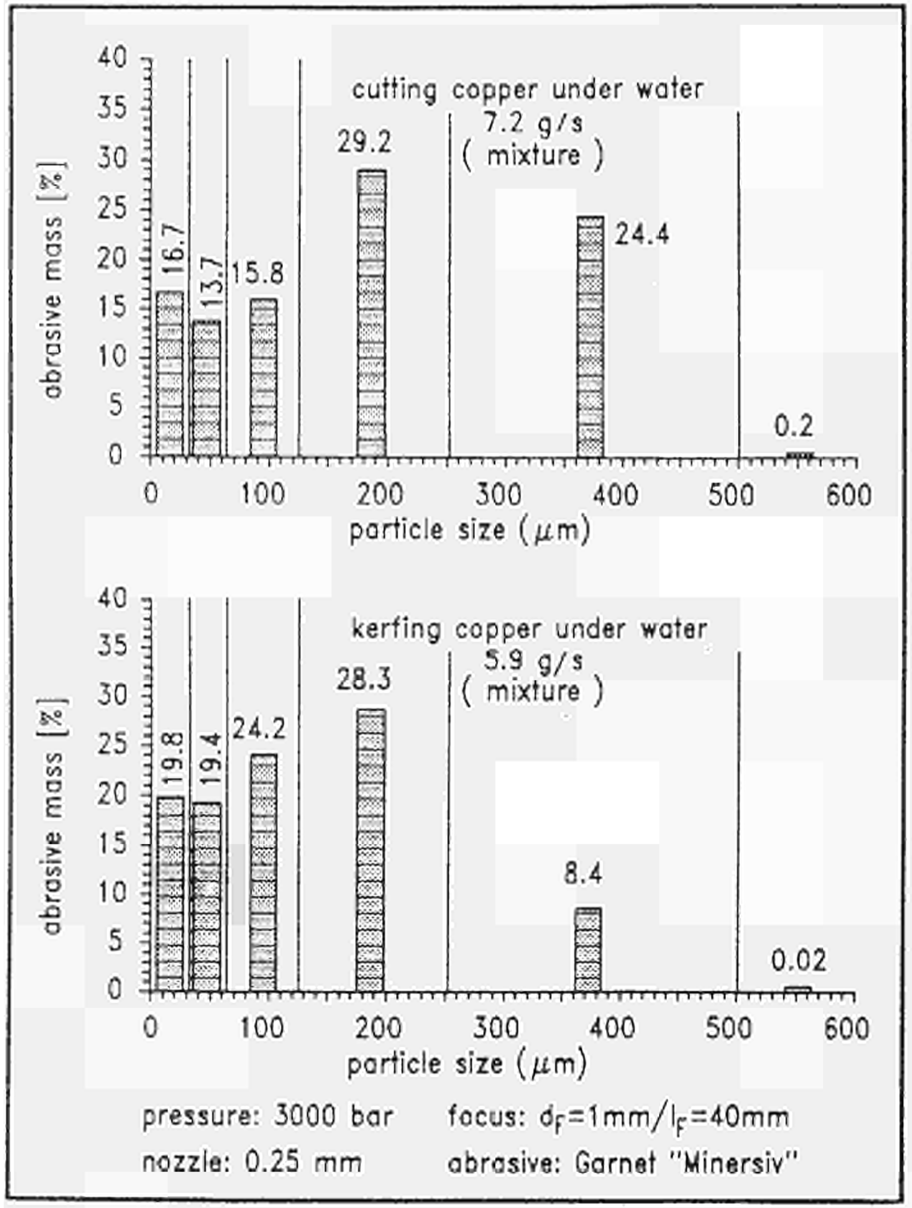


Figure 2 Size distribution

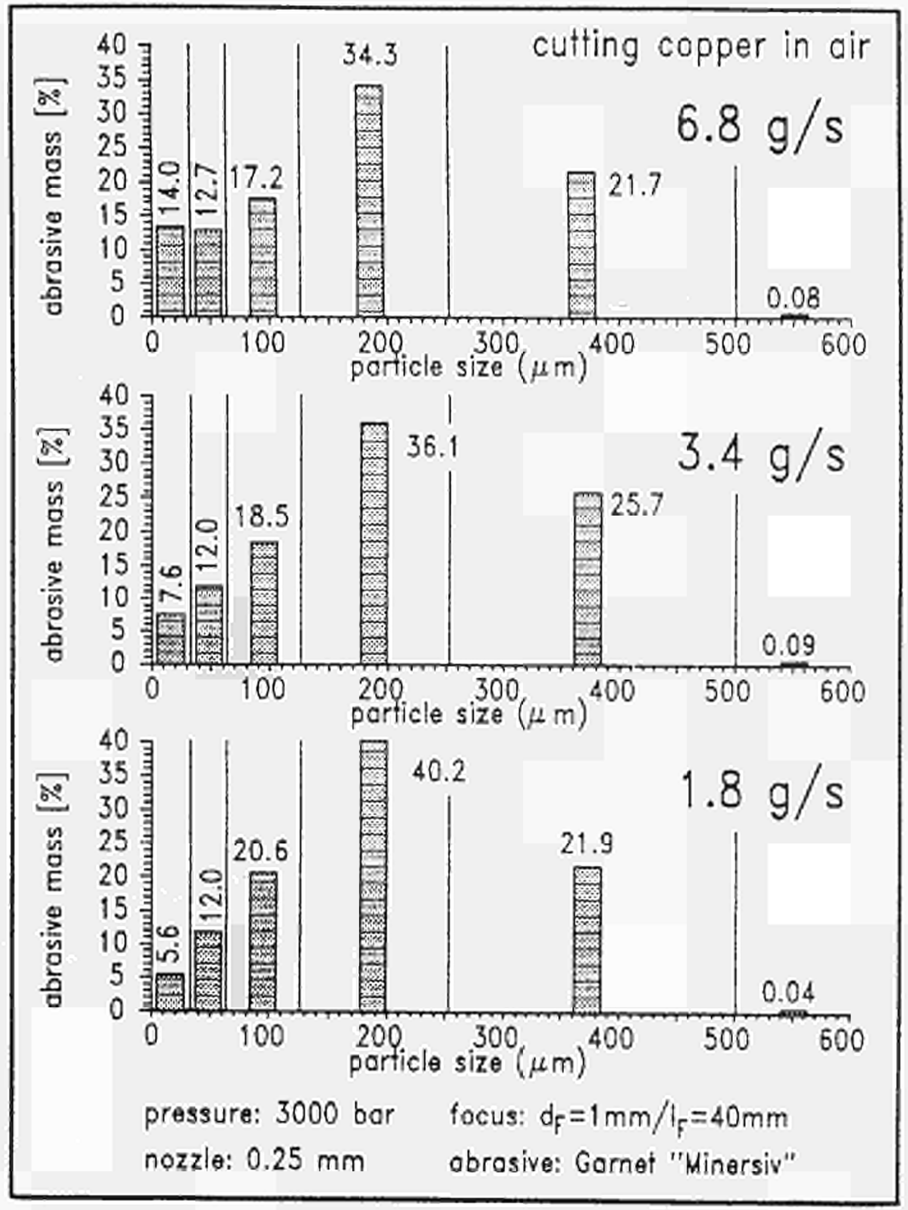


Figure 3 Size distribution

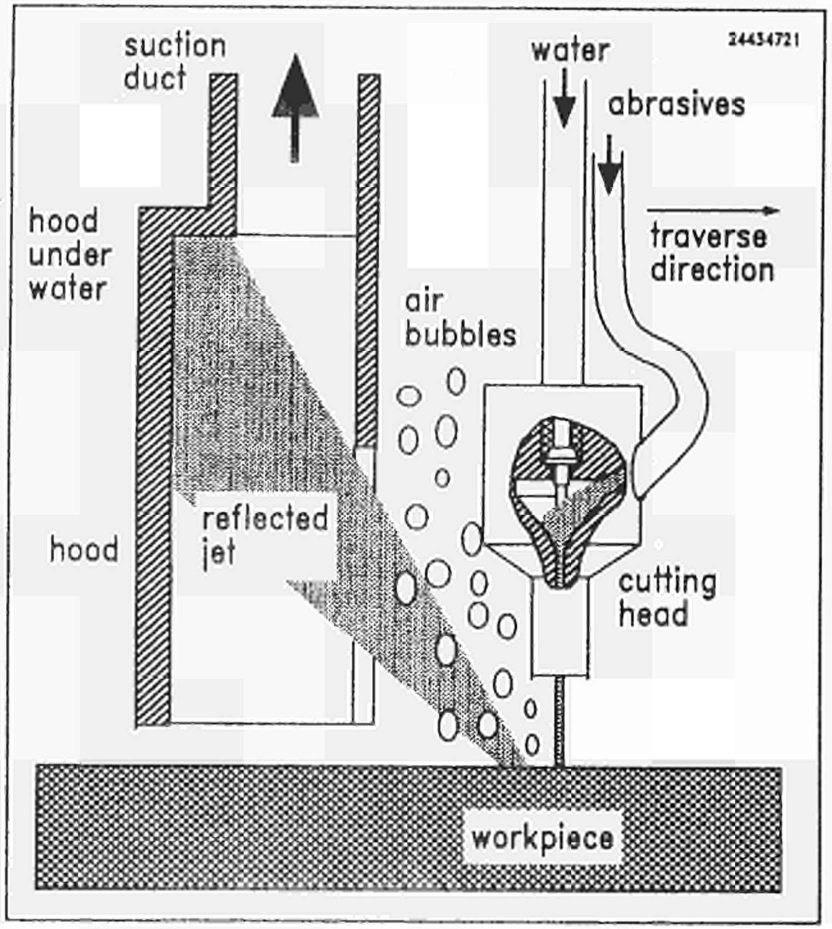


Figure 4 Hood

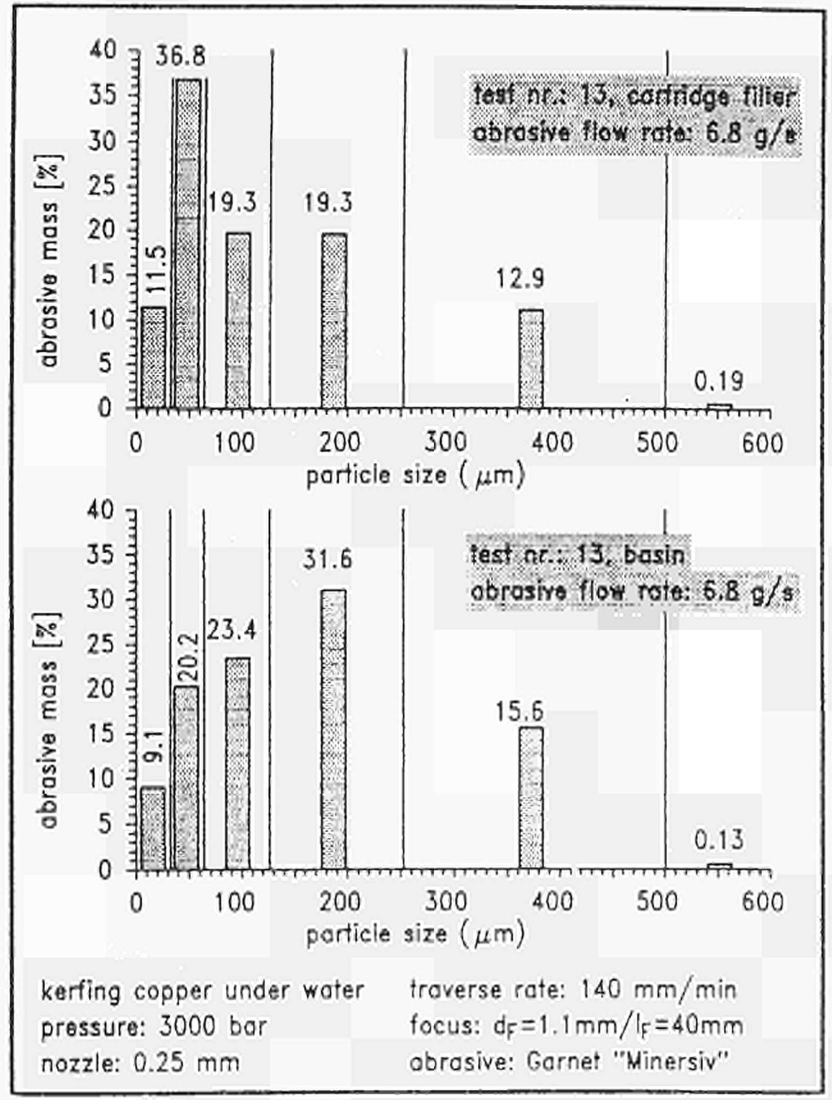


Figure 5 Size distribution

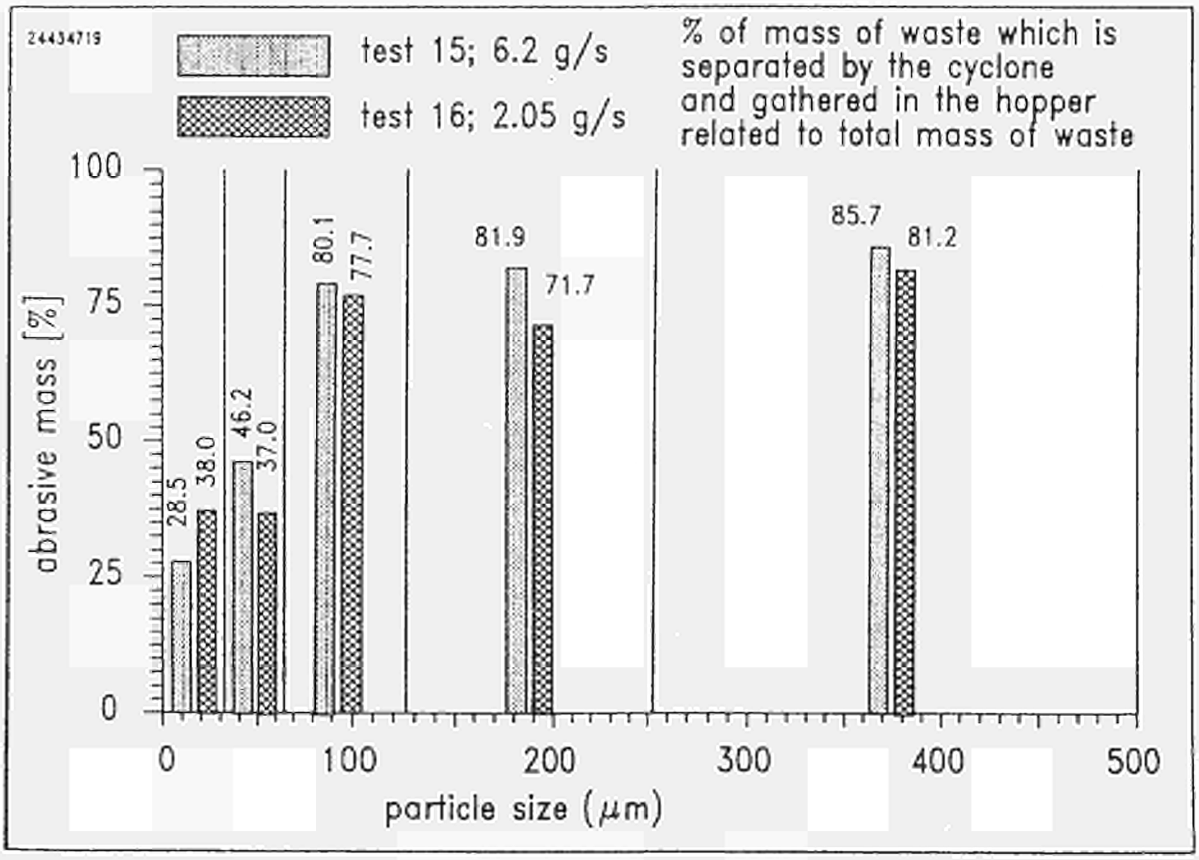


Figure 6 Comparison of test nr. 15 and 16

3.3. STEEL CUTTING USING LINEAR-SHAPED CHARGES

Contractor:	OTO MELARA	
Contract No.:	FI2D-0010	
Work Period:	July 1990 - June 1993	
Project Manager:	G PEZZICA,	
	Phone: 39/187/40 91 28	Fax: 39/187/42 10 26

A. OBJECTIVE AND SCOPE

Various types of cutting charges already exist, but mainly for cutting of few millimetres thick material.

The research work will therefore focus on the development of a high performance cutting charge minimising the damages to surrounding structures for the dismantling of thick-walled steel components (ranging between 10 to 250 mm thickness), e.g. pipes, reactor pressure vessels. The work will include studies and experiments at small and large scale, as well as a study to possibly eliminate or minimise undesired secondary effects caused by the projection of splinters at high speed.

Specific data will be produced on costs, work time and secondary waste arisings from the application of this steel cutting technique.

It is expected that the project will result in an economical and dose-rate tolerant cutting technique particularly suitable for dismantling work in inaccessible places.

B. WORK PROGRAMME

B.1. Determination of basic charge parameters

B.1.1. Theoretical assessment to characterise high performance cutting charges.

B.1.2. Manufacture of charges and execution of tests.

B.1.3. Analyses of the experimental data compared with the theoretical results, conclusions on first phase.

B.2. Optimisation of the cutting charges

B.2.1. Theoretical assessment to further optimise important parameters.

B.2.2. Manufacture of charges and execution of tests with measurements of blast effects in the air, of ground vibrations, photographs from an ultra-rapid framing camera and of flash X-ray tubes.

B.2.3. Analyses of the experimental data with a view to specify high-performance charges.

B.3. High-performance cutting charges specifications and tests.

B.3.1. Theoretical assessment of the final configuration of high-performance cutting charges and specification of 8 tests (in order to determine the scaling law).

B.3.2. Manufacture of charges and execution of large-scale tests in special areas allowing large amounts of explosives.

B.4. Final evaluation including specific data on costs, work time and secondary waste arisings.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

In March 1992, 6 penetration tests have been performed (B.2.2) and the related experimental data has been analysed (B.2.3). Taking into account the results obtained, 10 further tests were defined, aiming at the final optimization of the liner thickness and the stand-off distance; the tests were performed in July 1992 (B.2.2).

Two further charges have been manufactured, and one of these was tested in November 1992 with the aim of studying the free-jet configuration with flash X-rays photography. The related experimental data was analysed (B.2.3., subphase 2).

On the basis of the previous results and the ones obtained in 1992, a study of the observed fracture phenomena was started in collaboration with University of Pisa (Nuclear Mechanical Dept.). To this end, some numerical two-dimensional simulations of the penetration process of a laminar copper jet against a steel target have been performed, and two supplementary tests will be carried out in order to support the theoretical studies. This work, not foreseen at the beginning of the project, can be considered as an extension of Phase B.2. because it is actually an optimization work.

Progress and results

1. Manufacture of charges and execution of tests with measurements of blast effects in the air and flash X-rays photographs (B.2.2)
   1.1. **The 6 tests scheduled in B.2.2. subphase 1**, were performed and the results are reported in Table I. The basic concepts arisen from the analysis are as follows:
   - the unconfined charges with 20% less of Octol explosive (NC1) seem more effective than the reference (NCO) ones (see fig. 1a and 1b). In fact, it seems possible, with the NC1 charge, to cut about 200 mm of mild steel with 20.6-21.2 g of explosive per centimeter of charge length, instead of 25.3 g/cm as for the reference charge;
   - using a less powerful explosive (like TNT), no target penetration is obtained;

   1.2. **The 10 tests scheduled in B.2.2.**, were performed and the results are reported in Table II. The preliminary conclusions of the analysis are as follows:
   - the optimum liner thickness is 1.5 mm;
   - the stand-off (distance between the charge basis and the target) can be between 80 and 100 mm.

   The measurements of the air blast around the charges revealed peaks of overpressure in the order of 1 bar at 3 metres and 0.14 bars at 9 metres.

   1.3. **The analysis of the results of the tests carried out in November 1992** gives following jet configuration:
   - average velocity = 3,200 m/s
   - length = 90 mm;
   - at about 100 mm from the charge initial position, the jet front was apparently not broken.

   Further blast measurements were done in order to compare the confined with the unconfined charges and the values obtained were fairly similar. The peak values of overpressure in the air were in the order of 3 bar at 1.7 metres and 1 bar at 3 metres from the charge. The unconfined charges are to be preferred both for their easier fabrication and for safety reasons, as their fragmentation is practically non-existent. In figure 2, the results of all blast measurements are summarised and the unconfined and confined charges compared.

2. Study of the fracture phenomena (B.2.)

   The aim of this study is the optimization of the linear cutting charge, taking into account the phenomena of the fracture induced by the shock wave, generated by the jet impact on the target surface, which travels inside the target material.

Some numerical two-dimensional simulations have been performed, in which the status of the material when the jet penetration was completed has been detected: the initiation of the fracture seems independent from the target size.

Other simulations were planned in order to update the fracture model which is under development at the Nuclear Mechanical Department of the University of Pisa.

Moreover, to confirm the initial numerical results, two more tests against a target block larger than the standard one will be carried out in March 1993.

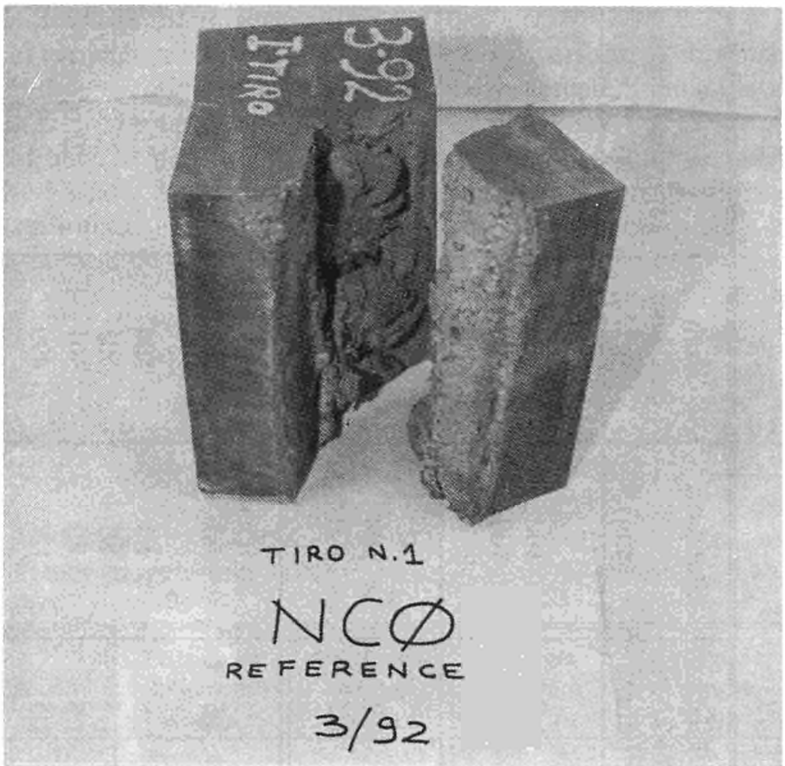
Table I. Results of tests with unconfined charges (March '92)

Base width = 50 mm (liner thickness = 1.5 mm)
Stand-off = 100 mm (target: mild steel)

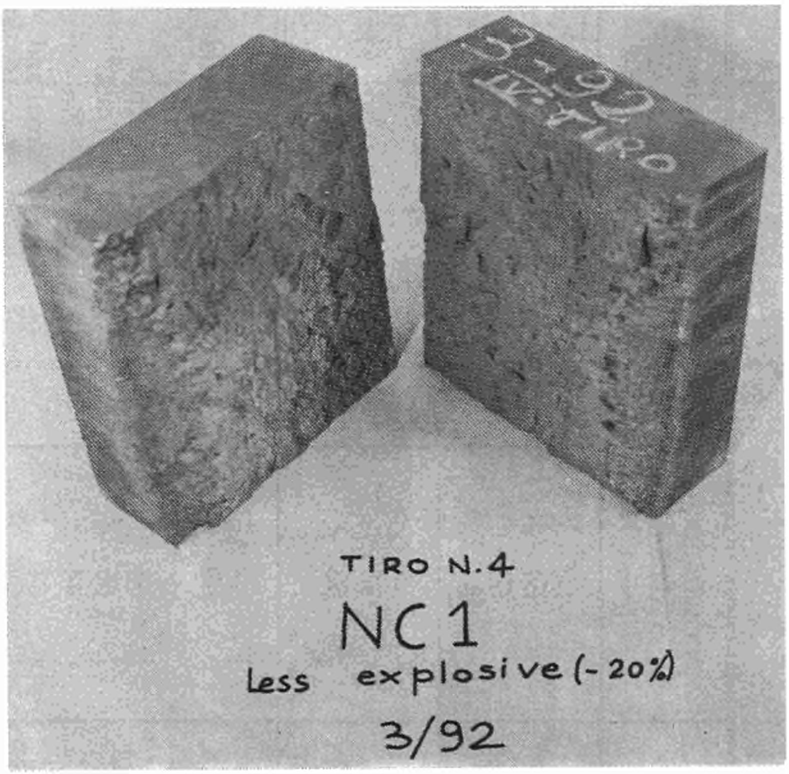
Charge configuration	Penetration maximum (mm)	Penetration average (mm)	Target fracture	High explosive (g)	High explosive length (g/cm)
NCØ OCTOL	60 63	57 60	PARTIAL PARTIAL	506 506	25.3 25.3
NC1 OCTOL (Less explosive)	56 54	51 51	TOTAL TOTAL	411 424	20.6 21.2
NC1 TNT (TNT explosive)	NO DETONATION			390 392	19.5 19.6

Table II. Results of tests carried out in July '92

Charge configuration	Penetration maximum (mm)	Penetration Average (mm)	Target fracture	Liner thickness (mm)	Stand-off (mm)	High explosive (g)	High explosive/length (g/cm)
NCØX OCTOL (Reference)	64 60	60 56	Partial Total	1.7 1.7	100 100	504 506	25.2 25.3
NC1X OCTOL (Less explosive)	35 52	30 47	No Partial	1.7 1.7	100 100	428 432	21.4 21.6
NCØ OCTOL	60	59	Partial and lateral	1.5	60	504	25.2
" "	62	60	Partial	1.5	60	504	25.2
" "	66	62	Total	1.5	80	501	25.0
" "	44	33	No	1.5	80	501	25.0
" "	54	49	Partial and lateral	1.5	120	499	24.9
" "	57	52	Total	1.5	120	500	25.0



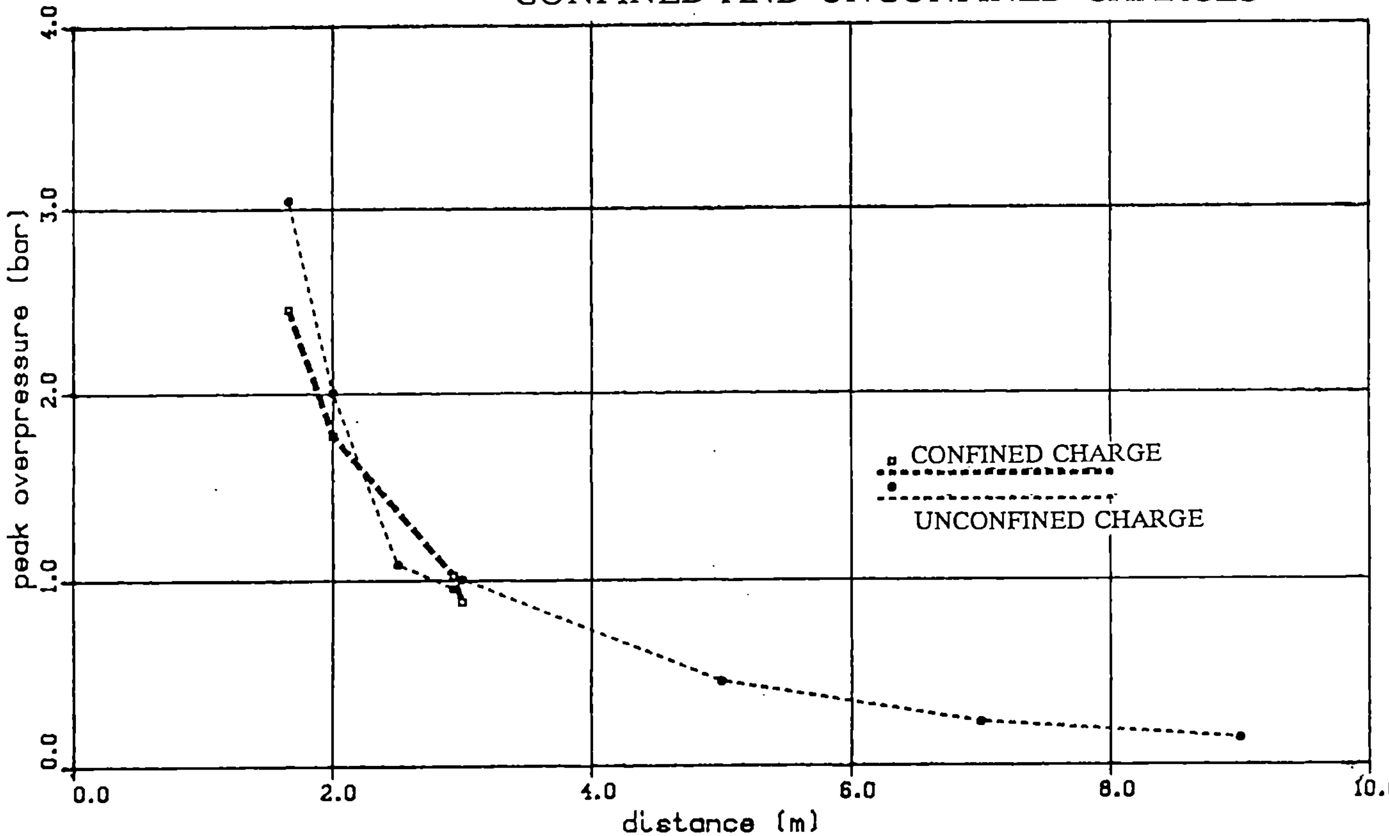
a)



b)

Figure 1: a) Unconfined reference charge. b) Unconfined charge with 20% less of explosive.

FIGURE 2. BLAST EFFECTS IN THE AIR FOR CONFINED AND UNCONFINED CHARGES



3.4. EVALUATION OF STEEL CUTTING TECHNIQUES (CONSUMABLE ELECTRODE, PLASMA TORCH, ARC SAW, GRINDER, RECIPROCATING SAW)

Contractors:	CEA Valrhô, CEA-Sac
Contract No.:	FI2D-0013
Work Period:	October 1990 - June 1993
Coordinator:	Ch LORIN, CEA/DCC/UDIN
	Phone: 33/66 79 63 04 Fax: 33/66 79 64 32

A. OBJECTIVE AND SCOPE

The project relates to industrial-scale testing in air of various relevant cutting tools. Its originality is a comparison between tools in the same normalised conditions of use.

The main purpose of this work is a comparison of different cutting techniques for the same working conditions in order to determine the real cutting time, to improve the knowledge of the cutting tools, and to evaluate the generated secondary wastes, cost aspects and the radiological impact.

The work requires an inactive testing cell, as well as appropriate materials and tools: the cell, located in an inactive testing station at CEA/Fontenay-aux-Roses, is an airtight room in which it is possible to work in a controlled atmosphere. Carbon and stainless steel plates with thicknesses of 10, 30 and 50 (or 60) mm with exactly known composition of the radioelements will be cut; the cutting tools which will be used are arc air, plasma torch, arc saw, circular disc and reciprocating saw.

Meetings will be arranged with partners after each tool test in order to improve their execution; therefore, the tests are carried out one after another. It is envisaged to cooperate in specific areas with the Universität Hannover and with the French industry.

The potential benefits of these tests are the protection and security of workers, a decrease of the volume of waste effluents and a better use of the tools themselves for future decommissioning work.

B. WORK PROGRAMME

B.1. Preparation of the testing cell (CEA-Sac)
B.2. Cutting under inactive conditions with selected tools and materials (CEA-Valrhô)
B.3. Cutting under simulated radioactive conditions (CEA-Valrhô)
B.4. Secondary waste analysis after each specific cut (CEA-Sac)
B.5. Final evaluation of the cutting techniques assessed, including the cost of the basis tool, the associated logistic, the consumable part, the radiation exposure to workers and research of relevant radionuclides in the cell (All).

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

In the first annual report, the facility where the cuts take place, the analytical techniques have been described. In the second annual report, the test procedures have been emphasized in order to compare the tools in the same conditions (at the steady state). All the measurements relating to the cutting with the grinder, the alternating saw,the plasma torch and the arc air are finished. The study of the arc saw is still in progress. The comparisons concern the using range of the tools, the wear of the most exposed parts, the cutting speed, the production of secondary emissions with their repartition (sedimented dross, attached slag, deposits on the cell walls, aerosols in the exhaust duct) and their characterization.

Progress and results

1. Preparation of the testing cell (B.1.)

The preparation of the testing cell has been described in the previous reports.

2. Selected tools and materials (B.2.)

Mild steel and stainless steel plates with thicknesses of 10, 30 and 50 mm have been cut.

The selected tools used in these experiments are common tools that we can find currently on dismantling sites and not prototypes or issued from a special manufacture except for the arc saw. The characteristics of the tools are as follows:

Alternating saw

trademark:	FEIN
blade length	400 mm
teeth number per cm:	6
tooth height:	1 mm
blade nature:	stainless steel
rate:	2.5 blows per second
working counter weight:	5 kg
working angle with the piece:	45°

Grinder

trademark:	BOSCH
energy:	electric
wheel trademark:	BARCUT
wheel diameter:	300 mm
wheel thickness:	4 mm
rotation speed:	5000 R.p.m
equivalent input:	2200 W
equivalent output:	1550 W
weight:	6 kg
cutting position:	gravity position

Plasma torch

trademark:	SAF
type:	NERTAJET 200
working voltage:	120 V
working intensity:	200 A
plasma gas:	Argon
flow rate of gas:	60 l/min

nozzle diameter:	2 mm
working stand-off:	7 mm
working position:	gravity position

Arc-air

working voltage:	40 V
working intensity:	450 A
electrode nature:	carbon
electrode diameter:	6.35 mm
working stand off:	1 mm
working position:	gravity position

Arc saw

trademark:	PROTOTYPE
working voltage:	44 V
working intensity:	1200 A
wheel nature:	FLUGINOX
wheel diameter:	320 mm
wheel thickness:	5 mm
rotation speed:	300 R.p.m
working angle with the piece:	45°

The grinder and the alternating saw are not adapted for cutting steel plates with thicknesses equal or superior to 30 mm. The arc-air for thicknesses superior to 30 mm has to be used several times in the same kerf to cut through. The plasma has been a satisfactory cutting tool for almost all thicknesses, it was too powerful for a thickness of 10 mm for which the optimal cutting speed could not be reached. The fastest tool is the plasma torch which has a cutting speed double compared to this of arc-air, multiplied by 10 compared to this of grinder and multiplied by 50 compared to this of alternating saw for 10 mm thickness steel plates. These ratios increase (between plasma torch and arc-air or alternating saw) for 30 mm thickness steel plates.

Figure 1 shows the mean surface cutting speed (surface = cut length x thickness) obtained for the five tools. This surface cutting speed is not very dependent upon the cut thickness and emphasized that the plasma torch is the fatest tool.

The wear of some parts of the tool is important for three tools (wheels of the grinder and the arc saw, electrode of the arc-air) and increase rapidly with the cut thickness (figure 2).

3. Simulated conditions (B.3.)

The composition of the cut plates and of the worn parts of the tools are known and some chemical analyses on the aerosols collected on the sampling filters are in progress.

These chemical analyses on stable isotopes (Fe, Ni, Cr, Mn) will give information about the behaviour of the radioisotopes of the same elements.

4. Secondary waste analysis (B.4.)

Some partial conclusions have been yet indicated in the second annual report and can be completed.

- The arc-air is the tool which produces the most important mass of wastes (mainly sedimented dross), it produces about two times more mass of wastes that the grinder, three times more than the plasma torch and five times more than the alternating saw for a 30 mm steel plate.The sedimented dross are linked to the dimension of the kerf and to the wear of the tools. For a given tool, the width of the kerf increases with the cut thickness except for the arc saw (figure 3).

. The deposits on the walls of the cell are more significative for the grinder thanfor the other tools. This is probably due to the showers of sparks induced by the grinder which eject particles towards the walls.

. The alternating saw produces much less aerosols than the other tools. For stainless steel, the plasma torch induces less aerosols per cut length than the arc-air and the grinder. For mild steel the arc-air produces more aerosols per cut length than the plasma torch and the grinder.

The size distribution of aerosols are often bimodal and sometimes trimodal showing thus the main phenomena of formation (mechanical action and fusion).

The thermal tools, like plasma and arc-air, produce more submicronic aerosols than mechanical tools.

The mass mean diameter of aerosols due to the cutting of stainless steel is smaller than this of aerosols due to the cutting of mild steel by plasma torch.

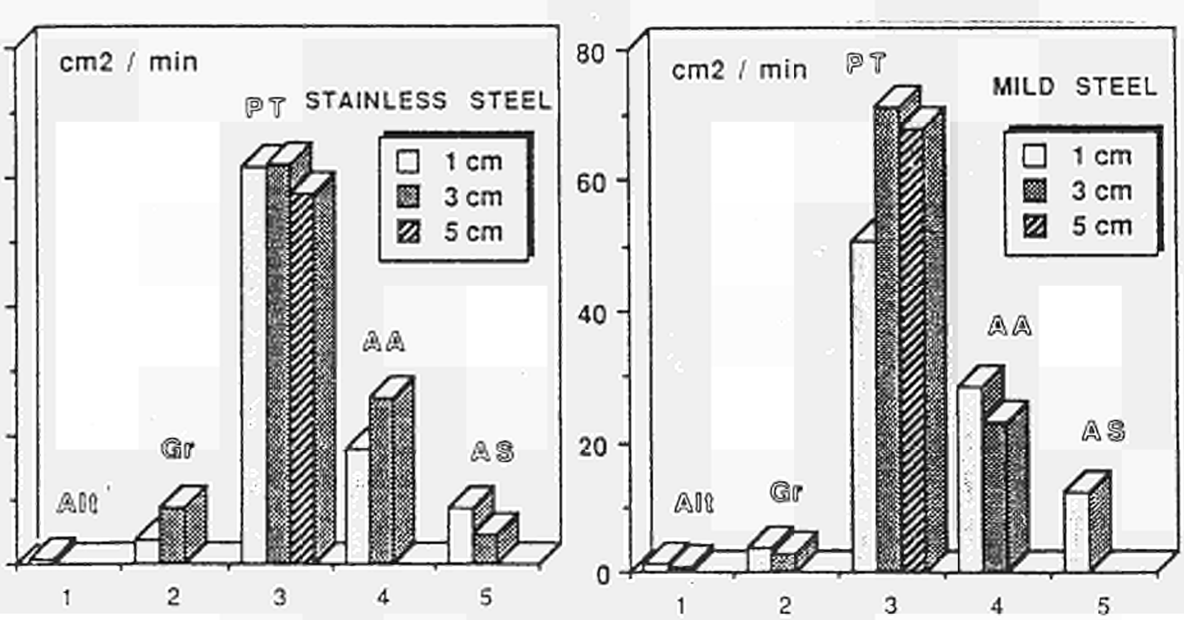


Figure 1 - Mean surface cutting speed

Alt : Alternating saw
Gr : Grinder

PT : Plasma Torch
AA : Arc-Air
AS : Arc-saw

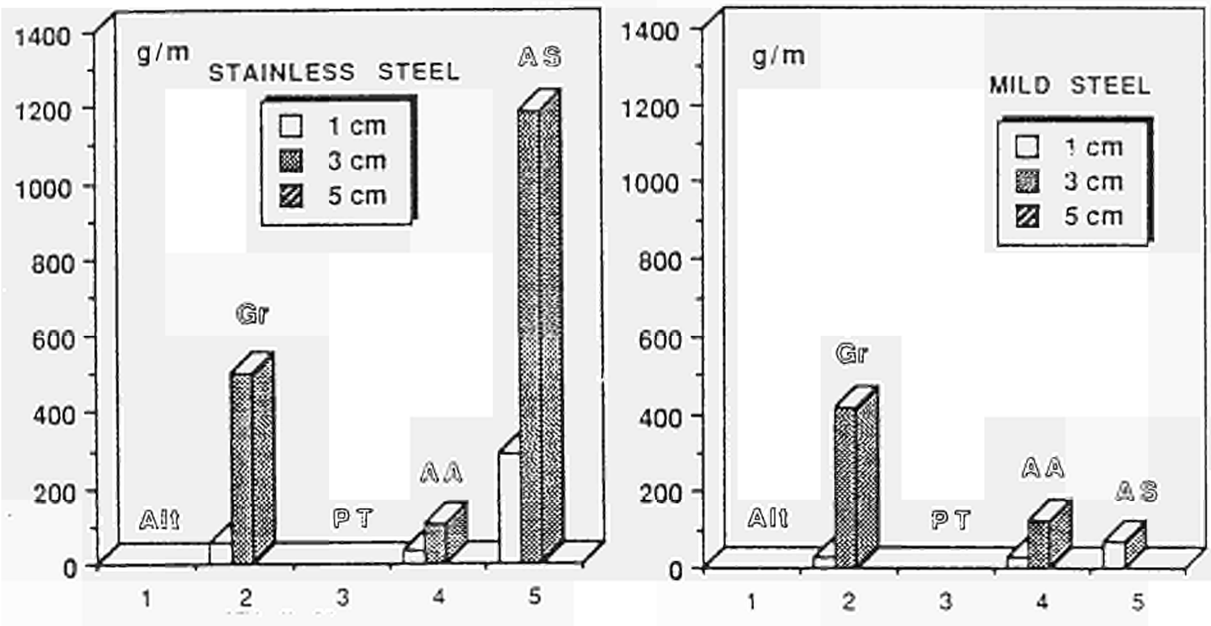


Figure 2 - Wear of the tool

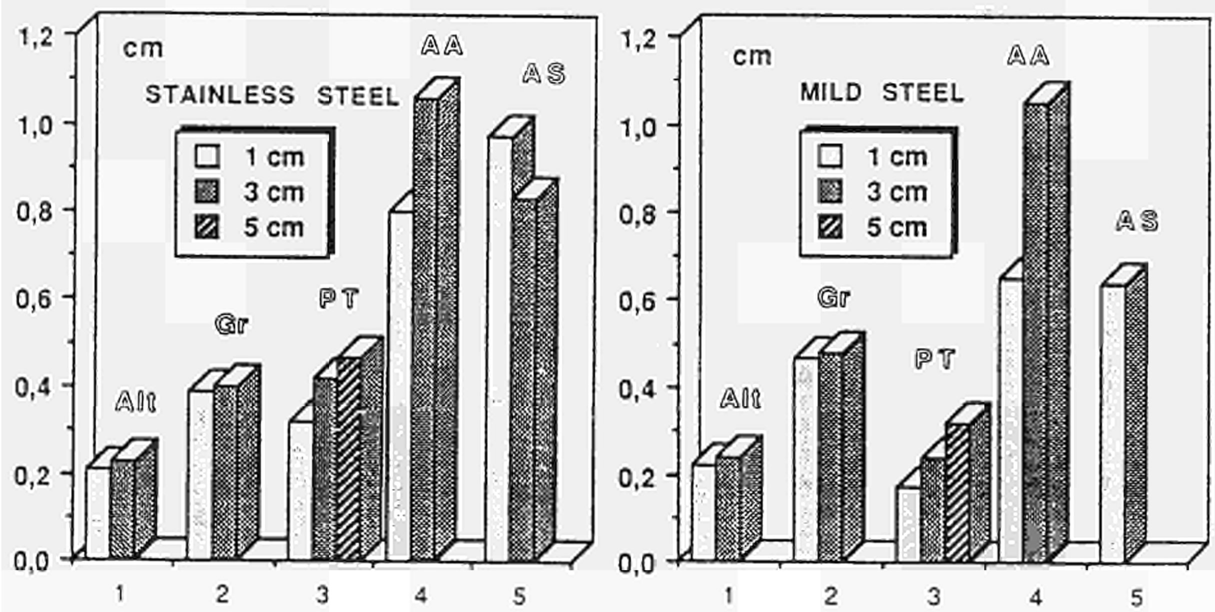


Figure 3 - Width of the kerf

Alt : Alternating saw
Gr : Grinder

PT : Plasma torch
AA : Arc-Air
AS : Arc-Saw

3.5. UNDERWATER THERMAL CUTTING TECHNIQUES AND ASSOCIATED REMOTE-CONTROLLED MANIPULATOR SYSTEMS

Contractors:	CEA-Cad, UH-IW, RWTHA, CEA-Sac
Contract No.:	FI2D-0019
Work Period:	July 1990 - December 1993
Coordinator:	J P DUFAYET, CEA Cad, SST/SEMO
	Phone: 33/42 25 72 11 Fax: 33/42 25 65 63

A. OBJECTIVE AND SCOPE

This project aims at improving underwater thermal cutting techniques and their remote control. The main objectives are to cut greater thicknesses and improve the safety of operation, e.g. assess harmful by-products, protect workers, assist the operator during operations.

So far, underwater thermal cutting of steel has been achieved up to 70 mm thickness. This project has the objective of achieving cutting up to 200 mm. Sensors and associated systems studied in laboratory will be applied in a semi-industrial installation.

The work involves an experimental investigation in the laboratory of each contractor followed by real-case applications under non-radioactive and radioactive conditions in the former Pegase reactor in Cadarache.

The project is a follow-up of work performed in the 1984-88 EC programme (contracts FI1D-0037, -0007 and -0039).

B. WORK PROGRAMME

B.1. Preliminary tasks (CEA-Cad)
B.1.1. Detailed requirements and objectives of the project
B.1.2. List of parameters and ranges to be studied
B.1.3. Specifications of sub-systems
B.2. Development of the plasma torch and adaptation of the moving device (CEA-Cad)
B.2.1. Improvement of the performances of the plasma torch
B.2.2. Adaptation of the moving device
B.2.3. Integration of the sensors into the torch handling system
B.2.4. Cutting tests with measurement of effluents
B.3. Development of other tools (UH-IW)
B.3.1. Optimisation of cutting parameters of plasma saw and consumable electrode
B.3.2. Control systems usable with the manipulator of CEA-Cad
B.3.3. Cutting tests with measurement of effluents
B.4. Development of control systems for sensor-controlled piloting of the handling system for the tools and the process parameters (RWTHA)
B.4.1. Improvement and application of inductive sensors
B.4.2. Process control and piloting of the tool handling system
B.4.3. Interfacing between the sensor system and the handling control system
B.4.4. Function testing in the laboratory
B.5. Preparation of radioactive samples taken from nuclear installations (all)
B.6. Final tests in Cadarache (all)
B.6.1. Transport of the systems to Cadarache and installation on the manipulator
B.6.2. Cutting tests on non-radioactive representative models
B.6.3. Tests with samples prepared under B.5.
B.7. Final evaluation and recommendations (all)

C. PROGRESS AND RESULTS

SUMMARY OF MAIN ISSUES

The work carried out in 1992 was connected with three aims in relation with the topics of work programme packages B2 B3 B4.

At the Pegase facilities the main activity concerned the starting of the whole installation with manual control. First tests on thick plates with maximum power conditions (240V, 1000A) have been realised with the TD 600 N SAF plasma torch and the results analysed.

With regards to the other tools developed by IW, tests with an improved plasma saw have been made showing better results referring to the form of the kerf. Cutting tests with the consumable electrode have been carried on. In order to increase the sectile material thicknesses, a new consumable electrode tool was designed and constructed. Underwater tests with the help of this device were successful on 80 mm diameter circular solid rods. Another cutting tool was also tested, called Contact Arc Metal Cutting (CAMC), with this technique greater cutting thicknesses will be easily obtained.

As regards the control system the microprocessor control unit, including low-level solftware modules and motor control hardware, was constructed.

First tests in real conditions have been made after transporting the control unit to Cadarache and installing it in the Pegase facility. This equipment was returned to the Aachen laboratory to make the required improvements.

Progress and results

1. Development of the plasma torch and adaptation of the moving device (B2)

After connecting all circuits (electricity, water, operating gas), a great part of the activity has been focused on starting the whole installation with the manual control unit. Figure 1 shows a general view of this installation. It includes the main following components : the tank (capacity 7775 l, water depth about 2m), the motorized manipulator holding the different tools (plasma torch, consumable electrode, plasma saw, CAMC), the ventilated containment (volume $36m^3$, flowrate ventilation $3000m^3/h$), control cabinets, orientable immersed video camera for cutting observation.

The new cables for the electrical torch supply allow a 1000A intensity to be obtained with insignificant losses. Then a first range of tests consisted in observing SAF torch behaviour when the intensity is increased to 1000A.

Various tests were carried out with Argon as operating gas. At the maximum power of the 4 Miller sets, we could cut an 85 mm thick stainless steel plate (figure 2), reaching a cutting intensity of 950A and a cutting voltage of 240V. But the tests could not be continued on account of a over-rapid wear of the torch tuyere.

The continuation of the test programme will consist in improving the SAF torch and optimizing operation parameters such as gas flowrate and cutting speed in order to achieve steady operating conditions. After this, a higher cutting voltage will be implemented in order to obtain larger cutting thicknesses.

2. Development of other tools (B3)

As problems arose in producing a kerf wide enough for the plasma saw to dip in, a modified saw was designed and constructed, including an additional gas flow in the torch body without raising its size (figure 3). Compared to the standard plasma saw, the tests showed better results referring to the form of the kerf, but still the kerf is very sometimes unsteady, which leads to collisions with the nozzle. Secondary arcs may then distroy the nozzle. Summarizing the results of tests leads to the conclusion that this cutting technique seems impossible to achieve remotely.

Cutting tests with the consumable electrode on model No.2 (tube ø 110/80 mm with an inner bolt of ø 54 mm) were not successful with a ø 4 mm wire. In order to increase the sectile material thickness, a new consumable electrode tool was designed and constructed (figure 4). This system allows to feed diameter wires up to ø 4 mm at variable speed. A special mechanism with a graphite electrode makes it possible to cut the wire right in front of the nozzle by means of short circuit, which is necessary, if the wire is welded to the workpiece because of malfunction. A central control unit can operate the power source and the water pump as well as the speed of wire feed. The overall consumed wire or the speed of wire feed may optionally be read from the display. Tests with the help of this device were successful on circular solid rods (ø 80 mm) under water. Two power sources will be connected in parallel, so that further tests can also concentrate on cutting with more energy.

Because of the complications in cutting large wall thicknesses, another cutting tool was tested, called "Contact arc metal cutting", in short form "CAMC". The principle is based on a system of electrode and workpiece, which are connected in short circuit with high current by means of contact. This cutting technique was already tested on the Cadarache-models. A new modular-constructed system uses a second water pump with high pressure. A favourite method to date is the so-called "Guillotine-technique". According to this, a flat electrode is laterally guided and slowly driven through the workpiece. The electrode can also be used like a sword, as was done on the model of the reactor vessel (figure 5).

3. Development of control systems for sensor-controlled piloting of the handling system for the tools and the process parameter (B.4)

The investigations of the applicability of inductive sensors have been finished. The 15 mm diameter inductive sensor BAW-018-PF-1K from the German company Balluff has been selected for this application as final solution. The sensor in its orignial version is only resistent against low radioactive emissions. Balluff explained that up to now, no tests in highly radiated surroundings have been carried out. They were not able to specify the exact lifetime of the sensors in radioactive environments with estimated dose rates up to 10^3 Gy/h. The construction drawings of the sensors were sent together with a list of materials used for an estimation of the maximum allowable dose rates. The plans were sent to CEA Saclay in order to obtain an estimation of the sensor's lifetime in radioactive surroundings, with regard to the materials used. CEA Saclay reported that in all cases tests under radioactive conditions would be necessary to evaluate the applicability of the sensors.

The work on process control mainly dealt with the development of the automatic control unit, where three possibilities for manipulator movement were implemented. The movement can be controlled completely manually either in axes coordinates or in cartesian world coordinates in a teaching phase and then start the cutting process with an automatic movement. For that purpose the manipulator is moved manually to the corresponding teaching points in the working area of the manipulator. After the process has started, the manipulator moves accurately along the taught paths at a programmable speed. The third possibility integrates the inductive sensor into the teached procedure. After teaching several approximate coordinates of the workpiece, a sensor run starts. Using predefined directions for searching, the sensor determines the exact positions of the workpiece surface. The different steps of the sensor-based teaching procedure are shown in Fig.6.

Software modules for the manual and automatic movement of the 4-axis manipulator were developed and tested. A suitable user interface was implemented for comfortable operation of the manipulator and its functions. A graphical environment with a menu-driven command and data input system was developed. With the help of this interface the operator is able to control all important system states and to feed data for movement and process control in a very easy and well-arranged way. During cutting, the process is supervised by the operator who obtains all the necessary information about the workpiece, travel speed and manipulator and workpiece geometry from the system controller via a LC-Display, embedded in the user interface.

The hardware developed for automatic process control is shown in two photographs in Fig.7 The first photo shows the developed manipulator controller with the user interface (Buttons and LC-Display), the motor power drivers and a ventilation system. The remote control unit is shown in the second photograph. The manual drive of the manipulator and the operation of the menu is done with the two axis analogue joystick. Several buttons are used for distinguishing between menu operation and movement control. An additional emergency stop button has been integrated to allow a fast reaction in troublesome situations.

Transport, installation and first tests at Cadarache :

After the transport of the automatic control unit to Cadarache, the system's electrical installation into the manipulator hardware started and the functions could be tested in the real environment.

The first tests of the control unit in cooperation with the manipulator hardware showed several hardware difficulties which could partly be solved at the facility in Cadarache. The remaining bus access problems could then be solved in the laboratory in Aachen.

During tests following the installation, the parameters of the motors, gears and sensors were adjusted to the special conditions of the manipulator and set to the software variables. Resonance frequencies were detected, at which the stepper motors fall out of step due to the necessary high torques and thus generating incorrect movements. This difficulty was solved excluding a narrow band of frequency from the possible motor velocity, which has in reality no influence upon the possible manipulator velocities. Tests during the cut with the plasma torch showed that no disturbances of the manipulator control could be detected during operation. A series of successive tests of the whole software package followed.

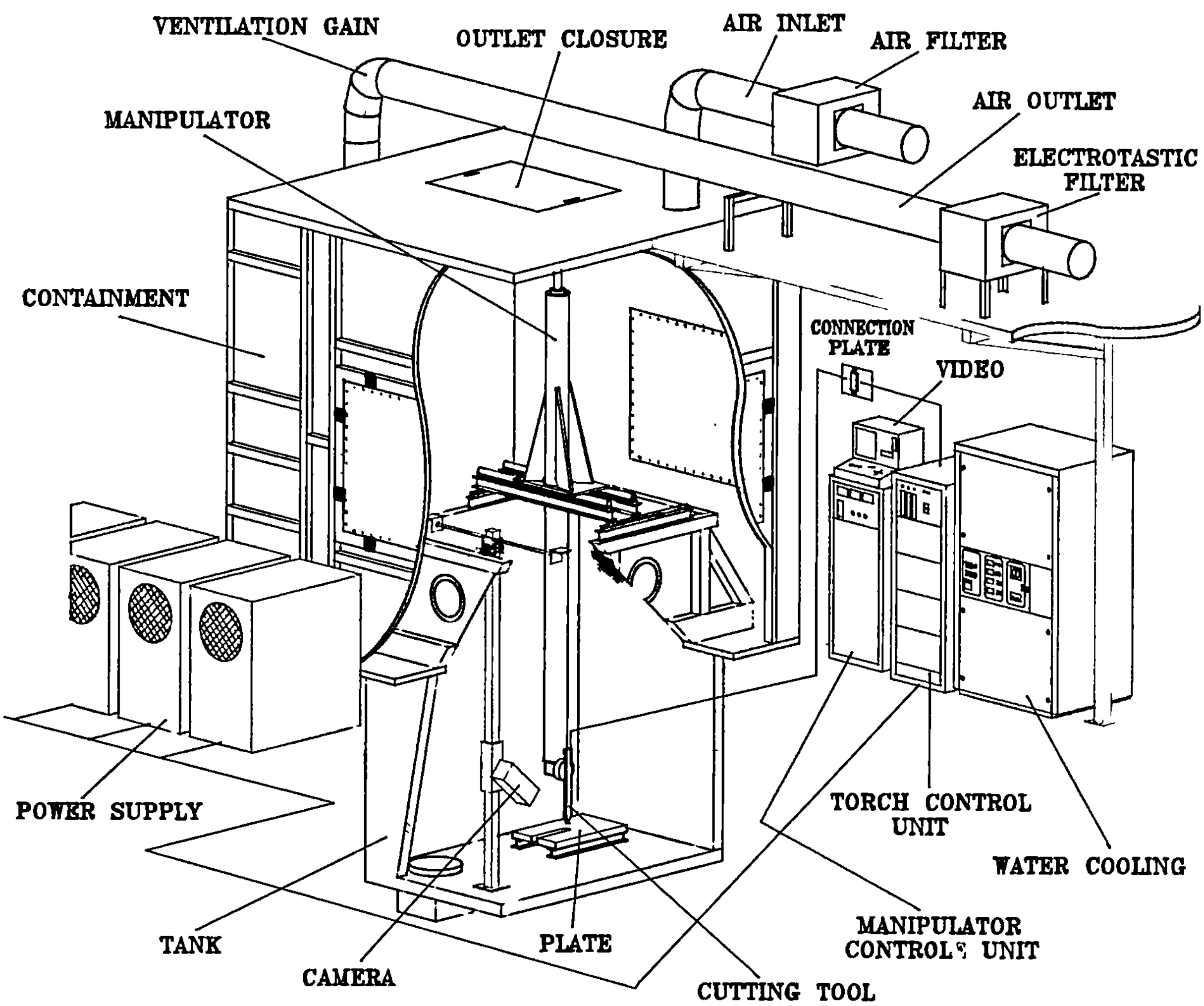


Figure 1 : General view of the installation at the Pegase facility

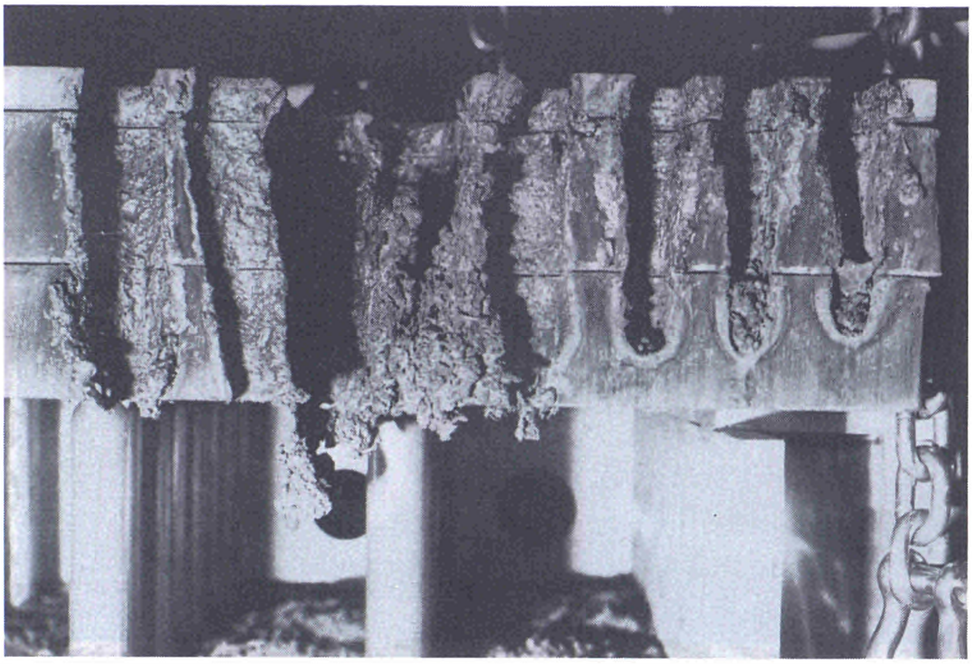


Figure 2 : 85 mm thick stainless steel plate with various cutting kerfs at intensities varying from 600 A to 950 A

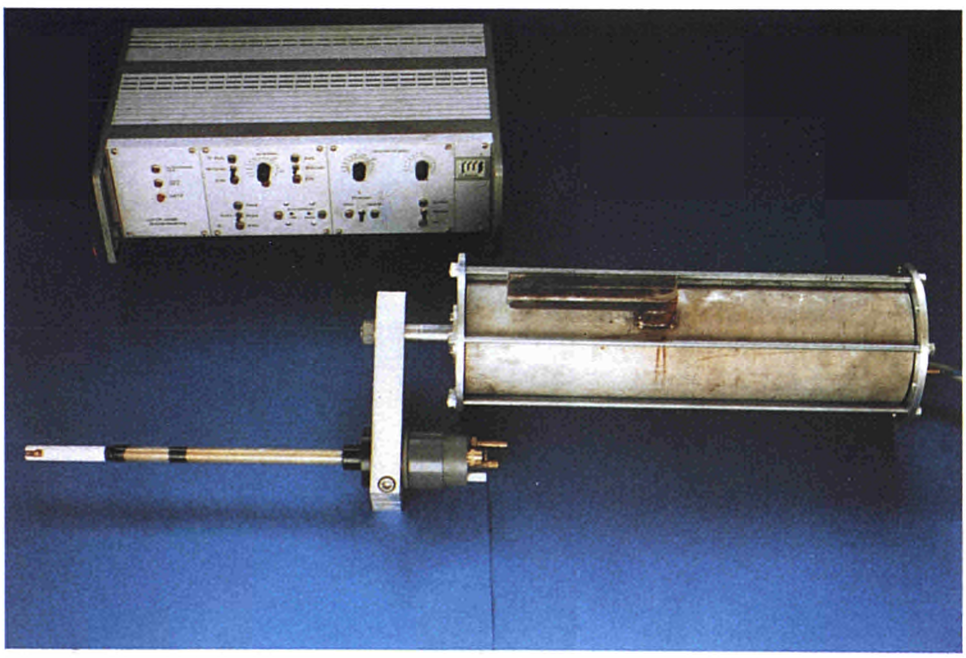


Figure 3 : Modified plasma saw with moving device and control unit

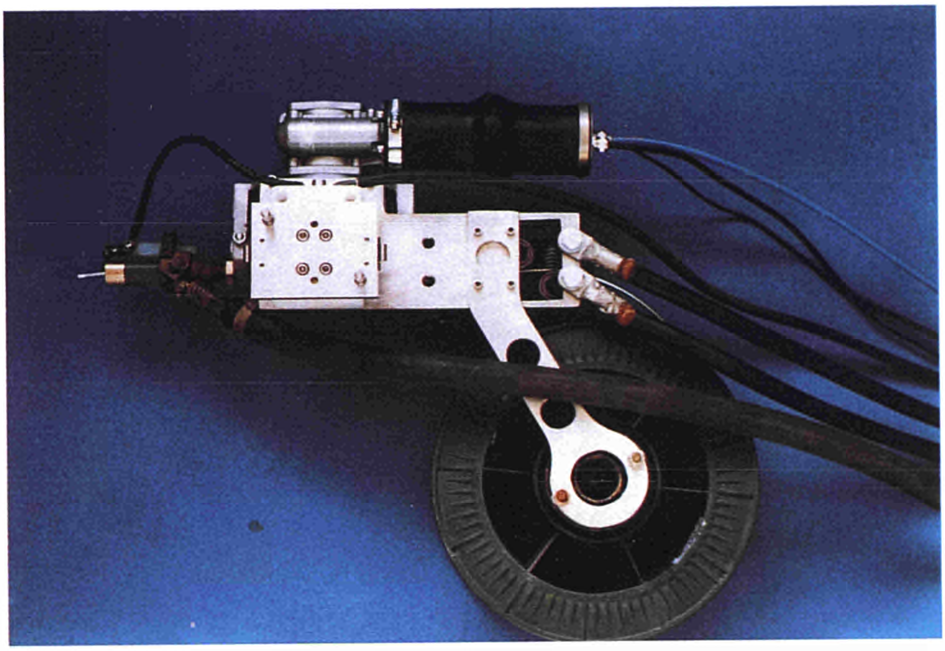


Figure 4 : Consumable electrode tool, including torch body with wire cutting system, motor, gear, wire feed unit and flange, that fits to the Cadarache manipulator

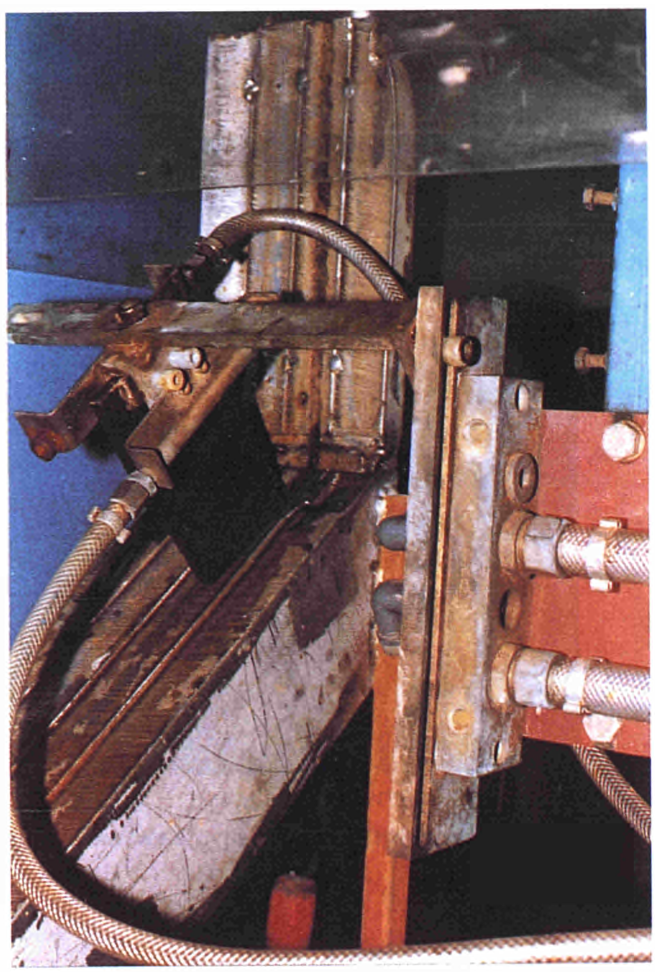


Figure 5 : Set up for cutting the model of the reactor vessel. The electrode was moved vertical down first to cut the reactor wall with a thickness of 25 mm. The device was then moved with an inclination referring to the reactor bottom with a thickness of 130 mm.

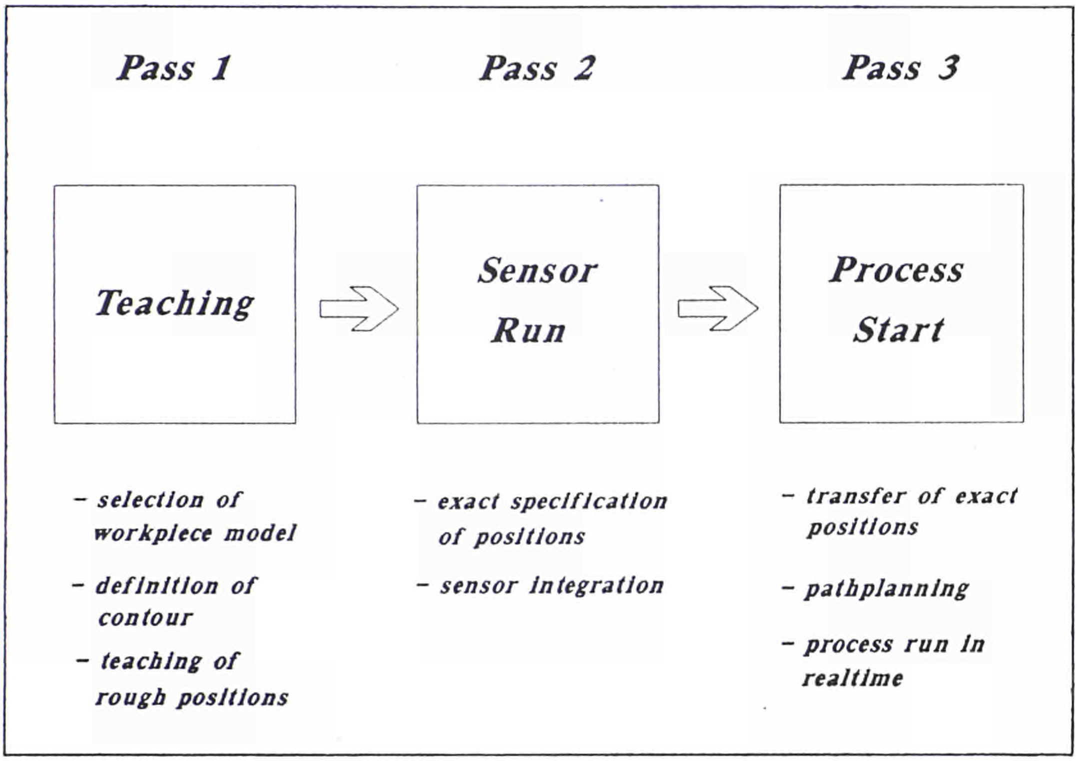


Figure 6 : Stages of sensor-based teaching

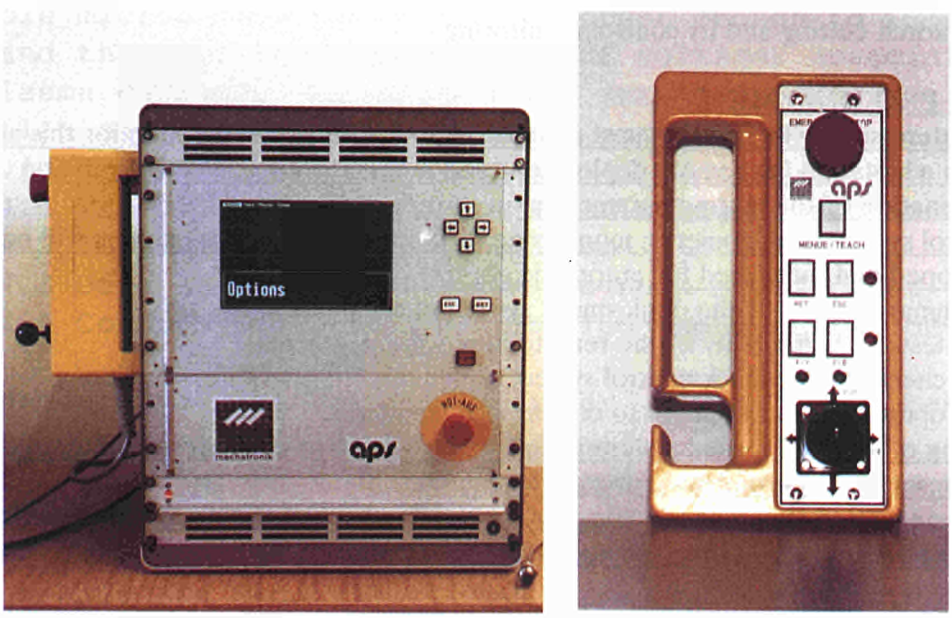


Figure 7 : Manipulator controller and remote-control unit

3.6. DEVELOPMENT OF A PLASMA ARC TORCH AND CONTROL/MONITORING TECHNIQUE FOR THE INTERNAL CUTTING OF SMALL BORE PIPEWORK

Contractors:	AEA Wind	
Contract No.:	FI2D-0026	
Work Period:	July 1990 - June 1993	
Project Manager:	S J WHITE, AEA Wind.	
	Phone: 44/9467/72437	Fax: 44/9467/72409

A. OBJECTIVE AND SCOPE

During decommissioning of nuclear facilities, small bore pipework needs to be cut remotely up to distances of 10m with access through the top of the pile cap. Often, due to the close packing of the pipework, the cutting operation must be performed internally through the bore of the tube. In the absence of direct viewing and manual access, there is a requirement to develop techniques for the cutting process and methods for monitoring and remotely controlling its operation and ensuring its effectiveness. The plasma torch process has been selected as the cutting method based on economic considerations and on its reliability and effectiveness in remote and manual operations.

The objective of the project is to develop:

- techniques based on non-contact sensors which can monitor and remotely control the progress and effectiveness of the cutting process;
- a small plasma torch capable of being inserted in the bore of < 50 mm internal diameter pipework with remote deployment (up to 10 m distance) under automatic control;
- the deployment system which can be located on the pile cap, positioned above each pipe in turn and lowered to a predetermined depth to perform a complete circumferential severance of the tube in one pass.

The work will include cutting trials of the complete system in a full-size mock-up of a reactor gas manifold.

The AEA Northern Research Laboratories will take into account the experience gained elsewhere and particularly at the "Institut für Werkstoffkunde der Universität Hannover" on plasma arc torch cutting and its control/monitoring.

B. WORK PROGRAMME

B.1. Literature survey to find the most suitable plasma cutting combination for this application.
B.2. Torch adaptation for remote deployment and automatically controlled rotation
B.3. Examination of the cutting parameters on representative pipework.
B.4. Control system developments; monitoring technique and feedback system will be designed, developed and interfaced for automatic control.
B.5. Preliminary testing of the deployment system in small-size mock-up
B.5.1. To test the workability of the remote deployment system.
B.5.2. To check the feedback control system under remote operation conditions.
B.5.3. To optimise the equipment to commercial standards.
B.6. Testing of the deployment system in full-size mock-up to evaluate the optimised system in a representative decommissioning environment.
B.7. Final evaluation including specific data on costs and radiological impact on work force and working area, working time and secondary waste arisings.

C. PROGRESS OF WORK AND RESULTS

Summary of Main Issues

This year saw the completion of the tasks associated with B2 (Torch Adaptation Design Study), B3 (Examination of cutting parameters) and B5 (Preliminary testing of deployment system in a small size mock-up). Due to operational requirements the cutting of the small bore staytubes were required mid 1992. The consequence of this requirement was to bring forward the development and testing of the plasma torch and its XYZ deployment system. During August 1991 a total of 130 tubes were successfully cut and inspected. During this operation a range of signals were recorded for signal conditioning evaluation. The work relating to task B4 was expanded to include transducer and signal conditioning development. The need for this expansion was due to the fact commercially available microphones were not robust enough to stand continuous use in the proposed environments. A piezo-electric ultrasonic transducer was used based upon those developed within AEA Technology for high temperature applications.

Progress and Results

1. B4 Control System Developments

Progress to date has been related to the development of a suitable transducer to record the signal and the associated control algorithm. During the initial data collection trials it became apparent that the commercially available microphones would not be robust enough to withstand the high temperatures and HF voltages associated with plasma cutting. To allow for a more detailed definition of requirements the tasks under B4 were sub-divided as follows:

- B4.1 Transducer and Signal Conditioning Development
- B4.2 Experimental Data Collection
- B4.3 Data Analysis and Control System Development
- B4.4 Control System Implementation and Integration

A piezo-electric ultrasonic transducer was developed based upon the experience of using such devices within AEA Technology for high temperature applications. The main area of concern with the transducer was the low frequency response characteristics, such devices are normally used in the high Kilo and Megahertz range rather than the potential sub Kilohertz range required for this application.

Following a series of trials a suitable microphone was identified which gave a good match with the original reference microphone in the same frequency band. This device will now be used in all subsequent work.

Problems related to the interference produced by HF voltages produced during arc-striking have been overcome by careful use of earthing and screening plus the use of an optical data link between the transducer and the microprocessor based controller. The link now digitises the accoustic data and transmits the information via an optical fibre to the controller situated at a reasonable distance away from the cutting site. Trials have been successfully carried out to demonstrate that the digitisation process is fast and accurate enough to retain the acoustic data of interest.

2. B4.2 Data Collection

To test and develop the control algorithm test data is required, a number of recordings have been taken of the cutting process. A series of different cut profiles were recorded to identify the different acoustic signatures between 'cut' and 'no-cut' operations. This information is currently being used to develop the control algorithm.

3. B4.3 Development of Algorithm

Good progress has been made during the last 12 months of the development of the control algorithm. The recordings from the data collection phase were played into an analogue analysis system. This system consisted of 13 band pass filters connected in parallel. The signal from each of these pass bands was then rectified and smoothed to give a DC level as an output. These outputs were then digitised and the data stored on disc. The amplitudes to identify a dominal frequency band common to all of the signals from the different types of cut. Recent work is concentrating on identifying a suitable signal range to be implemented into the algorithm.

Additional developments on the electronics have also taken place with the requirement that the front-end electronics be battery powered and therefore isolated from the mains supply. Reductions in power consumption have been achieved by a factor of approximately 10. This has been achieved by using an analogue means of transmitting the signal along the optical link.

4. B4.4 Control System Implementation and Integration

The system will be based upon an industrial PC for the system implementation. This will replace the original multibus based system which will prove increasingly difficult to support. Tests using an industrial PC have taken place in the cutting environment to test its susceptibility to HF interference. To date the system has performed well.

5. B6 Testing in Full Size Mock-Ups

During all phases of data collection full scale mock-ups have been used. Actual cutting data from the reactor has also been used following the decision to bring forward the operational cutting programme. It is currently hoped that additional testing of the system can take place on the reactor before the completion of the contract.

3.7. DEVELOPMENT OF A STEEL CABLE TO CUT HIGHLY REINFORCED CONCRETE WITH MINIMISED WATER CONSUMPTION

Contractor:	Diamond Service
Contract No.:	FI2D-0027
Work Period:	September 1990 - July 1992
Project Manager:	A BOSELLI.
	Phone: 39/523/822 447 Fax: 39/523/822 630

A. OBJECTIVE AND SCOPE

The project is aimed at the development of a steel cable charged with diamond pearls with particular cutting qualities, considerable mechanical resistance and minimum consumption of water or other cooling mixtures. The cable should be particularly suitable for cutting highly reinforced concrete structures of nuclear installations. The main objective is the control of secondary waste generation (cooling and severed concrete) during the cutting operations. The cuts can be carried out at various distances from operator to structure and therefore offer considerable security and protection of work force. The cutting time and the derived radiation exposure to operators will be evaluated on a uncontaminated concrete structure.

The contractor will carry out cuts on concrete structures of a USSR nuclear plant; the obtained results will be compared with those obtained within the framework of this project.

B. WORK PROGRAMME

B.1. Development of a high resistance diamond pearl

B.2. Development of suitable materials for the cable vulcanisation

B.3. Preparation of the test mock-up and of a representative concrete block

B.4. Selection and improvement of a suitable steel cable

B.5. Assembling of cable components (light steel cable, diamond pearls, springs and spacers)

B.6. Cutting tests on non-contaminated concrete structures

B.7. Final evaluation, taking into account cable consumption, costs of the technique, cooling water consumption, secondary waste arisings, radiation exposure, and a comparison with the cutting work carried out in the USSR.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The investigations carried out led to the improvement of the diamond wire for cutting reinforced concrete. The quantity of coolant used could be drastically reduced by adding some EKOSOL to the cooling water.

Progress and results

During the first cutting tests, carried out with poor coolant, an excessive and uneven consumption of the diamond beads was noted. It was therefore necessary to modify the diamond bead, using for the new one synthetic diamond 20/30 mesh and increasing by 20% the Tungsten Carbide content in order to improve hardness and cutting capacity of the beads (B.1).

Cutting tests on highly reinforced concrete carried out with a steel cable charged with 40 diamonds beads/mt. and covered with plastic (Desmopan) gave quite satisfactory results and confirmed that this type of vulcanization (Fig. 1 + 2) reduces the risks of breakages of the cable and, as a result, reduces radiation exposure to operators due to reparation (B.2).

To avoid breakages of the steel cable, copper joints were replaced by steel joints, external Ø 8 mm, length 20 mm, taken from high pressure hydraulic pipes (B.4, B.5). The coolant was gradually reduced adding an emulsible oil (EKOSOL) to water. The best results were obtained adding 5% EKOSOL to water; 1 m^2 of the test mock-up was cut in $\pm$ 150 minutes and $\pm$ 230 l of coolant consumed (B.6).

The diamond cable can be used for cutting nuclear components, where the cable can wrap the structure to be cut. To achieve this, it is sometimes necessary to drill the structure for the passage of the cable, with subsequent generation of secondary waste (B.7).

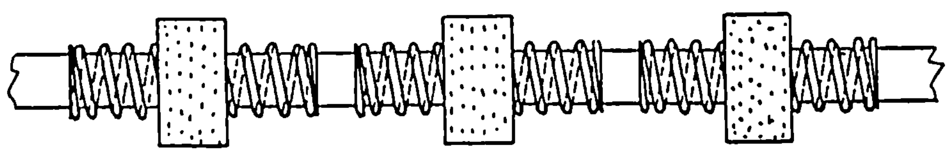


Fig. 1: Assembling of wire before vulcanization

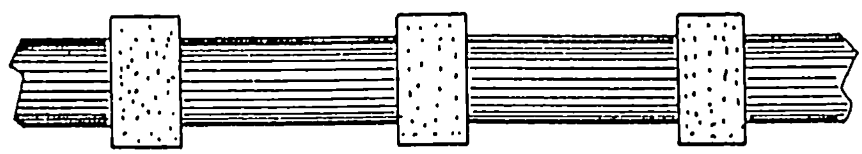


Fig. 2: Wire after vulcanization

3.8. ASSESSMENT OF STATE-OF-THE-ART CO LASER TECHNOLOGY AS AN IMPROVED DISMANTLING TOOL

Contractors: AEA Culh., DLR Stuttgart
Contract No.: FI2D-0028
Work Period: September 1990 - November 1992
Coordinator: J H MEGAW, AEA Culham.
Phone: 44/235/46 42 15 Fax: 44/235/46 41 38

A. OBJECTIVE AND SCOPE

The main objective of this project is to carry out a laboratory-scale experimental investigation of the capabilities and potential advantages of carbon monoxide (CO) lasers, compared with carbon dioxide (CO_2) lasers. Previous studies on CO_2 lasers for decommissioning indicate that they can operate as elegant and flexible tools, but there are limitations with regard to cutting performance, and a need for articulated mirror-based beam delivery systems. The present project is motivated by: reported Japanese results indicating CO laser cutting performance significantly superior to that of CO_2; and the potential for use of optical fibre beam delivery at the shorter wavelength (5 μm cf 10.6 μm).

The partner organisations (which are currently engaged in developing CO lasers in the power range up to approximately 1 kW) will carry out complementary investigations, using CO and CO_2 beams, on steels, concrete and graphite concerning: the nature of the beam-workpiece interaction and how it differs at the two wavelengths; assessment of the respective cutting capabilities. It is expected that the work will: provide the sole European source of such information; enable quantification of possible technical and economic advantages of CO lasers for decommissioning; provide information on the parallel Japanese programme (where it is reported that CO lasers of $\geq$ 20 kW are under development); make recommendations on a strategy regarding possible future use of CO lasers for decommissioning and commercial exploitation thereof.

The responsible partners for work on structural steel and graphite and for work on stainless steel and concrete will be AEA and DLR, respectively.

B. WORK PROGRAMME

B.1. Assessment of beam-workpiece interaction for CO laser, and comparisons with CO_2 laser (All)

B.2. Assessment of CO laser cutting capabilities and comparisons with CO_2 laser (All)

B.3. Final evaluation showing quantified differences in materials-processing capabilities of the two lasers, specific data on costs, secondary waste produced and radiological impact on workforce and working area (All)

C. PROGRESS OF WORK AND RESULTS OBTAINED

1. Summary of main issues

All work on the programme has now been successfully completed. In Phase B1, the beam handling studies have been consolidated to show the sensitivity of the CO laser to water vapour not only in the beam line transmission tubes but also in the air–filled components within the laser cavity. The investigations of the beam–work piece interaction plasma have been extended to demonstrate: the threshold CO laser beam intensity for plasma ignition is at least twice that of CO_2 laser radiation; the shorter wavelength of the CO laser has the benefit of about 60–70% less absorption in a plasma plume than a CO_2 laser; on the other hand, for certain welding operations, the presence of plasma can assist energy coupling to the metal. Direct measurements of the absorption of CO and CO_2 laser irradiation by solid and molten steel and stainless steel have been undertaken by DLR and AEA using integrating sphere equipment. These proved to be challenging experiments but nevertheless the results at both institutes are in agreement, showing a small but significant increase in the absorption of CO laser radiation by stainless steel as the surface transforms from the solid state into a liquid state during exposure to the laser radiation. In Phase B2, CO and CO_2 laser cutting trials have been carried out by both partners on a range of materials which are relevant to decommissioning. In the light of the maximum laser power obtainable from the CO lasers, AEA has concentrated on lower power, thin section cutting whilst DLR worked on thicker section using power up to 400W. In Phase B3, evaluations of the cutting data have been undertaken and the full results, together with its implications regarding secondary waste, will be presented in the project final report. The data show the degree of difference in cutting performance between CO and CO_2 lasers depends on the work piece material and the cutting conditions.For the purposes of the present report, only a few of the cutting results are presented from those experiments showing a difference for the laser types.

2. Progress of work

2.1 Beam handling

The previously–reported work under this heading included beam profiling and studies of long–distance beam propagation, the latter covering the effects of atmospheric water vapour on beam transmission. Further work has been carried out at AEA concerning water vapour, primarily because the laser output was found to fluctuate on days of high humidity. Spectral analysis of the output beam indicated that the strongest bands of spectral lines exist around 5.3μm, 5.4μm and 5.6μm, but that two lines from the last two bands change intensity from zero to maximum at the same rate as was observed with the pyroelectric laser power meter (which indicated a 10% modulation in total power at a time constant of about one second). Turning the beam line purge off reduced the relative intensity of all lines (the total power reduced to about 50%) while the power variations as measured with the pyroelectric detector increased to about 30%. These results suggest that the power variations cannot be caused by water vapour in the beam line alone — there is also an effect from the laser itself. It was subsequently found that saturating the 'roof top' mirrors enclosure with water vapour totally extinguished the variable lines in the laser spectrum (and caused a large reduction in total power). It can therefore be concluded that the observed variations are explained partly by residual water vapour in the laser beam path and partly by water vapour traces within the optical resonator. It is therefore necessary to employ air of adequately low dew point; the present work (see also previous progress report) suggests that a dew point value of –40° should be aimed for.

2.2 Plasma effects

In many practical applications of high power lasers, the beam–work piece interaction is characterised by the appearance of an intense plasma which absorbs the beam and, generally, reduces the amount of useful power available for processing the work piece. The present investigations have addressed three aspects, comparing CO and CO_2 lasers: (i) threshold intensity for plasma ignition (DLR); (ii) experimental determination of the attenuation of a laser beam in traversing a plasma plume (AEA); (iii) the role of plasma during welding (DLR). These are discussed in turn.

(i) For a laser beam focused on a stainless steel target in argon, and depending on the interaction time of the laser beam, the processing range starts at low intensities with the onset of a plasma which enhances the coupling efficiency of the laser radiation to the metal surface. The upper limit of the intensity is given by the separation of the argon plasma from the metal surface which then absorbs all the laser radiation and prevents it from coupling to the target. As the absorption coefficient of the plasma is inversely proportional to the square of the wavelength of the laser radiation, the processing range of the CO laser radiation is expected to be shifted towards higher intensities compared to the CO_2 laser and so higher processing velocities should be possible. The previously–reported DLR experiments, using a CO_2 laser beam, showed that at at a power intensity of $1.23 \times 10^6 Wcm^{-2}$ no plasma is observed,and at $2.5 \times 10^6 Wcm^{-2}$ a plasma is ignited and deep welding occurs. These experiments have now been repeated with the CO laser, and using a 'best form' focusing lens yielding a maximum intensity of $4.6 \times 10^6 Wcm^{-2}$. The results indicate that up to this level, no argon plasma is ignited. Thus a CO laser beam may be used at focused intensities up to approximately twice that of a CO_2 laser without igniting a plasma.

(ii) The AEA work on absorption of a probe laser beam by welding plasma plumes has been consolidated and extended. The 5 kW laser CL5 created a weld (with associated plasma plume) in a rotating pipe sample and the plume was probed using either the CO laser beam (delivered to the welding workstation via the 20 m beam line) or a low power CO_2 laser beam, produced by intercepting part of the welding beam with a small scraper mirror. The investigation and optimisation work on transmission through the 20 m beam line resulted in the availability of probe powers up to 30W, where the signal to noise situation was good. A range of conditions were investigated for creation of the welding plasma. The 10.6μm laser power was typically 4.5 kW, the shield gas was argon, and pipes of aluminium and stainless steel were used. In general, the absorptivity measured by the probe beam was extremely small, typically < 2%, at 10.6μm and vanishingly small at 5μm. Experiments were also carried out using a Tungsten Inert Gas welding torch as replacement for the welding laser beam. The torch conditions were adjusted to give a surface melt representative of that obtained with the laser beam and this provided a relatively stable plasma for probing with the 5 and 10.6μm beams although uncertainties were present due to fluctuations of plasma and probe beams. However, best estimates suggest that absorption at 5μm and 10.6μm are less than 5% and (15±5)%, respectively. These are not inconsistent with the indications from the laser plasma measurements and it thus appears that absorption of the CO laser beam is approximately one third of that for the CO_2 laser beam. This demonstrates a valuable benefit of the CO laser and is in line with the x4 reduction predicted from inverse bremsstrahlung theory.

(iii) Additional information about the influence of plasma on beam–work piece interactions has also been obtained from experiments at DLR. Melt–welding experiments were carried out using the CO and CO_2 lasers at 400W (same spot size) on stainless steel for different work piece speeds using different shielding gases. At low speeds (corresponding to high line energies = laser power/welding velocity) the CO_2 laser welds show the highest absorption with argon as shielding gas. In this case, the argon plasma enhances the coupling efficiency of the laser beam to the metal. On the other hand, at high welding speeds where the influence of plasma coupling is less effective, the CO laser beam is coupled more effectively to the target material due to its higher absorption coefficient for 5μm wavelength radiation. As no plasma is ignited at this wavelength, the welding results are nearly independent of the kind of the shielding gas.

2.3 Beam absorption by metal (B3)

Both AEA and DLR have used the integrating sphere technique to measure the absorption by stainless steel at different temperatures around its melting point. Schematic diagrams of the apparatus at DLR and AEA are shown in Figures 1 and 2 respectively. The powers P_1 and P_2 measured by detectors D_1 and D_2 are proportional to the beam powers reflected from, and incident on, the sample; the temperature–dependent reflectivity $R(T)$ is given by $R(T) \propto P_1/P_2$. The constant of proportionality between $R(T)$ and the quotient of the two signals can be obtained from the results at room temperature where R is known from the literature. The results of experiments at DLR are shown in Figure 3 where the quotient signal is plotted as a function of time.The central region corresponds to the beam–on time for the laser pulse. The arbitrary values and spikes before and after the pulse are due to division by values of P_2 near or equal to zero. During the laser pulse, the reflectivity shows a slight decrease, starting at 0.9 (the room temperature value) down to 0.84 shortly before the melting point of the surface is reached. The melting point is characterised by a small step in the reflectivity halfway through the laser pulse, where R rises to about 0.87. This rise at the melting point is expected from theoretical considerations and has been verified up to now with experiments using CO_2 lasers. Corresponding results from AEA are presented in Figure 4. The detector signals were recorded on a LeCroy 9400 oscilloscope which enabled calculation and display of the quotient. Shown are the signals from the reference detector D_2, the reflected signal, and their quotient. The signal from the reference detector is negative because its amplifier also inverts the signal. Because of this, the quotient is also inverted. The quotient shows, after a fast initial rise due to non-identical time constants of the detectors, a decrease for about 1ms, where a step increase is observed (indicated by a small line in each of the traces), and melting takes place. Assuming an initial value of 0.9 for the reflectivity, it decreases to about 0.81 just before melting, and then increases again to about 0.88. Given the difficult nature of the experiments, there is good agreement between the DLR and AEA results, and they confirm theoretical predictions and help understanding of the beam–work piece interactions during laser processing.

2.4 Cutting trials and evaluation (B2 and B3)

Cutting trials in a range of materials have now been completed at AEA and DLR. The approach was to choose conditions so that, insofar as was possible, comparisons could be made between cutting performance for the two wavelengths at comparable beam powers and spot sizes. In general the performance indicator was the threshold cutting speed i.e. the fastest speed for which a beam of given power would permit complete penetration (and separation) of the workpiece. Reflecting the laser powers available, AEA have concentrated in thinner section samples whilst DLR has undertaken work on thicker sections at power levels up to 400W. The data show that the degree of difference in cutting performance between CO and CO_2 depends on the workpiece material and the cutting conditions. For the purposes of the present report, only a few of the cutting results are presented for those experiments showing a difference for the two wavelengths.

2.4.1 Work at AEA

Using a CO gas mixture, the single–fold DC excited slow flow laser at AEA can operate at approximately $5\mu m$ wavelength for power levels in the range 160–200W. However, since reproducibility was crucial to these experiments, the level was set at 100W. The resulting power at the workpiece, for a transmission system comprising four mirrors and a focusing lens, was 60W. Half of the total length of the same laser was used with a CO_2 gas mixture to produce the $10.6\mu m$ beam. The laser beam was focused vertically downwards by a lens on to the work pieces which were transported on an electrically driven linear table having programmable speed. The cutting gas assist nozzle was of the standard type, coaxial with the beam. The CO laser beam was focused by a lens of 50mm focal length, whilst in the case of CO_2 two alternative lenses, of focal length 50 and 100mm were used. The CO beam spot diameter lay between the values obtained with the two lenses in the CO_2 case. Cutting performance comparisons were achieved by interpolating or normalising the CO_2 results to a value corresponding to the equivalent CO beam spot size; the normalisation was based on an energy balance in which, for constant power, the amount of material removed per unit time was considered a constant, determined by the product: thickness × speed × spot diameter. The actual measurements of focused beam spot size were carried out by scanning a knife–edge through the beam whilst measuring the transmitted power. The diameter of a Gaussian beam is the distance between the knife–edge positions that transmit 84.1 and 15.9% of the total power. Figure 5 shows the results from beam scanning the three focus conditions. The families of curves in the upper part correspond to different positions along the beam axis; the beam diameters, calculated by this method, are plotted in the lower part of the diagram to yield the beam envelope. The spot sizes are shown, together with other cutting parameters, in Table 1. In collecting cutting data at low power, it is particularly important to take into account the statistical nature of the process. Accordingly, the following procedure was adopted. With a single sample strip in position under the cutting head, a hole was pierced through the material and subsequently the linear slide brought into motion to a predetermined speed. The performance was noted as either Cut or No cut, depending on whether sustained penetration took place. This procedure was repeated between 10 and 15 times in quick succession, resulting in a 'cutting result ratio'. A cutting result of unity means that all cuts were successful, while a cutting result of zero means that none of the attempted cuts were successful (no penetration). In practice, it can be expected that when the cutting threshold speed is approached, the cutting result ratio will more or less gradually reduce from unity to zero. To illustrate the method, and to indicate results for one of the non–metals, data is presented in Figure 6. The cutting result ratio is plotted against speed for 1mm thick silicon nitride. It is seen that, for a ratio <1, CO yielded faster speeds. For the threshold cutting speed at ratio 1, and normalising the CO_2 results based on the CO spot size, the following is obtained:

Threshold CO cutting speed, v(CO)	$2.6mms^{-1}$
Normalised threshold CO_2 cutting speed, $v(CO_2)$:	$1.5mms^{-1}$
$v(CO)/v(CO_2)$	1.7

Further results, described and discussed in the Final Report, were obtained for a further six materials (3 metals, 3 non–metals).

2.4.2 Work at DLR

The cutting experiments with the CO laser radiation were done with a workstation, built at DLR. The station consists of a one–axis speed–controlled table and a small beam guiding system. The table was located as near as possible to the laser, and the beam guiding system used cooled plane copper mirrors. The cutting experiments with a wavelength of 10.6μm were performed with a 1 kW longitudinal flow, DC excited CO_2 laser, which is commercially available. The working station in this case was a commercial CNC–controlled two–axis table. The focusing optics were commercially available transmission optics. The diameter of the focal point is equal for both lasers, as are the maximum power density and the medium power density. The focusing details are given in Table 2, and a focus intensity profile, obtained with a PROMETEC beam analyser, is given in Figure 7. The nozzle used for the oxygen supply was a standard type with an orifice diameter of 0.8mm. The nozzle was arranged in such a way that the distance between orifice and surface of the workpiece was 0.5mm at all the runs. The gas pressure was measured inside the housing of the focusing optics. In all the runs the position of the focal point was on the surface of the sample. The polarisation of the beam is parallel to the cut direction. The pressure of the working gas was adjusted to the type and thickness of material. Some essential features of the cutting experiments are summarised below.

Materials:	mild steel, stainless steel, concrete
Thickness range:	0.5–5mm
Cutting power range	100–400W
Cutting speed range	$0.2–9mmin^{-1}$
Assist gases	oxygen, nitrogen
Assist gas pressure range	1–7bar

The threshold cutting speed in these experiments is defined as that giving complete separation of the sample. When using a non–reactive gas (N_2) there is a sharp boundary between cut and non–cut region. The use of a reactive gas (O_2) gives a much more complicated situation and the definition used here is that there should be no dross blockage at the bottom of the cut. The full set of results are described and discussed in the final report. Here, only one set for the non–reactive (N_2) gas cutting of mild and stainless steel is presented in Figures 8 and 9 respectively for a range of thicknesses. In this case there is significant improvement in cutting speed for CO operation, particularly in thinner sections. Simple modelling of the process has been carried out based on power input to the work varying with the angle of inclination of the leading edge of the kerf. This angle was assumed to be given by beam spot diameter over work piece thickness i.e. all the beam was absorbed on the leading edge. Using existing data for the variation of absorption with angle, and its scaling with temperature, the power absorbed could be calculated. This was set in balance against that required for melting the kerf material and supplying the heat conduction loss. The predicted cutting speeds are in good agreement with the experimental measurements for stainless steel (Figure 9); discrepancies in the case of mild steel are believed to be due to vaporisation effects not yet included in the model.

Table 1. Cutting conditions of the AEA trials.

Laser wavelength	5μm and 10.6μm nominal (CO and CO_2 laser respectively)
Laser power	(60±2)W at either wavelength
Focusing lens	plano-convex ZnSe, A/R coated for 5μm, focal length 50mm and 100mm
Optical aperture	f/3.6 and f/7.2
Focus position	nominally on cutting surface
Gas assist	dry air
Gas assist pressure	(4.5±0.1) bar-g
Nozzle	conical, 1.6mm diameter circular exit
Nozzle stand-off distance	(1.0±0.1)mm above cutting surface

Table 2. Focusing details of DLR cutting trials.

	Focal spot size (μm)	max. power density (Wcm^{-2})	med. power density (Wcm^{-2})
CO laser f=63mm	252	1.6×10^6	7.5×10^5
CO_2 laser f=127mm	240	1.6×10^6	8.0×10^5

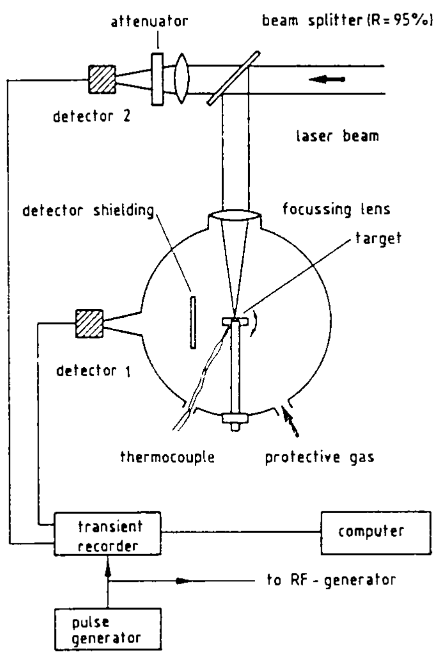


Figure 1. The integrating sphere equipment used by DLR.

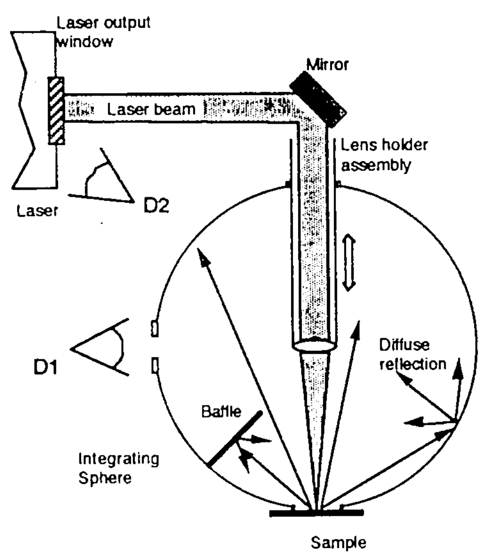


Figure 2. The integrating sphere used by AEA.

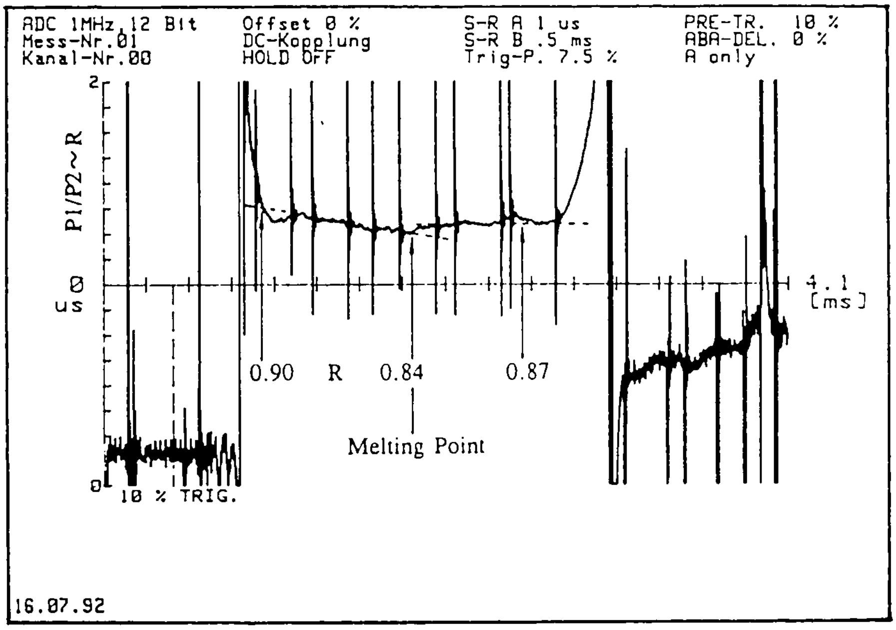


Figure 3. DLR stainless steel reflectivity measurement results.

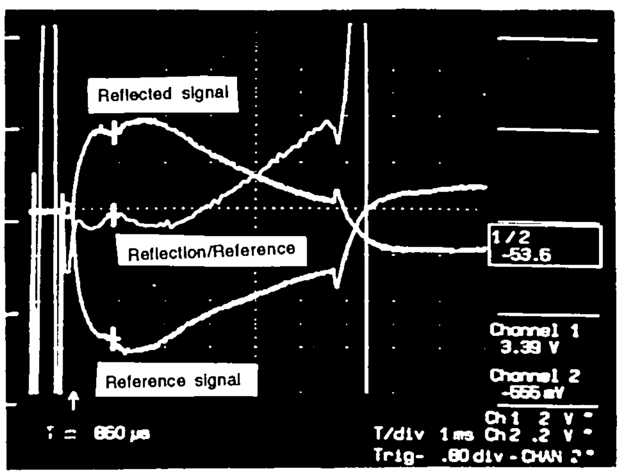


Figure 4. AEA stainless steel reflectivity measurement results.

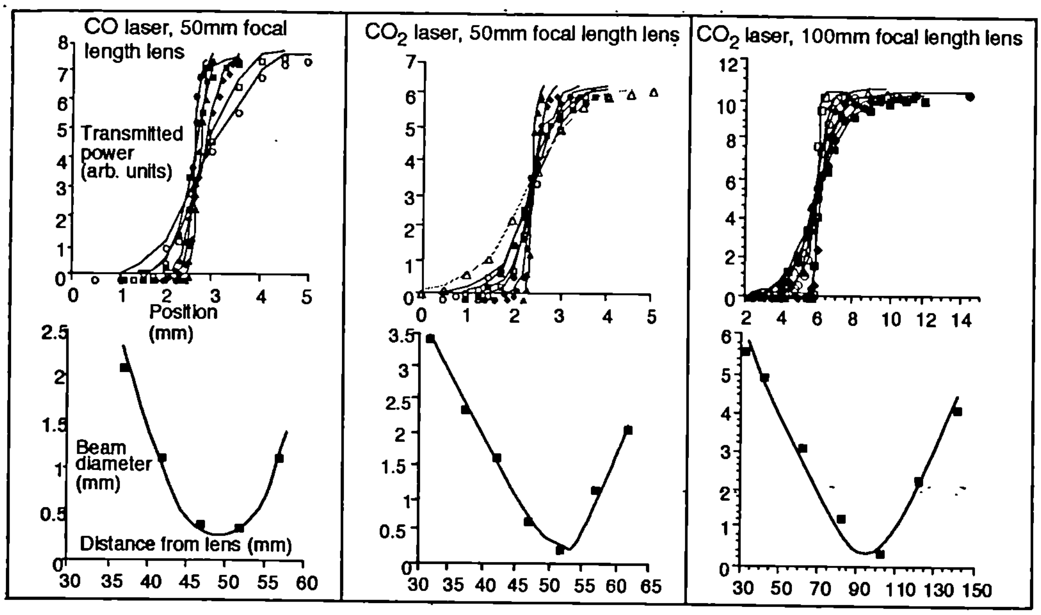


Figure 5. Spot size measurements (e^{-2} power contour) of CO and CO_2 laser configurations at AEA (the transmitted power units of the top graphs are arbitrary and cannot be compared).

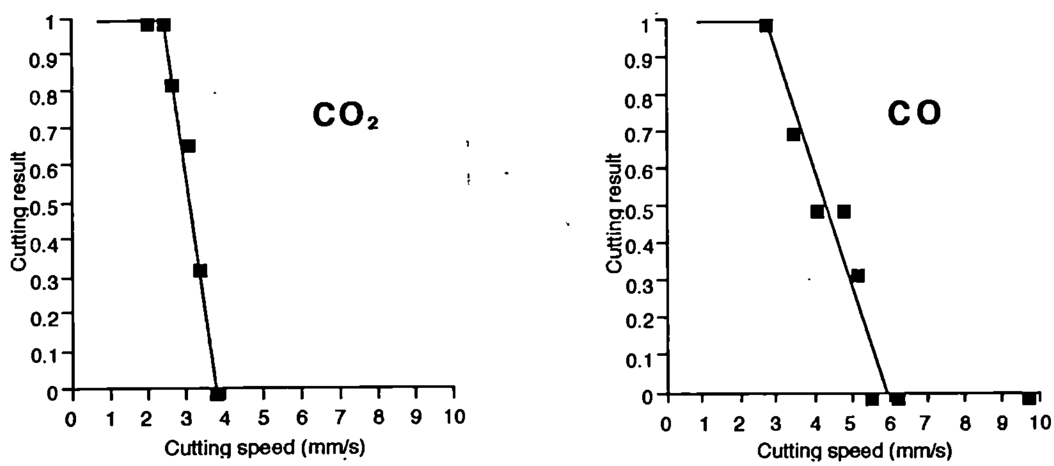


Figure 6. Cutting results from AEA for silicon nitride.

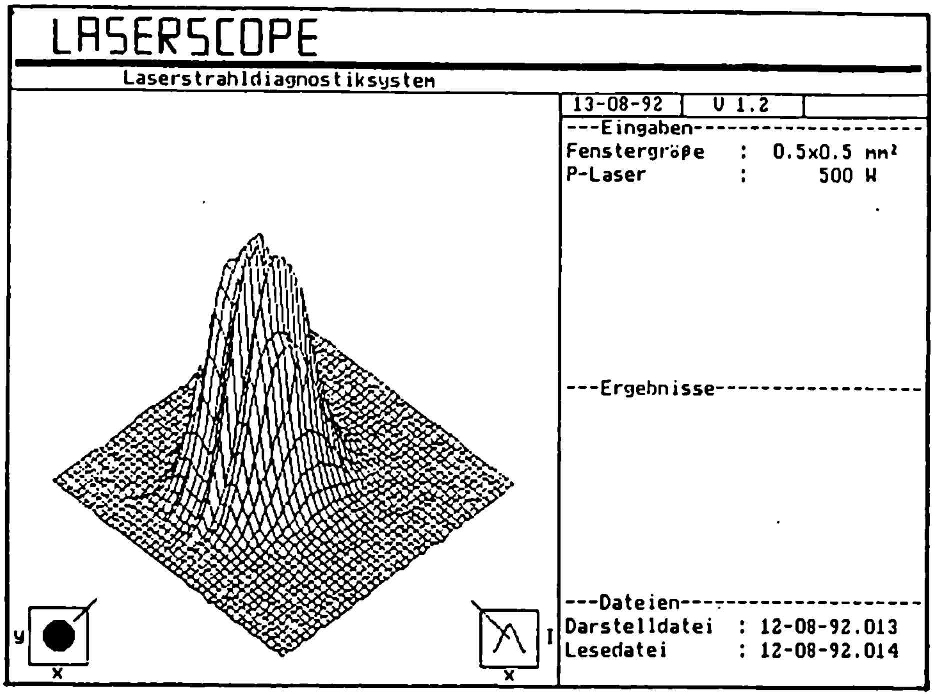


Figure 7. DLR focal spot intensity profile, measured with a PROMETEC beam profiler.

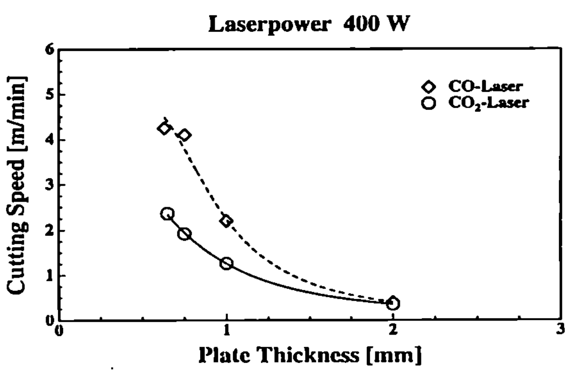


Figure 8. DLR cutting experiments on mild steel with nitrogen.

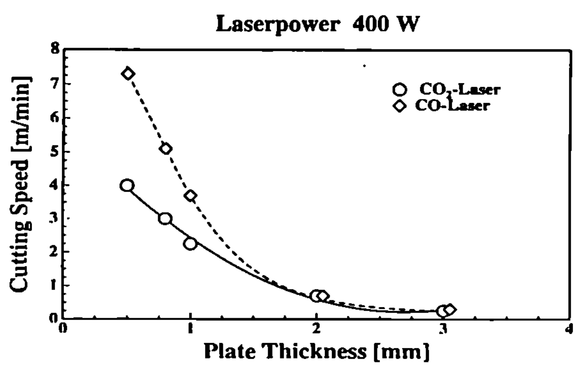


Figure 9. DLR cutting experiments on stainless steel with nitrogen.

3.9. CUTTING OF CO_2 PRIMARY CIRCUIT PIPES OF G2/G3 REACTORS USING EXPLOSIVE CHARGES

Contractors:	CEA Valrhô, COMEX, EPC
Contract No.:	FI2D-0036
Work Period:	September 1990 - December 1993
Coordinator:	Ch LORIN, CEA/DCC/UDIN
	Phone: 33/66 79 63 04 Fax: 33/66 79 64 32

A. OBJECTIVE AND SCOPE

The project is part of the dismantling of the primary cooling circuit (CO_2) of the G2/G3 gas-graphite reactors, composed of valves, blowers and tubes (diameters 800, 1200 and 1600 mm with respective thicknesses of 10, 15 and 20 mm), using explosive charges. It includes technical studies, experimental investigations and tests at industrial scale, carried out under real radioactive conditions.

The innovation of this project is the use of shock waves, produced by the explosive charges, to remove the inside contaminated oxide layer of the tubes.

The use of explosive charges seems beneficial because allowing to cut remotely large activated or contaminated items under improved protection and safety conditions for workers and with a minimum of secondary waste arisings.

The data output will mainly be related to:

- the necessary time to carry out dismantling operations using explosive charges and their respective costs,
- the safety and radiation exposure of personnel involved in the operations,
- the effectiveness of shock waves for decontamination purposes.

B. WORK PROGRAMME

B.1. Assessment of basic cutting parameters

B.1.1. Definition of cutting power of dihedral-shaped charges (EPC)

B.1.2. Establishment of the agreement files (All)

B.1.3. Preliminary test series on flat steel plates (EPC)

B.1.4. Calculation of the minimum quantity of explosive for each thickness (EPC)

B.2. Pre-test series with bounded steel samples (simulating tube sections) (EPC)

B.3. Definition of pyrotechnic devices (EPC, COMEX)

B.4. Detailed engineering study of validation tests

B.4.1. General assessment of the test conditions (COMEX, EPC).

B.4.2. Definition and design of auxiliary equipment required during cutting operations (COMEX, EPC)

B.4.3. Selection of representative items to be cut (All)

B.5. Validation tests on G2/G3 tubes

B.5.1. Definition of test procedure as needed for agreement by authority (CEA)

B.5.2. Preparation of the test area (All)

B.5.3. Validation tests: 27 cutting operations on 800, 1200 and 1600 diameter tubes) (All)

B.6. Final evaluation of all relevant data collected, e.g. specific data on costs, radioactive job doses, working time and secondary waste arisings (All).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

Cold trials using optimized cutting fuses have successfully been carried out; they have enabled the confining screen to be specified and tested as well.

First discussions with the French safety authorities have led to the further obligation to prove the resistance ability of the installations neighbouring the test area. With this in view, the test cell and the auxiliary equipment have been chosen and their resistance to shock wave calculated. The test equipment will be manufactured while expecting the hot test authorization.

Progress and results

Detailed engineering study of validation tests (B.4)

General assessment studies (B.4.1)

Figure 1 shows the environment in which the operations will be performed. The room giving access to the test cell is located in the basement of G2 reactor, the test cell is located under a road. The crane of the reactor main building will be used to transport the pipes.

The concrete structures calculations show that the stress must not exceed 0.2 bar. So, the hot tests will start using small quantities of explosive in order to measure the real response of the confining screen. The explosive quantity not to be exceeded will then be determined.

Auxiliary equipment (B.4.2)

Figure 2 shows the equipment which will be required during the operations. The crane and the reinforced door have been specified and are going to be manufactured. The blasting equipment was manufactured and its resistance tested in open air with up to 600 g of explosive.

Selection of representative items to be cut (B.4.3)

Because of the concrete structure resistance, the explosive quantity is restricted to about 200 g per shot. In order to limit the number of shots per pipe, the hot tests will be carried out with only the smallest pipes (i.e. 800 mm diameter), so that each cutting fuse will be cut 1/6 of the circumference and more than 20 shots will be necessary to decontaminate a pipe (1.3 m in length) by means of the shock wave.

The maximum contamination level was evaluated at 600 Bq/cm^2 (Co-60: 95%, Cs-137: 5%).

Validation tests on G2/G3 tubes (B.5)

Definition of test procedure as needed for agreement by authorities (B.5.1)

The corresponding safety files are in the achievement phase.

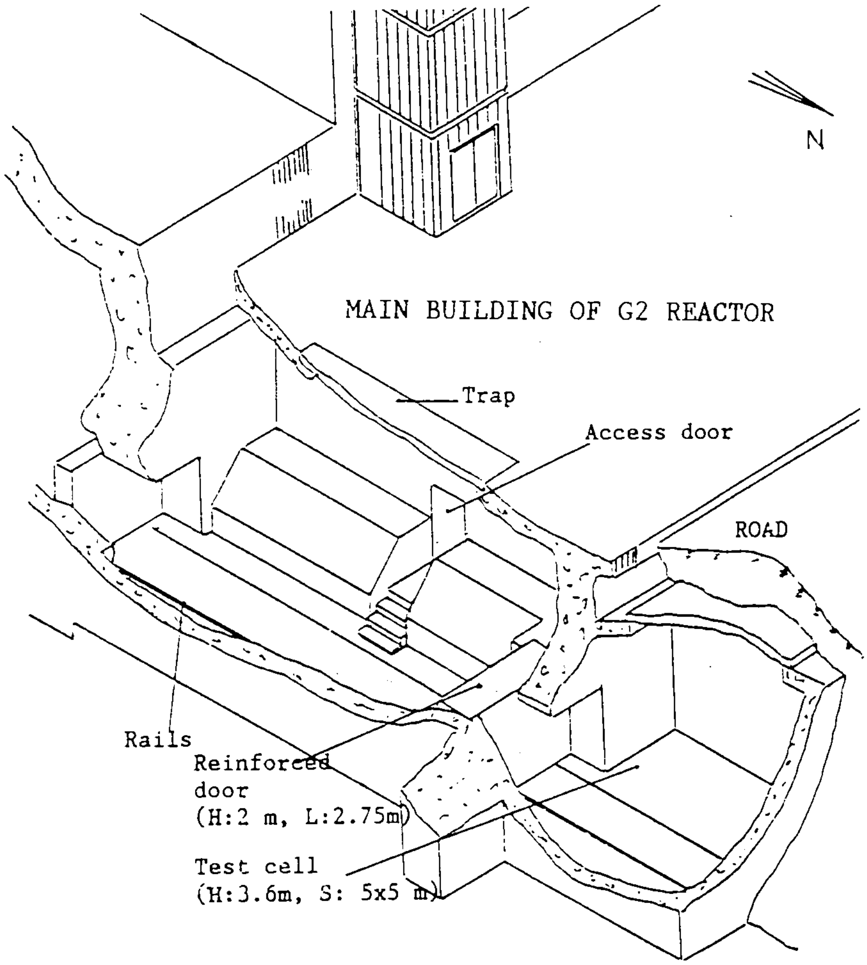


Figure 1 : Test area

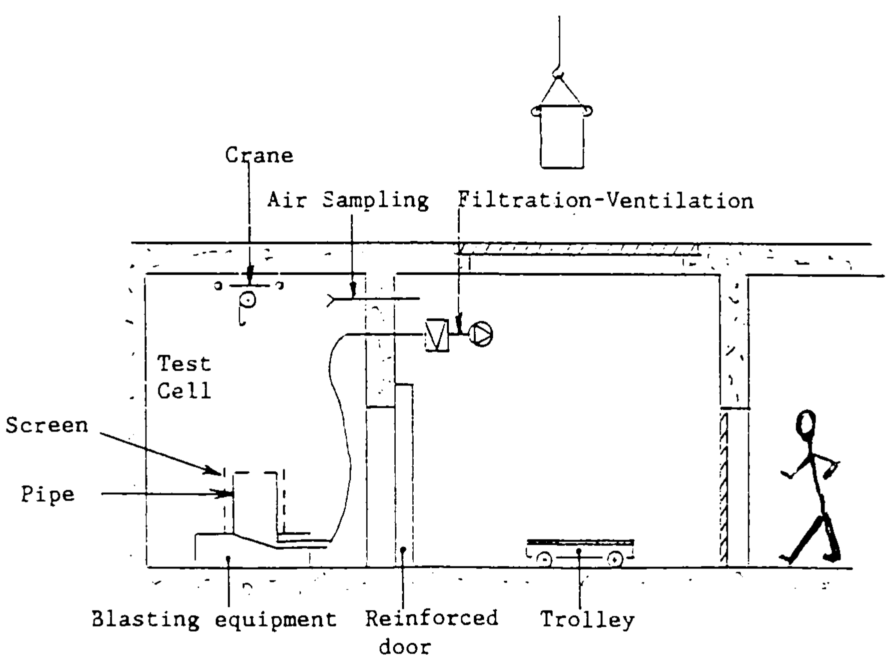


Figure 2: Main apparatus

3.10. UNDERWATER LASER CUTTING OF METAL STRUCTURES

Contractors: CEA-FAR, Radius
Contract No.: FI2D-0047
Work Period: January 1991 - March 1993
Coordinator: C CHARISSOUX
Phone: 33/1/69 08 62 87 Fax: 33/1/69 08 75 97

A. OBJECTIVE AND SCOPE

The feasibility of underwater cutting with a CO_2 laser has been demonstrated under 0.5 m of water, using a pressurized oxygene jet to eliminate the water between focusing nozzle and the piece to be cut. The laser beam can therefore interact with the piece without obstruction. The research work aims at demonstrating this technique under 10 m of water with a view to its application for PWRs.

The work includes the determination and optimisation of the relevant cutting parameters (e.g. power, max. cutting depth, cutting speed) under various water depth on non-active stainless steel components and the establishment of a data base specific to underwater cutting.

CEA developed laser systems for dismantling tasks in the framework of the previous Community research programme (contract FI1D-0013). Radius Engineering is working on powerful laser systems (up to 5 kW).

As laser cutting gives very small and proper kerves, a substantial reduction of swarfs and aerosols can be expected compared with other thermal cutting techniques.

B. WORK PROGRAMME

B.1. Conception of an underwater laser head able to cut up to 10 mm of stainless steel and being easily replaceable

B.2. Manufacturing of a 3 kW CO_2 laser head specified in B.1.

B.3. Mechanical and optical testing of the laser head in air up to 1.5 kW with subsequent conceptual adaptations, if any.

B.4. Manufacturing of the experimental device including water basin of 10 m depth and aerosol recuperation

B.5. Functional underwater cutting tests (little water depth)

B.6. Cutting under 10 m water (same programme as in B.5.)

B.7. Development of the remote system to control the alignment between laser head and piece to be cut.

B.8. Computer-assisted optimisation tests with respect to main cutting parameters (e.g. laser power, cutting speed, gas pressure and quantity, kerf width, effluent generation) for stainless steel plates of 5 to 40 mm thickness

B.9. Evaluation of effluents' generation with respect to the B.8. tests

B.10 Establishment of a specification document for the laser system as well as for the cutting technique

B.11 Evaluation of the safety, costs and radiological impact of the technique including cost of equipment and cost per one meter of cut work.

C. PROGRESS OF WORK & RESULTS OBTAINED

SUMMARY OF MAIN RESULTS

Following the remote cutting tests, realized with the 500 W laser (in 1991), the 1992 work programme concerned the cutting tests in air and under 0.5 m of water, using a 5kW CO_2 laser.
The laser power was fixed at 2.5 kW at the workpiece, which enabled to cut plates corresponding in thickness to the majority of the parts considered for dismantling applications (stainless steel 316, thickness > 10 mm).
These tests have shown that the kerf width is smaller under water than in air. On the other hand, the maximum cutting speed under water is about 40 % lower than the cutting speed in air. Some preliminary experiments under 7 m of water seem to show a better quality of the cut surfaces compared to the preceding results obtained under 0.5 m water.

1.B5 Functional underwater cutting tests

To explore the range of thicker cuts (>10 mm) a CO_2 laser of 5 kW has been optically connected with the experimental site. Fig. 1 shows the beam delivery system including two folding mirrors to bring the beam inside the underwater cutting head.

2. Cut parameters

This series of tests was oriented to a comparison between cuts realized under 0.5 m water and in air, all other parameters being identical.
The laser power was set at 2.5 kW at the workpiece, which enabled to study cut thicknesses typical for dismantling applications.
Three thicknesses of stainless steel (10, 20 and 30 mm) were cut under water (0.5 m) and in air. Table 1 specifies the cut parameters.

Lasertype	Laser 5 kW
Laser power at workpiece (W)	2500
Focal length (mm)	250
Focus position relative to the workpiece (mm)	+ 4.5
Nozzle geometry (mm)	0.8 x 4
Oxygen pressure (bar)	4
Oxygen mass flow rate (l/mn)	80
Distance between nozzle and workpiece (mm)	1
Thickness of sample (mm)	10, 20, 30

Table 1 : Cut parameters, Laser 5kW

3. Comparison of cut quality, underwater and in air

For these trials, the curves of Fig. 2 present the maximum cutting speed as a function of sample thickness. At each particular value, a cutting speed is found which is 40 % lower in the underwater case than in air case. Qualitatively, the generated burrs show a spherical form for the underwater cuts, having maximum diametre of 2 to 3 mm.

4. Visual aspects of the cuts

In Fig.3 pictures are given of the cut surfaces. As can be seen, there is little difference between the cuts under water compared to the cuts in air.
The pictures of Fig.4 and Fig.5 present the width of the kerf, respectively from the top side and from the back side. Table 2 summarises the obtained results.

	In air			underwater		
Sample number	1	2	3	4	5	6
Thickness (mm)	10	20	30	10	20	30
Cutting speed (m/mn)	0.7	0.2	0.08	0.3	0.09	0.03
Kerf width top side (mm)	0.6	0.8	0.8	0.7	0.7	0.7
Kerf width back side (mm)	1.3	1.5	1.8	1.3	0.8	1

Table 2 : Results obtained at 2.5 kW

The top side of the kerf shows a nearly constant width (0.7 mm) for each thickness, independent of the cutting circumstances (ambient vs. water). On the other hand, the kerf width at the back side is of the order of 1.5 mm for the air cuts and of the order of 1 mm for the underwater cuts. In this way, one can expect a reduction in aerosols and splatter generation with the underwater technique.

5. Preliminary cutting tests under 7 m of water

At high oxygen pressures (> 2 bars), problems arose related to the water column. The small section of the column (Ø 70 mm) produced the creation of water bursts which went upside in the column, together with the oxygen. It showed necessary to install an expansion chamber at the upstream side of the column to trap the bursts and protect the measurement equipment for the aerosols (Fig. 1). Moreover, above 7 m of water, the problem was not resolved. Therefore, we will limit the water layer to a thickness of 7 m instead of 10 m as was foreseen originally. Preliminary tests under 7 m of water will be conducted on samples with 20 mm thickness (steel 316). These trials seems to indicate a better cut quality compared to the earlier tests under 0.5 m water (Fig. 6). A test programme is currently under execution to confirm these preliminary conclusions and to further characterize the quantity and the nature of the splatters and the aerosols.

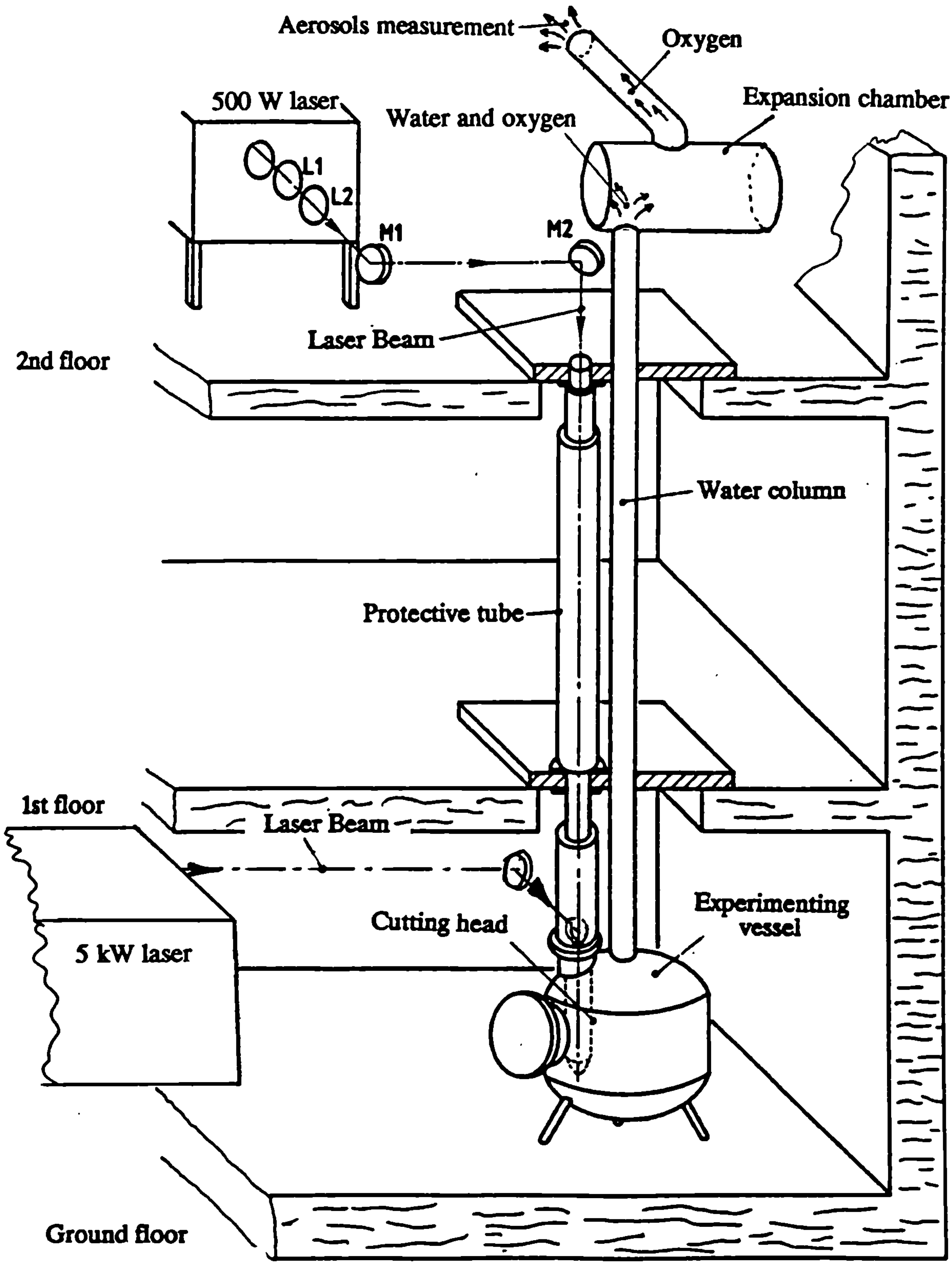


Fig. 1 : General presentation of the system

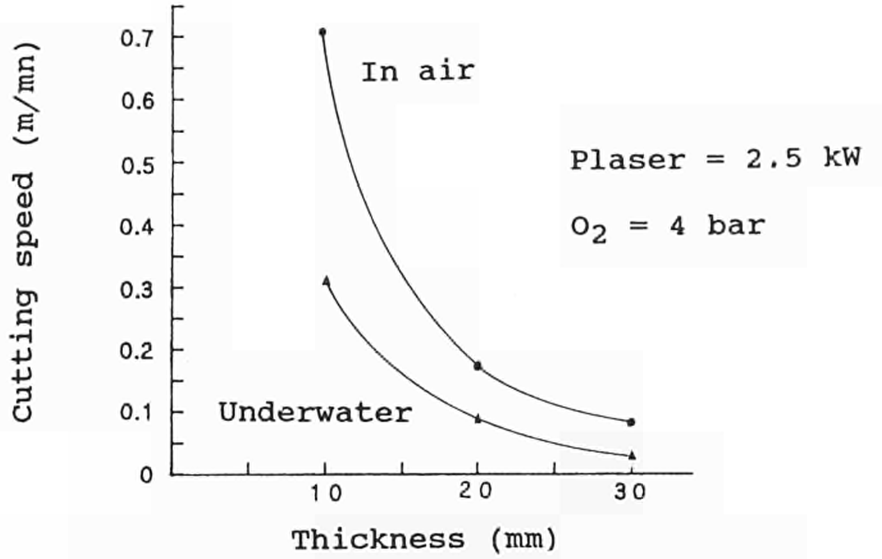


Fig. 2 : Cutting speed vs. thickness

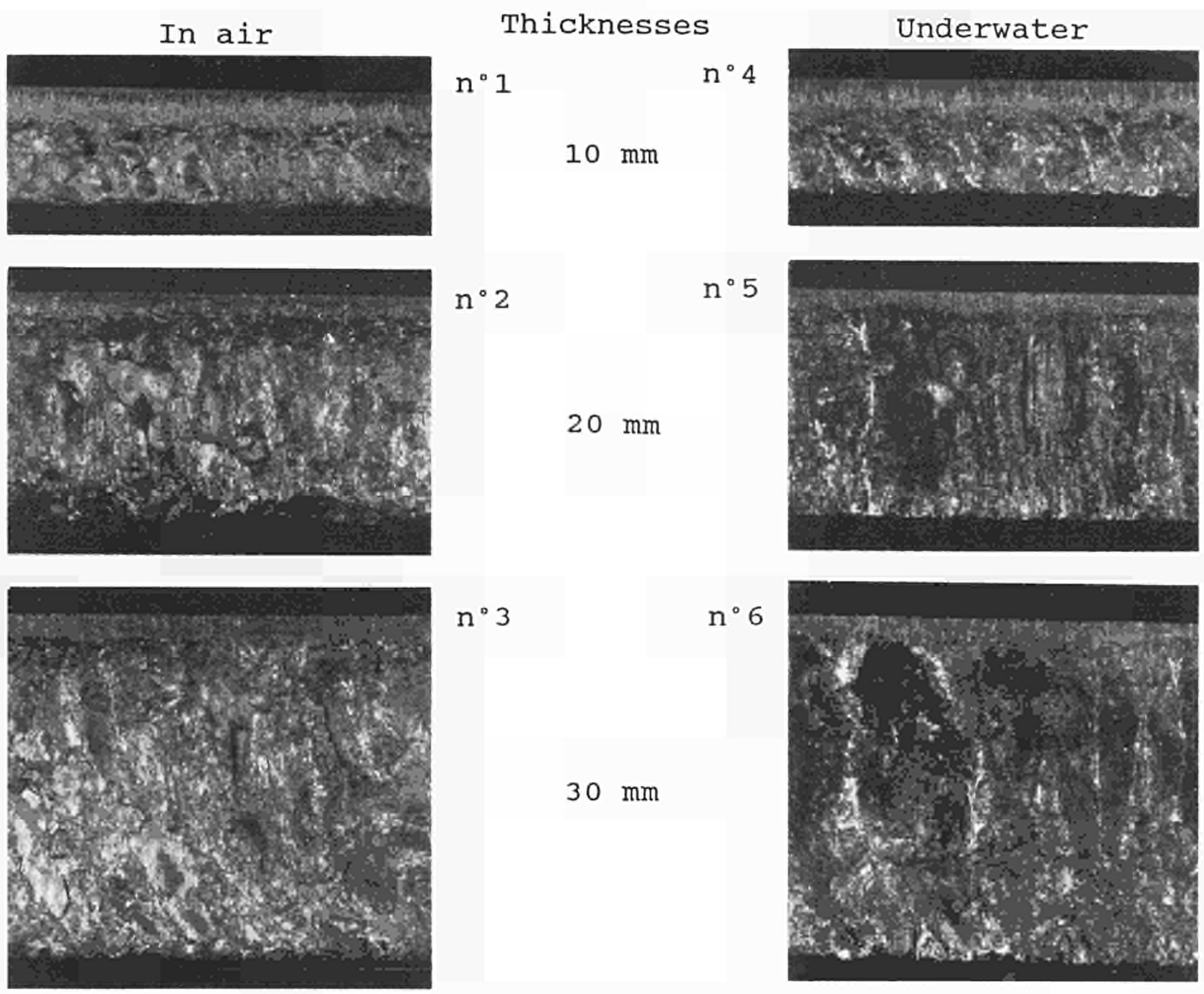


Fig. 3 : Appearance of CO_2 laser cutting

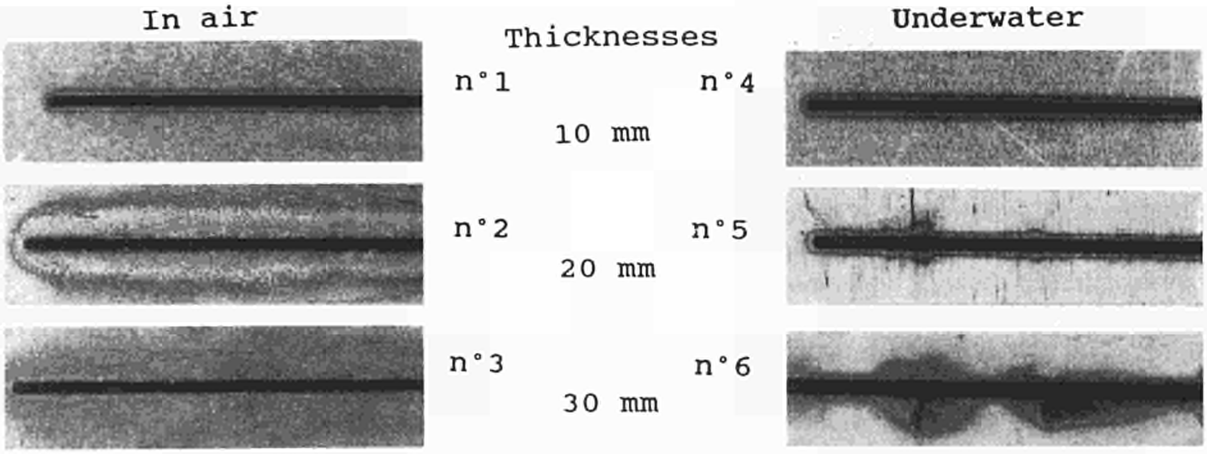


Fig. 4 : Top side appearance of CO_2 laser cutting

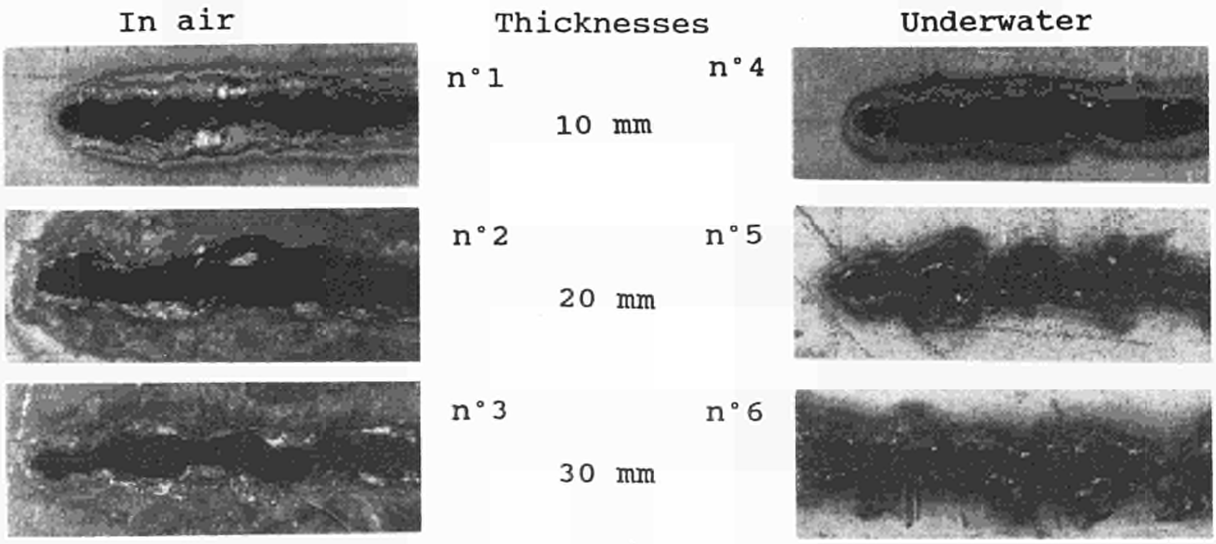
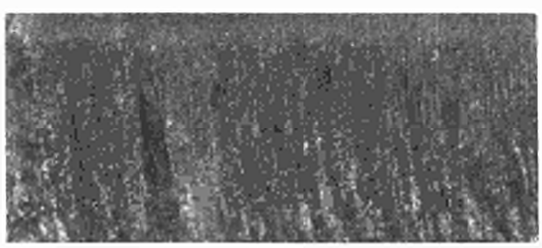


Fig. 5 : Back side appearance of CO_2 laser cutting



Thickness:20mm

Plaser = 2.5 kW

O_2 = 4 bar

Fig. 6 : Appearance of CO_2 laser cutting under 7 meter water

3.11. OPTIMISATION OF PLASMA TORCH ELECTRODE DESIGN FOR NUCLEAR DISMANTLING TASKS

Contractors: AEA Culham, ARC, Goodwin, TWI
Contract No.: FI2D-0049
Work Period: April 1991 - March 1993
Coordinator: R S G PARKER, AEA Culham
Phone: 44/235/463 685 Fax: 44/235/464 300

A. OBJECTIVE AND SCOPE

The plasma arc cutting torch has major role to play in the dismantling of nuclear facilities due to its advantages compared with other cutting techniques. However, it does have one serious drawback which is the comparatively short life of the plasma torch electrode. The replacement of the used electrode could have major cost and safety implications when the torch is operating remotely in a radioactive environment. The main objective of this project is therefore to improve the life of the electrode by at least a factor of 2, so as to reduce the cost and occupational radiation exposure when using this technique.

Novel plasma torch electrode designs and materials will be assessed, plasma modelling for the optimisation of electrode characteristics applied, the novel electrodes manufactured and their performance tested on typical nuclear dismantling tasks. Recommendations from the result of this project will be made, which will allow European plasma torch manufacturers to retain a lead in the world market.

Experience obtained using plasma arc cutting for decommissioning of the Windscale Advanced Gas-Cooled Reactor, and the work being performed at the "Institut für Werkstoffkunde der Universität Hannover" will be taken into account.

B. WORK PROGRAMME

B.1. Literature survey to obtain current information on plasma arc torch design, with particular reference to electrode life enhancement.

B.2. Requirements study to identify the necessary or desirable properties and features of the electrode material and the scope of the theoretical work to be carried out.

B.3. Identification of potential electrode materials with specialists of the Harwell Laboratory to select the materials to be included into the test programme.

B.4. Plasma arc process modelling and design of electrode/nozzle.

B.5. Testing existing methods and electrodes in order to allow for comparison with the tests performed on the new electrodes (B.6.).

B.6. Manufacture and testing of selected new electrodes to determine their performance, and comparison with those on the existing electrodes (B.5.).

B.7. Review of test results on the existing and new electrodes, comparison with the theoretical work and modifications to the electrodes to further improve their performance.

B.8. Manufacture and testing of revised electrodes as defined in B.7..

B.9. Final evaluation of new electrode designs with respect to the potential benefits to the dismantling of nuclear facilities, including specific data on costs, waste arisings, working time and related radiation exposure of workforce and working area.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The identification of potential electrode materials has been completed (B.3.)

The plasma arc process modelling has been modified by agreement among the collaborators (B.4.).

For the testing of existing methods and electrodes, a database of customers experience with Hafnium electrodes is still being compiled (B.5.).

As regards the manufacture and testing of new electrodes, field tests on ceramic sprayed nozzles are currently being undertaken. Ru + Y_2O_3 electrodes have been received and are ready for test. ZrB_2 electrodes have been ordered. The ordering of doped Hf electrodes is imminent. (B.6.).

Progress and results

Identification of potential electrode materials (B.3.)

Further literature search, the opinions the the collaborators and discussions with representatives from the Advanced Materials and Theoretical Studies departments at AERE Harwell produced a new list of candidate electrode materials. The electrode materials selected for further study are as follows:

a/ Zirconium Diboride, ZrB_2, the subject of an expired patent for this application.

b/ Refractory metals doped with rare earth oxides, namely Hafnium and Lanthana, Hf + La_2O_3; Hafnium and Ceria, Hf + Ce_2O_3; Ruthenium and Yttria, Ru + Y_2O_3. Lanthana should improve electrode life and make for easier arc starting, Ceria and Yttria should improve electron emission.

c/ Ceramic spraying the electrode holder to inhibit the formation of a crater at the insert/electrode holder interface, which results in electrode erosion and eventual failure.

Machined inserts of TiB_2 and LaB_6 were considered to be unsuitable for use above 1500°C. Lanthanium Chromite, Lanthanium Strontium Chromite, Depleted Urania, Molybdenum Disilicide, Zirconia, SiC and Magnesite were less favoured materials that were discussed but not considered as potential candidate test materials.

Suppliers of the materials identified in a/ and b/ have been found and manufacturing routes have been established.

Plasma arc process modelling and electrode design (B.4.)

A survey of previous plasma modelling studies was conducted and gave the following conclusions:

a/ With the limited time and resources available, it would not be possible to complete a full study of plasma-electrode processes.

b/ A two-dimensional analysis of the thermal contours of the cutting head would be possible but the output from any analytical work would be of no practical value unless it was closely allied to experimental and operational observations.

c/ There is a lack of reliable data on the novel materials; heat transfer properties on the Cu/Hf interface are not available and there seems to be no practical solution to the removal of oxide from the Hf insert.

The collaborators agreed that it appeared more cost-effective to carry out experimental modifications to the nozzle/electrode holder design and carry out tests. It was felt however, that a theoretical justification for any improvements found would be valuable so thermal modelling may follow laboratory trials of successful modifications.

Improved cooling was seen as the most important feature to investigate, features to investigate were:

a/ Protrusion length of electrode into coolant

b/ Thermal contact resistance between electrode and holder

c/ Thermal ratcheting of the electrode/holder and its effect on b/.

A check using hand calculations has shown that thermal ratcheting does not occur in either the electrode or the holder. The power generated at the Hf, the Cu holder and the Hf/Cu interface due to ohmic heating is low, less than 1 watt in each case.

A limited number of tests with the Hafnium insert protruding into the coolant to improve the electrode cooling have been completed. The results from about ten tests showed a reduced electrode life, though this is not conclusive as poorer Hafnium to copper contact may have been a contributory factor.

Testing existing electrodes (B.5.)

Laboratory testing of the Hafnium electrodes has been carried out by Arc Kinetics using plasma welding equipment to establish a controlled basis for comparison with new electrodes. However, it is felt that these trials were unrepresentative of real operating conditions and field test results are to be the final basis of comparison. A database of customers experience with Hafnium electrodes is continuing to be compiled by Goodwin Air Plasma.

Manufacture and testing of new electrodes (B.6.)

The collaborators had agreed that the planned standard laboratory tests were not a good simulation of the field cutting conditions, so field tests were being done to short list the various designs and laboratory testing would be done on the most promising candidate designs. This was the reverse of the planned procedure but it was agreed that it might produce sensible results more quickly. Results from the first field tests are expected in March 1993.

a/ Some trials with ceramic sprayed nozzles have been completed. No problems in arc striking were found and although there had been some flaking of the ceramic, preliminary tests indicated that erosion of the copper was being inhibited at the Cu/Hf interface. At present, there are insufficient results from which to make any firm conclusions. Field tests are continuing.

b/ A small number of sample Ru + 20% Y_2O_3 electrodes have been obtained. These have been fitted into the Goodwin electrode holders and trials are to be carried out as far as possible with the limited number of electrodes available.

c/ A supplier of Zirconium Diboride ZrB_2 has been found and an order of sample electrodes has been placed. Delivery is expected at the end of April 1993.

d/ It has been found that Goodwin Air Plasma's Hf suppliers had been conducting work, independently, on doped Hf electrodes. La_2O_3 doped electrodes were reported to have oxidized too rapidly, CeO_2 doped electrodes were reported to be brittle and difficult to fabricate.

The general opinion of the collaborators is that these results could not be considered as definitive and that the planned programme of work on novel materials should continue. The placing of an order for the manufacture of sample Hf + La_2O_3 and Hf + CeO_2 electrodes is imminent. Delivery is expected in mid-May 1993.

4. AREA No. 4: TREATMENT OF SPECIFIC WASTE MATERIALS: STEEL, CONCRETE AND GRAPHITE

A. Objective

In the dismantling of nuclear installations, large amounts of radioactive metal, concrete and - in the case of gas-cooled reactors - graphite will arise. This waste must be suitably conditioned for disposal or recycling. The area has been strictly delimited to preclude overlapping with the Community research programme on radioactive waste management.

B. Subjects of the research performed under the previous programmes (1979-88)

Research work performed mainly related to:

- the treatment of dismantled material such as steel, copper and brass by melting with a view to its possible recycling/reuse; the reduction of its volume; its decontamination (e.g. elimination of actinides);
- development and assessment of techniques for coating metal and concrete parts in order to immobilise surface contamination; assessment of treatment techniques for radioactive concrete;
- comparative assessment of various modes of treatment and disposal of radioactive graphite; development of a conditioning technique for radioactive graphite bricks for shallow land disposal.

In all these investigations, due attention has been paid to the necessity of adapting treatment techniques to final waste destinations.

C. Programme 1989 to 1993

Melting of very low-level radioactive steel scrap from Light Water Reactor components, to produce new nuclear components, is already becoming industrial practice and is not expected to need further research. Further work is required, however, in relation to steel scrap originating from other types of nuclear installation, e.g. alpha-contaminated material, and non-ferrous scrap.

Further development is also needed for concrete and graphite waste, i.e.:

- volume reduction of contaminated/activated concrete;
- metallic coating of graphite parts by ionic deposition to fix radionuclides;
- recycling of the reinforcement steel in concrete.

D. Programme implementation

Five research contracts relating to Area No. 4 were in progress during 1992, from which one was completed in the same year.

4.1. INDUSTRIAL-SCALE MELTING OF TRITIUM-CONTAINING STEEL FROM NUCLEAR INSTALLATIONS

Contractors:	SG, NIS
Contract No.:	FI2D-0014
Work Period:	July 1990 - May 1993
Coordinator:	D HOLLAND, SG
	Phone: 49/2151/894-290 Fax: 49/2151/894-345

A. OBJECTIVE AND SCOPE

The objective and scope is to subject steel scrap, coming from decommissioning, to (among others) a special tritium removal treatment in order to reduce the amount of waste which has to be disposed.

The proposed work consists in trapping the tritium released from scrap during the heating and melting process in a specially adapted exhaust system of the melting facility. For this project the already existing facility has to be modified, adapted and tested. The estimated steel quantities for the treatment amount to ca. 18 Mg with exhaust gas streams of about 5,000 m^3/h.

B. WORK PROGRAMME

B.1. Material choice: Laboratory evaluations on samples coming from parts of the NPP Niederaichbach will provide a representative charge of material. (all)
B.2. Radiological measurements: which consists in choosing the tritium elimination device in accordance with the measurement techniques. (all)
B.3. Modification of the existing plant with integration of the new components. (SG)
B.4. Determination of tritium release during heating up at different temperature steps. (SG)
B.5. Evaluation of the released tritium activity. (all)
B.6. Documentation of the collected results. (all)

C. Progress of Work and Results

Summary of Main Issues

In 1992, thirteen Tritium gas bottles were chosen for the experiment with a total weight of 750 kg. As primary information it was said that the activity level should be far below 200 Bq/g so that a direct use in the Siempelkamp foundry - CARLA plant - would have been possible. Control tests showed that the average activity was about 1,000 times higher than expected. To solve this problem, it was decided to add small amounts of this highly active material to not contaminated scrap to obtain an average activity level of approx. 200 Bq/g. In parallel, 18 Mg of Tritium contaminated waste was gathered. Besides Tritium, other nuclides such as Cobalt 60, Iron 55, Nickel 63 and Caesium 137 are contributing to the average activity of 35 Bq/g. Tests were made to calibrate the equipment.

The results of these tests showed that there were big differences between the test in the laboratory and tests in four different institutions, which were involved to make sure that the results were true.

Due to this problem, it was not obvious for the authorities that the accepted level for Tritium containing radioactive material in the Siempelkamp plant was kept. Special papers had to be made and special licenses to be granted to be able to melt the Tritium contaminated steel scrap. At the end of the year, all difficulties had been solved so that in the beginning of 1993 the melting campaign could be planned.

B.2. Radiological Measurements

The results of the tests made by different laboratories have been collected by NIS, Niederaichbach. The determination of Tritium in the material samples was as follows:

- material sample no. 1, including an oxide layer	820 Bq/g
- material sample no. 2, without oxide layer	770 Bq/g
- material sample no. 3, oxide layer removed by grinding	440 Bq/g
- material sample no. 4, oxide layer removed by brushing	below detection limit

These results are approx. 10 times higher than the results of the other samples which were taken from KfA, Jülich.

A test at KKN laboratory showed that the cladding of the heat treatment furnace was hygroscopic. Due to this fact, the scheduled test procedure was changed as follows; the heat treatment furnace was brought to the proposed operational temperature and the sample was put into the preheated furnace.

For the test melts, this procedure was chosen and the basic results of the tests were as shown below. Initial activity level of all samples was approx. 3,000 Bq.

200°C

- determined as $HT0/T_20$, detected	267 Bq
- possible as HT/T_2, detected up to the detection limit of the Tritium monitor	2,200 Bq
- residual activity after heat-treatment	2,800 Bq/g.

700°C

- detected as $HT0/T_20$	238 Bq
- possible release of HT/T_2 up to the detection limit of the Tritium monitor	2,200 Bq
- residual activity after heat-treatment	1,900 Bq

1,100°C

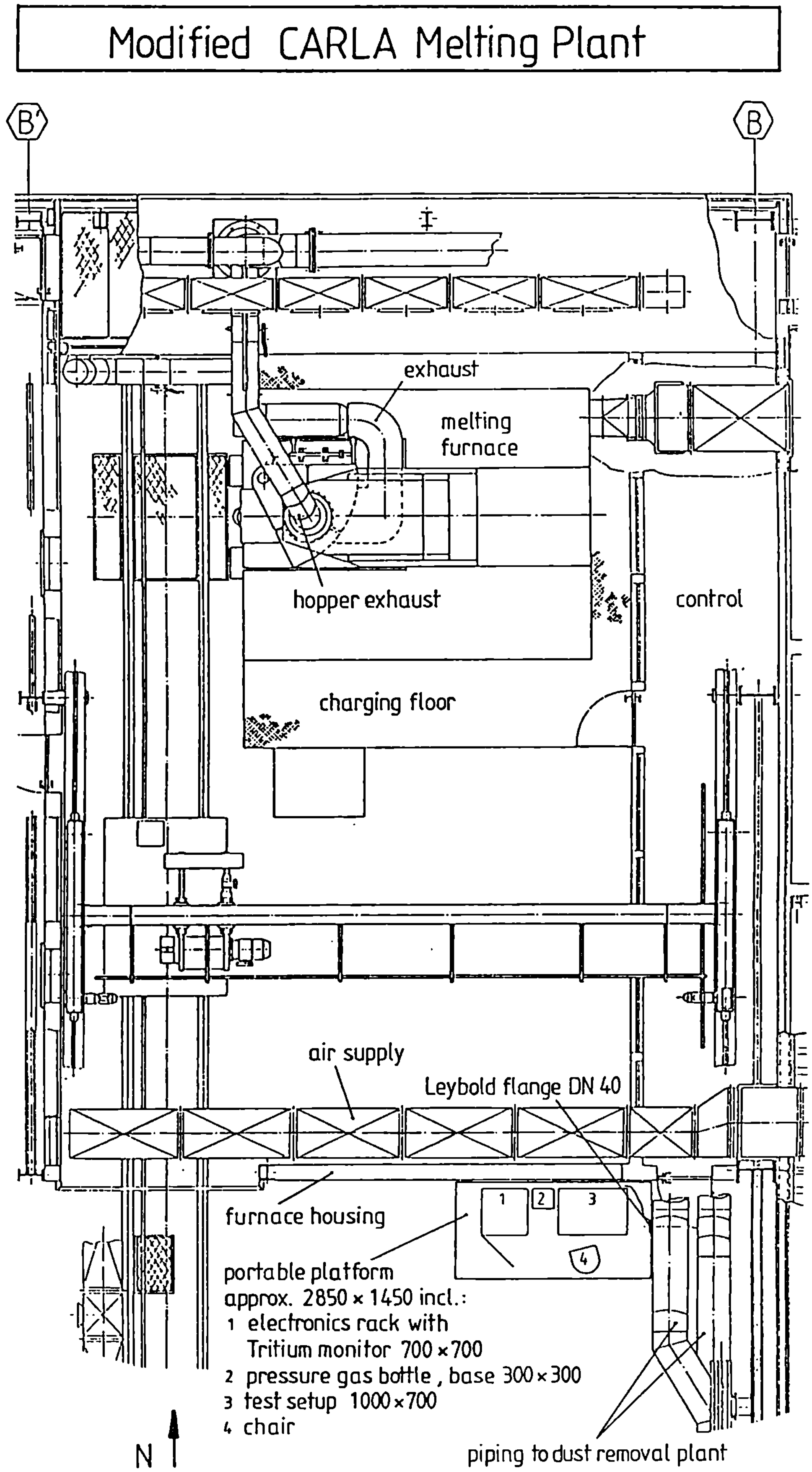
- detected as HT/T_20	235 Bq
- possible HT/T_2 release up to the detection limit of the Tritium monitor	2,200 Bq
- residual activity after heat treatment	1,800 Bq

The problem why the results of the test plants did not fit to the proposed results lied in the fact that the chosen test facility was not suitable for this special type of measurements. An other step was necessary to obtain comparable results at different laboratories and with different ways of heating the steel scrap.

B.3. Modifying the Existing Plant

After having chosen the measurement equipment and testing it, the components for the measurement device to be integrated into the existing SGR-plant were chosen. Figures 1 and 2 show the test area.

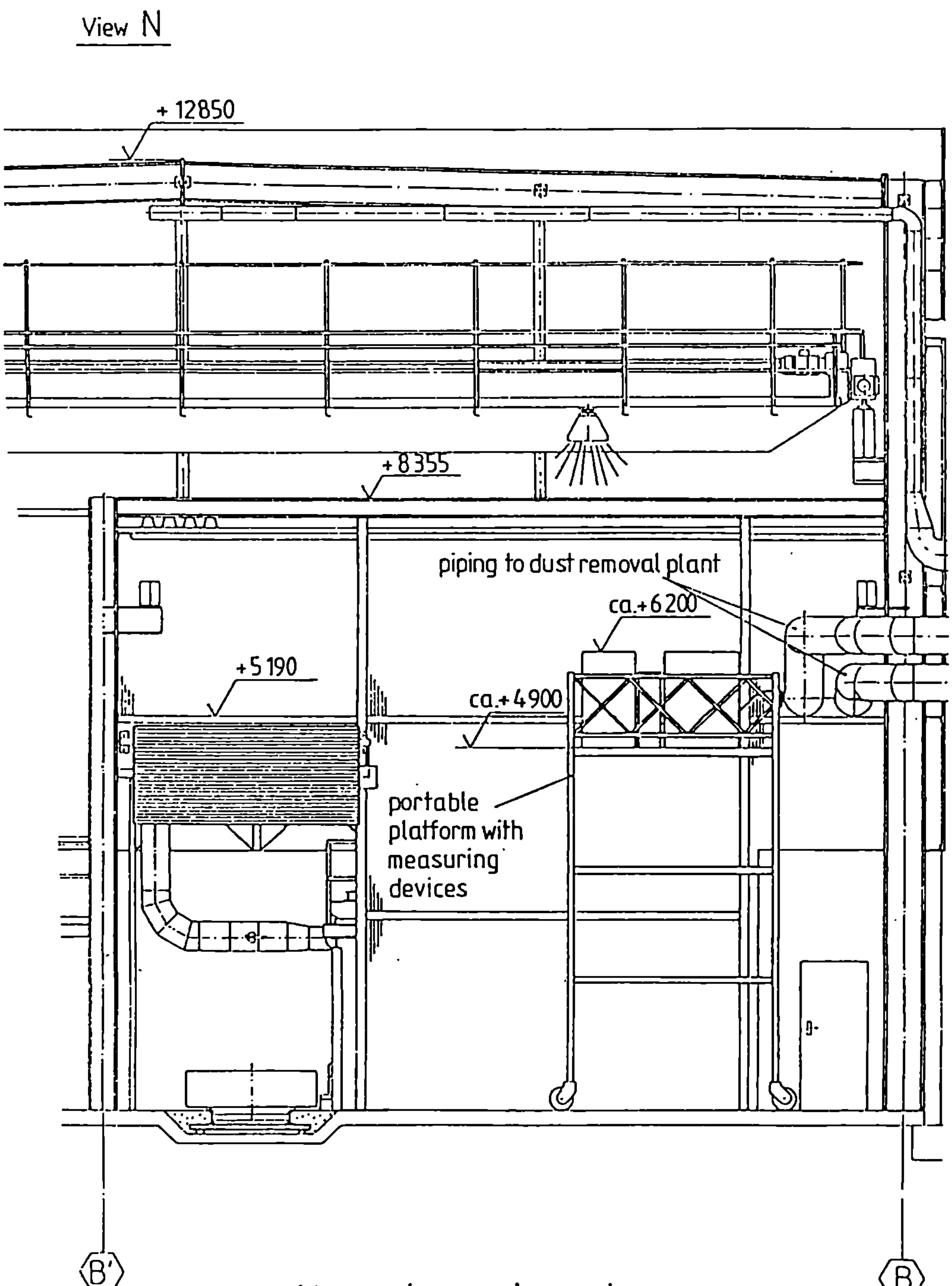
7 Mg of uncontaminated scrap have been spread on 26 drums and are ready for the bypass tests at Siempelkamp. The material will become contaminated for the melting test with 200 g Tritium contaminated material of the gas bottles per drum. To be sure of the Tritium activity of the bottle material with which the scrap to be melted was contaminated, parallel samples with a mass of 10 g of the sample material were sent to NIS, so that parallel results were obtainable from an external laboratory.



Modified CARLA Melting Plant
B'
B
exhaust
melting furnace
hopper exhaust
control
charging floor
air supply
Leybold flange DN 40
furnace housing
1
2
3
4
portable platform
approx. 2850 × 1450 incl.:
1 electronics rack with
Tritium monitor 700 × 700
2 pressure gas bottle, base 300 × 300
3 test setup 1000 × 700
4 chair
N
piping to dust removal plant

Figure 1.

Modified CARLA Melting Plant



Measuring equipment
for CARLA - bypass test
(Tritium)

Figure 2.

4.2. DEVELOPMENT OF A PROCESS FOR VOLUME REDUCTION OF CONTAMINATED/ACTIVATED CONCRETE WASTE INCLUDING PILOT-SCALE TESTING WITH ACTIVE WASTE

Contractors:	KEMA, Taywood
Contract No.:	FI2D-0015
Work Period:	July 1990 - December 1993
Coordinator:	H A W CORNELISSEN, KEMA, Arnhem
	Phone: 31/85/56 61 04 Fax: 31/85/51 58 35

A. OBJECTIVE AND SCOPE

This work concerns the development of a semi-technical scale test installation for separation of concrete constituents. As only a relatively thin layer of concrete structures will be contaminated or activated, the proposed process consists in a further volume reduction of the material to dispose off by separation of the radioactive constituents (cementstone) from the supposed non-radioactive part (aggregates) of this removed concrete cover.

The material that will be conditioned originates from decommissioning activities at the Kahl nuclear power plant.

The research programme could be useful for developing an industrial-scale manufacturing process. Furthermore, the experience gained in this field by Taywood (CEC contract FI1D-0042) will be applied to solidification.

B. WORK PROGRAMME

B.1. Selection of a separation technique: determined by the importance of the activation/ contamination of the concrete constituents. (KEMA)
B.2. Determination of process variables for the conceptual design of the test installation. (KEMA)
B.3. Design of a small-scale transportable test installation. (KEMA)
B.4. Construction of the test installation. (KEMA)
B.5. Testing and optimisation of the installation with non-radioactive concrete. (KEMA)
B.6. Verification with radioactive concrete. (KEMA)
B.7. Immobilisation and solidification of concrete debris. (Taywood)
B.8. Evaluation of the results with respect to equipment, costs, released activity etc. (KEMA)

C. Progress of work and results obtained

Summary of main issues

In normal quality concretes, the volume of the porous cementstone is approximately 30%, while the remaining part consists of dense aggregates. Tests have shown that contamination primarily penetrates in the cementstone. Separation of the porous and dense components of concrete will therefore result in substantial volume reduction of radioactive waste. This is beneficial for economical and environmental reasons.

KEMA has developed, designed and constructed a pilot plant scale test installation to separate aggregates from contaminated concrete.

The first test-runs with the installation showed that some minor modifications proved to be necessary for dust-free operation. This mainly concerned the valve construction and operation of the process containers.

After that tests with no contaminated concrete were performed. It was concluded that by separation the original amount of "contaminated" concrete was reduced to 37% of the input material.

Progress and results

1. Test runs (B.5)

Based on a thermal treatment (up to about 600-700 °C) followed by milling and sieving, a separation installation was designed and constructed in the KEMA laboratories. A top view of this installation is shown in figure 1.

The first test-runs with the installation showed that some minor modifications proved to be necessary for dust-free operation. This mainly concerned the valve construction and operation of the process containers.

Well defined, not contaminated, concrete (150mm) cubes were made with maximum grain size of 31.5 mm. The 28 days compressive strength was about 40 N/mm^2. From the concrete composition and the sieve line of the quartz aggregate, the amount of cementstone and sand with grain size <1mm can be estimated. The findings are given in table I.

Table I: Data of Concrete Mix used for the Test-Runs

component	amount
1. Portland cement	320 kg/m^3
2. quartz sand <1mm	450 kg/m^3
3. formed cementstone (calculated)	400 kg/m^3
cementstone and sand <1mm (2+3)	850 kg/m^3
fine material ratio*	35%

* (850/2400)x100%

It was calculated that the amount of fine material <1mm was 35% of the total mass of concrete. The specific mass of the concrete was taken as 2400 kg/m^3.

Nine test-runs were executed, in which the concrete parts were subjected to temperatures in the range of 650-700 °C for 3 to 4.5 hours. The milling time was set between 1 and 2 hours. After sieving over 1 mm, the amount of "clean" material proved to be 63%, and the amount of "contaminated" residue was 37% (standard deviation 3%). Because of crushing of some aggregate particles during operation, this ratio of 37% is slightly higher than theoretical value as given in table I.

It can be concluded from the results that by separation the original amount of "contaminated" concrete was reduced to 37% of the input material.

The residue has to be conditioned as radwaste. As part of this research project (B.7.), methods for solidification are being studied by Taywood [1]. Important is that the volume is kept as low as possible. This seems to be feasible by the addition of hydration activators to the mixture of powdered cementstone and sand.

The test installation will be further optimized by tests with no contaminated concretes. In the summer of 1993 the installation will be shipped to an ongoing decommissioning project in Germany for verification tests with radioactive contaminated concrete (B.6.). The findings will be reported by the end of 1993. For the tests in Germany detailed work procedures were formulated and reported [2].

2. Cost benefits estimation (B.8.)

Operation of the test installation as described in this paper shows that volume reduction of contaminated concrete by separation is an interesting option. For large scale operations, however, modifications are necessary for instance in the direction of an integrated continuous process. The economical benefits are dependent on many factors such as costs of disposal of contaminated concrete and the definition of radwaste especially the corresponding release levels. In the discussions in Europe the release levels vary somewhere between 0.1 and 1,0 Bq/g.

Based on rough estimations a cost-benefit analyses was made for concrete separation installations having capacities of 1 and 2 m^3 contaminated concrete per day. The investment costs were calculated by extrapolation of the costs of the test-installation. The results of the analyses are presented in figure 2. It can be seen that after 2 to 5 years operation the benefits become dominant.

References

[1] JULL S., "Immobilisation of Active Concrete Debris generated from Decommissioning of Nuclear Power Stations". Taywood Report 1303/91/5782, 1991.

[2] PEEZE BINKHORST I.A.G.M., TALEV D.K., "Work Procedures for separation of Contaminated Concrete". Reportnumber 20250-KET/R&B 92-1114, 1992.

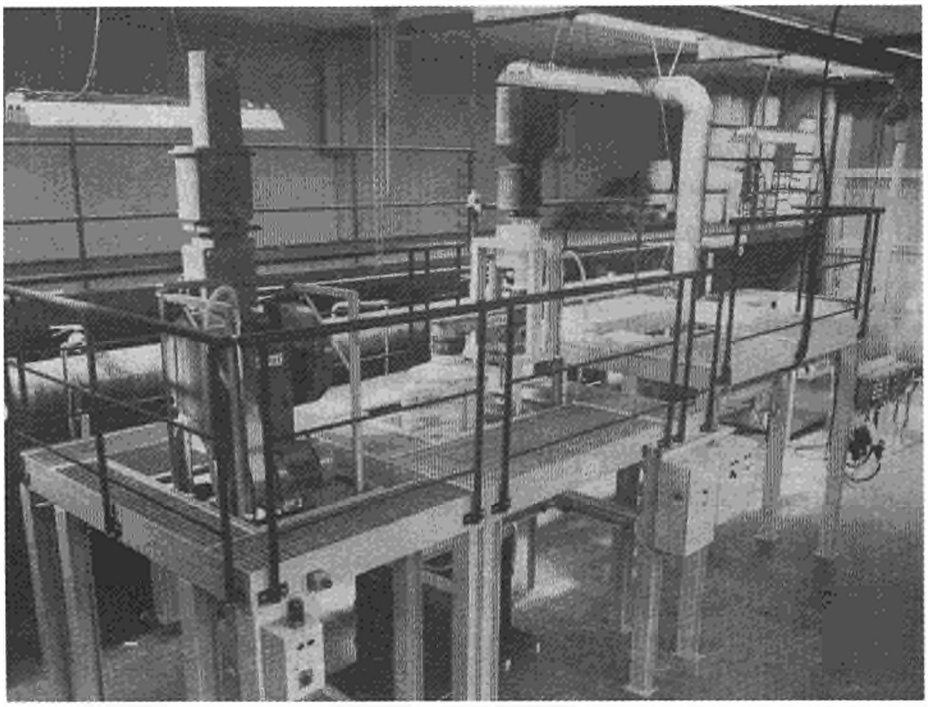


Figure 1: Top View of the KEMA Test Installation

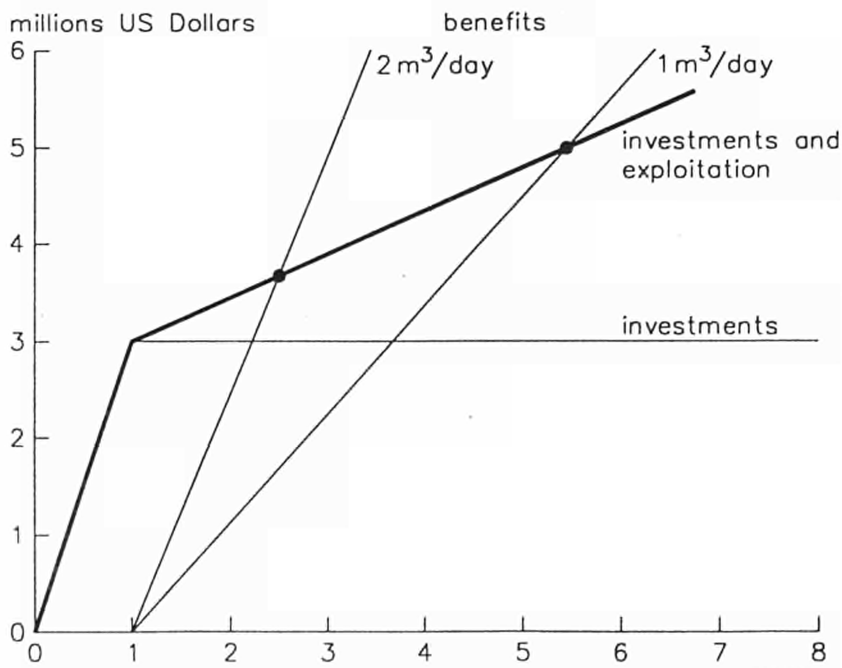


Figure 2: Cost Benefit Estimation of Concrete Separation Plants with Capacities of 1 m^3 and 2 m^3 Concrete

4.3. TREATMENT AND CONDITIONING OF RADIOACTIVE GRAPHITE FROM NUCLEAR INSTALLATIONS

Contractors: CIEMAT, UDA
Contract No.: FI2D-0017
Work Period: July 1990 - December 1993
Coordinator: A ESTEBAN DUQUE, CIEMAT, Madrid
Phone: 34/1/346 62 19 Fax: 34/1/346 60 05

A. OBJECTIVE AND SCOPE

The objective of the laboratory-scale investigations is the development of chemical processes for the treatment of radioactive graphite for its safe storage. It consists in:

- previous extraction of radionuclides (mainly tritium) to decrease the radioactivity of graphite;
- fixation of radionuclides (mainly C-14) to avoid their leaching during the storage of graphite;
- impermeabilisation of graphite by metal coating for its transport and storage.

Extraction of the radionuclides with chemical agents will be done before the metallising process for fixation, in order to minimise leaching of radioactive products during storage, followed by standard leaching tests. The radioactive graphite will be procured from the experimental reactor JEN-1 and the gas-cooled reactor Vandellos-I.

B. WORK PROGRAMME

B.1. Removal and/or fixation of radionuclides
B.1.1. Investigations on radioactive and inactive sample structure and texture using different analysing techniques. (all)
B.1.2. Testing of appropriate chemical agents on samples with regard to their possible decontamination and/or immobilisation features. (UDA)
B.1.3. Study of radionuclide removal, mainly titium.
B.1.4. Study of radionuclide fixation.
B.1.5. Characterisation of treated samples using methods from subtask B.1.1.
B.2. Metal coating of graphite by ionic deposition.
B.2.1. Characterisation of samples similar to B.1.1. (all)
B.2.2. Performance of process parameter studies for metal coating applications on inactive samples. (all)
B.2.3. Chemical modification of radioactive surfaces. (CIEMAT)
B.2.4. Metallisation of inactive samples. (all)
B.2.5. Metallisation of radioactive samples. (CIEMAT)
B.2.6. Characterisation of the treated samples concerning chemical properties and thickness of the metal layer, porosity of the surface etc. (all)
B.3. Leaching experiments with the metallised specimen. (CIEMAT)
B.4. Assessment of results and conclusions.

C. Progress of work and results obtained

Summary of main issues

In this period of time the study of the content and the behavior of volatile compounds has been continued.

The hydrocarbons contained in the Vandellos 1 reactor sleeves, have been quantified according to their desorption temperature and those which are extracted under 300 °C have been identified. Some of them coming from the environment are fixed on the surface of graphite.

The thermal desorption of hydrocarbons and the kinetic process, under and over the working temperature in the reactor, has been studied.

In general terms it has been found that organic compounds are desorbed at least up to 1000 °C if the process takes place in inert atmosphere. If it takes place in presence of oxygen the desorption finishes at 500 °C, when the graphite burns.

The absorption and desorption kinetic of water (the other molecule capable to leach tritium) from the graphite at different temperatures also has been studied.

The possible removal of tritium by adsorption and desorption of water in order to produce an isotopic exchange, has been simulated using unirradiated graphite collecting the water with silica gel.

The development of copper coating process was started last year. In order to eliminate the pores in the metallic layer, it was necessary to study the conditioning of graphite's surface.

In the nickel metallization, the stability of the bath and the influence of the additives in the covering rate and in the structure of the covering have been studied.

The characterization of the coating has been carried out using rugosimetry, electronic microscopy and optic microscopy of metalographic cuts.

Progress and results

1. Removal and/or fixation of radionuclides (B.1.)

The organic compounds contained in the non-irradiated graphite from the Vandellos-1 reactor sleeves, have been quantified according to their desorption temperature (figure 1) and those which are extracted under 300 °C have been identified (table 1). The desorption processes and their kinetic parameters have also been studied.

To identify the organic matter it was extracted using two methods: extraction with carbon sulfide and controlled thermal desorption.

The characterization of the products has been carried out by gas chromatography and mass spectrometry and the desorption studies by thermogravimetry (TG) and differential scanning calorimetry (DSC). The data processing has been done with a specific software for thermal analysis.

It has been found that some desorbed compounds come from the environment because they are concentrated in the surface. They have been found in the atmosphere and it has been proved that they are adsorbed by the graphite when it remains some time in an atmosphere which contains these products.

Over 300 °C they have been quantified but nor characterized due to the limitations of the equipment.

Up to 400 °C the existence of compounds have been detected which are desorbed at different temperatures due to their different boiling point, modified by capillarity effects according to the porosity on the surface of graphite. In the kinetic study (plotting the reaction rate constant versus temperature following the Arrhenius law) several desorption processes, which confirms this, can be seen (Figure 2).

From approximately 400 °C up to 830 °C a continuous desorption process has been detected which fits to a single kinetic model (Figure 3). The DSC confirms the existence of different processes under and over 400 °C.

Maybe there are molecular groups chemisorbed in the surface and not inside the pores, which could come out above that temperature. This would justify its kinetic behavior. If this groups existed they would be unsaturated and would tend to chemisorb tritium as the hidrocarbons desorbed round to 400 °C.

Over 830 °C there is a continuous loss of weight which fits to a kinetic process of zero order. It has been found that it is not an oxidation process due to the entrance of air, nor to a systematic error of the equipment, but it can be justified in no way.

The absorption and desorption kinetic of water at different temperatures has been studied. By orientation, at 25 °C the desorption is 9000 quicker than the adsorption.

The removal of tritium by water adsorption with posterior desorption and adsorption in silica gel has been simulated, with unirradiated graphite.

2. Metallic coating of graphite by ionic deposition (B.2)

The influence of several variables in the deposition rate of the copper coating has been studied: metallization time, temperature and surface activation time. The deposition rate, after a short starting time, remains constant through all the process, is proportional to the temperature and increases when already the graphite is reactivated.

In the beginning it was found that there were failures in the initiation of the metallic process and that the coated samples had many pores. The first problem was due to the absence of reactive points in the graphite surface. It was solved with a previous treatment of the surface. Seventeen different compounds have been used: organic and inorganic with different polarity and different functional groups to obtain different solubility effects and oxidation potentials. The influence of the surface treatment upon the amount of deposited catalyst and its surface distribution has been studied (figure 4). The second problem has not been solved yet and is in relation with the kind of deposit (crystalline or amorphous). Nevertheless promising results have been obtained with the conditioning of the surface.

The metallization with nickel has been studied: The influence of additives in the bath versus the covering rate, structure of the covering stability of the bath. The following reducers were used: sodium hypophosphite, hidracine and borohydrure. And as additives: none, acetone, thiourea, ammonium molybdate and lead nitrate. In general the samples covered with nickel are compact and amorphous.

The metallized samples have been characterized by rugosimetry, electronic microscopy and optic microscopy of metallographic cuts. The kind, size, morphology and distribution of the crystal metal has been studied.

3. Leaching experiments (B.3)

The radioisotopes leaching tests have been started with graphite from Vandellos obtained in the wet way. It has been used uncovered graphite and concrete covered graphite. The ISO-6391 leaching is being observed with deionized water as leaching liquid. The test will last for about a year. At present 100 days have elapsed, so that only partial results have been obtained. The radioisotopes from 10 gaseous and liquid samples have been analyzed.

More test will be carried out using graphite obtained in the dry way and with metallized graphite.

Table I. Detected organic compounds

COMPOUND	B.P. (°C)	COMPOUND	B.P. (°C)
Benzene	80	Methyl-decane	196
Toluene	110	Dodecane	216
Xylene	139	Naphtalene	217
Pyrene	156	Trichlorobenzene	219
Propyl-benzene	159	a-terpineol	221
Butyl-butyrate	165	Tridecane	234
Alkyl-benzene	< 170	Methyl-naphtalene	244
Decane	174	Tetradecane	252
Trimethyl-benzene	175	Pentadecane	270
Limonene	177	Hexadecane	28

Table II. Kinetic data of volatiles desorption in graphite powder

Range calculated (°C)	Kinetic data		% weight
25 to 400°C	-	-	~ 0.4
400 to 830°C Range 0.1-0.8	n	0.52	~ 0.7
	ln K_o	2.18	
	E_a	76.34<	
830 to 1000°C	-	-	~ 0.3
Isothermic desorption 20-50 min 1000°C Range 0.1-0.8	n	- 7.48	~ 2
	ln k	-0.04	

Kinetic equation: $\frac{d\alpha}{dt} = K_o e^{-\frac{E_a}{RT}} (1-\alpha)^n$

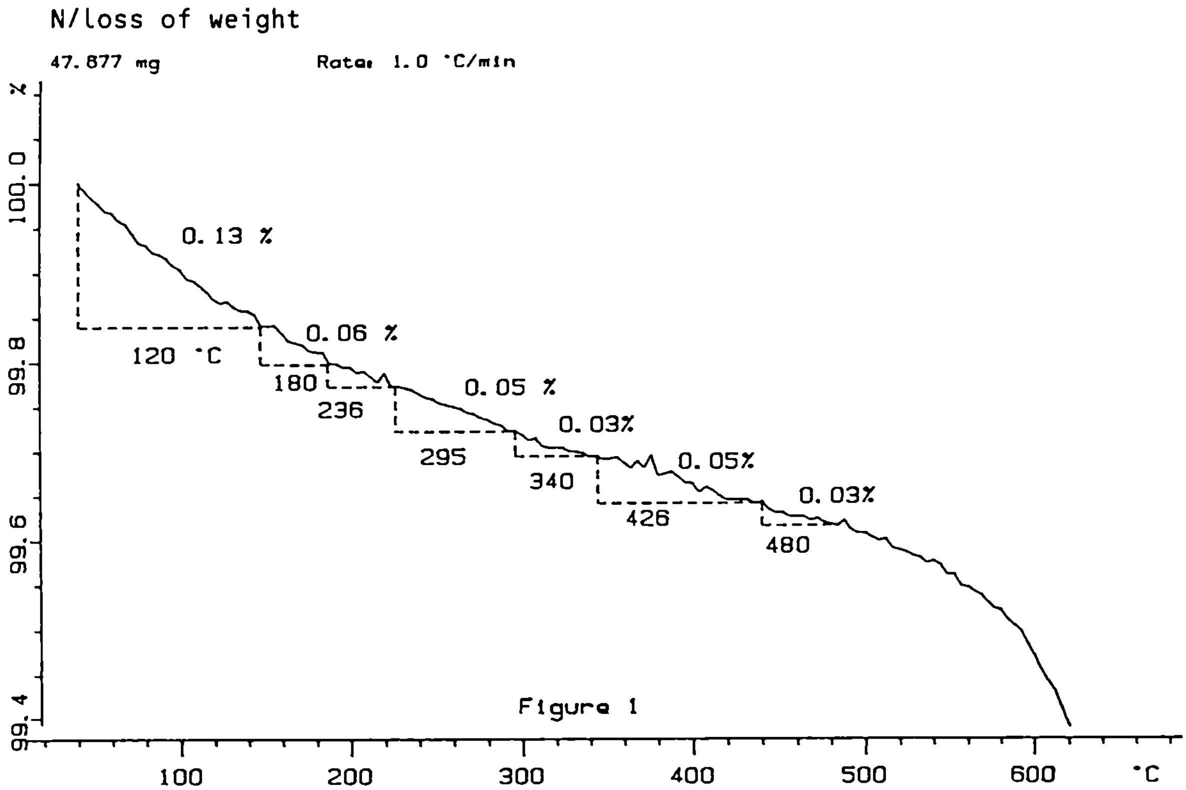


Figure 1

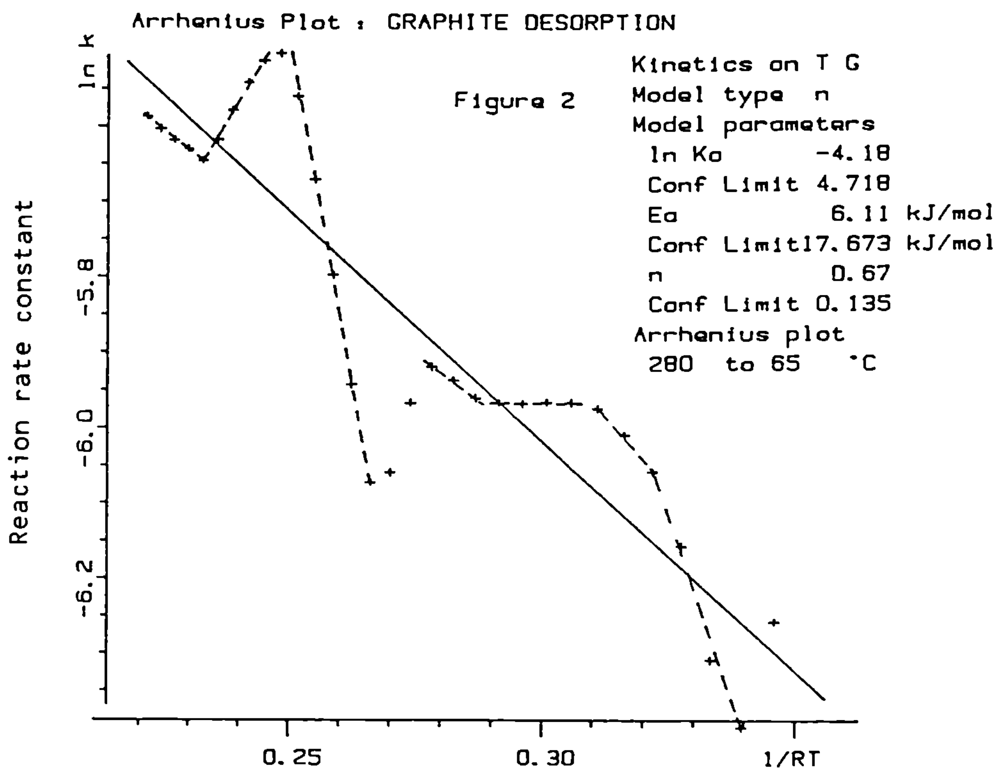
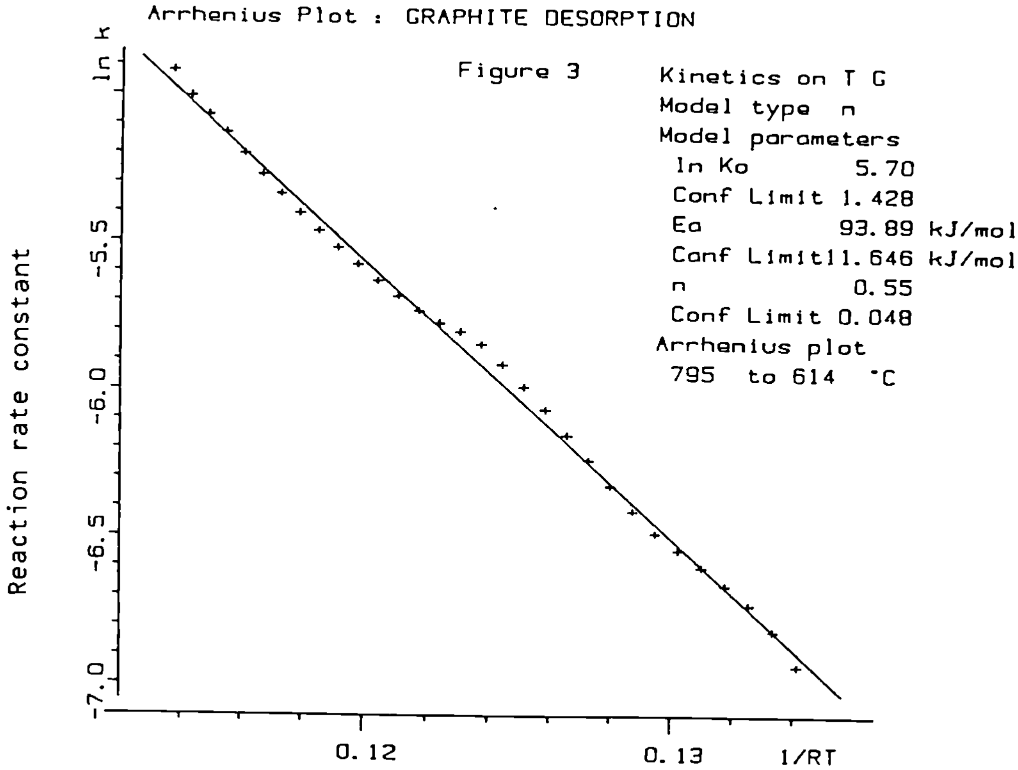


Figure 2



SURFACE TREATMENT
SENSITIZATION WITH NICKEL CHLORIDE

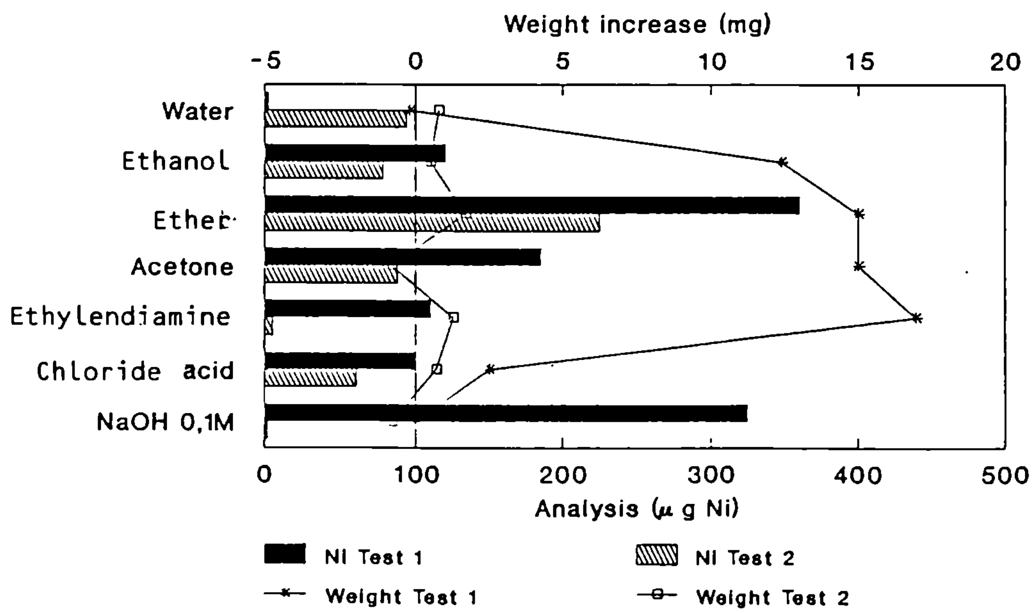


Figure 4

4.4. RECYCLING OF ACTIVATED/CONTAMINATED REINFORCEMENT METAL IN CONCRETE

Contractors: Bureau A+
Contract No.: FI2D-0021
Work Period: September 1990 - June 1992
Coordinator: H H KOOLEN, Bureau A+
Phone: 31/47/50 17 400 Fax: 31/47/50 33 264

A. OBJECTIVE AND SCOPE

A large part of activated or contaminated steel and copper arising from decommissioning of nuclear installations could be recycled, as aggregate or reinforcement in concrete for new nuclear installations. The object of the study is:

1) choosing the type, amount and form of the metals to be used;
2) analysing the possible process to transform the metal into smaller particles and producing high grade concrete;
3) finding out the possible applications of different concrete qualities within the field of nuclear applications.

The first part will be a literature review, the second part will consist in laboratory experiments with non-radioactive metals, and the third part will be a desk study.

During the study, specific data about the process costs will be estimated. This research programme has a strong relationship with the melting technique developed by SG (FI1D-0016 and 0059) and could have interactions with the separation technique studies by TNO/KEMA (FI1D-0068).

B. WORK PROGRAMME

B.1. Literature study on metal waste types, quantities and activation/contamination levels in order to select potential processes for waste transformation.

B.2. Conduction of a specific test programme on combinations of different metals and metal forms with concrete and mortars.

B.3. Evaluation of the results and survey of possible applications.

C. Progress of work and obtained results

Summary of main issues

During the year 1992 all 3 tasks were finished and as a result the final report was made. For task B1 the desk research for steel quantities and production techniques was finished and also a literature study for the amounts of concrete used in the building of nuclear power plants was performed. For task B2 the testprogram by doing some tests on shrinkage and creep for the most promissing steel/concrete mixture was completed. Also some calculations on the radiological behaviour of this type of concrete was performed.

During the above mentioned period also a feasibility study on the production and application of concrete with 100% sand replacement by steel granules was conducted.

In the following the activities and the results of all 3 tasks are specified.

Progress and results

1. Marketing study (B.1.)

In this part of the research-program a desk and marketing study is carried out in which the amounts, types and kinds of radioactivity of these metals, used in nuclear installations, are examined.

Furthermore an inventory of possible processes to convert metals into smaller particles, (which can be added to concrete in order to make a construction quality concrete) was finished.

Also information about the amounts of concrete used in the construction of nuclear powerplants was gained.

The literature revealed that the amounts of contaminated steel are depending on factors such as type of reactor, time after shutdown, operation time etc. It was not possible to gain exact figures on the amounts of steel per type of reactor, therefore literature based assumptions were made. During decommissioning the following amounts of steel are set free:

- for a PWR about 3,000 t;
- for a GCR type 5 about 7,000 t;
- for a GCR type 4 about 3,000 t;
- for a BWR about 4,500 t.

These assumptions are concerning large commercial nuclear powerplants in the EC. From operational reactors also about 30 tonnes of contaminated steel per year per reactor will be set free.

Further assumptions are:

- operational period of a powerplant : 30 years;
- start decommissioning : 5 years after shut down;
- decommissioning period 5 years

Based on these assumptions it can be concluded that the amounts of steel arising in the future from existing large commercial powerplants in the EC ranges from 3,500 up to 40,000 t/a. See figure 1.

In the first semester of 92 also some additional information was gathered on techniques in order to produce steel particles such as fibres and granules. Also new kinds of fibres were found. Unfortunately it was in this phase of the testprogram not possible to perform some tests on these fibres.

The conclusions which can be drawn from this study are:

- The melting technique of activated steel is well developed, including controlling the exhaust products and safety measures.

- Also the production of small granules up to 8 mm is a well known process with low costs.
Besides melting, the production of milled fibres or small dish-formed granules is possible. This technology however is not yet developed sufficiently. As for melting the activity level of the steel will not be reduced. Therefore in this technique, only steelscrap with a low activity level should be used.

In order to get some information on the amounts of concrete which are used for the building of a nuclear powerplant a literature study was performed. The best application of concrete with contaminated steelscrap is to use it where it gets activated or contaminated anyway. The amounts of concrete which get activated or contaminated during operation of an NPP varies between 10,000 t and 45,000 t.

2. Research on metal-concrete-composites (B.2.)

In order to get some information on long term deformation of the concrete made according to the (most promising) recipe of the previous testprogram, tests for shrinkage and creep were performed during the first semester of 92. This recipe in which we replaced 100% of the sand by steel granules was tested according to the Dutch concrete standard. Compared to blanc concrete (B25) shrinkage and creep are substantial better. Compared to concrete with the same compressive strength (B55) the creep is still substantial better, the shrinkage is about the expected value for B55 concrete.

During the last period calculations on the radiological behaviour of concrete with contaminated steel granules were also performed, i.e. on the shielding capacity and the exposure rate. The more sand is replaced by steel granules the better the shielding capacity. See figure 2.
Concerning the exposure rate of concrete with contaminated steel granules, calculations were performed with a contamination of ^{60}Co and ^{137}Cs and a source strength of 1 Bq/g and 5 Bq/g. The results showed that for walls thicker than 20 cm the exposure rate does not increase anymore. See figure 3.

3. Feasibility study (B.3.)

From task B1 and B2 it was concluded that concrete with the addition of steel granules (100% sandreplacement) is the most promising combination. On this combination we conducted our feasibility study using the capital value analysis model. This model is an existing method for which a software version was developed by our company.

The starting points for this study are:
- the calculations have been made for an imaginary company which includes the activities melting and production of steelgranules, situated anywhere in Europe and combined with a "mobile" concrete factory situated on the building site of a new nuclear power-station.
- the supply of the contaminated/activated steelscrap is assumed to be provided by other companies.
- for the melting station a capacity of 2,000 t/a is chosen. Since this capacity is only fractional compared to the available steel quantity as well as to the outlet possibilities into the concrete market, no further market research had to be executed. On the other hand one should be aware of the fact that a large capacity probably leads to a more efficient production and therefore a more favourable project evaluation.

Two calculations have been made. One for the concrete market for building applications (power stations and storage buildings for disposal) such as foundations, walls, floors, roofs, seperations and the biological shield. The other calculation was made for the market of concrete waste containers. This distinction has been made since the market value of the steel containing concrete is completely different for both markets.

From the calculations could be concluded that the capital value development is positive for the application of steel granules in concrete for the building of new nuclear powerplants, as well as for the manufacturing of radwaste containers; particularly the containermarket is interesting. The volume of this market, which has not been investigated, could be interesting for an output of 2,000 t steel per year. These positive results are calculated with a safe margin. For instance safe assumptions are made for the purchase price of steel and the selling price of concrete. Furthermore it seems that the furnace capacity of 2,000 t/a is not an optimum capacity. (See figure 4)

4. Conclusions:

The overall conclusions for this study are:

- from a point of view of workability, compressive strength and the amount of steel in the concrete, by replacing 100% of the sand in concrete by steel granules, gives the best results.
- by using this concrete for the conctruction of new nuclear powerplants, it is possible to re-use about 6,000 - 26,000 tonnes of steel per plant.
- by using this concrete it is also possible to construct thinner walls because the shielding capability is much better.
- out of several possibilities to produce steel particles, (task B1) the melting technique together with a granule production technique was opted for.
- based on a capacity of the melting plant of 2,000 t/a it turns out to be a feasible project. This applies concrete applications in new nuclear power stations and storage buildings and even stronger for the application in concrete waste containers.

Figure 1: Quantities of contaminated/activated steel in the future coming from reactors which are now in operation in the EC (rounded up at 500 t.).

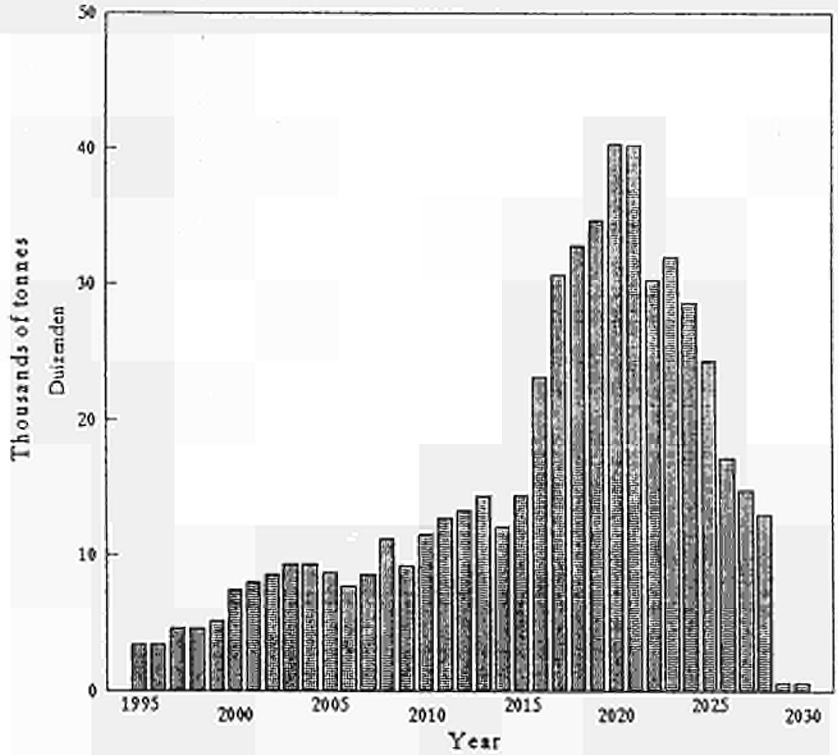


Figure 2: Calculated wall thickness in order to get a reduction to 1/10 of the exposure rate of 1MeV.

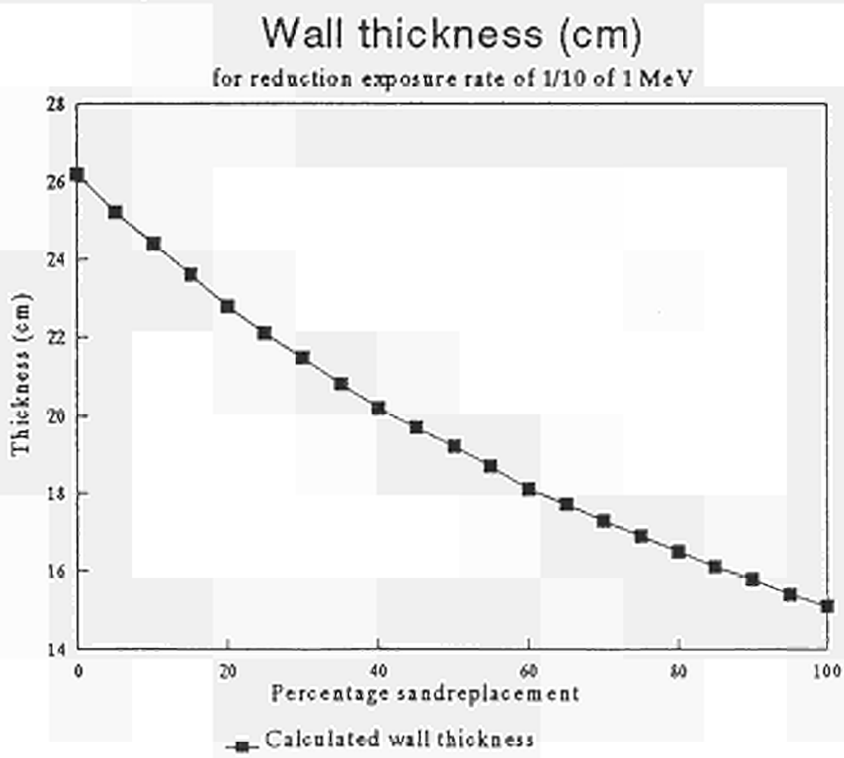


Figure 3: Exposure rate at 1 m distance and a source of 1 Bq/g (μR/h)

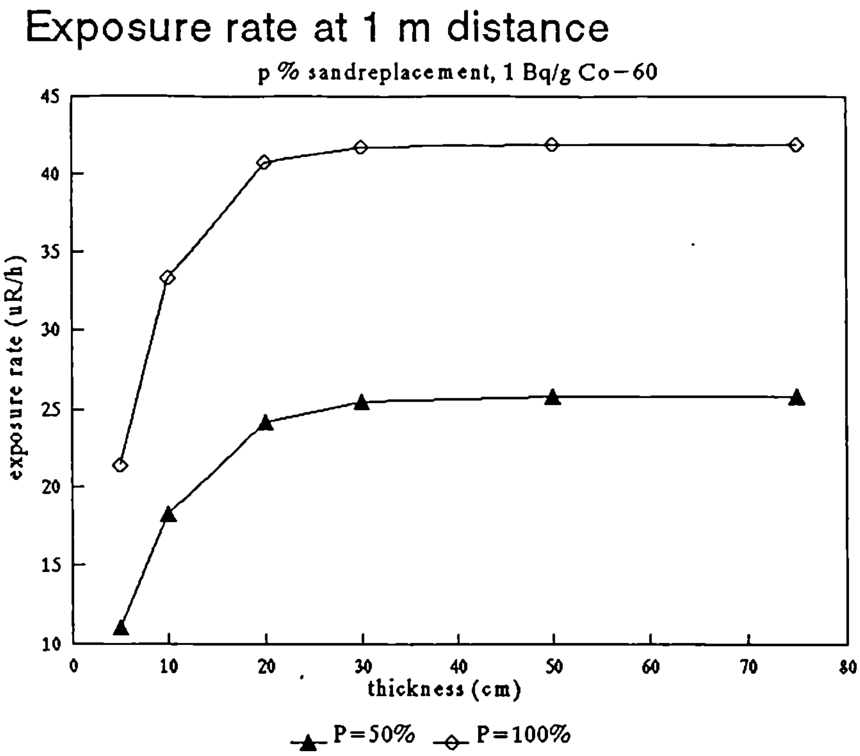
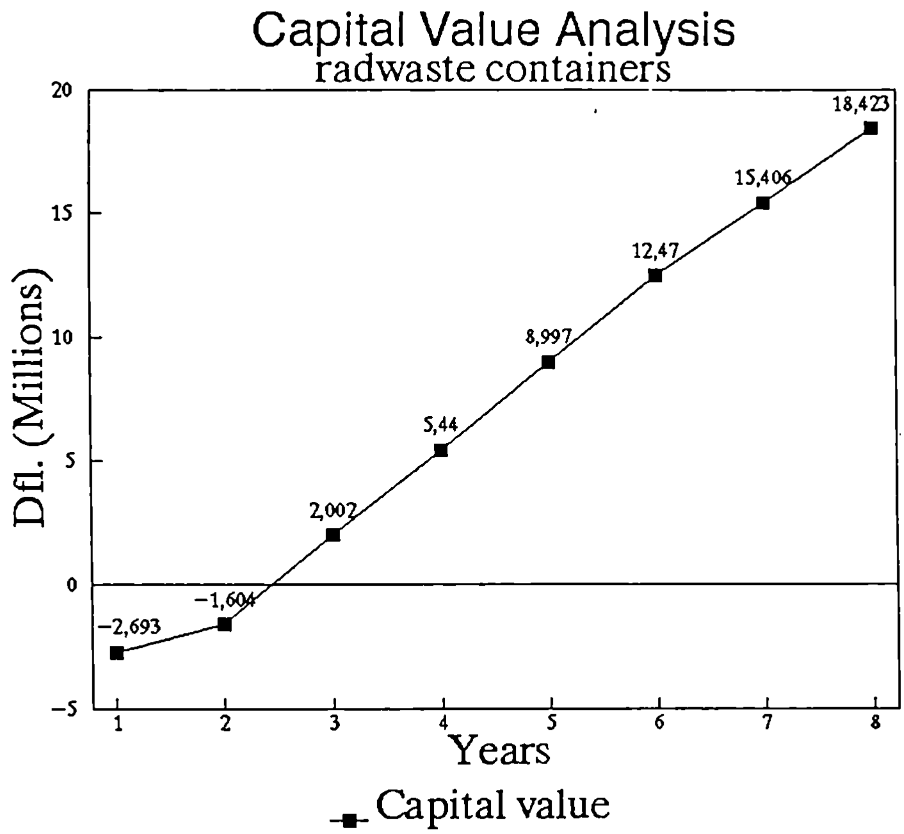
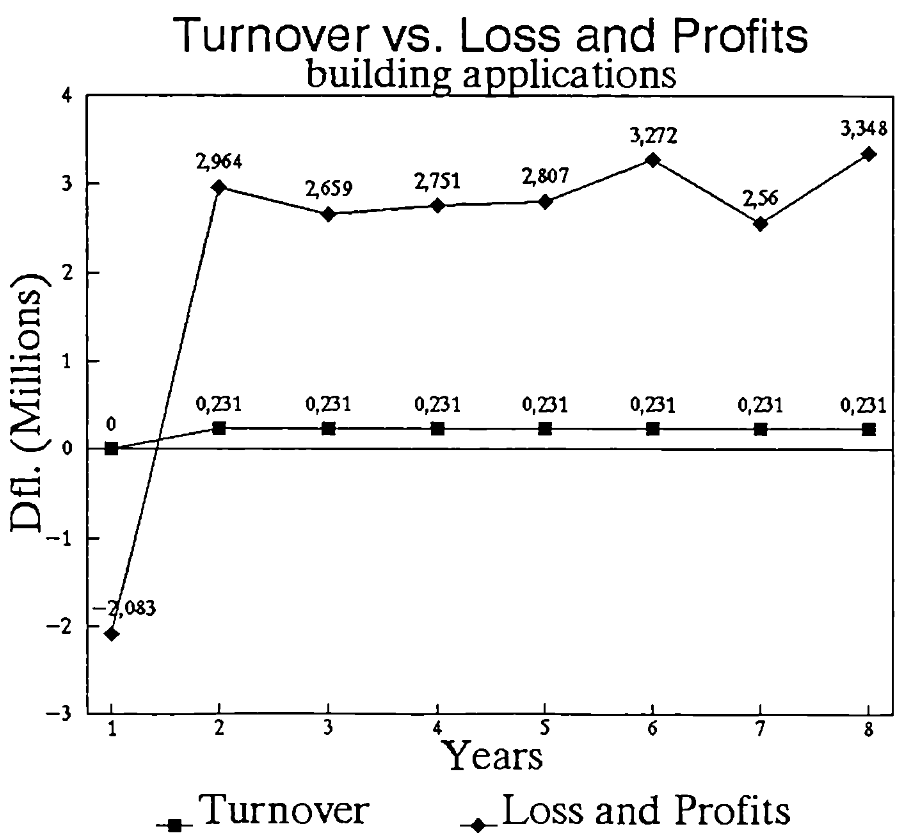


Figure 4: Figures a and b show the capital value for the project.
a) for building applications in the nuclear field
b) for radwaste containers
The capital value may be considered as the present value (in this project expressed in Dfl.) of an imaginary company exploring the process of making steel granules and a steel-concrete mixture. The capital value containes detailed information on stocks, taxes, instalments, the difference between economical, technical and fiscal depreciation , etc. As can be seen the value develops positivily for both applications.



Capital Value Analysis
radwaste containers
Dfl. (Millions)
20
15
10
5
0
-5
18,423
15,406
12,47
8,997
5,44
2,002
-1,604
-2,693
1
2
3
4
5
6
7
8
Years
Capital value



Turnover vs. Loss and Profits
building applications
Dfl. (Millions)
4
3
2
1
0
-1
-2
-3
2,964
2,659
2,751
2,807
3,272
2,56
3,348
0
0,231
0,231
0,231
0,231
0,231
0,231
0,231
-2,083
1
2
3
4
5
6
7
8
Years
Turnover
Loss and Profits

Figure 4

4.5. INVESTIGATIONS ON RECYCLING OF RADIOACTIVE NON-FERROUS ALUMINIUM AND COPPER BY MELTING PROCESS

Contractors: Siemens-KWU, SG
Contract No.: FI2D-0037
Work Period: December 1990 - November 1993
Coordinator: K.H. GRÄBENER, Siemens-KWU
Phone: 49/69/807 36 45 Fax: 49/69/807 20 66

A. OBJECTIVE AND SCOPE

The research work aims principally at developing a method to refine contaminated Al and Cu scrap to a product that enables unrestricted reuse in conventional industrial process.

Parameters such as heating rate, temperature, slag former, surrounding atmosphere will be varied to get optimum conditions for decontamination by melting.

The behaviour of the most relevant isotopes will be investigated and the possibility of melt decontamination on Al and Cu will be examined. For the treatment of Al, co-operation with CIEMAT, Madrid, will be established (contract No. FI2D-0023).

The organic coatings on various Cu items represent a special handicap. Investigations will be made on how the radioactivity is distributed between metal and coatings, whether the separation prior to melting is necessary or not and how harmful gaseous effluents can be managed.

In preceding works, the melting technique was already assessed for steel (contracts Nos. FI1D-0044 and FI1D-0016).

B. WORK PROGRAMME

B.1. Arrangement between CIEMAT, Madrid/Siemens-SG to co-operate in aluminium melting.
B.2. Installation of an inductively heated furnace with exhaust system. (SG)
B.3. Procurement of representative contaminated Al and Cu samples. (Siemens)
B.4. Treatment of Cu. (SG)
B.4.1. Investigations on metal coating separation and gamma-nuclide distribution.
B.4.2. Basic melting experiments with observation of radiation and contamination of workers and working area.
B.4.3. Supplementary melting experiments with varying melting conditions.
B.4.4. Determination of radionuclide distribution in slag, metal, dust and coating.
B.5. Laboratory-scale melting experiments with Al. (Siemens)
B.5.1. Optimisation of melting conditions.
B.5.2. Determination of radionuclide distribution.
B.5.3. Investigations on recycling of the salt melts.
B.6. Melting of Al in an industrial furnace. (SG)
B.7. Derivation of specific data on costs, radioactive job doses, working time and secondary waste arising from the above items. (all)

C. Progress of Work and Results Obtained

Summary of Main Issues

Until now, the melting experiments were performed with aluminium scrap supplied by the Siemens service group. This scrap was mainly contaminated with Co-60 and Cs-137. But in future aluminium scrap contaminated with uranium will arise. Therefore we examined the decontamination of aluminium artificially contaminated with metallic and oxidic uranium. The results showed that only a small part of the uranium was dissolved by the metallic melt. The main part was retained on the outer surface.

B4 Treatment of Cu

B4.1 Investigations on metal coating separation and gamma nuclide distribution

The removal of various coatings from copper scrap and industrial-scale melting of copper were performed by Siempelkamp. In contrast to the planning, this work was supported by investigations of the Siemens laboratory. The distribution of gamma contamination among the individual fractions after separating the isolation materials by blast air was determined. The most important gamma activity was that of K-40. The results are listed in Table 1.

The activity content in three copper samples from melting campaigns at Siempelkamp were also estimated. The alpha- and the gamma-emitting radionuclides could be measured without difficulty. Only in the case of the two pure beta-emitting nuclides, C-14 and Ni-63, were the detection limits higher than the limit values for unrestricted reuse. For more sensitive measurements, chemical separation of the radionuclides is necessary. An extensive description of estimations of low-level contents of alpha-, gamma- and beta-emitting nuclides is given in /1/.

B5 Laboratory-scale melting experiments with Al

B5.1 Optimization of melting conditions

For this work we used contaminated Al scrap which was supplied by the Siemens service group. We optimized the melting parameters by performing melting experiments with different additives and melting temperatures. The optimum melting parameters that could be achieved and that were sent to Siempelkamp are shown in Table 2.

B5.2 Determination of radionuclide distribution

The fractions which resulted during the melting experiments are: metallic melt (ingot), recycled salt, unsoluble residue, crucible and air filters. The radionuclide content was determined for each fraction.

The decontamination factor for each radionuclide was calculated by dividing the total activity of the charge by the activity of the metallic phase. The results of the activity measurements and the decontamination factors are given in Tables 3 and 4. For Co-60 a decontamination factor of 25 and for Cs a decontamination factor of about 3 were estimated. Besides estimating the amount of the radionuclides, the homogeneity of the radionuclide distribution was checked by autoradiography. In all cases in which original scrap and salt was used for melting, a uniform distribution of activity was found inside the melt without hot spots.

All of these examinations were performed with aluminium mainly contaminated with Co-60 and Cs-137. In future, when nuclear facilities are decommissioned, certain amounts of aluminium scrap contaminated with uranium have to be expected. Therefore two melting experiments with aluminium which was artificially contaminated with metallic and with oxidic uranium were started. The results showed that metallic and oxidic uranium were retained on the outer surface of the metallic phase. Only a small part was homogeneously dissolved in the metallic phase and in the slag.

Part of this homogeneously contaminated Al melt was remelted with fresh salt. Through this second melting the uranium was nearly entirely transferred to the slag.

B5.3 Investigations on recycling of salt melt

One aim of this work is to minimize the volume of secondary waste and to examine the possibilities for recycling the salt slag. A simple and effective way is to dissolve it in water, to filter the resulting solution and to evaporate the filtrate. The salt obtained in this way can be recycled to the melting process. The activity of the contamination is concentrated in the filter residue. An exception is the behaviour of the radionuclide Cs-137 which is enriched in the recycled salt.

B6 Melting of Al in an industrial furnace

Within the framework of the EC research project concerning the recycling of radioactive aluminium and copper scrap, 10 Mg of aluminium are to be recycled for unrestricted reuse.

Large-scale tests are to be performed to verify the method for decontaminating aluminium through melting developed by Siemens in laboratory experiments.

The aim of these activities is to ensure that the aluminium is as free as possible of radioactive materials after melting, is below the allowable limits recommended by the Radiation Protection Commission (SSK) and can therefore be removed from the nuclear cycle.

In accordance with Point B5 of the EC contract, Siemens performed laboratory tests to determine the optimum conditions - e.g. through variation of temperature, additives and dwell time - for a large-scale melting test.

As described in the first report by Siemens (Annual Progress Report 1991) a furnace lined with corundum was used. The second report by Siemens stated the following parameters as representing optimum conditions for a large-scale test:

Siemens Test Conditions

Melting temperature:	720 °C
Quantity of additives:	5 wt. % of scrap
Composition of additives:	45 wt. % NaCl
	45 wt. % KCl
	10 wt. % CaF_2
Dwell time:	30 minutes
Purge gas:	air
Crucible material:	carbon

The laboratory tests revealed that the activity is enriched in the liquid salt phase arising during melting. The decontamination efficiency achieved in the melting tests was approximately 90 %.

Therefore, both the furnace lining and the additives were varied as shown below:

Siempelkamp Melting Parameters

Melting temperature:	680 to 730 °C
Quantity of additives:	as stated by manufacturer
Composition of additives:	
Cleaning agent AL-SM:	45 wt. % NaCl
	40 wt. % KCl
	15 wt. % cryolite
Purge gas:	
Degasifying agent AL-T42:	250 l of purge gas mixture, sodium-free
Degasifying agent AL-C19:	300 l of nitrogen
Crucible material:	steel crucible with corundum liner

Out of the total quantity of molten aluminium, 78.4 % was found to have a gamma activity for the nuclide Co-60 of less than 1 Bq/g. Other gamma emitters could not be detected.

Since beta emitters as well as "other nuclides" were stated on the lists accompanying the drums, the measured data were extrapolated for the nuclides possibly still remaining in the melt. Even if this calculated activity is included, 74.4 % still remains below the limit for industrial reuse of the Commission's recommendations for radiation protection.

In this connection it should be mentioned that these recommendations are based on a potential distribution scenario only for steel. Since, however, the present case involves nonferrous metals which are subject to other distribution scenarios, a special release limit for unrestricted or restricted reuse must be agreed upon with the licensing authorities. This has already been practised by the Bavarian authorities for melted brass condenser tubing. A maximum allowable value of 0.5 Bq/g, with a surface contamination of 0.5 Bq/cm^2 for unrestricted release, was approved for such cases.

During the entire melting campaign, 127 kg of particulate matter with an activity of 3360.2 kBq were produced. This equals 0.65 % of the quantity of aluminium that was melted. On account of the relatively high impurity content of the supplied material, the quantity of slag and other waste material amounted to 8.34 % with an activity of 94510.58 kBq.

Literature

/1/ Haas, Paus, Hofmann
Freimessung metallischer Materialien aus der
Schmelzdekontamination
Siemens Bericht S531/92/114.

Table I: Results of Gamma Activity Measurements

Cu isolation
Total mass: 1551 g

Total activity before separating with blast air

Nuclide	KeV	Bq	Bq/g	%
Cs-137	661	11.5	0.0074	100
Co-60	1173 1333	86.4 100.2	0.056 0.065	100 100
K-40	1461	911.9	0.588	100
U-235	144 185	< 5.49 < 5.50	< 0.081 < 0.0035	100 100
U-238	1001	< 125.8	< 0.081	

Fractional activities after separation with blast air

a) Heavy fraction 327 g, 86 wt %

Nuclide	KeV	Bq	Bq/g	%
Cs-137	661	7.27	0.0055	63
Co-60	1173 1333	47.8 50.0	0.036 0.0377	55 50
K-40	911	1155.6	0.8708	126

b) Light fraction 183 g, 12 wt %

Nuclide	KeV	Bq	Bq/g	%
Cs-137	661	2.73	0.0149	24
Co-60	1173 1333	49.4 53.5	0.2700 0.2923	57 53
K-40	1001	82.5	0.4508	9

c) Fine-grained fraction 23 g, 2 wt %

Nuclide	KeV	Bq	Bq/g	%
Cs-137	661	2.67	0.1130	22
Co-60	1173 1333	20.6 18.7	0.8957 0.8130	24 19
K-40	1001	100.0	4.3488	11

5. AREA No. 5: QUALIFICATION AND ADAPTATION OF REMOTE-CONTROLLED SEMI-AUTONOMOUS MANIPULATOR SYSTEMS

A. Objective

Because of radiation fields, some decommissioning tasks must be performed with remote control, in order to minimise occupational exposure. This requirement forms a major technical challenge in decommissioning.

The objective of this research is to qualify and adapt remote-controlled semi-autonomous systems for manipulation of decommissioning tools and instruments.

B. Subjects of the research performed under the previous programmes (1979-88)

Remote-controlled manipulation systems did not form the subject of a Project Area of its own, so far, but limited activities in this field were performed under Projects No. 2 (Decontamination) and No. 3 (Dismantling techniques).

C. Programme 1989 to 1993

Remote-controlled semi-autonomous manipulators should be adapted and tested, in order to qualify and improve their performances with typical decommissioning tasks and tools. For this purpose, existing components and sub-systems should be used and adapted as far as feasible. This concerns in particular sensing systems and computer programmes for semi-autonomous process control, which form important aspects of the research. Special attention should be paid to highly repetitive time-consuming operations, e.g. decontamination and clearance measurements of large surface areas of premises.

D. Programme implementation

Six research contracts relating to Area No. 5 were in progress in 1992, from which one was completed in the same year.

5.1. ROBOTIC SYSTEM FOR DISMANTLING OF THE PROCESS CELL OF A REPROCESSING PLANT

Contractor: ENEA, CRE Trisaia
Contract No.: FI2D-0006
Work Period: October 1990 - September 1993
Project Manager: P MATALONI
Phone: 39/835/97 43 94 Fax: 39/835/97 42 92

A. OBJECTIVE AND SCOPE

Most reprocessing plants, at the end of their lifetime, consist of small shielded cells, inside which the process equipment is installed. The plant philosophy required the operator to enter the cells for any maintenance interventions; the cells are usually accessible from a top corridor through openings closed by shielded plugs.

The present research projects aims at testing a robotic system that can dismantle the equipment of a small cell of this type and remove cut parts from the cell without any direct intervention of the operator. The envisaged robotic system consists of a servomanipulator (MASCOT IV) and a hoist installed inside a containment box; the box has the purpose of avoiding the dispersal of contamination both during the cutting operations and during the transfer of the cut parts to the conditioning cell.

The robotic system will be tested using a mock-up of the dissolution cell of the EUREX plant, built according to the criteria of small shielded cells.

B. WORK PROGRAMME

B.1. Design and construction of a mock-up of the dissolution cell of the EUREX plant

B.2. Design and construction of a containment box and installation of the MASCOT IV servomanipulator

B.3. Non-radioactive testing of the robotic system with dismantling operations, using the cell mock-up.

B.4. Non-radioactive testing of the robotic system with simulated cell decontamination operations, including simulated smear tests.

B.5. Specific data on costs of the system and its radiological impact on work force and working area.

C. Progress of work and results obtained

Summary of main issues

The activity carried out during 1992 has concerned the construction of the robotic system; the system consists of the MASCOT IV servomanipulator installed inside a containment box.

The construction of the containment box has been completed.

The construction of the mock-up of the Dissolution Cell of the EUREX Plant has been completed also.

The installation of the robotic system and the cell mock-up inside the rooms G43-101 of the ITREC Plant has been delayed to the beginning of 1993.

Progress and results

1. Design and construction of a mock-up of the Dissolution Cell of the EUREX Plant (B.1.)

The construction of the mock-up of the Dissolution Cell of the EUREX Plant has been completed; the mock-up is shown in Figure 1.

The positioning of the cell mock-up in the room G43 of the ITREC Plant has been delayed to the beginning of 1993, after the assembling of the robotic system inside room 101.

2. Design and construction of a containment box and installation of the MASCOT IV servomanipulator (B.2.)

The robotic system consists of a MASCOT IV servomanipulator installed inside a containment box. The box is positioned above the cell mock-up and communicates with it by means of an opening closed by a double lid system.

The mechanical construction of the containment box has been completed; for an overall picture see Figure 2 (the box was built without the side walls to allow a continuous control by the operators during the displacement of the MASCOT). The construction of the electric and pneumatic circuits has been delayed to January 1993.

The installation of the MASCOT IV (at the end of the telescopic column) has been delayed to February 1993, after the installation of the containment box inside room 101.

3. Non radio-active testing of the robotic system with dismantling operations, using the cell mock-up (B.3.)

In waiting for the construction and installation of the robotic system, some experimental tests have been carried out to test the MASCOT and the cutting tools.

The MASCOT, shown in Figure 3, was installed in a Technological Hall.

The MASCOT showed some defects during the first tests, especially as regards its advanced functions; the defects have been corrected.

Two cutting tools have been considered for the experiment: a grinder and a reciprocating saw; to avoid mechanical reaction on the servomanipulator, suitable supports have been studied and built (some supports are shown in Figure 4).

FIG. 1 - THE CELL MOCK-UP

FIG. 2 - THE ROBOTIC SYSTEM

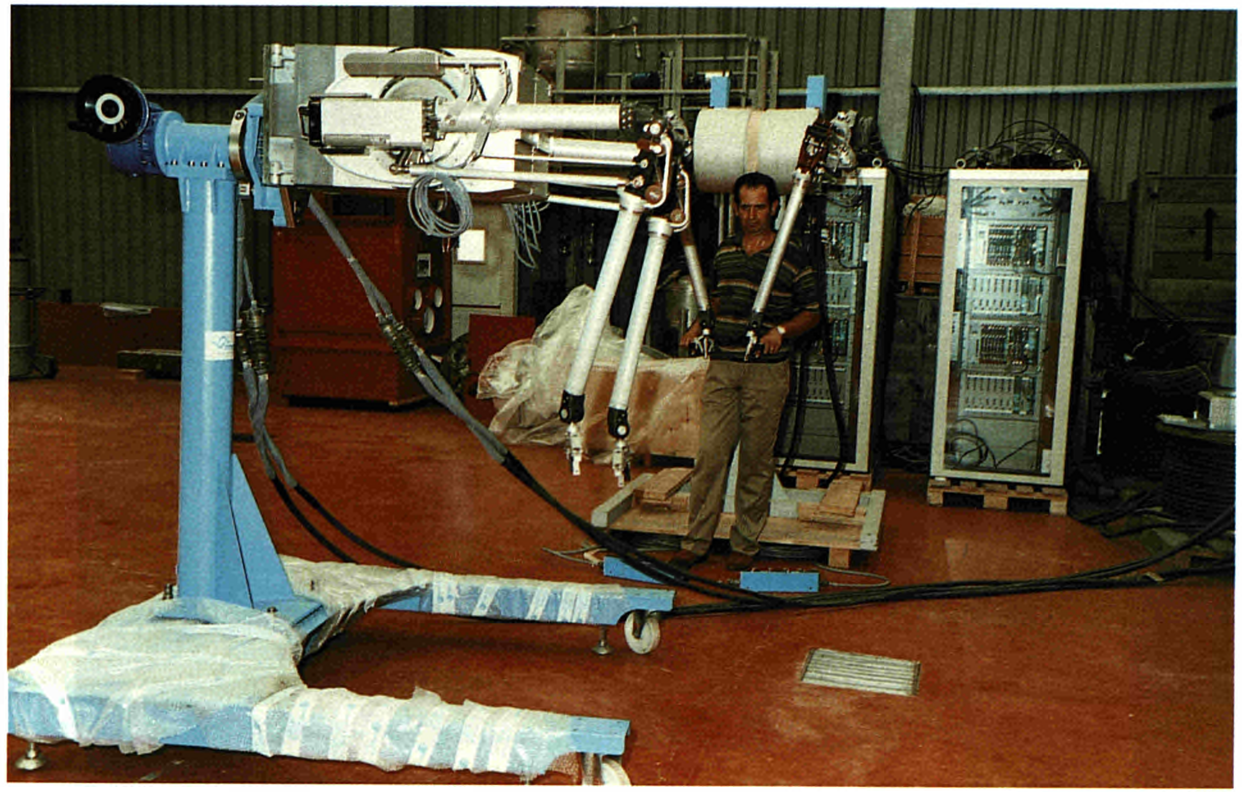


FIG. 3 - THE MASCOT IV SERVOMANIPULATOR

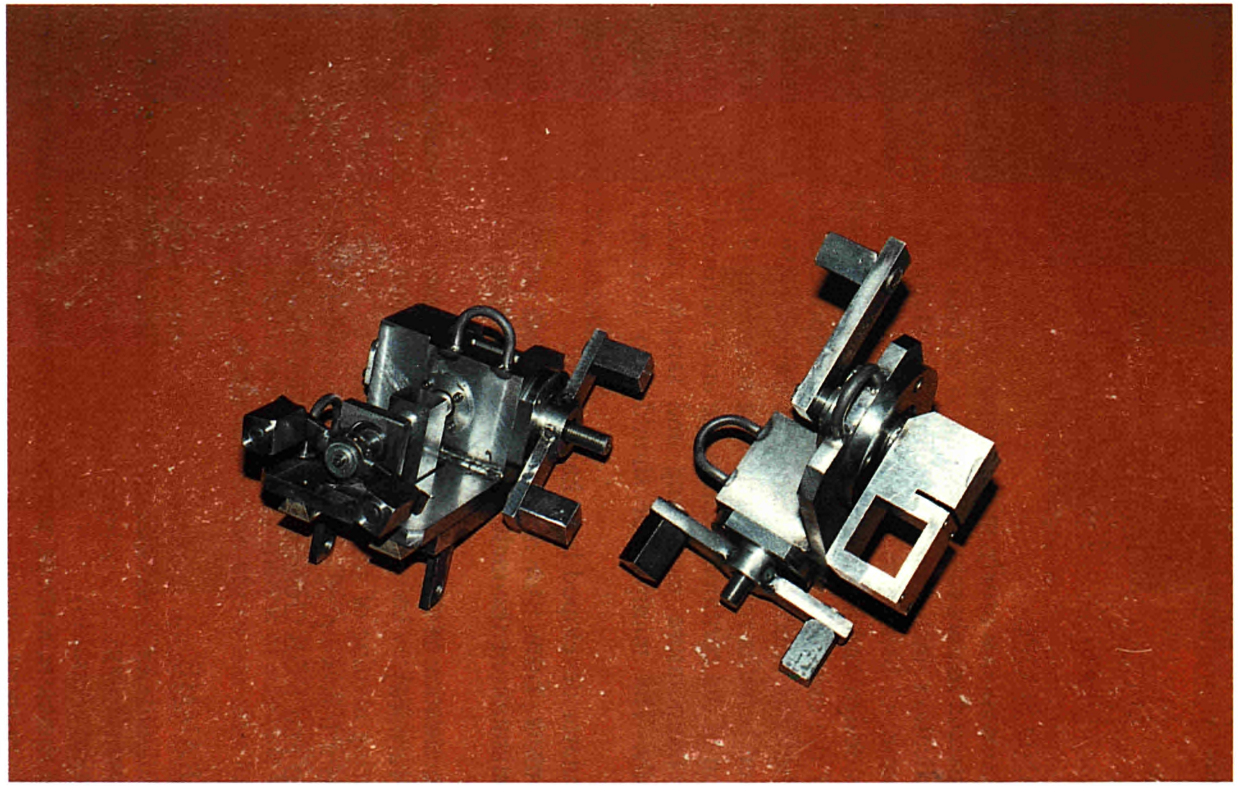


FIG. 4 - SUPPORTS FOR THE CUTTING TOOLS

5.2. DESIGN, CONSTRUCTION AND TESTING OF A MANIPULATOR FOR REMOVING SLAG, MEASURING TEMPERATURE AND TAKING SAMPLES DURING MELTING OF RADIOACTIVE METAL

Contractors: ANSALDO, Siempelkamp
Contract No.: FI2D-0008
Work Period: July 1990 - March 1993
Coordinator: M CIARAVOLO, DNU/RTL, ANSALDO S.p.A.
Phone: 39/10/655 88 46 Fax: 39/10/655 87 99

A. OBJECTIVE AND SCOPE

The work consists essentially in the improvement of an existing melting procedure for radioactive materials and mainly relates to:

- a system specification including a preliminary study to identify the most appropriate manipulator system,
- the components design and manufacturing,
- modifications of the existing melting plant for components housing,
- installation of the components and testing of the system.

The expected benefits relate mainly to a reduction of the radiation dose to the melting staff and a reduction of the contamination in the area surrounding the furnace - operations such as slag removing, temperature measuring and samples taking being nowadays carried out completely manually. The manipulator should also increase the efficiency of the melting technique.

The manipulator developed here has thus to:

- keep people away from the furnace while it is open, in order to avoid their radiation/contamination by the melt, in particular through inhalation of radionuclides leaving the melt;
- reduce the contamination of the surroundings of the furnace (the volatile nuclides like caesium leave the open furnace);
- remove dust during melting of zinc-plated metal.

The work is a follow-up of previous EC contracts (FI1D-0016, -0047 and -0059) under which Siempelkamp and KGB Gundremmingen developed the melting facilities TAURUS I, II and CARLA.

B. WORK PROGRAMME

B.1. System requirements such as basic operations, environmental conditions, interfaces will be specified (Siempelkamp)

B.2. System definition (Ansaldo)

B.2.1. Selection of the basic concept, performing the three operations required, and comparison with a single-purpose device.

B.2.2. Definition of main manipulator operations required, i.e. scumming, sampling, and temperature measurements of the furnace melt bath.

B.3. Design of the defined system components (Ansaldo)

B.4. Manufacturing and shop testing of components (Ansaldo)

B.5. Modification of the existing facility (Siempelkamp)

B.6. System installation and testing in the Siempelkamp melt shop CARLA (All)

B.6.1. Cold tests, e.g. tool changing, manipulator working autonomy, at ambient temperature.

B.6.2. Tests at operational thermal conditions.

B.6.3. Tests with radioactive material < 74 Bq/g, i.e. carbon steel, stainless steel, steel plates covered with zinc, brass, copper and aluminium.

B.7. Final evaluation with regard to costs, melt time, safety, occupational radiation exposure and radioactive emissions to the environment (All)

C. PROGRESS OF THE WORK AND OBTAINED RESULTS

Summary of main issues

Starting from the system functional architecture, the activities carried out in this period have seen the progress of the design of the electro-mechanical part and the definition of the operative sequences.

The architecture of the control system has been defined enabling software development, together with hardware definition to satisfy interfacing requirements with the rest of the plant.

A number of tests have been performed in the CARLA plant, gaining useful information for gripper and control system design; moreover different systems for determining the level of the melting bath surface have been investigated.

A 1:2 scale mock-up of the gripper was built and tested, thus demonstrating its capability to perform slag removal and showing the need for some modifications which have been implemented in the final design.

Manufacturing activities have started and have been completed for both mechanical part and control hardware, allowing the beginning of integration and the scheduling of the workshop functional tests.

On-site modifications to the existing plant have also been defined and carried out to allow the mounting of the manipulator.

Progress and Results

1. Design of the defined system components (B.3)

The detail design phase has led to the definition of both mechanical part and control system, substantially confirming the validity of the functional structure developed in the conceptual design phase.

As extensively described in the annual report 1991, the system consists of a jib crane able to swivel for a 240° angle to allow positioning of the trolley-mounted vertical telescopic mast above all the working positions (see Figure 1).

Apart from some pneumatic-actuated cylinders, all main movements, such as crane rotation, trolley motion and mast raising/lowering, are performed by kinematic chains driven by brushless servomotors controlled in position and speed by means of absolute encoders; brakes and microswitches plus mechanical stops at the end of the stroke are also foreseen as safety measures.

The telescopic mast is provided with a Sommer WW180 automatic changing tool device allowing connection with the gripper or with the other tools foreseen,the changing tool device can also rotate with respect to the telescopic mast to allow the gripper blades to cut the slag which may solidify on the crucible wall.

A number of tests have been directly performed in CARLA plant to acquire useful information, in particular:

- temperature measurements during operation were performed with the available gripper for accurate definition of gripper design conditions,
- real process control times were measured to determine the required control system response times
- different systems for melting bath level measurement were investigated; a ultrasonic distance measuring device, detecting surface level with an accuracy of ± 5 mm, was selected and worked properly during a four weeks melting campaign.

A series of tests (see Figures 2 and 3) were also made using the gripper mock-up to verify the correctness of design choices providing a feedback for some small design changes.

A safety analysis to prevent critical conditions during plant operation was performed, leading to the implementation of safety features addressed to the solution of problems caused by hypothetical accidents and malfunctions: a double rope lifting system was chosen to avoid gripper uncontrolled descent in case of rupture, while a separate supply back-up mast lifting hoist has been foreseen to avoid gripper melting in case of power failure when the latter is immersed in the crucible. A thermocouple located near to the gripper-actuating leverage and a load cell constantly monitoring the load acting on the mast lifting ropes give input signal for internal safeties intervention.

The control system (based on a Siemens S100 PLC) allows all the manipulator's movements to be remotely controlled by the operator either in the automatic or in the manual mode.

The manipulator should usually work in the automatic mode with step-by-step sequences and the possibility of switching to the manual control, maintaining active the safety functions; two different emergency conditions are foreseen, the software emergency stops all the movements, bringing the telescopic mast in a safe condition at its upper position in case it is located over the crucible, while the hardware emergency cuts the power supply.

The various operative sequences are:

- slag removal and depositing into the slag drum
- gripper coating by protective varnish
- taking of samples from the melting bath
- bath temperature measurement by pyrometer immersion
- gripper cleaning by hammering.

The choice of the sequence to be carried out is performed by video pages, which also allow the setting of operating parameters and their visualization during the working phase.

For safety reasons the melting bath level, which is the reference parameter for the slag removal cycle, is directly acquired from the CARLA plant PLC by means of a serial line, avoiding possible operator mistakes.

The control system hardware consists of a power rack for the motor drivers and a console housing the PLC and the graphic monitor; they will be located in front of the window in the CARLA plant control room to allow direct view of the operations.

A simplified graphic representation on the PLC monitor allows the operator to follow manipulator movements, enabling an on line check of the operations.

2.Manufacturing and shop testing of components (B.4)

After the detail design, a contract for the manufacturing of the mechanical part has been issued; commercial components purchasing and manufacturing have been carried out throughout the year and completed in December 1992, allowing some preliminary tests of the manipulator mechanical functionality on an ad hoc test structure (see Figure 4).

An analogous itinerary has been followed for the control system enabling completion of the software programme and its functional validation by simulation; at the same time control hardware (see Figure 5) has been assembled and verification of power racks cabling made.

The integration between the control system and the electro-mechanical part will be carried out in the month of January 1993 and functional tests will follow to allow the shipment of the manipulator to Siempelkamp within the end of February.

3.Modification of the existing facility (B.5)

Taking into account functional, interface and plant layout requirements, the support structure for the manipulator was designed and built, enlarging the existing foundry platform and positioning the various anchor plates necessary for jib crane and other equipment mounting.

Cable layout has also been defined as well as power racks and control panel location.

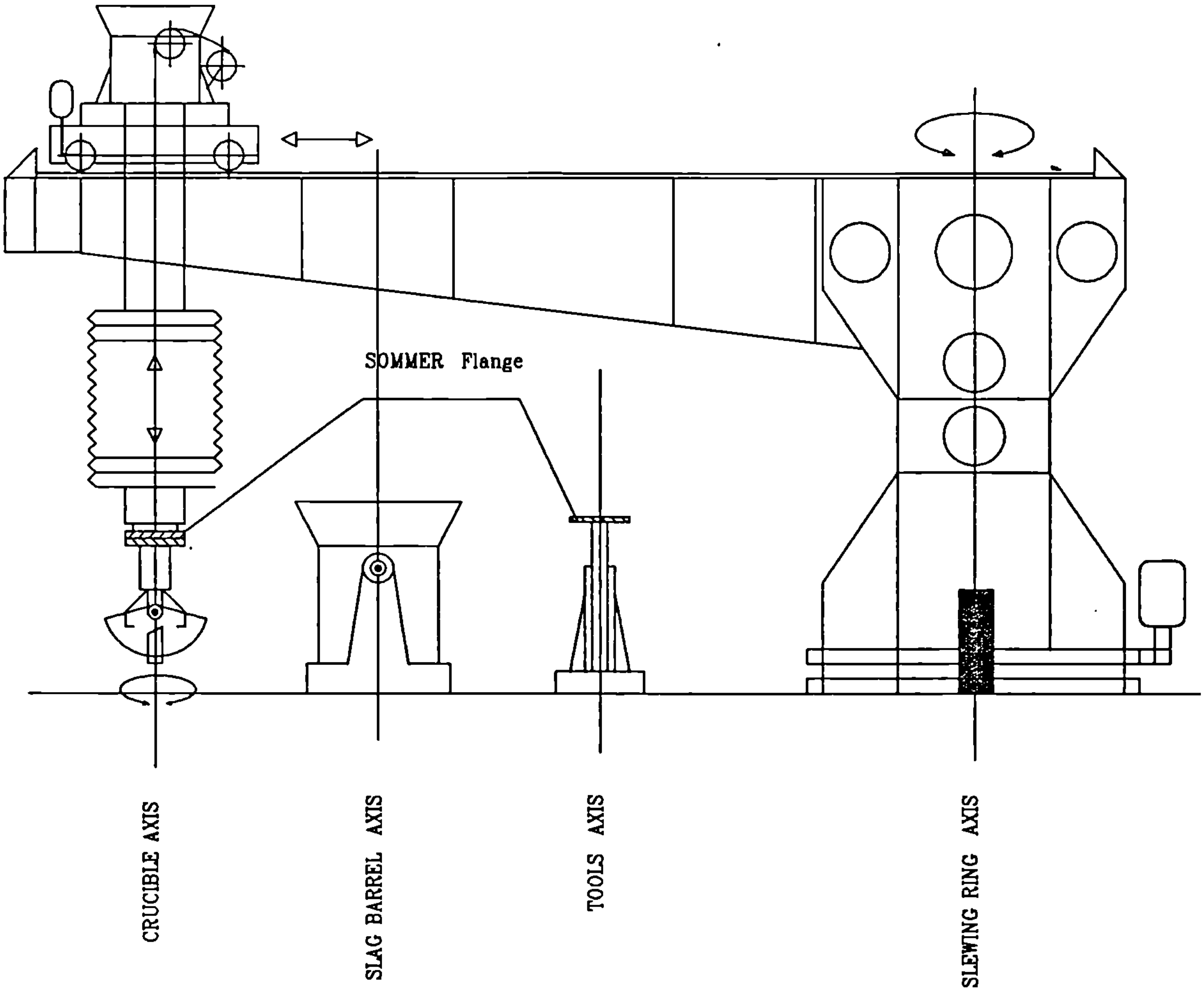


Figure 1: Jib crane manipulator

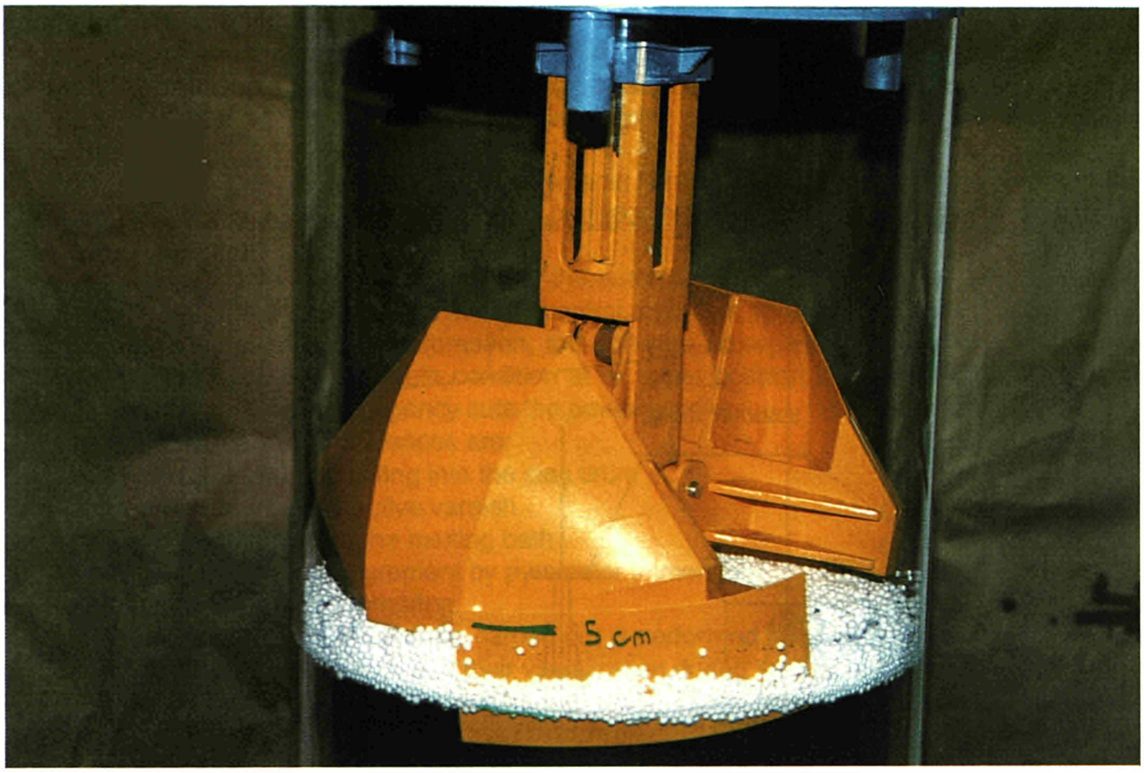


Figure 2: Gripper mock-up test: Immersion

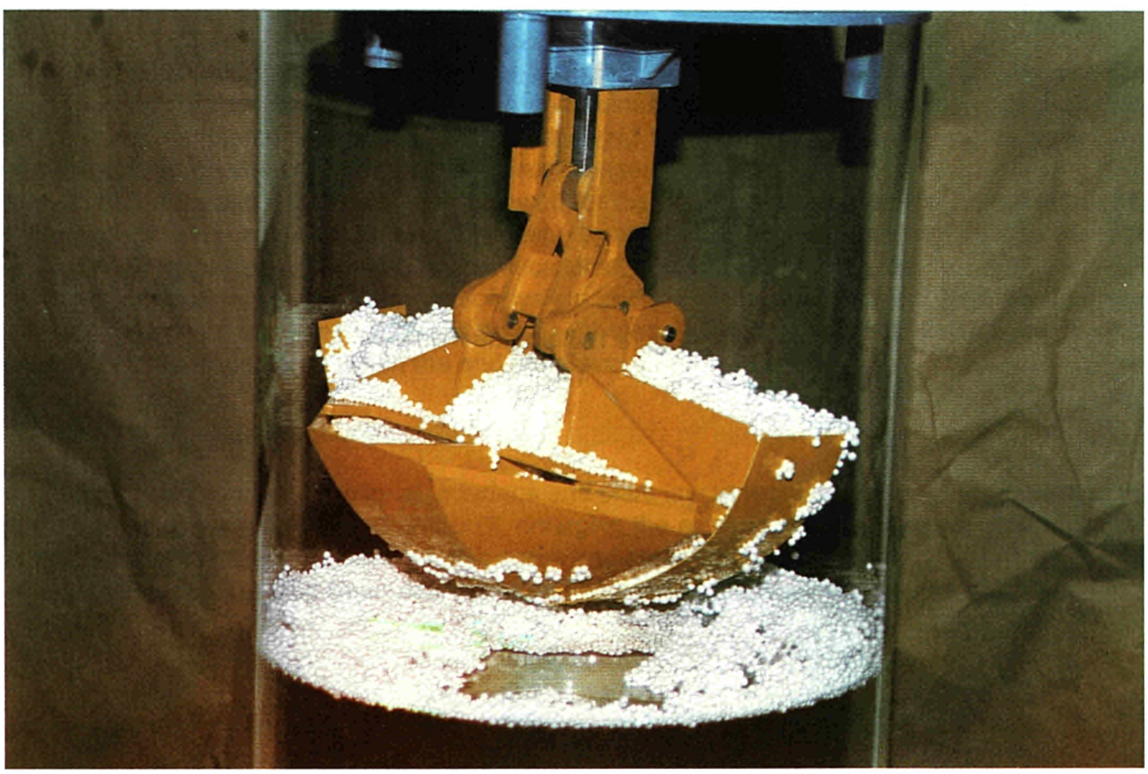


Figure 3: Gripper mock-up test: Slag taking

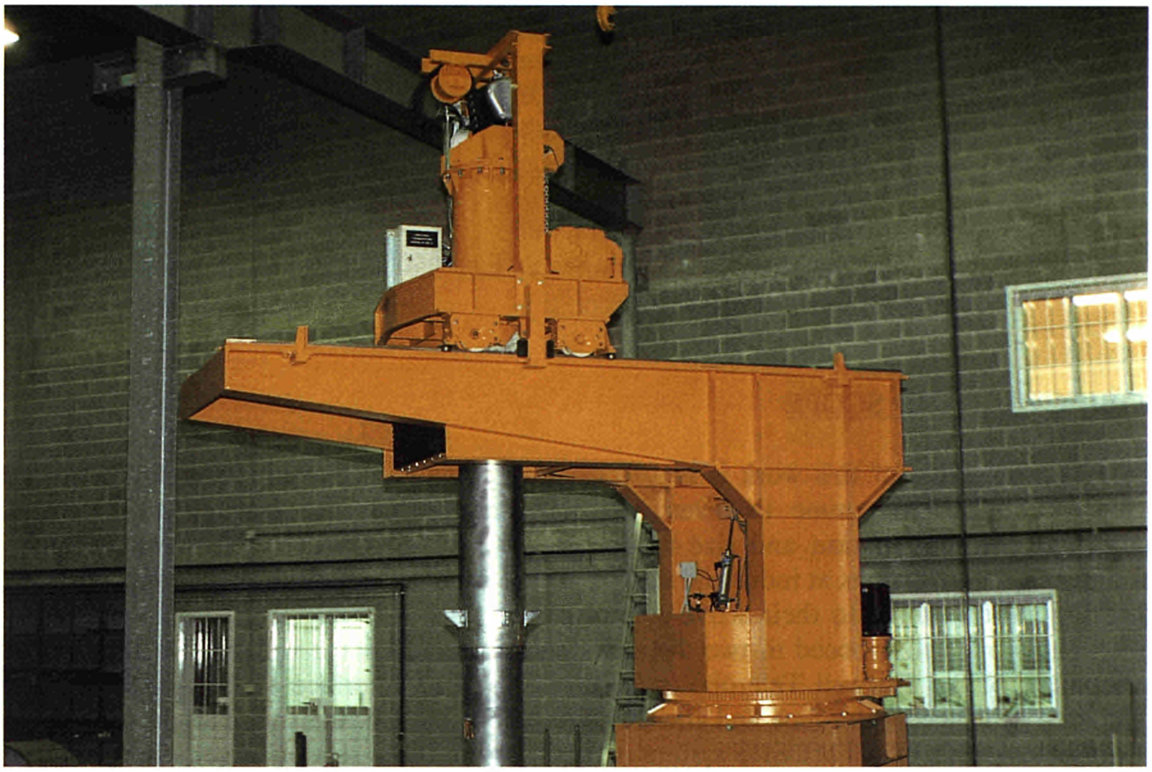


Figure 4: Assembled Manipulator at workshop

Figure 5: PLC Monitor

5.3. TELEROBOTIC MONITORING, DECONTAMINATION AND SIZE REDUCTION SYSTEM - TMDSRS

Contractors: AEA Harw., SCK/CEN
Contract No.: FI2D-0012
Work Period: July 1990 - December 1993
Coordinator: M H BROWN, AEA Harwell
Phone: 44/235/434 691 Fax: 44/235/436 138

A. OBJECTIVE AND SCOPE

The objective of this work is to use existing equipment developed under the Harwell Nuclear Robotics Programme to investigate and demonstrate the feasibility of telerobotic monitoring, decontamination and size-reduction systems (TMDSRS). The work will include experimental investigations at industrial scale, and use sample workpieces of an appropriate size and configuration similar to their active counterparts.

The work will proceed in two distinct stages. The first stage will involve the continued development of the Harwell Telerobotic Controller and its interface to NEATER, a Nuclear Engineered Advanced Telerobot, and ancillary equipment and mechanisms. This development will allow in-active trials of a TMDSRS system on each of the three sets of target workpieces (B.3., B.4. and B.5.). The work will be carried out in the Harwell Robotics Demonstration laboratory.

The second stage of the work (B.6.) will involve active trials of one of the areas demonstrated in the first stage. The selection of the appropriate application will ensure that a safe, useful and representative active trial can be accomplished.

This development will reduce man-Sv and costs of decommissioning projects. Greater efficiencies in placing or deploying decontamination tools and in cutting and packing waste will improve waste disposal strategies, and reduce waste arisings. Data on cost benefits will be produced in submissions made to justify the selection of a suitable project for the active trials (phase 2). Cooperation on sensors with SCK/CEN Mol is included in the work programme.

B. WORK PROGRAMME

B.1. Control system extension to work effectively with each of the three non-active applications.
B.2. Electropolishing head unit development and irradiation tests (AEA)
B.2.1. Requirements analysis for the electropolishing head unit (AEA)
B.2.2. Requirements analysis for the sensor functions (SCK/CEN)
B.2.3. Selection of sensors to meet the requirements analyses of B.2.1. and B.2.2. (SCK/CEN)
B.2.4. Design and construction of the integrated head unit (AEA)
B.2.5. Irradiation tests of the integrated head unit (SCK/CEN)
B.3. Decontamination of different surfaces; radiation monitoring, electropolishing and registration software (AEA)
B.4. Clearance monitoring developments (AEA)
B.5. Glovebox size reduction developments
B.5.1. Analysis of subsystems susceptible to radiation damage (SCK/CEN)
B.5.2. Tests on subsystem components in the gamma irradiation test facility at the BR2 reactor (SCK/CEN)
B.5.3. Tool and operational software development (AEA)
B.5.4. Tool change adaptation and cutting tasks demonstration jointly with a range of tools (AEA)

B.6. <u>Active decommissioning trials in the appropriate active area</u>

B.6.1. Pre-trial analysis of the radiation environment (AEA)

B.6.2. Active trials including the NEATER carrying out of a task or set of tasks (AEA)

B.6.3. Support for active trials to reduce the probability of failures (SCK/CEN)

B.7. <u>Economic analysis of TDMSRS and its radiological impact on work force and working area</u>

B.7.1. Pre-active trial cost-benefit analysis to establish economic advantages of telerobotic operations (AEA)

B.7.2. Post-active trials analysis on costs, incurred dose burdens, working and exposure times of ancillary operators, and estimates of secondary waste arisings (AEA).

C. Progress of work and obtained results

Summary of main issues

An electropolishing Head Unit [EHU] plus alternative sub-assemblies and material samples have been irradiated and early results of the associated post irradiation examinations issued. Another EHU has been attached to the wrist of a Puma robot in order to demonstrate telerobotic electropolishing techniques for decontaminating flat metal surfaces. In addition, scanning software has been developed to support robotic as well as teleoperator controlled electropolishing. Clearance monitoring hardware and the necessary software have been designed to assess the feasibility of increasing the monitoring ahead by using a lightweight pole to extend the reach of the robot.

Active trials of the telerobotic size-reduction of plutonium contaminated gloveboxes have begun. Before the start of the trial an economic, environmental and safety assessment was carried out. Initial results from the trial have verified costings predicted in the economic analysis.

Progress and results

1. Background

The TMDSRS is based on three main building blocks: a Nuclear Engineered Telerobot [NEATER], A Telerobotic Controller [TRC] and a six axis Bilateral Stewart Platform Joystick [BSPJ] to facilitate man-in-the-loop operation. NEATER is based on Stäubli Unimation's clean room robot, the Puma 762 CR. It is modular for ease of maintenance, fully sealed for prevention of contamination and radiation tolerant to 1MGy. An 80486 PC based TRC provides an interface for unilateral and bilateral input devices and hosts sensor based control algorithms such as those required for decontamination. The TRC is linked to the VAL II robot controller via a high speed Ethernet link running SLAVE protocols.

2. Electropolishing Head Unit [B2]

A prototype electropolishing head and sample materials of polymer seals, piping and mounting components have been prepared for irradiation [1]. Proximity sensors needed for positioning of the head with respect to the surface to be decontaminated have been selected. Different targets have been irradiated in the Co-60 gamma irradiation facility at Mol at a mean dose rate of 340 Gy[Si]/h up to a total dose of 300 kGy [Si] in stages of 1, 3, 10, 30, 100, 300kGy.

Initial post irradiation examinations show that inductive proximity sensors, have sufficient tolerance. Silicone and Viton 0-rings show degradation phenomena starting at 10 kGy and 100 kGy respectively [Figure 1]. The weak point in the present design is the attachment of the front seals to the head. The seals are glued in square grooves, and the choice of a proper glue is critical. Leaks appeared at a dose of 3 kGy. Alternative designs and choices of glue have been studied. Further characterisation of the mechanical efficiency of the seals and of the chemical resistance of the different materials with respect to gamma dose are in progress [2].

3. Surface decontamination [B3]

Two inactive demonstrations of robotic decontamination using an electro- polishing head unit are planned. One will operate in robotic mode and demonstrate the decontamination of medium to large flat metal surfaces. The other demonstrator will operate in teleoperator mode to demonstrate the feasibility of decontaminating small metal surfaces in cluttered environments.

In initial teleoperator trials the head unit was placed in contact with a contaminated [simulated] surface using the Bilateral Stewart Platform Joystick [BSPJ]. By sensing, through the joystick, the contact forces reflected from the surface, the operator was able to ensure that all of the head unit face is in contact with the surface before applying a vacuum and commencing the electropolishing process.

In preparation for the robotic demonstration of surface decontamination trajectory and scanning software has been designed, coded and successfully tested in the Harwell Telerobotic Controller [HTC]. Two trajectory modes are provided: Joint Interpolation and Straight Line Control. In Joint Interpolation mode the robot will move to a point-X, which may be specified in Joint coordinates or Cartesian coordinates. During this move each robot joint starts and finishes at the same time. In Straight Line mode the robot moves in a straight line between two points in World or Tool mode.

For robotic decontamination, the robot will move under Joint Interpolation control to a starting position specified in Joint coordinates. This ensures that the robot is in the correct configurations before commencing a scan. The scan area is specified through the HTC touch sensitive screen by selecting:

- start and finish of first line
- a point on the last line
- a scan interval dictated by the width of the electropolishing area.

4. Clearance Monitoring [B4]

The aim of this work package is to develop a lightweight robotic system to scan a radiation monitoring head over large concrete surfaces to confirm or otherwise the removal of active concrete. To extend the scanning area of the system, the compliant monitoring head will be deployed at the end of lightweight pole held in the robot wrist.

A dynamic analysis has shown that poles beyond a certain length and mass will degrade system performance. Excessive pole and head unit inertia can lead to control loop instabilities and motor stalling. In performance trials, although the system remained stable, secondary oscillations were superimposed upon the desired trajectory. Limit cycling effects arising from interaction of system non-linearities, such as wrist backlash, with the pole inertia were considered to be the most likely source of these oscillations.

The inertia threshold, beyond which the robot joint servos start to become unstable, was determined to be $2.6 kgm^2$. The pole and head unit was designed to satisfy this inertia constraint, the aim being to minimise the head unit weight in order to maximise pole length. Head unit weight is dictated by the need to house a commercially available beta gamma dose rate meter and attach the head to the pole via a universal joint. This joint is required to ensure that the face of the head unit remains in a plane parallel to the surface of the wall throughout the scan. Manufacture of the pole and head unit assembly is complete. It meets the specified pole/head unit weight of 1.3kg and length of 1.3m. To keep the weight down the pole was fabricated from composite materials.

The initial proposal for scanning kinematics design strategy was considered to be unduly complex and expensive to implement. A much simpler strategy has emerged from an analysis of the heuristic approach adopted by human operators when executing similar extended reach scanning tasks. In this new approach the algorithm mimics the procedures employed by the human arm. The software to execute this strategy has been coded and evaluation trials will commence shortly.

5. Active Trials [B6]

An analysis of the radiation environment, expected to arise from the size reduction of 200 plutonium contaminated gloveboxes, has been performed. The gloveboxes are up to 30 years old and each contains up to 50 grams of thinly spread plutonium based contamination. During the size reduction process, which is scheduled to take five years, the facility is expected to become contaminated with 100gms of plutonium material ie 0.5gms per processed glovebox. Over the last 30 years a wide variety of plutonium mixtures have been handled in these gloveboxes. Although this variety complicates the prediction of radiation environments, the averaging of data from two distinct, but commonly used, categories of plutonium mixtures should provide a reasonably accurate estimate. The categories comprise plutonium derived from either low burn-up cycle - 1 fuel or high burn-up cycle - 6 fuel.

Plutonium (Pu) derived from cycle - 1 fuel consists mainly of 239-Pu with very low percentages of other isotopes while plutonium derived from cycle - 6 fuel contains a rich variety of radioactive isotopes. Both fuels, especially cycle - 6 fuel, contain small quantities of the isotope 241 - Pu which decays to 241 - Am with a 14 year half-life and then to 237 - Np with a 432 year half-life. 241 -Pu emits a 20 keV beta particle while 241 - Am emits both a 5 MeV alpha particle and a 60 keV gamma photon.

Predicted annual gamma and neutron dose rates at 1 metre from the bulk of the contamination are provided in the table below:

	Gamma Dose Rates		Neutron Dose Rates	
	Gy/Yr		mGy/Yr	
Plutonium Type	Year - 1	Year - 25	Year - 1	Year - 25
Cycle - 6	40	184	33	182
Cycle - 1	4	19	4	19

The radiation environment of the size-reduction facility is generated from short range alpha/beta/gamma/neutron radiation and long range gamma/neutron radiation. The short range radiation is only of importance when the contamination is in contact with susceptible components or materials. Predicted gamma dose rates are not high enough to cause significant damage to the size-reduction facility in its five year life time. The nuclear engineered robot [NEATER] and the associated size-reduction support equipment have radiation tolerances -1MGy and 10 KGy respectively - well in excess of that required.

Safety features have been set up to protect operators, plant and containment from damage. Telerobotic decommissioning improves operator safety by reducing the number of man entries to the active area from five to one, hence reducing the risk of ingested dose and exposure to ionising radiation. To provide operator protection during man entry, the robot operating area is isolated from the rest of the active area by a fence capable of stopping airborne debris such as swarf particles, broken saw blades etc. Robot protection is provided by the installation of torque limiting software to limit the force exerted, in the event of a collision with the environment or associated plant, to an acceptable level. It is essential that an equipment malfunction or operator error should not give rise to a breach of the containment. Consequently, the size reduction system is positioned so that all windows, the weakest part of the containment, are outside the reach envelope of the robot and are protected from robot accelerated projectiles by safety barriers located on the active side of the windows.

Installation started in mid-June and was finished by the end of August. Tool change positions were taught prior to installation because of the difficulties and added expense of teaching under active conditions. To improve reliability tool positions were taught using an improved calibration procedure. This technique enabled 4500 tool changes to be executed, without failure, in performance validation trials.

Active trials of the telerobotic glovebox size-reduction system started in September 1992. To date the trials, which have been performed in an active facility at AEA, Harwell, have been relatively trouble free. Initially, some difficulties were experienced in hole saw cutting efficiency. This was traced to incorrect hole saw manufacturing procedures and has now been rectified. It was also observed that operation of the hole saw caused the robot to stall in certain arm configurations. Operators now avoid stalling by adjusting the lift, rotate, tilt table [LRT] to obviate the need for these weaker arm configurations. The best long term solution is to reduce the weight of the drill which at 15kg is above the 10kg continuous duty payload of the robot.

6. Cost benefit analysis [B7]

Telerobotic glovebox decommissioning costs have been estimated and compared with the costs incurred by traditional manual methods. Costs are based on the size-reduction of a 500 litre Type GP 220 steel box of dimensions 1000mm x 750mm x 680mm. Each entry to the active area requires a team of six people: two pressurised suit workers, one clean dresser, one active dresser, one controller and a health physicist.

For telerobotic and manual glovebox decommissioning procedures, the cost of size-reducing a single box was broken down against costs for: labour, oversuits, waste volume, waste drums, drum liners, PNC monitoring, paperwork, and amortisation.

Labour and waste disposal costs dominate size-reduction costs and both make a significant contribution towards the higher cost of the manual method. Telerobotic methods generate lower waste volumes, and hence disposal costs, because the system facilitates a higher waste packing factor. Amortisation costs of the telerobotic system and its installation have been added to the cost of the telerobotic size reduction of a single box. The cost estimate predicts a cost saving of £5k per box and a total saving of £1M over 200 gloveboxes.

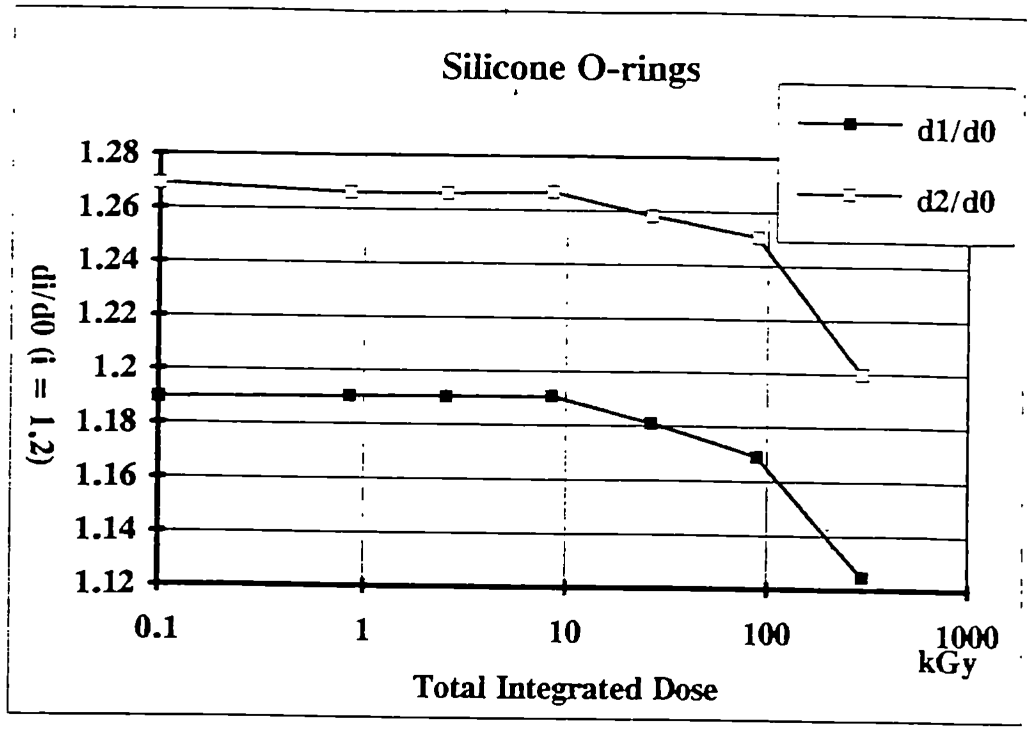
A preliminary analysis of telerobotic size-reduction efficiency was performed after four months of active trials. In this time 20 plutonium contaminated gloveboxes have been decommissioned, the number being dependent on resource availability rather than size-reduction efficiencies. In the pre-trial economic analysis it was assumed that two operators would be required to run the process and it would take one day to size-reduce a simple box from cell entry to fixing the lid on the waste drum. A simple box is defined as one that has no external pipework or metering and from which all internal equipment has been removed.

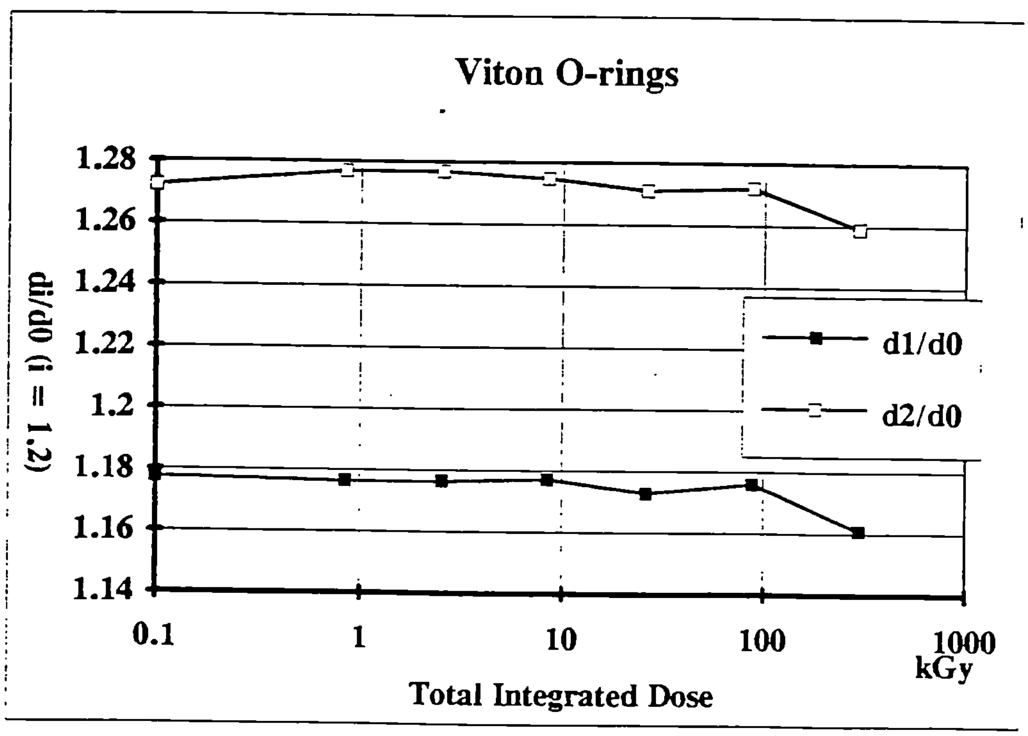
During the active trials it was found that the size-reduction process could be executed with only one operator it took two days to size-reduce a typical box. As the typical box had both internal and external fittings to complicate the decommissioning process the two days size-reduction time compares favourably with the one day predicted for the simple box.

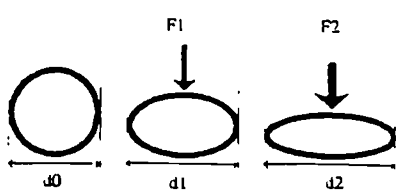
References

[1] COENEN, S, CEN/SCK Report 4.12 [1992]

[2] COENEN, S, CEN/SCK Report TEL/G4002/92-56/SC [1993].







d0 = diameter when no force is applied on O-ring
d1, d2 = diameter when force is applied on O-ring
(F1 < F2)

Figure 1: Test results of Viton and Silicone 0-rings

5.4. ADAPTATION AND TESTING OF A REMOTELY CONTROLLED UNDERWATER VEHICLE

Contractor:	AEA Wind.
Contract No.:	FI2D-0025
Work Period:	July 1990 - June 1993
Project Manager:	P W WORTHINGTON, AEA Windscale
	Phone: 44/9467/72414 Fax: 44/9467/72409

A. OBJECTIVE AND SCOPE

Preparatory work to decommission the Windscale piles is being carried out. As part of this work, fuel debris and contaminated silt is to be recovered from two water-filled, 100 m long, 2.7 m x 2.1 m section fuel transfer ducts which connect the piles to the fuel storage pond.

As no commercial equipment exists which has been purpose-designed for generic nuclear applications, the main object of this work is to adapt and test a remotely controlled underwater vehicle provided with manipulators, on-board television systems and remote sensors for surveillance purposes. This includes the development of an underwater remotely controlled vehicle for use in nuclear applications, the development of techniques for fuel handling and silt recovery, and the assessment of equipment durability, radiation tolerance and of decontamination problems, e.g. removal of contaminated concrete surfaces, etc.

The potential benefits of this work programme will be reduced doses to workers, decontamination and inspections, reduced secondary waste and reduced decommissioning costs due to lower labour input.

Information on costs, occupational exposure, work time and secondary waste arisings will be made available.

The work will include non-active trials in a full-size mock-up.

B. WORK PROGRAMME

B.1. Specification of the underwater vehicle, to allow manufacturers to tender.

B.2. Vehicle manufacture, at the tenderer's work.

B.3. Preparation of the full-size test mock-up of the entrance area to the water duct.

B.4. Adaptation and testing in the mock-up of the vehicle/system on its ability to perform the various decommissioning tasks.

B.4.1. Dry testing of the complete system.

B.4.2. Wet testing of the complete system.

B.5. Development in the active environment of Windscale Pile No. 1.

B.6. Final evaluation will include specific data on costs, work time, occupational exposure and the secondary waste arising from the technique.

C. Progress of work and obtained results

Summary of main issues

The Vehicle has been manufactured (B.2.). It was delivered to AEA Technology in April 1992. The Dry Test Facility has been constructed (B.3.). Work is proceeding in the test facility on the Vehicle development programme and operator training (B.4.1.). The Wet Test facility is under construction. Use of the Vehicle for fuel recovery operations (B.5.) is forecast for November 1994.

Progress and results

1. Vehicle manufacture, at the tenderer's works.(B.2.)

The ordering and manufacture of the underwater Vehicle was delayed because of the need to fully address the use of the Vehicle in making a safety case to satisfy the UKAEA Nuclear Site Licence. Vehicle manufacture commenced in August 1991. The Vehicle was delivered to Windscale site in April 1992.

The original proposed Vehicle would have been a prototype designed specifically for this application. A decision has been made by the AEA to purchase a Vehicle built up of proprietary items that will be adapted to work in the water duct.

2. Preparation of the full size test mock-up of the entrance area to the water duct.(B.3.)

It was decided in early 1991 to have two separate test facilities, one dry and one wet because much of the early development programme could be carried out in a dry facility. Design work for the Dry test facility was completed and the dry test facility has been constructed at Windscale. Construction of the dry test facility was completed in accordance with the original contract programme.

The dry test facility is a simulation of a 3m long section of the water Duct. It is designed to be light-tight so that a simulation of conditions in the water duct can be attained.

The Wet Test Facility is under construction and will be available for Vehicle and silt recovery trials in early 1993.

3. Adaptation and testing in the mock-up of the Vehicle/system on its ability to perform the various decommissioning tasks.(B.4.)

Work is in progress on Vehicle development in the Dry Test Facility. A number of areas are being addressed;

Training programmes have been designed for the proposed operators of the Vehicle. The training is aimed at simulating the operations that will be done during fuel and silt recovery in the Water Ducts. All operators have completed the Vehicle manufacturers initial training course.

The training programmes consist of practice periods followed by regular witnessed tests. Early indications are that the operators quickly become familiar with the vehicle's operation because the time taken to complete the witnessed tests reduces rapidly. Early indications are that the operators reach a plateau at the minimum time they take to complete a witnessed test. Further training is in progress.

(2) Camera systems: An overview camera mounted on the rear of the Vehicle monitors all Vehicle operations. An additional camera has been identified and fitted to the manipulator to give a better view of the gripper during operations.

(3) Vehicle impact on Water Duct: An assessment of the impact forces imposed on the water duct walls by accidental collision of the Vehicle or the manipulator is required. A suitable device for the measurement of these forces is fitted inside the Dry Test Facility. Trials are yet to start on this area.

(4) Manipulator jaw gripping force: The gripper force will be variable between 0 and 500N. It will have to be limited to avoid damaging fuel elements when they are being recovered. A system has been designed to control the gripper force and will be incorporated onto the Vehicle during 1993.

(5) Vehicle detection system: An equipment survey has identified a suitable sonar device to enable the distance that the Vehicle has travelled along the water duct to be accurately measured. This will assist in the record keeping of fuel element locations when they are recovered.

The same device is capable of identifying fuel elements buried in the silt that exists in the Water Ducts. This will enable the Vehicle to continue working when silt becomes re suspended in the water.

(6) Simulation of water duct silt: A simulation of the water duct has been prepared. This will be used in the Wet Test Facility to study dispersion effects of the Vehicle operations and problems with ingress of contamination. This simulated silt will also be used in silt pumping trials.

4. Wet testing of the complete system.(B.4.2.) The design of the Wet Test Facility is complete. Construction of the facility is in progress with a completion date of February 1993.

5. Development in the active environment of Windscale Pile No.1.(B.5.)

This phase of the programme will commence in November 1994.

6. Final evaluation will include specific data on costs, work time, occupational exposure and the secondary waste arising from the technique.(B.6.)

This phase of the programme will be completed in June 1995.

5.5. TEST OF LONG-RANGE TELEOPERATED HANDLING EQUIPMENT WITH DIFFERENT TOOLS FOR CONCRETE DISMANTLING AND RADIATION PROTECTION MONITORING

Contractors:	KfK, KA, AEA Harw., BAI
Contract No.:	FI2D-0032
Work Period:	October 1990 - December 1992
Coordinator:	K MÜLLER, PHDR/HT, KfK
	Phone: 49/7247/82 43 43 Fax: 49/7247/82 43 86

A. OBJECTIVE AND SCOPE

An existing advanced handling system (EMIR) will be used as a carrier system for various devices for concrete dismantling and radiation protection monitoring. It combines the advantages of long reach and high payload with highly dexterous kinematics.

This system will be enhanced mechanically to allow the use of different tools. Tool attachment devices for automatic tool exchange will be investigated as well as interfaces (electric, hydraulic, compressed air, cooling water and signals).

The control system will be improved with regard to accuracy and sensor data processing. Programmable logic controller (PLC) functions for tool control will be incorporated. The free field of the EMIR will be used to build a mock-up that allows close simulation of that scenario without radioactive inventory. Aged concrete will be provided for the integration tests.

Finally, the economical and technical effectiveness of the different methods will be assessed/evaluated.

B. WORK PROGRAMME

B.1. Basic concept investigation

B.1.1. Interface specification between tools and EMIR (KfK)
B.1.2. Investigation of tool attachment devices for an automatic tool exchange system (KfK)
B.1.3. Setting up of test parameters (All)
B.1.4. Literature review concerning tool holders, adapters and tool replacement (KA)
B.1.5. Selection of the tool replacement system (KA)
B.1.6. Microwave equipment; design concept and interface specification (AEA)
B.1.7. Literature review on automation and measuring (BAI)
B.1.8. Selection of the type of radiation detector (BAI)
B.1.9. Definition of contaminants (BAI)
B.1.10 Design of the mechanics involved (BAI)
B.1.11 Electronics design for a noisy and dirty environment (BAI)
B.1.12 Conception of the hardware requirements for the computing system (BAI).

B.2. Development of tools

B.2.1. Development of a tool positioning sensor (KfK)
B.2.2. Design and manufacture of a sensor equipment (KfK)
B.2.3. Examination of kinematic requirements (KfK)
B.2.4. Enhancement of control system (KfK)
B.2.5. EMIR hardware enhancement (mechanical and non-mechanical interfaces) (KfK)
B.2.6. Setting up and optimisation of test parameters (KfK)
B.2.7. Adaptation of a commercial tool replacement system or development of an appropriate system (KA).
B.2.8. Development of a tool replacement adapter system suited to EMIR requirements (KA)

B.2.9. Installation of radiation measuring instrument plug connectors in the adapter (KA, BAI)
B.2.10 Tool holder rack design and development (KA)
B.2.11 Tool adapter plates, tool store and tool replacement equipment manufacturing (KA)
B.2.12 Assessment of treatment of specific cutting effluents (KA)
B.2.13 Provision of representative microwave equipment (AEA)
B.2.14 Manufacturing of measuring system for representative alpha-beta isotopes; software development adapted to detectors (BAI).

B.3. <u>Adaptation of tools and experiments</u>
B.3.1. Setting up of a representative test mock-up (KfK)
B.3.2. Tool integration and testing (All)
B.4. <u>Data evaluation</u>
B.4.1. Evaluation of test results concerning EMIR (KfK)
B.4.2. Evaluation of test results concerning mechanical tools (KA)
B.4.3. Microwave data evaluation (AEA)
B.4.4. Measuring system qualification (BAI)
B.4.5. Final evaluation and recommendations including specific data on costs, work time and occupational exposure and estimates of secondary waste arisings (All).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary

In 1992, the free field test site was installed with all components as tool bench; tools and their coupling plates; concrete testbeds for the experiments and auxiliary equipment for the microwave device.

Tool integration and testing took place with delay because the wiring and cabling along Emir arm for all participants took much more time than foreseen.

The experiments were solved with a certain delay end Oct/Nov., so the data therefore are partly still under evaluation.

The cooperating institutions performed the design reports of their tool developments, and first resulting statements arrived. They will be put together in the final report of EMIR work in detail.

Progress and Results:

KfK-activities /1/

1 Carrier system

The carrier system is an existing, hydraulically actuated large-range manipulator. It was enhanced in the course of the EMIR-project as a carrier system for decommissioning tasks. The enhancement comprised the following areas:

- Extension of the hydraulic circuit
- Task specific sensor systems
- Enhancement of the manipulator control.

1.1 Hydraulic extension

The original hydraulic circuit of EMIR had the sole purpose of driving the machine.

The tools to be adapted to EMIR, as the hydraulic hammer, the hydraulic shear, the core drill and the coupling units of the tool exchange system required additional hydraulic pressure of varying levels for actuation.

So on the last arm of the boom two hydraulic blocks were added.

Block I controls the tools and reduces the EMIR pressure of 32 MPA to a tool pressure of 15 MPa. The servo-valves allow the control of the flow between 0 l/min - 70 l/min. The hydraulic shears pressure level of 63 MPA is generated on the B-plate of the tool.

Block 2 controls the coupling units. The EMIR pressure of 32 MPa is reduced to 13 MPa. The functions of the coupling units are initiated by means of two solenoid-controlled valves.

1.2 A-plate and B-plates

The A-plate with all components like weather-proof housings for the LASER sensors, the inclinometer and the B-plates for all partners have been manufactured at the KfK shop floor and mounted on EMIR-wrist (**Fig. 1**) A-plate or onto the specified tool (B-plate).

1.3 Sensors

The manipulator control is based so far on 12 Bit resolvers for angle measurements in the joints and pressure sensors to measure the hydraulic pressure in the cylinders. To accommodate the positioning and orientation accuracy requirements of the tool exchange system and of the tools in operation, the instrumentation of the machine has to be considerably enhanced.

A) Upgrade of swivel joint resolver
The azimuthal positioning accuracy depends heavily on the accuracy of the resolver of the swivel joint in the tower axis of EMIR. This accuracy is crucial for docking process since the tool bench is set up in radial direction with respect to EMIR. To improve this parameter, the resolver was equipped with a 16 Bit resolver electronic unit. This upgrade improved the positioning accuracy in azimuthal direction from about $\pm$ 30 mm to about $\pm$ 5 mm.

B) Installation of inclinometer
In order to achieve utmost parallelism of A- und B-Plate during the docking process, a two axes inclinometer was installed on the A-plate.

C) Installation of four Laser proximity sensors
As the internal sensor systems of EMIR, the afore-mentioned resolvers do not allow for sufficient positioning accuracy in the docking process and during approach of the concrete surface, the enhanced manipulator control relies on sensor information that provides the distance relative to the approached object. The distance data are provided by four LASER proximity sensors mounted in the four corners of the A-Plate. These LASER sensors are high performance devices with a range of 100 - 2000 mm and an accuracy of $\pm$ 3 mm.

The purpose of these four LASER sensors is twofold:
- In the docking process the LASER sensors measure the distance to the tool bench, thus allowing for an approach in small incremental steps.
- In the approach of a concrete surface proximity, data provided are evaluated to derive the orientation of the wrist relative to the concrete surface.

D) Installation of force/torque sensor
A force/torque sensor array developped earlier in KfK (pat. pending) was integrated with the tool exchange system. This force/torque sensor has a range of 0 - 20KN and 0 - 20 KNm. It is mounted at the very end of the arm. The main purpose of the force torque sensor is to allow force control of the manipulator during docking and during processing of the concrete surface.

1.4 Manipulator control

The EMIR manipulator control system was upgraded to accommodate the requirements of the decommissioning application. This implies:

- Coarse motion and fine motion control for docking.
- Sensor data fusion. Application of different sensor systems to evaluate the position and orientation of the wrist.
- Data exchange with the contamination monitor computer, in order to calculate a map of contamination distribution on the concrete surface.

2. Tool exchange system

The tool exchange system consists of a rigid steel frame, equipped with tool support frames on springs. All tools have their specific position and can be taken and laid down easily (**Fig. 2**).

KAH activities:

All tools were manufactured and worked within the specified values with good accuracy. The detailed design is described in /2/.

Summary of Results
The B-plates for all KAH tools (also AEA and BAI) were successfully coupled and the tools tested for the remote-controlled application.

Hydraulic Shears
By the first cutting attempt, the hydraulic shears rotated in their guide so that the cutting edges of the jaws became aligned on the test pipe axis and did not cut effectively.
The three bolts through the guide, initially intended as a means of radial adjustment for the shears, were pulled up against the housing of the shears thus eliminating any torsional movement of this tool.
Subsequently, up to diameter 2", pipes were cut without the slightest difficulty.

Hydraulic Hammer
The hydraulic hammer was tested on two 15 cm thick concrete slabs of quality B25 each with a single row of reinforcement.
For the first test the slab (edges vertical) lay in the horizontal plane on supports. With the chisel axis vertical, the hammer approached the test slab from above.
Two holes were hammered through this slab in 12 and 20 seconds respectively. No detrimental effects to EMIR (resonance) could be observed during this test.
For the second test the slab was held at an angle of 45° to the longer side and could be approached from below.

Crown Drill
The crown drill was tested on two 15 cm thick concrete slabs of quality B25 each with a single row of reinforcement.
The first test utilised the horizontal slab previously used for the hydraulic hammer tests.
The holes of diameter 201 mm were bored through the slab and reinforcement, each requiring approximately 5 minutes.

For the second test the slab was held vertically (edges horizontal) and approached with the axis of the crown drill normal to the slab face.

Again two holes of 201 mm diameter were bored through the slab and reinforcement, each requiring approximately 10 minutes.

Conclusions
- The B-plates and all accessories (centering pins and coupling units) were adequately designed for their intended function. (Load carrying capacity and stiffness).
- The geometry and positioning of the coupling and centering units on the A/B plates matched up exactly.
- The coupling/decoupling procedure and the simultaneous coupling of the supply lines functioned without problems.
- The fixing capacity of the modified vacuum plate for the crown drill was substantially more than required for this particular task.
- The control panel functioned in all respects; opening/ closing of valves, activating electromagnets, control of tool performance.

More detailed experimental data will be provided in the final report.

AEA-activities: /3/

Tuning. The microwave tool was handled by EMIR without problems despite of 10% overweight. The method suffered a little from dust arising within the beam guide and auxiliary equipment was sensitive to weather (humidity).

First results are:

The tool was first placed at the concrete face, and run at minimum power (10 kW) for manual adjustment of the tuner. This was successful, as subsequent tests gave less than 2 kW reflected power.

Spalling. Table 1 gives all the results. Test 1 was a continuation of the above tuning work, and at the same spot. An 8-shaped spall resulted, indicative of the standing wave pattern in the shroud. The lower part of this hole shows how penetration terminated, 40 - 45 mm deep, at horizontal metal members. This showed that the whole panel was reversed, with the deep, vertical reinforcement at the back.

Test 2 was the most important of the series, as it attempted a traverse motion in which the microwave seal was laid partly across the rough surface generated by Test 1. Only inconsequential local microwave leakage was observed (Table 1) and the test proceeded steadily. It was stopped after 7 explosions only for of caution reasons.

Test 3 was a standardised repeat of Test 1, using a fresh face rather than one preheated by tuning work at 10 kW and with a duration of ten explosions to ensure that the volume of debris would be easily accommodated within the lower part of the shroud. It proceeded as planned (see Table 1). The final explosion was exceptionally violent and the microwave horn was afterwards found to have its upper edge dented by debris.

Test 4 was intended to be another traverse test, partly overlapping the Test 2 site in a sideways direction. However, placement problems with EMIR were particularly marked on this occasion, and the test itself was later stopped, owing to repeated trips of the generator, signalled as 'generator arcs'.

Table 1
Summary of test results for microwave tool attached to EMIR at KfK

Test number		1	2	3	4
Duration		16 min.	132 sec.	185 sec.	18 min.
Number of explosions		4	7	10	0
Test terminated at/or by		Temporary arc	Caution	"10-bang" test	Generator arc
Power level	kW	10 and 50	40	40	variable
Power, maximum	kW	50	40	40	40
Energy used, approx.	kWh	4.2	1.47	2.06	1.47
Debris swept up,appr.	kg	4.14	2.47	3.3	0
Volume concr.rem.	10^{3} m^{3}	1.76	1.047	1.4	0
Efficiency	litres per kWh	0.419	0.71	0.68	0
Radio interference, dB over 1 μV/m		10	not measured..................................		
Point leak/seal	mW cm^{2}	0.5 to 2	< 0.5	< 0.5	< 0.5

Demolition efficiency. Table 1 shows that spalling was proceeding with an efficiency of 0.4 to 0.7 litres removed per kWh expended, which is very satisfactorily.
Microwave emission. Leakage was checked with personnel monitors at all times but nothing was found apart from highly localised point sources close against the Knitmesh seal. There was only one of these found on each test. Radio interference was checked on Test 1 by KfK staff, using a sensitive multi-channel, omni-directional aerial system placed at the standard test distance of about 30 m. No harmonics were found, while at 896 MHz only 1 dB above 1 μV/m was registered. The maximum permitted level is 50 dB above 1 μV/m. As the dB scale is logarithmic, it is evident that at present emissions are well within all international limits for interference with radiocommunications.

Manoeuvrability. EMIR succeeded in carrying the tool without trouble, thus fulfilling the main objective regarding a mobile tool. Positioning capability was affected by the magnitude of the load however.

Conclusions
A microwave demolition tool has been successfully mounted on the EMIR long-range manipulator. In addition, tests have been carried out that have met the objectives. Thus,

- Manipulation with EMIR is more than adequate as regards strength capability, though some improvements are to be sought during later development as regards the manoeuvrability and position control.
- Effectiveness of the new tool horn as regards spalling efficiency is rather better than predicted.

- Microwave leakage is at present under very good control, giving insignificant radio interference even with only the shielding provided on the tool itself. Tools with continuous debris discharge have yet to be evaluated.

Fig. 3 shows the microwave tool mounted on EMIR arm.
Fig. 4 shows the horn and sealing in detail

BAI-activities: /4/

Detector Description
For the detection of ALPHA and BETA contamination on large surfaces, a gas-filled proportional counter is best suited.

The selected detector for the BAI 9313 contamination monitor is the LB 6358 Alpha-Beta counter, manufactured by Labor Berthold GmbH. This proportional counter is the latest state-of-the-art for alpha beta surface contamination monitoring. Furthermore, the high voltage, the preamplifier and integral discriminator are integrated in the detector.

The contamination monitor BAI 9313 uses six (6) LB 6358 proportional counters in order to enlarge the total active measuring area. The detector gas supply is daisy-chained, and has an ON/OFF controlled gas flow. This minimises the gas consumption significantly. Minor modifications are made to improve the gas flow.

Intelligence of the Monitor
It is very important that the contamination information can be relayed to a logging computer with noise-free signal transmission. Therefore, the monitor will contain all the required intelligence to process the detector signal information. Between the remote computer and the monitor, a minimum of electrical signals are used.
At any time, the monitor may be disconnected or connected without danger.

Mechanics
The BAI 9313 is equipped with a special metal flanges, compatible with the EMIR telemanipulator arm end. On this flange, snap lock electrical couplings systems are foreseen with a mechanical guidance. The telemanipulator is programmed to grab the detector, controlled by a 486 computer. The detector, when not in use, is docked in a special constructed tool-bench.

A study metal grid protects the Mylar window from the six proportional counters. The distance between detectors and contaminated surface is as small as possible to optimize the contamination detection. This grid is mounted on four spring-loaded switches, at the four corners of the monitor housing. These switches sense the monitor collision and feed a signal back to the remote computer.
The weight for the total monitor is approximately 50 kg. The special interface plate to the EMIR telemanipulator weights about 80 kg.

EMS Provisions
Although no precise data is available from the setup, significant Electro-Magnetic Radiation levels can be expected in general during decommissioning works. Therefore special precautions have been taken to optimize the Electro-Magnetic Susceptibility (EMS) of the detector. Basically, all BAI equipment is designed conform to IEC 801-2 to 801-5. These specifications are in accordance with VDE 0871 and CISPR-22 recommendations.
The electronics has a double metallic shield of 100%. The low voltage input has a double, current compensated EMI filter. The data communication output is RS 422 over a shielded industrial cable.

Computer System

Data Communication
The data communication protocol between BAI 9313 and computer system is an IN-HOUSE standard called F^2C. Over 350 instruments are installed in the world with this data communication system. F^2C is an instrumentation protocol for asynchronous point-to-point transmission. Simple status information, instrumentation parameters, measuring results, commands, as well as large data volumes, are handled efficiently.

The transmission speed is 9600 bps FDX 8N1.
The central computer system controls the detector measuring cycle.

The minimum hardware requirements are:

PC-AT	DOS 3.2 or higher
Conventional memory	640 kB
Disk	3.5"
Hard disk	10 Mb
Monitor	VGA Colour
Serial interface	COMI RS 232 connection to BAI 9313 detector system
Serial interface	COM2 RS232
Mouse	PC bus IRQ5

Computer Software
The Software is written in TURBO PASCAL and runs on any IBM-compatible machine, type PC AT. The software is mouse-driven and extremely simple to operate.

The screen represents the concrete surface which is to monitor, and is divided into the coordinates, corresponding to the physical locations on the concrete wall (**Fig. 5**). Each time a contamination measurement is finished (Alpha and Alpha/Beta cycle), the results are compared with the contamination levels 0.4 and 0.04 $Bq.cm^{-2}$. The contamination level is converted using a colour scale of 4 grades between grey and red and the coordinate will be coloured conform to the activity level.

6. Testing

The EMIR telemanipulator arm is first guided to the contamination monitor device, which is docked in a tool bench. Via smooth path control, the arm end is coupled with the contamination monitor. Mechanical and electrical contacts are locked automatically. Stainless steel test plates with embedded radioactive sources are mounted at random on the concrete wall, covered by an aluminium plate. The EMIR telemanipulator will scan the concrete wall in a predefined sequence. The monitor will measure the 'contaminated areas' and report the readings to the remote computer system.

Two source types are used:	1 nCi	(37 Bq)	Am241
	3 nCi	(111 Bq)	Sr 90

The sources are embedded in stainless steel plates (100x100 mm). The sources are sealed such that the activity levels 0.4 (Beta) and 0.04 (Alpha) $Bq.cm^{-2}$ can be tested with the monitor. (s. **Fig. 6**)
All results are obtained with 30 s counting time. The data is collected in open air, with the test plates in contact with the detector grid surface.

Evaluation of the results

1. Alpha minimum detection levels.

Alpha background count rates are very low. The results are prone to the resolution of the digital counters. Any significant increase above zero background, can be measured as a contamination level.

When a 10% accuracy is accepted (about 100 counts), the 0.04 Bq.cm-2 clearance level can be measured in about 300 seconds assuming that the contamination is smeared over an area of minimum 300 cm^2.

2. Beta/gamma minimum detection levels.

Background count rates are typically lower when detectors are faced to the wall. In practical situations however, the natural activity of the concrete wall may influence the readings as a background contribution. (Spiers & Gibson). The minimum detection contamination level, is = 0.04 $Bq.cm^{-2}$.
This detection level is about 10 times lower than the recommended clearance levels by EC and others. Background levels have to be measured on the location.

8. Conclusion

The ruggedness proved to be very good. The mechanical strength of the detector setup was no problem, even when inaccurate positioning lead to higher forces then foreseen. The detector gas supply autonomy is largely sufficient for decontamination monitoring jobs of several days.

The active detector surface should be increased. The telemanipulator interface plate can be an obstacle for corners. High humidity had little or no effect on the detection.

Time schedule and work programme

All points of the scheduled work programme are solved now, except point B.4.5. "Final data evaluation" which are to be included into the final report. This report will discuss the main design aspects and the results obtained. It will be distributed March 93 as a draft to CEC.

Internal reports received:

/1/	Auslegungsbericht EMIR H. Gengenbach; E. Ernst	Dez. 92 HT-Bericht 23-93
/2/	Design report KAH-Tools A.J. King	Dez. 92 HT-Bericht 21-92
/3/	Microwave Demolition tool for EMIR P.F. Wace; R.A. Shute	Dez. 92 Draft-paper SMIRT 12
/4/	Alpha-Beta Contamination monitor F. Withagels BAI	Jan. 93 HT-Rep. 22-93

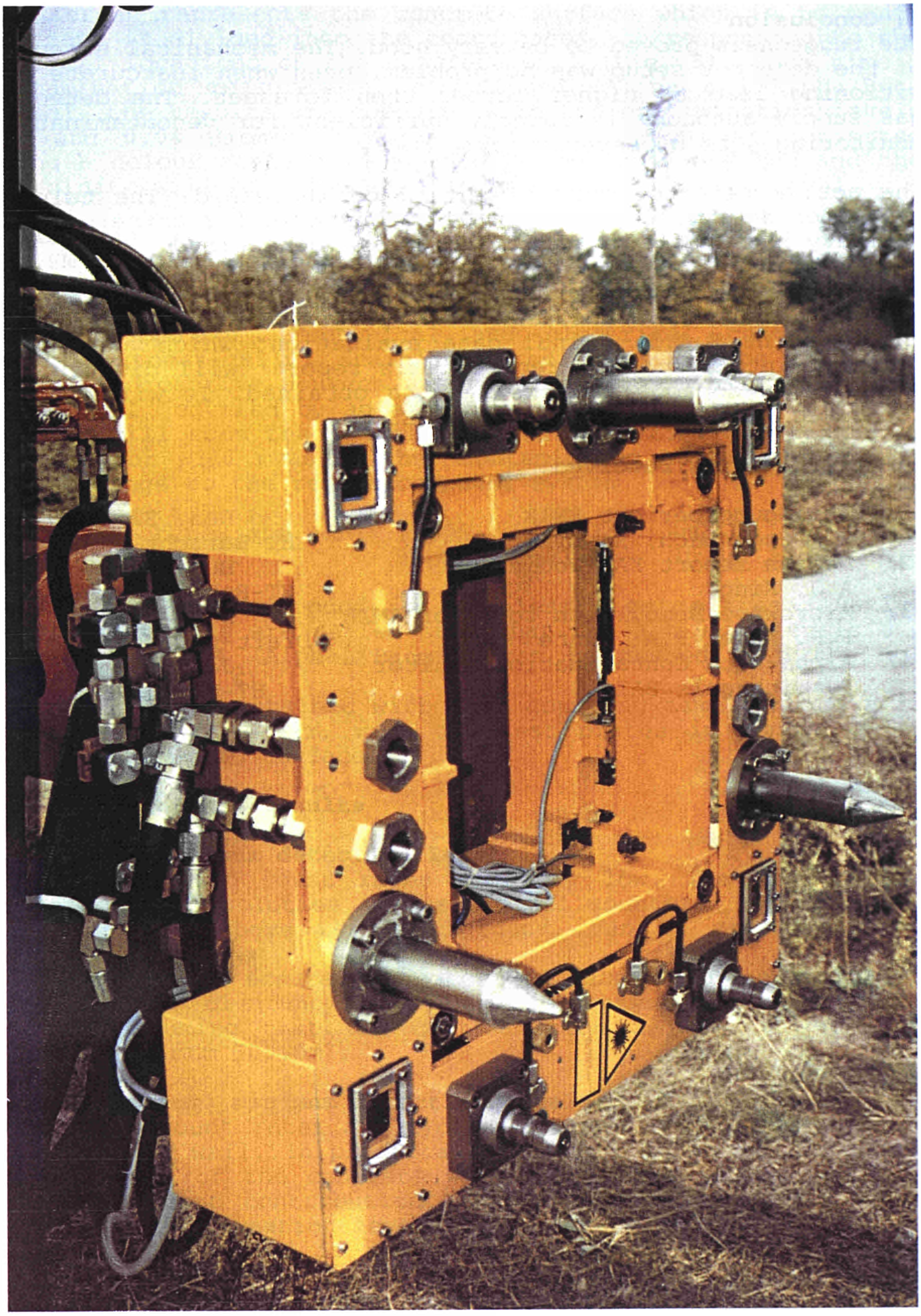


Fig. 1: Complete A - plate with installations

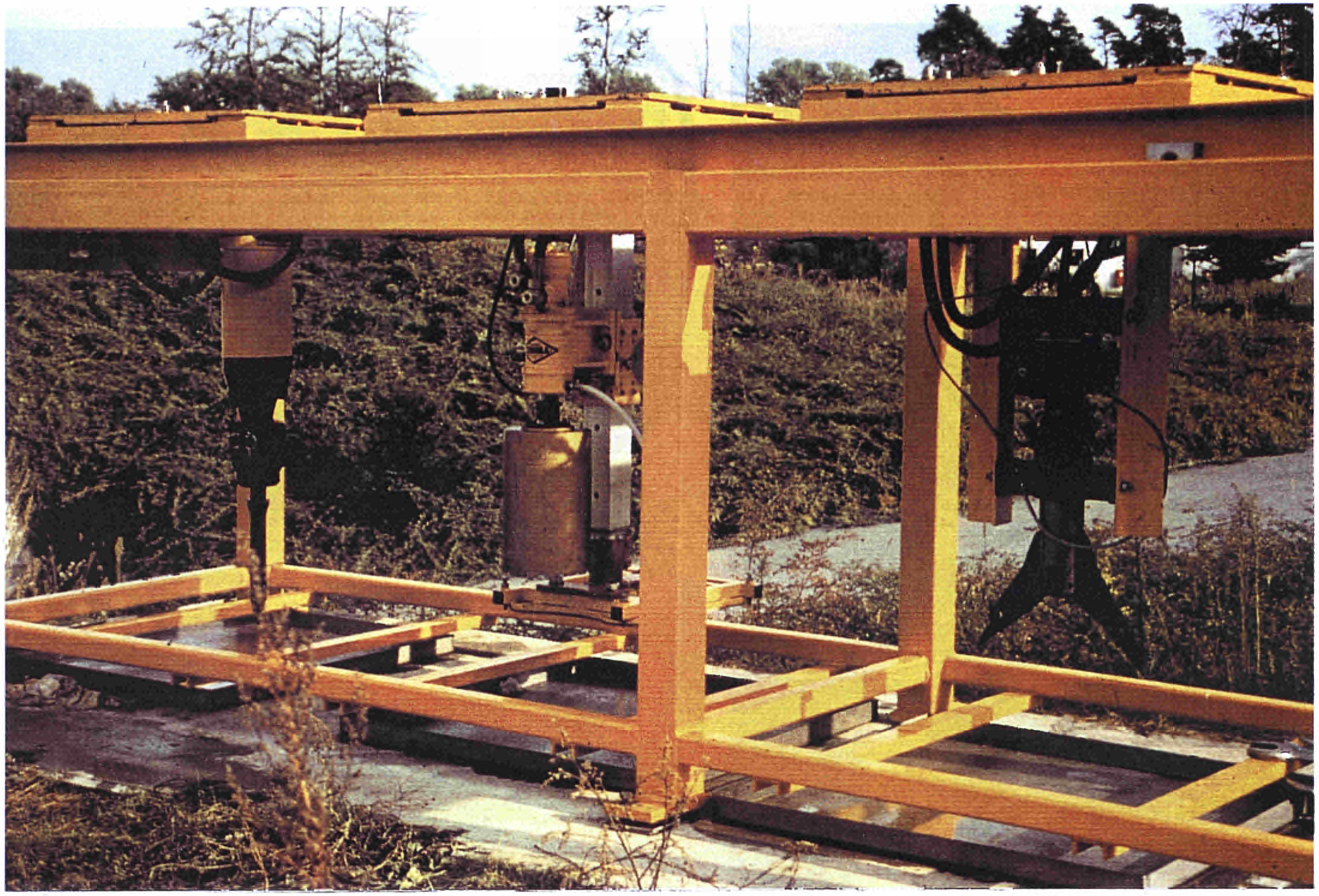


Fig.2: Tool bench with tools

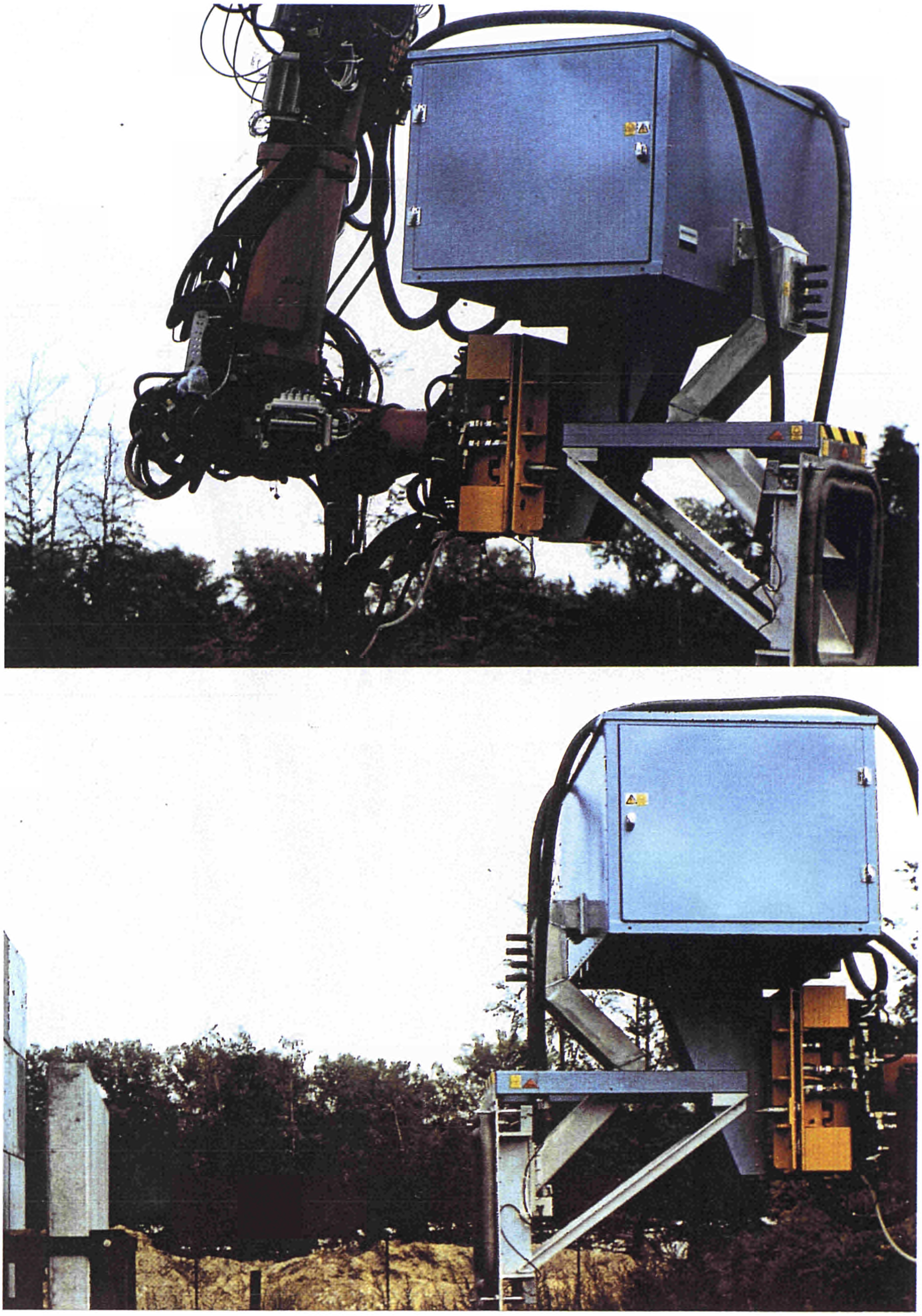


Fig. 3: Microwave tool with horn

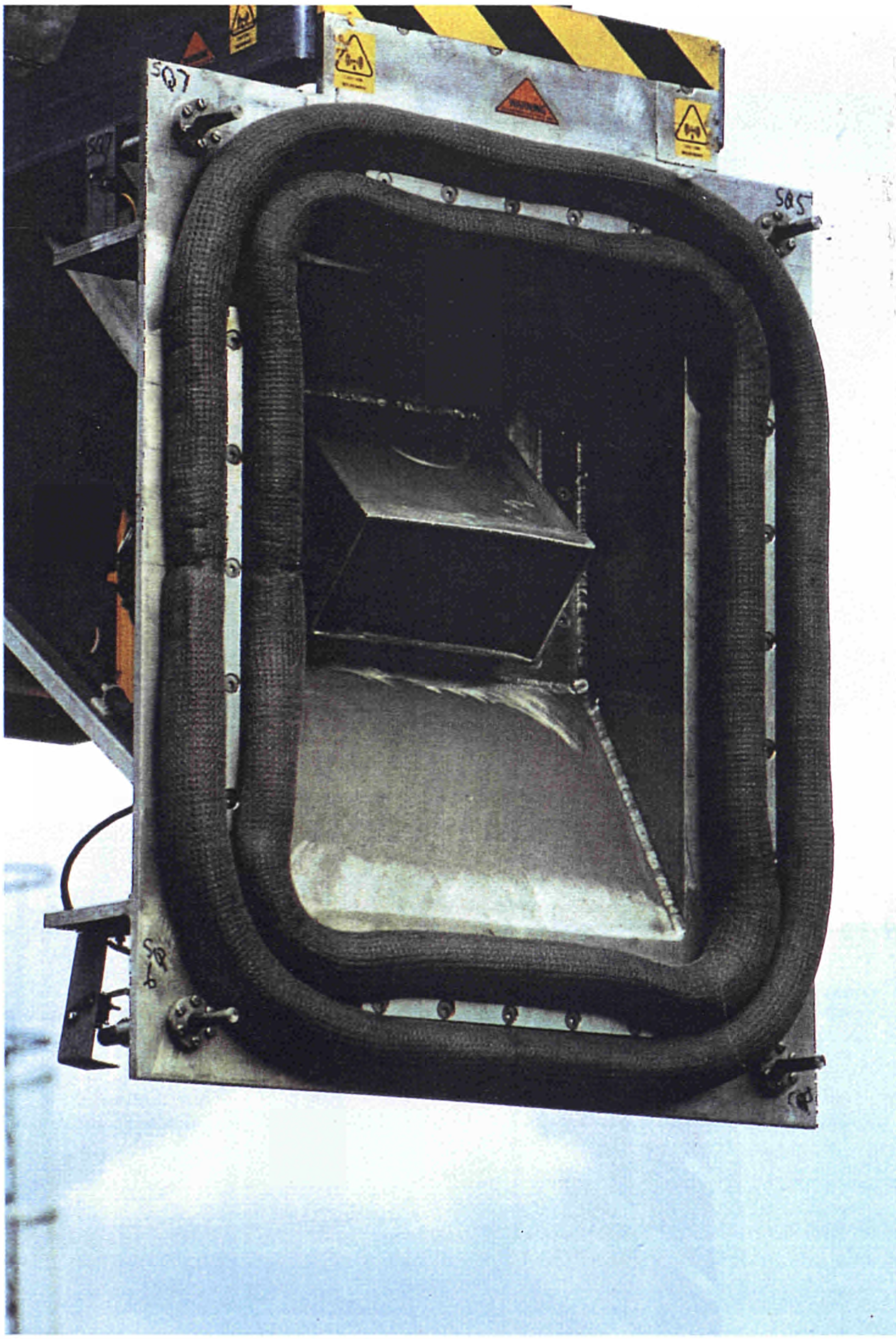
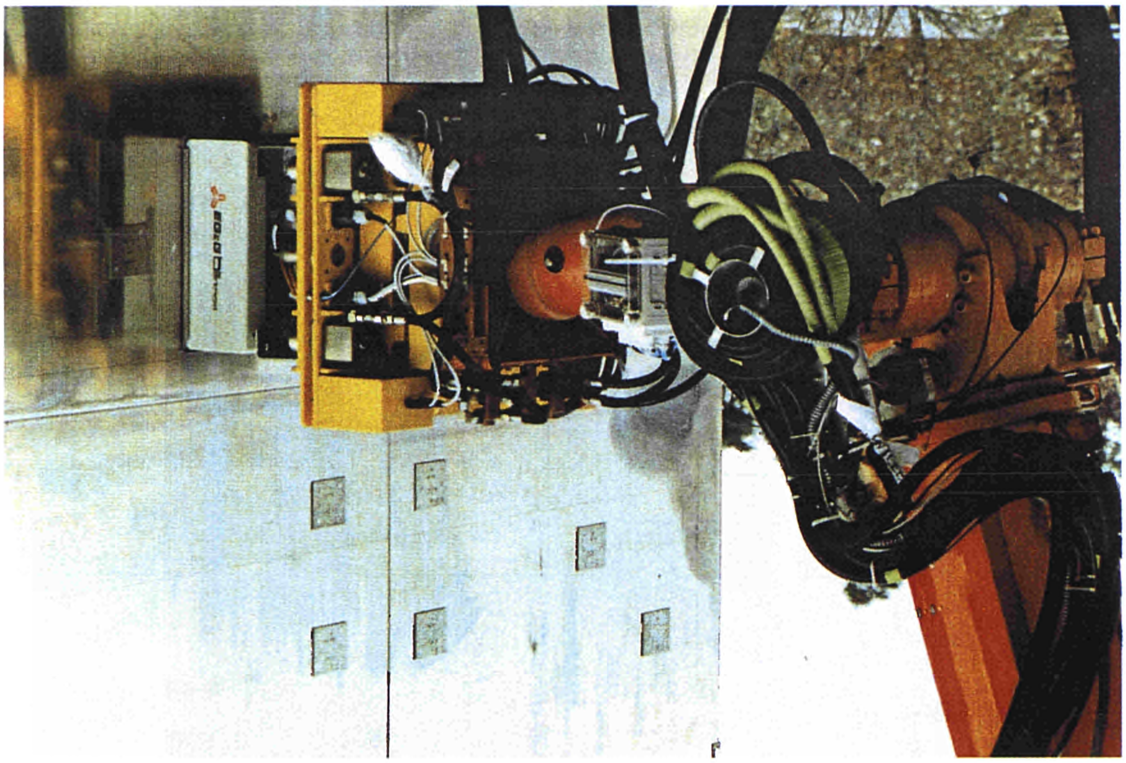


Fig. 4: Horn and sealing



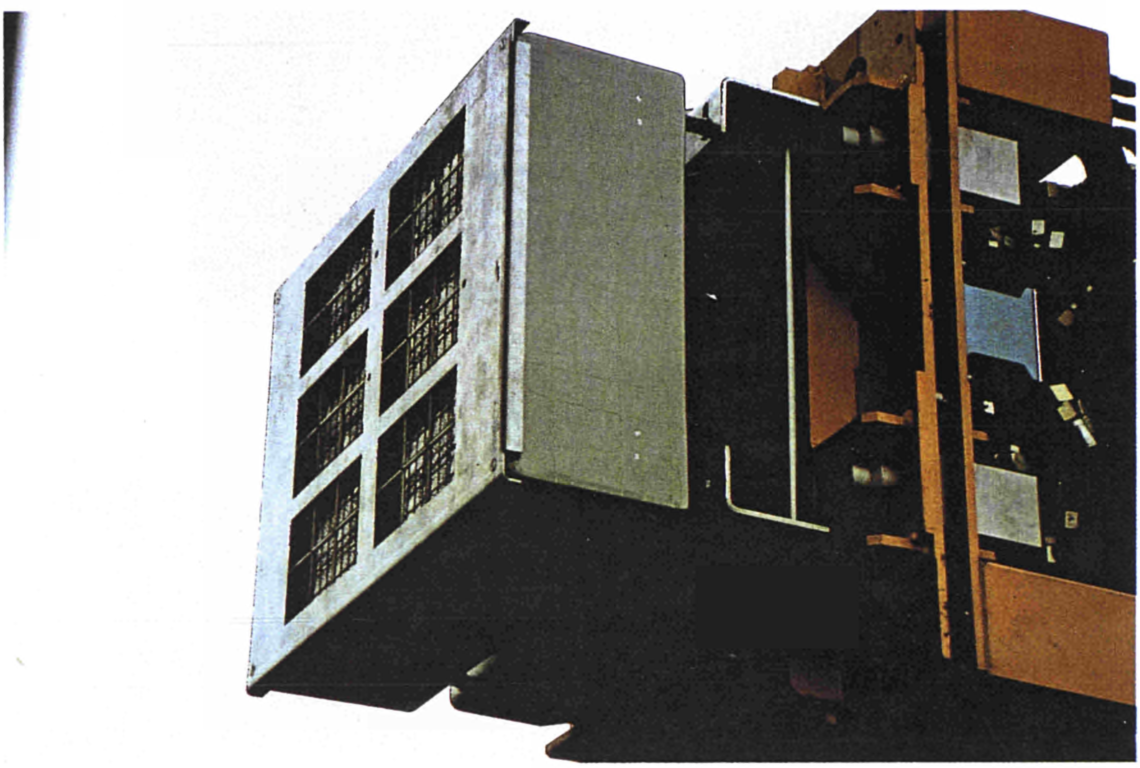
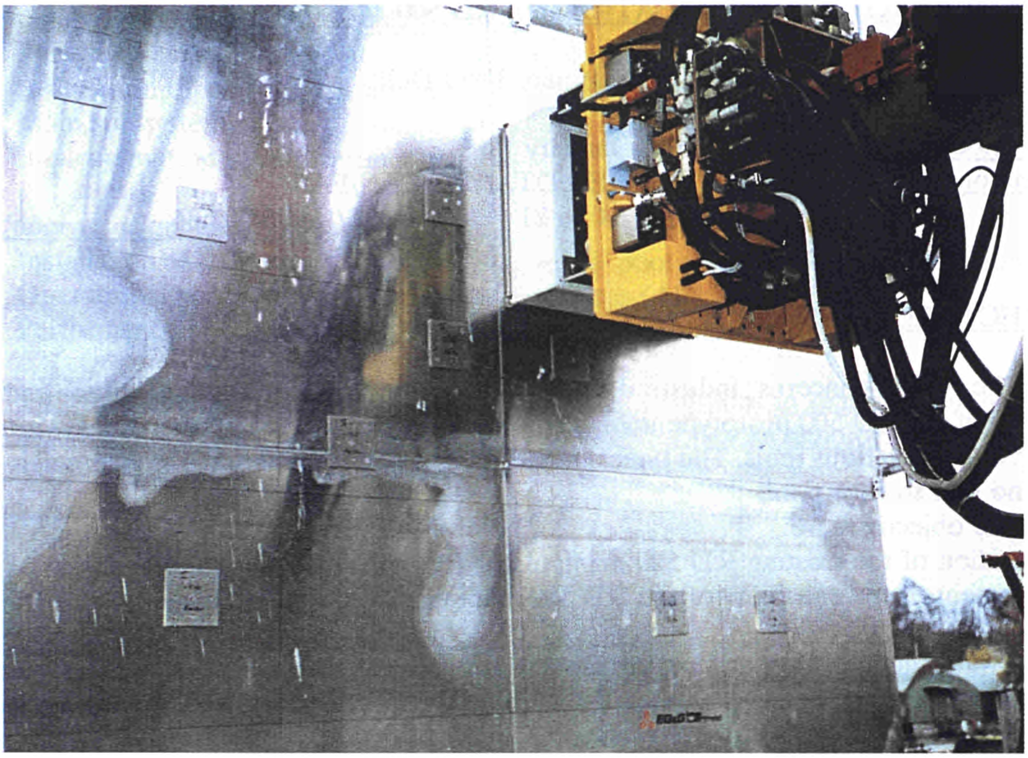
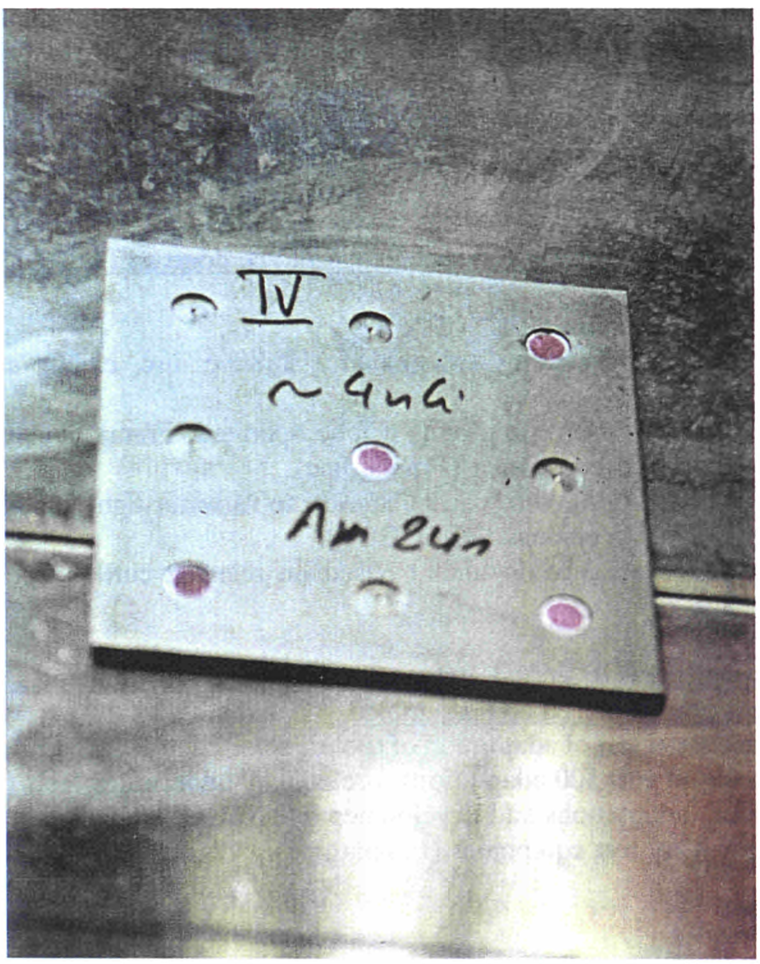


Fig.5: Monitor array at the wall



a)



b)

Fig. 6: Plate equipped

a) with test sources

b) detail of source

5.6. UNDERWATER QUALIFICATION OF RD 500 MANIPULATOR

Contractors:	CEA FAR, Framatome, TNO Delft
Contract No.:	FI2D-0041
Work Period:	October 1990 - January 1993
Coordinator:	M MAUPOU, CEA/DTA/UR, CEN/FAR
	Phone: 33/1/46 54 86 21 Fax: 33/1/46 54 02 36

A. OBJECTIVE AND SCOPE

The work concerns industrial-scale underwater experimentation in non-radioactive conditions of the RD 500 prototype telemanipulation system, which has been already extensively tested in air with various tools. The typical nuclear dismantling environment concerned is a LWR vessel and fuel storage pool.

The objectives are:

- Adaptation of the existing RD 500 manipulator for underwater dismantling tasks;
- Assessment of the capability of the RD 500 manipulator to operate under water with various tools;
- Underwater qualification and performance assessment of a new ultrasonic imaging system;
- Qualification of the complete system by an in-field application and definition of an industrial underwater RD 500 system.

The research work will assess the feasibility of underwater dismantling operations, the performance of the computer-assisted modes of control and the assumption that the RD 500 system can be more effective than hands-on work in relevant decommissioning environment.

The CEA-UR will coordinate the research work. Subsidiary companies of the CEA and Framatome (SNE La Calhène and ATEA) will perform specific technical adaptations on the RD 500 systems and the underwater qualification tests.

B. WORK PROGRAMME

B.1. Identification of underwater requirements and specification to be done on the RD 500 and the vision system.

B.1.1. Identification of relevant underwater tasks (CEA).
B.1.2. Selection of appropriate tooling systems (plasma arc, abrasive disc, electro-erosion) (CEA).
B.1.3. Definition of test mock-ups on which the tooling will be operated (Framatome, CEA).
B.1.4. Specification of the auxiliary test equipment (Framatome).
B.1.5. Specification of RD 500 adaptations, with particular view to its water-tightness (CEA).
B.1.6. Specification of the optical vision systems (TNO, CEA).
B.1.7. Drafting of a qualification procedure document based on relevant cutting operations (CEA).

B.2. Preparation of the preliminary tests in air and under water; the basic hardware and software will be developed/adapted, manufactured and assembled

B.2.1. Study, manufacturing and shop test of adaptation of tooling selected in B.1.2. (Fr.+ CEA)
B.2.2. Design and manufacturing of RD 500 adaptations; preliminary underwater tests (CEA).
B.2.3. Vision systems acquisition, adaptations and developments (TNO, CEA).
B.2.4. Manufacturing of the auxiliary test equipment (Framatome).

B.3. Preliminary testing of the complete system

B.3.1. Individual air and underwater testing at each partner's laboratory (All).
B.3.2. Installation of simplified test mock-ups for main sub-system testing in air (Fr.+ CEA).
B.3.3. Main sub-systems testing (Framatome, CEA).
B.3.4. Implementation of improvements (All).

B.4. Underwater qualification tests

B.4.1. Installation of the various equipments in a water pool at ATEA/Framatome (Framatome).
B.4.2. Operational verifications of the complete system (All).
B.4.3. Performance of the qualification tests as defined in B.1.7. (Framatome)

B.5. Final evaluation and specifications with respect to conditions in real dismantling projects; evaluation of the costs of an industrial RD 500 system and of its radiological impact on work force and working area (All).

C. Progress of work and results

Summary of main issues

In the beginning of 1992 the project statement was the following :

- 3 generic underwater dismantling tasks associated with their process equipment (tools, vision, pressurized system, mock up,...) have been defined, they consist of :
  - a mock-up of the lower internal structures of PWR vessel and an electro-erosion tool, for a task of bolt removal ;
  - a mock-up of the thermal baffles of the outlet sodium pipes of RAPSODIE and a plasma torch, for a task of plate cutting ;
  - a mock-up of the bolted connection of the grid on the RAPSODIE vessel and a grinding tool for a task of bolt assembly cutting.

- An ultra sonic vision system was defined to aim at the provision of an image when the optic vision is disturbed by environmental conditions.

The work performed during the year 1992 mainly includes the tasks B2. and B3. It consists in :

* manufacturing and testing the RD500 underwater adaptations
* supplying and modifying the cutting tools, plasma torch and grinding machine
* manufacturing and testing a specific electro erosion tool
* designing and manufacturing the ultrasonic camera
* manufacturing the mock-ups.

After preliminary individual tests of these various subsystems an integrated test with the robot and the tools has been performed in the "Minerve" pool (CEA facility in Fontenay aux Roses).

These tests have permitted to define all the computer assisted teleoperated (CAT) functions necessary to operate the dismantling tasks, to program the robot with these functions, and to check the performances of the overall system.

These tests have shown the global system validity (all the tasks have been performed with a good quality of result) and allowed the definition of the improvements necessary to optimize the demonstration.

Progress and results

1. Supplying, adaptation and test of the grinding tool (B.2.1) (B.3.1)

The chosen tool is a STANLEY hydraulic powered grinding machine, it is a specific design adapted to be operated underwater.

Individual tests to check the tool working and to determinate the operational parameters were performed. No specific problem appeared during these tests.

2. Supplying, adaptation and test of the plasma torch (B.2.1) (B.3.1)

The plasma torch and its power alimentation are manufactured by NERTAJET. A specific underwater working kit was adapted to the torch. Various tests in a water tank were performed to fix the parameters. We have decided to choose non-nominal working conditions to simplify the operation. (We don't use water vortex but the cutting speed is notably decreased). The plasma gas is formed by argon (60 %) and nitrogen. (40 %) with a flow of 50 to 60 l/mn. The cutting speed is 400 mm/mn.

3. Study, manufacturing and test of the electro erosion tool (B.2.1) (B.3.1)

A specific design of the tool, taking into account the accessibility constraints in the internal structures of the reactor, has been made.

After its manufacturing, tests to check its mechanical motions and water-tighness of its casings were performed.

Machining operations with the tool in normal and demineralized water permitted to conclude to the necessity of a demineralized water injection around the electrode to obtain a correct working of the tool. This improvement made, the operational parameters were defined. The duration of a bolt machining is 30 to 55 minutes depending of the location precision of the tool in front of the bolt. The flow of demineralized water is 40 l/mn.

4. Study and manufacturing of the mock-up (B.2.4) (B.3.1)

For the qualification three mock-ups are needed associated with the three tools. They are compatible with a common support basis and used one after the other. The three mock-ups and the support were studied and manufactured. The consumable pieces are removable and the same equipment will be used for the final qualification.

5. Study, manufacturing and test of RD500 Adaptations (B.2.2) (B.3.1)

Two kinds of adaptations are taken into account :

- Water tightness and pressurization

The adaptations mainly consist in adding to the robot the pressurized system (air supply, regulator, safety valve...) and a water proof umbilical, the sealing interface being on the electric connection near the robot. The safety is provided by an internal pressurization of 200 mbar. All these adaptations have been tested in air before a complete verification during the integrated test in the pool.

- Automatic tool exchange

To realize a dismantling task it should be necessary to use different tools. To use robotic assistance the reference of the tool with respect to the robot must be known with a good precision. Since the standard gripper cannot offer this quality, tools and robot are fitted out with a pneumatic clamp for all the manipulations. To simplify the operator activity, every tool gripping or removal being automatic, a support rack for the tools was set up on the robot basis.

All these improvements were defined, manufactured and tested in the MINERVE pool.

6. Vision systems acquisition, adaptation and developments (B.2.3) (B.3.1)

- Optical vision

The two robot cameras were purchased and fixed one on the body, the second on the forearm. The cameras are connected to the robot internal wiring.

- Ultrasonic vision

the ultrasonic camera, is a three dimensional camera that will serve as a supplementary vision system to a set of conventional video cameras. The system will be qualified for the tasks ; removal of welded keys securing bolts by means of the electro erosion tool, and cutting of metal sheets as performed by a plasma torch.

The core of the acoustical camera is the Scanner 250 of the PIE medical corporation. The probe selected for this application is based on a linear-array technique. To emulate a mechanical scanning along this linear array an electronic focusing of acoustical energy is used. To produce three dimensional images, a mechanical translation system is being developed that is controlled by three DC motors. Two of these motors are used for the (limited) rotation of the array, needed to keep this one in the correct position (i.e parallel to the metal surface of the object). The third one performs the scanning [Figure 1].

The computation of the data as well as the man machine interface is supported by a SUN Sparc station. A question concerning the MMI is : how to present a three dimensional volume of data in such a way that a human operator can interpret it ?

Different ways of displaying have been tested :

* reduced two dimensional images with two kinds of data (traveltime image, or reflecting image) [Figure 2] ;
* Perspective image, giving a presentation of three dimensional information, with the difficulty for the moment, that only the parallel frontal surfaces are displayed. A solution to this problem is to use a fore-knowledge of the bolt shape, and to calculate the exact position to generate a picture of how the bolt would look like from any point in space. An additional advantage of this procedure is that the position information will be necessary for a future tool work automatization.

7. Preliminary underwater tests with RD500 and the tools (B.3.2) (B.3.3)

To check the compatibility of all the subsystems as well as the system performances a three months test compaign has been done, using the CEA pool "MINERVE" [Figure 3].

The result of this compaign is to demonstrate the three dismantling tasks executed with the real procedure. The principal operator assistances are the following :

- for each tool ; the robot automatically takes and puts back the tool in its rack ;
- for the grinding tool ; after the cutting plan designation the robot locks its degrees of freedom out of this plan ;
- for the plasma torch ; after the learning of the trajectory (by palpation of six points of the cutting window). The robot automatically positions and moves the torch above the cutting line.
- for the electro erosion tool ; after the learning of the bolt plan the EE tool is automatically positioned perpendicular to this plan, a reference point is learning by vision in front of the bolt, and the tool is self positioned and "locked" by a force control on the plate.

The main lessons from these tests are :

- the easiness of operator tasks and the quality of the results once the procedures and parameters are adjusted ;
- the good working of the grinding and electro erosion tools ;
- the fragility and relative difficulty to operate the plasma torch ;
- the need to optimize the position accuracy of the electro erosion tool ;
- the need to increase the reliability of the tool exchange system ;
- the need of a maintenance concerning the robot water-tightness.

8. Final qualification preparation (B.1.7, B.3.4)

For each subsystem, the improvements have been defined, and their implementation have been started.

The definition and manufacture of complementary devices and mock-up adaptations have begun.

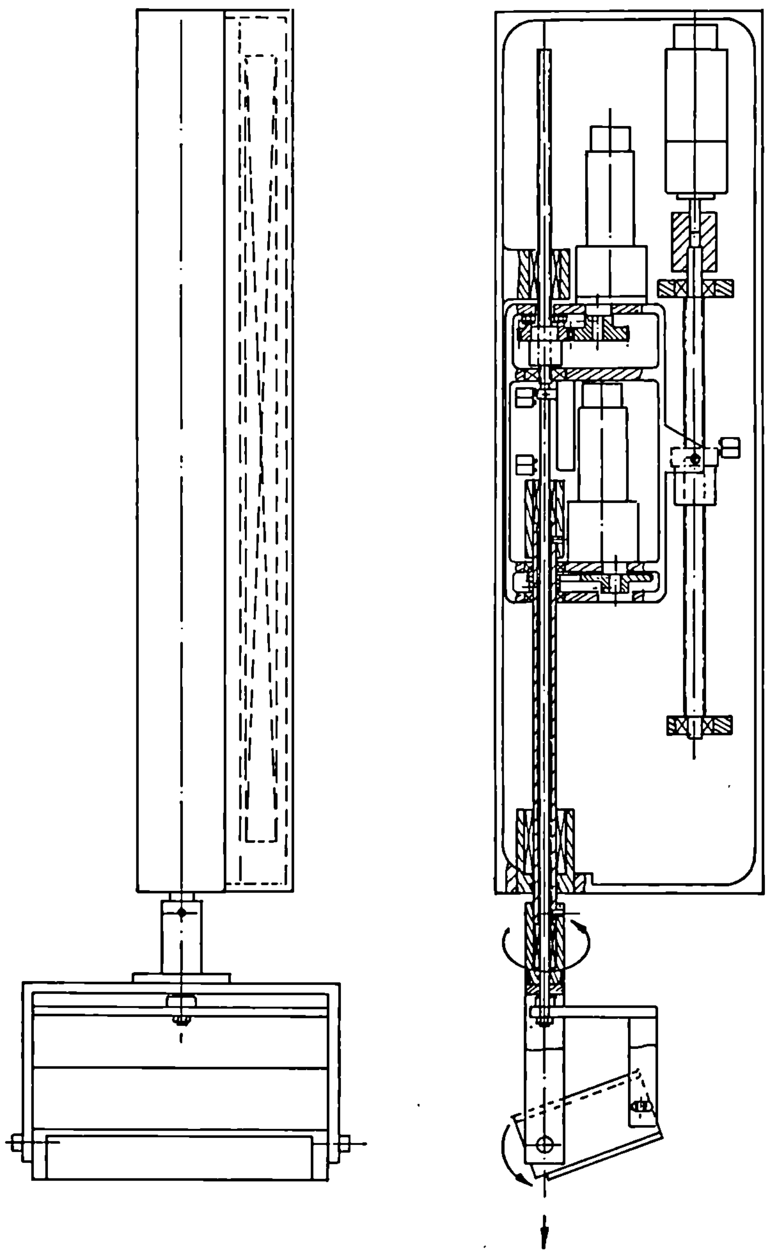


Figure 1 : Drawing of the system of DC motors used for the positioning and translation of the linear array

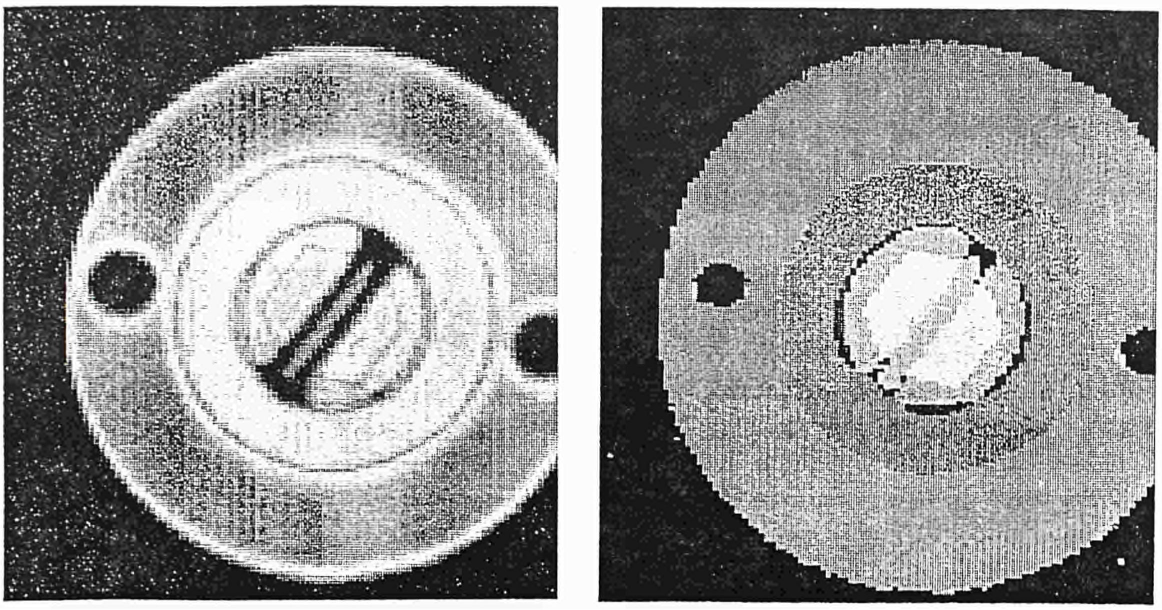


Figure 2 : Reflectivity (left) and Traveltime (right) image of the scanned mockup

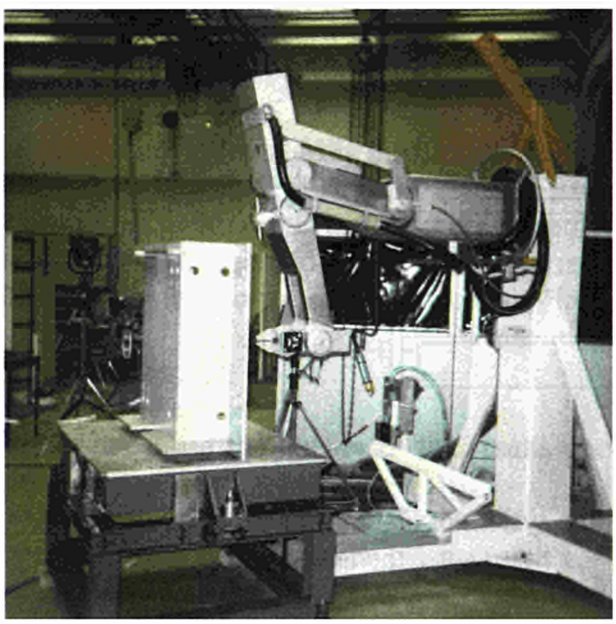


Figure 3 : RD500 Underwater test preparation

6. AREA No. 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 area 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. Subjects of the research performed under the previous programmes (1979-88)

Research has been performed in the following main areas:

- estimate of the quantities of radioactive waste arising from the decommissioning of typical nuclear installations, based on analysis of radioactive metal and concrete samples;
- study of strategies for the decommissioning of typical nuclear installations and for the conditioning/management of the radioactive waste arising therefrom;
- characterisation of the radioactivity associated with components and structures of various nuclear installations, with emphasis on long-lived radionuclides; in situ measurement techniques for the localisation and identification of radionuclides, including the case of mixtures of alpha, beta and gamma emitters;
- assessment of residual activity levels below which activated and/or contaminated parts could be reused and corresponding measurement methods.

C. Programme 1989 to 1993

Radioactivity measuring techniques should be improved/developed with particular regard to clearance procedures for materials, buildings and sites, including the case of mixtures of alpha, beta and gamma emitters. The quality assurance of clearance procedures should also be considered.
Strategies for the decommissioning of typical nuclear installations should be further studied, account being taken of the waste disposal facilities existing or planned in various member countries. Safety being one of the aspects to be considered, a methodology for evaluating the risk of decommissioning operations should be developed.
The evaluation of residual activity levels below which materials from decommissioning could be reused should be pursued, including consideration of statistical aspects.

D. Programme implementation

Nine research contracts relating to Area No. 6 were in progress in 1992, from which two were completed in the same year.

6.1. METHODOLOGY TO EVALUATE THE RISKS OF DECOMMISSIONING OPERATIONS ON NUCLEAR PLANTS

Contractors: AEA-Culcheth, NRPB, AEA-Windscale
Contract No.: FI2D-0030
Work Period: October 1990 - March 1993
Coordinator: G C MEGGITT, AEA Technology, Warrington
Phone: 44/925/25 42 24 Fax: 44/925/25 45 44

A. OBJECTIVE AND SCOPE

The theoretical work is composed of two distinct but complementary studies:

a) Waste management options: The theoretical study continues to develop an existing methodology to aid decommissioning waste management decisions, and to demonstrate the improved methodology by applying it to the prototype AGR Windscale reactor decommissioning waste for which the final management option has not yet been chosen. The main extension to the existing methodology (see final report on contract FI1D-0051 is to enable the incorporation of risks and uncertainties, rather than simply doses and environmental impact parameters. The improved methodolgy, like the existing one, will be applicable to decisions concerning the decommissioning of all types of nuclear reactors and could lead to reductions in radiation risks and financial costs, as well as promoting consistency between the approaches in various countries;

b) Decommissioning strategies: The work will aim at developing a comprehensive methodology to evaluate radiological risks to the public and workers from decommissioning of non-reactor nuclear plants. Such a methodology will allow the comparison of different decommissioning strategies from a risk point of view so that the benefits associated with, for example, delay in decommissioning to more advanced stages could be assessed.

B. WORK PROGRAMME

B.1.a. Development of a radiological risk evaluation methodology (NRPB) considering the uncertainties in models and modelling parameters.
B.2.a. Selection of the waste stream for an example application of the methodology (AEA)
B.3.a. Definition of the radionuclides inventory and their distribution in the selected waste stream. (AEA)
B.4.a. Definition of waste management options (AEA)
B.5.a. Estimation of financial costs for each of the management options. (AEA)
B.6.a. Calculation of doses and risks for individuals and the public. (NRPB)
B.7.a. Assessment of social and environmental impacts of waste management options. (NRPB)
B.8.a. Demonstration of the methodology by identifying the optimal management options. (NRPB)
B.9.a. Review of the results and check of their applicability to other decommissioning decisions. (all)
B.1.b. Definition of decommissioning phases of non-reactor nuclear plants. (AEA-Culcheth for the entire b-study)
B.2.b. Identification of techniques for carrying out decommissioning operations and their risk-bearing elements.
B.3.b. Identification of risk assessment procedures taking into account normal and possible accidental risks.
B.4.b. Evaluation of procedures for assessing the risks associated with leaving the plant under care and maintenance;
B.5.b. Examination of methods for the aggregation of risks associated with particular decommissioning strategies.
B.6.b. Demonstration of the identified methodologies to a non-reactor facility.
B.7.b. Final evaluation on the suitability and limitations of the identified methodologies.

C. Progress of work and results obtained
Summary of main issues
Part A : NRPB and AEA-Windscale

Substantial progress has been made on Task B.1A (development of a methodology for evaluating risks), but this task will not be considered complete until Task B.8A (demonstration of methodology) has been performed. Tasks B.2A (identification of waste stream), B.3A (definition of waste inventory), B.4A (definition of waste management options) and B.5A (estimation of financial costs) are essentially complete, unless (any additional specific data requirements arise). Task B.6A (estimation of radiological impact) is almost complete, subject to results being finalised. Progress on Task B.7A (assessment of other impacts) has been made, but further work will be necessary as Task B.8A proceeds. Work on Tasks B.8A (demonstration of the methodology) and B.9A (review of the methodology) has not yet begun.

Part B : AEA-SRD
All tasks (B.1B-B.7B) have been completed and preparation of the final report is currently underway. The principle task this year has been the application of the methodology to the decommissioning of non-reactor nuclear plant. Decommissioning risk assessment studies of a plutonium glove-box suite and an Intermediate Level Waste (ILW) store were undertaken as a demonstration of the methodology. The studies showed that the only major difficulties in applying probabilistic risk assessment techniques to decommissioning were in modelling some accident scenarios and in finding data appropriate for decommissioning. The reason for this is that hitherto in the nuclear industry resources have been devoted to obtaining risk assessment data for operational plant, rather than for decommissioning activities.

Progress and Results
Part A : NRPB and AEA-Windscale

1. Estimation of Financial Costs (B.5A)
Estimates have been made of the costs of managing the chosen waste stream, namely the graphite components (the moderator, reflectors and neutron shield) from the decommissioning of the Windscale Advanced Gas-cooled Reactor (WAGR). These calculations indicated total waste management costs (including waste conditioning, packaging, storage, transport and disposal) in the region of £20,000 per cubic metre of packaged waste for intermediate level waste (ILW) disposed of in deep geological repository, and of just over £1,300 per cubic metre for low level waste (LLW) disposed of at a near-surface facility.

2. Estimation of Radiological Impacts (B.6A)
Estimates have been made of the various radiological impacts associated with the management of the graphite wastes, based on the projected radionuclide inventories (from Task B.3A). Two generic waste disposal sites were considered: a deep geological repository for ILW and LLW if appropriate and a near-surface disposal facility for LLW. Two possible exposure scenarios were considered for each disposal site, namely undisturbed transport of dissolved radionuclides in ground water and direct inadvertent

human intrusion into the waste. These calculations were intended to provide data for the demonstration of the decision-aiding methodology (Task B.8A), rather than as a full assessment of disposal sites, and therefore a number of simplifying assumptions were made.

For the groundwater transport scenario, estimates were made of the individual and collective doses to members of the public from disposal of the graphite wastes to these sites. This was done using computer models to represent radionuclides transport in the geosphere and biosphere, and human exposure pathways. Global circulation and carbon-14 released to the biosphere was also modelled. Collective doses were calculated for four different populations (local, UK, European and global) and for three different time periods (up to 100 years, 100 to 10,000 years and 10,000 to 1,000,000 years). For the deep site, the calculations indicated that only chlorine-36 is likely to give rise to any doses, and that these will be small, peaking at around 5 nSv annual individual dose after about 700,000 years. For the shallow site the doses from all nuclides were estimated to be yet smaller (because the LLW inventory is much smaller than that of ILW), but arise much sooner than for the deep site, peaking at about 500 years. Collective dose commitments were also estimated to be relatively small - less than 1 man Sv.

The uncertainty associated with these calculations was addressed by performing an uncertainty analysis using parameter value distributions for a number of key uncertain parameters, such as groundwater velocity and radionuclides retardation factors. Thus, in addition to best " estimates" of the radiological impacts, distributions were also obtained indicating the possible range of values. These calculations indicated that the individual and collective doses might be significantly higher than the "best estimates", the 95th percentile values were up to an order of magnitude greater.

For the human intrusion scenarios, potential individual doses at different times after disposal were estimated using simple models of representative exposure scenarios. For deep disposal, the model represented inadvertent intrusion during exploratory drilling, and for near-surface disposal the scenario concerned the possible future redevelopment rates. The potential doses from these scenarios were much higher than those for the groundwater pathways - up to 1 mSv for intrusion into the shallow sites at 50 years - but the probabilities of such exposures were much less than unity.

Estimates have also been made of the doses likely to be received by workers in the course of waste management, but these have yet to be finalised.

3. Assessment of Social and Environmental Impacts (B.7A)

Relevant impacts have been identified, namely non-radiological risks (primarily conventional industrial risks), non-radiological environmental impact (such as noise, visual impact and traffic), "social" factors (such as local employment and public acceptability) and project-specific factors, in particular the value of the WAGR decommissioning as a demonstration project, and the consequent availability of funding and key staff.

Many of these impacts are difficult to quantify in the way that, for example, doses or costs are quantified. However, meaningful comparisons between different options can be made and quantitative judgements can be made about the relative importance of the differences between the options with regard to the different factors. This will be investigated more fully in the demonstration of the methodology (Task B.8A).

Part B : SRD

1. Methodology application (B.6B)

In last years annual report, a methodology for evaluating the radiological risks of decommissioning was described. In brief, this consists of identifying potential accidents and faults by the use of systematic hazard identification techniques such as Failure Modes and Effects Analysis (FMEA) and HAZard and OPerability Studies (HAZOPS). The annual radiological risk (R) from the identified hazard (ie accident or fault) is then evaluated from the formula R = FC, where F and C are the frequency (probability per year) and the radiological consequence of the hazard. The total risk is then the sum over all hazards.

Two radiological risk assessments were carried out as a demonstration of the methodology; an assessment of decommissioning plutonium glove-boxes and an assessment of a decommissioning plan for an Intermediate Level Waste Store.

Plutonium glove-boxes: Laboratory F, part of AEA Technology's Advanced Fuel Laboratory at Windscale, was designed for the fabrication of fast reactor fuels. The facility contains a large number of glove-boxes, and some of these are now surplus to requirements and will be decommissioned over the next few years. After post-operational clean-out, glove-box dismantling will be carried out inside a Modular Containment System (MCS) by personnel wearing full pressurised PVC suits. The MCS is a portable containment system consisting of stainless steel panels attached to a modular framework. Exit and entry is through an entry port containing a shower, and airborne contamination in the MCS is controlled by a dedicated ventilation system. Strippable coatings on the interior surface of the MCS simplify post-operative contamination control; the contamination is sandwiched between successive coatings. This can be peeled from the walls and disposed of as waste.

Accidents and faults were identified using the FMEA and HAZOPS techniques. Those with the potential to result in dose to decommissioning personnel, other on-site workers and members of the public were analysed in detail. The radiological risks were estimated after consideration of the frequency of the operations which could give rise to the faults, and the consequences were modelled assuming worst case conditions. The radiological risks were found to be acceptably low; typically an individual fatality risk of less than 10^{-5} year^{-1} for decommissioning personnel and 10^{-9} year^{-1} for other on-site workers and members of the public.

ILW Store: The Dounreay Wet Silo is used to store Intermediate Level Waste (ILW), generated from reprocessing fast reactor fuel and fuel cycle support operations. The waste typically comprises activated steel from irradiated core components and miscellaneous items such as contaminated gaiters, string and plastic and paper articles. The Silo is currently about 80% full and has several years of useful life remaining. At the moment, the fate of the Silo is undecided; various options are open and no decision has been taken. One plan which might be adopted is to build a new facility to remove the waste and package it into suitable containers for disposal. This plan has been assessed to provide an example of the methodology.

The new facility would consist of a shielded enclosure built on top of the Silo to allow waste removal, and waste packaging. The facility would be fully automated with waste packages transported through the plant by a powered trolley system. The various stages (plant construction, waste removal, waste packaging and Silo demolition) were analysed using the HAZOPS technique. Potential accidents or faults with radiological consequences for decommissioning personnel or others (on-site workers or members of the public) were identified. Radiological risks were estimated in terms of individual fatality $year^{-1}$. These were found to be acceptably low; typically an individual fatality risk of less than 10^{-6} $year^{-1}$ for all groups (decommissioning personnel, other on-site workers and members of the public).

2. Final evaluation (B.7B)

The example studies clearly demonstrate the feasibility of applying probabilistic risk assessment techniques to decommissioning operations. However it was found that difficulties can be encountered in two areas: data and consequence modelling. Data specific to decommissioning is limited because until recently the resources of the nuclear industry have been mainly devoted to plant design and operation. Modelling the radiological consequences of some accident or fault scenarios can be difficult because little experimental evidence exists to validate the assumptions.

6.2. DOSES DUE TO THE REUSE OF VERY SLIGHTLY RADIOACTIVE STEEL

Contractors: CEA-FAR, BS, SIEMENS BEW
Contract No.: FI2D-0031
Work Period: September 1990 - February 1994
Coordinator: Mrs H GARBAY, CEA-IPSN, Fontenay-aux-Roses
Phone: 33/1/46 54 73 41 Fax: 33/1/47 35 14 23

A. OBJECTIVE AND SCOPE

The scope of the study is the determination of doses due to the reuse or recycling of very slightly contaminated radioactive steel in case of mechanical and thermo-mechanical treatments applied to scrap when exempted from regulations.

The study will mainly be based on already available data both in the nuclear field and in the conventional scrap industry. Experimental investigations will be performed, as far as possible, on radioactive samples coming from nuclear installations being dismantled. The various treatments applied to scrap before its melting have not yet been studied and are of great interest. In particular, techniques used in scrapyards should be studied in the view of inhalation and external exposure injuries.

This study applies to a large quantity of steel arising from dismantling of nuclear installations (EUR 10052).

Benefits are expected as regards management and cost of radioactive waste arisings, protection still being secured. The results concerning contamination dispersion during cutting of scrap will be useful for the evaluation of future large-scale decommissioning operations.

B. WORK PROGRAMME

B.1. Discussion and documentation of the present regulatory situation. (BS-CEA)
B.2. Performance of steel cutting and aerosol sampling experiments observing industrial conditions. (CEA-Siemens)
B.3. Evaluation of inhalation risk in realistic situations. (all)
B.4. Determination of the radiological impact based either on bibliographic data or on experimental results. (BS-CEA)
B.5. Development of a stochastic programme to obtain the individual dose distribution. (BS)

C - PROGRESS OF WORK AND DETAILED RESULTS

Execution of the experimental programme (B2)

1/ Mechanical cutting experiments - cobalt and caesium.

The cutting experiments in inactive conditions, have been performed with a mechanical tool ; this is a 20 tons shear press which is placed in a glove box of about 0.6 m^3 volume. The air exhaust rate is 100 l/min.

The materials cut are pipes made of stainless steel on the one hand and carbon steel on the other hand, about 0.9 m length and 2 cm diameter ; a test is also carried out on rusty carbon steel ; the pipes are internally "contaminated" with inactive cobalt chloride ($CoCl_2$) and caesium chloride (CsCl).

The filtration system collects the aerosols in totality. In a first approximation the maximum aerodynamic diameter is estimated at about 10μm and the collected particles on the filter are supposed to be inhalable.

2/ Thermal cutting experiments - cobalt and caesium.

During the second 6 months period, cutting experiments have been conducted with oxyacetylen torch on carbon and rusty steel and with plasma torch on stainless steel.

These experiments were conducted in a ventilated room of 30 m^3. The air exhaust rate is 300 m^3/h.

Aerosols are sampled inside the experimental room, by a sampler determining particle size distribution above and below 10μm. They are also sampled by isokinetic sampling in the extraction line with an impactor and a diffusion battery which allow size distribution determination from 0.0075 μm to 15 μm.

3/ Thermal cutting experiments - uranium

During the experimental period from January to June 1992 continuing investigations were performed on austenitic and ferritic materials to determine the alpha radioactivity release during thermal cutting.

Using a modified experimental set-up, including a high volume cascade impactor, compared to the preceding experiments, in addition to the total alpha radioactivity release, particle size distributions of the generated aerosols could be determined.

Based on the available results presented in Madrid in October 1992, during the second experimental period 1992 (November to December) the following experiments were performed according to discussions with the other contractors :

- Contamination with UO_2-powder : cutting of austenitic and corroded material
- Contamination with U-nitrate solution : cutting of austenitic material

Extended background measurements and additional filter analyses were performed during this series of experiments.

Identification of risk (B3)

1/ Mechanical cutting experiments - cobalt and caesium

The element concentrations in the experimental room are respectively of the order of 0.3 $\mu g.m^{-3}/\mu g.cm^{-2}$ for caesium deposited on carbon or rusty steel, 0.04 $\mu g.m^{-3}/\mu g.cm^{-2}$ for caesium deposited on stainless steel, 1 $\mu g.m^{-3}/\mu g.cm^{-2}$ for cobalt deposited on carbon or rusty steel and 0.3 $\mu g.m^{-3}/\mu g.cm^{-2}$ for cobalt deposited on stainless steel. Cobalt is released at a higher level than caesium.

An extrapolation can be made in a first approximation for semi-industrial conditions ; the element concentrations are about two orders of magnitude less.

2/ Thermal cutting experiments - cobalt and caesium

The aerosol releases are of the order of 1 $mg.m^{-1}$ for carbon steel and rusty steel cut with an oxyacetylen torch and of the order of 3.6 $mg.m^{-1}$ for stainless steel cut with plasma torch. For any of the experiments the fraction of inhalable aerosols is around 75%, including 1/3 of aerosols which mass median aerodynamic diameters (MMAD) are lower than 0.2 μm.

3/ Thermal cutting experiments - uranium

Within the scope of the evaluation, linear dependence of the releases relative to the cut length could be confirmed (see Figs. 1 and 2). The following release codes C_R respectively for austenitic and ferritic materials resulted :

$C_{RA} \approx 4.\ 10^{-2}\ Bq.cm^{-1} / Bq.cm^{-2}$ (austenitic material)

$C_{RF} \approx 2.\ 10^{-2}\ Bq.cm^{-1} / Bq.cm^{-2}$ (ferritic material)

The determination of the particle size distribution reveales the highest values for a cut-off of approximately 3 μm as well as particle sizes lower than 0.42 μm.

The evaluation of the experiments of the 2nd experimental period 1992 will be performed in the first half of 1993.

Radiological impact (B4)

1/ Dose factors related to particle size distribution

The organ specific dose factors in IRCP 30 are based on an activity median aerodynamic diameter (AMAD) of 1 μm. Neglecting the influence of particle size the inhalation doses could be calculated using these factors. It has been shown that the mass median aerodynamic diameters (MMAD) for various segmenting technics in the scrap industry are not equal to 1 μm [1, 2]. Assuming, due to lack of data, that the AMAD is equal to the MMAD it is appropriate to introduce a correction factor for particle size into the dose equations. In IRCP 30 a method is given to correct the dose factors for different particle sizes. The correction factor depends on the nuclide and solubility class. A plot of the correction factor of U-238 as a function of AMAD for two solubility classes is shown in Figure 3.

2/ Mechanical cutting tool

The annual doses calculated for scrapyard worker cutting 10% of very low-level radioactive scrap are shown in table 1. Assuming the same release rate for uranium and plutonium than for cobalt, annual doses are also calculated for U-238 and Pu-239. The highest value obtained for Pu-239 is around 60 $\mu Sv.a^{-1}/Bq.cm^{-2}$.

3/ Thermal cutting tool

The experiments on thermal cutting tools are available for uranium and can be extrapolated to other alpha emitters. The annual dose due to inhalation of Pu-239 would give in industrial conditions a value of the order of 20 $\mu Sv.a^{-1}/Bq.cm^{-2}$.

Development of a stochastic programme to obtain the individual dose distribution (B.5)

Stochastic simulations were performed to estimate the doses resulting from segmenting of α-contaminated scrap. Data out of the literature [3, 4, 5], from Siemens BEW and CEA-IPSN were used in the simulations to estimate the release fraction from various segmenting methods. In the simulations four ways of segmenting the scrap are modelled. In table 2 the data are compared to the values used in the model.

In the 1991 annual report [5] it was shown that the small scrapyard is the most critical in terms of the doses received by the workers. For this reason only the small scrapyard was simulated. It was assumed that 200 tons/a of α-contaminated scrap was released to 3 scrapyards. On the average the scrap distribution was 128 tons/a, 57 tons/a and 15 tons/a to each of the three scrapyards. In the model the percent of scrap processed on the average by each of the four segmenting methods was : shear press 25%, torch 48%, saw 4% and grinder 23%. Each of the scrapyards employs 16 workers so that 48 persons can potentially be exposed. 1000 simulated releases of the 200 tons/a of α-contaminated scrap were made for each the exemption levels indicated in table 3 and the dose distributions calculated. In this table the preliminary results are presented. Values of less than 1 can be interpreted as the probability of occurrence. In the last column, the dose interval in which the maximum doses occurred and their probability of occurrence (in brackets) are shown for each of the exemption levels.

REFERENCES

[1] ANTOINE P., Proceeding of the ANS International Topical Meeting, Waste Management and Decontamination and Decommissioning, Niagara Falls, NY, 14-18 september 1986

[2] NEWTON G.J. et al., International Decommisionning Symposium, Seattle, october 1982

[3] STANG W. ; FISCHER A.; RUBISCHUNG P., Decommissioning of Nuclear Installations, EUR-12690 (1990)

[4] ONODERA J. ; YABUTA H. ; NISHIZONO T. ; NAKAMURA C. ; IKEZAWA Y.; J. Aerosols Sci., Vol. 22, Suppl. 1, 747-750 (1991)

[5] COMMISSION OF THE EUROPEAN COMMUNITIES, Nuclear Science and Technology, Annual Progress Report, EUR 14498 (1991)

TABLE 1 : DOSES IN SEMI-INDUSTRIAL CONDITIONS FOR THE MECHANICAL CUTTING TOOL

Radionuclide		DF_{inh} Sv/Bq	Nucl. conc. in air per surf.activity $Bq.m^{-3}/Bq.m^{-2}$	Dose rate for for one hour cut $\mu Sv/h^{-1}/Bq.cm^{-2}$	Annual dose for the worker $\mu Sv/a^{-1}/Bq.cm^{-2}$
Co 60	carb-rust	$5\ 10^{-8}$	$1.2\ 10^{-2}$	$7.2\ 10^{-4}$	$4.32\ 10^{-2}$
	stainless		$4\ 10^{-3}$	$2.4\ 10^{-4}$	$1.44\ 10^{-2}$
Cs 137	carb-rust	$8.7\ 10^{-9}$	$4.2\ 10^{-3}$	$4.4\ 10^{-5}$	$2.63\ 10^{-3}$
	stainless		$5\ 10^{-4}$	$5.22\ 10^{-6}$	$3.13\ 10^{-4}$
U 238	carb-rust	$3.2\ 10^{-5}$	$1.2\ 10^{-2}$	$4.61\ 10^{-1}$	$2.76\ 10^{1}$
	stainless		$4\ 10^{-3}$	$1.54\ 10^{-1}$	9.22
Pu 239	carb-rust	$6.7\ 10^{-5}$	$1.2\ 10^{-2}$	$9.65\ 10^{-1}$	$5.79\ 10^{1}$
	stainless		$4\ 10^{-3}$	$3.22\ 10^{-1}$	$1.93\ 10^{1}$

Table 2 : Measured release fractions compared to model parameters

	Parameter values used in the stochastic model				Experimentally measured	
segmenting method	minimum	most likely	maximum	units	values	source
shear press	0.1	0.5	2.0	Bq/m / Bq/cm^2	0.07-1.2	CEA
torch	0.005	0.02	0.3	Aerosol Release / Kerf Activity	0.002-0.2 0.02	[4] SIEMENS
saw	0.05	0.2	1.0	Aerosol Release / Kerf Activity	0.06-0.6	[4]
grinder	0.05	0.2	1.0	Aerosol Release / Kerf Activity	---	---

Table 3: Preliminary results from the stochastic simulations

Exemption Level	Number of persons exposed in the dose interval [μSv/a]			Dose interval where maximum dose was registered
Bq/cm^2	1 - 10	10 - 100	100 - 1000	μSv/a (prob. of occurrence)
0.04	6	1	0.006	100 - 126 (0.006)
0.05	6	1	0.012	126 - 160 (0.005)
0.1	7	3	0.074	250 - 320 (0.005)
0.3	8	5	0.72	630 - 790 (0.01)

FIGURE 1
Release of activity during thermal cutting

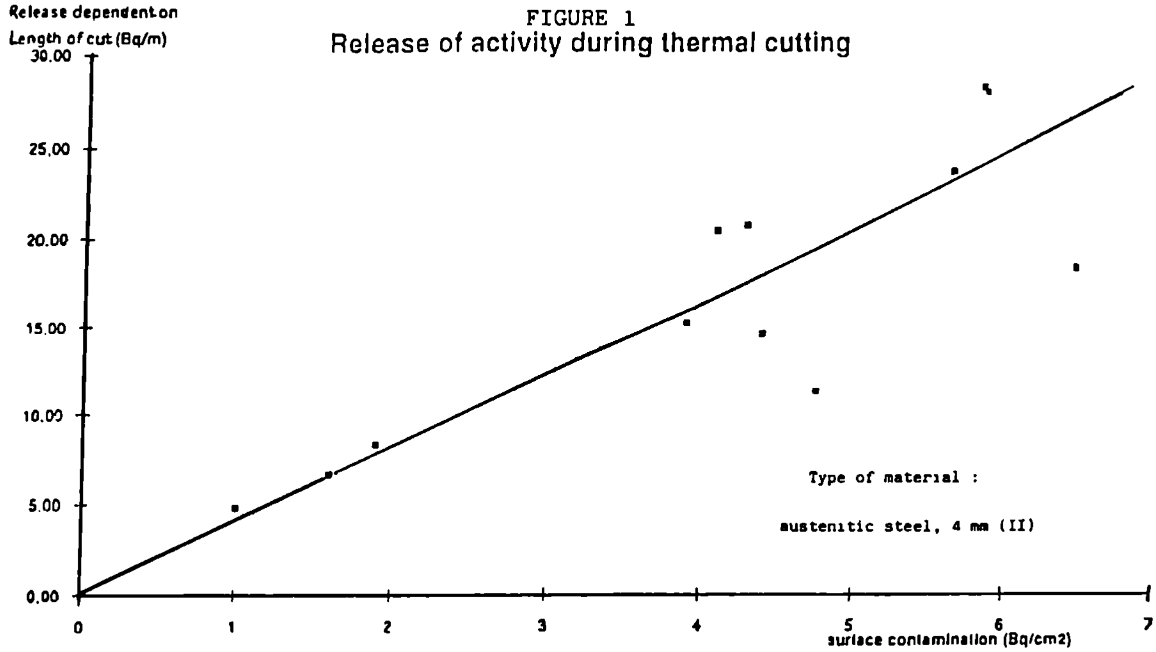
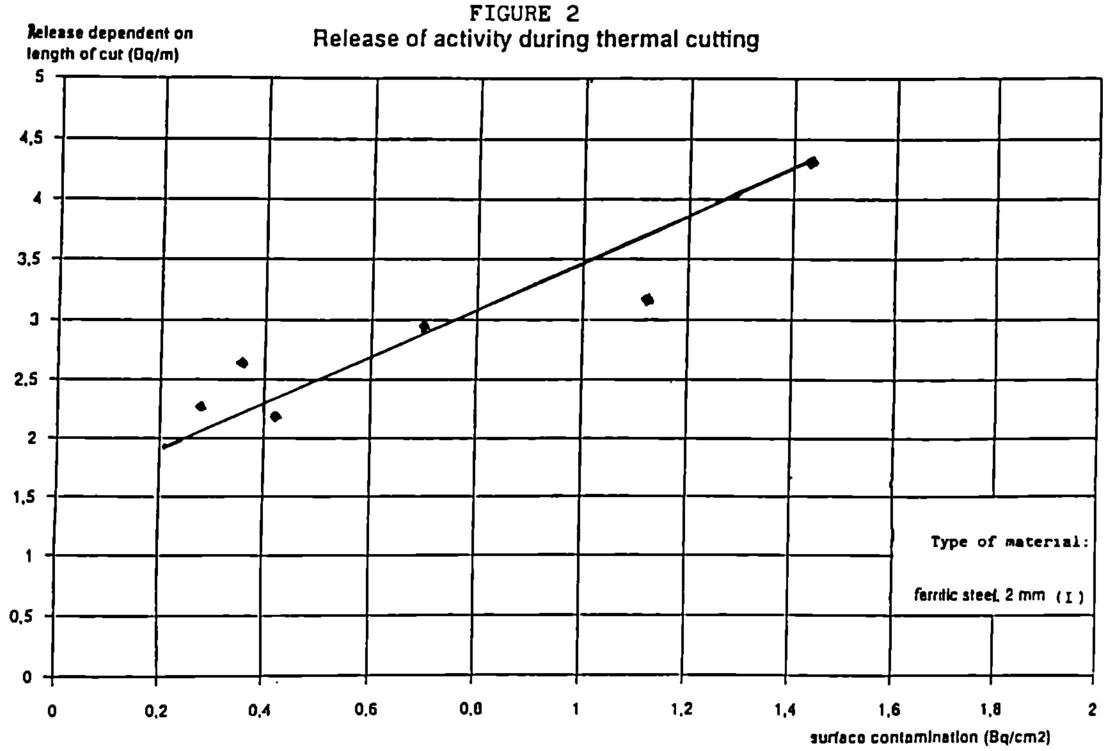


FIGURE 2
Release of activity during thermal cutting



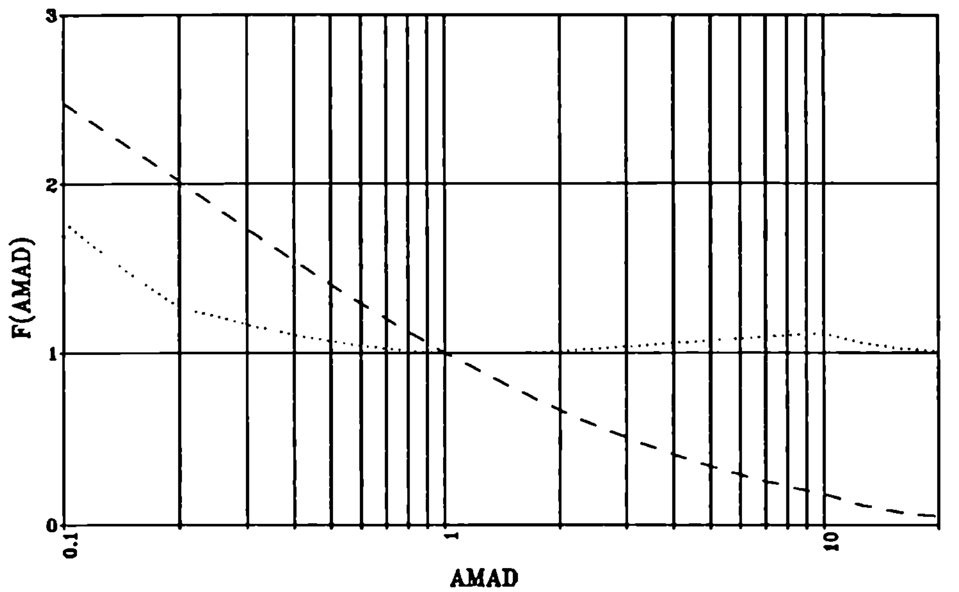


Figure 3	Correction factor F(AMAD) for U–238 and solubility class D(········), W(– –)	

6.3. QUICK MEASURING METHODS OF RADIONUCLIDES IN MATERIALS AND WASTES DURING DECOMMISSIONING OF NUCLEAR INSTALLATIONS

Contractors:	TÜV-SWD, FHGF
Contract No.:	FI2D-0033
Work Period:	September 1990 - December 1993
Coordinator:	L DIERKES, TÜV, Mannheim
	Phone: 49/621/395 530 Fax: 49/621/395 299

A. OBJECTIVE AND SCOPE

Under the ALARA guidelines of the German Radiological Protection Ordinance, it is necessary to know the exact amount of radioactivity and the radiological potential of the materials of installations to be decommissioned.

The objective of this work programme is to determine a correlation between the gamma and beta emitters (electron capture nuclides) by analysing the activation products and contaminants in reactor materials and in waste products. These informations are essential for determining the radioactivity released to the environment and for radiological protection of the public and the personnel.

The extracted material (e.g. iron) will be submitted to beta-activity measurements, followed by a gamma-activity determination. The correlation of both measuring methods should make it possible to reduce the determination of the total radioactive material quantity to gamma-spectroscopic analyses.

The work programme will be performed in contact with the Chemistry Division, Harwell Laboratory UKAEA, especially concerning the exchange of measuring methods.

B. WORK PROGRAMME

B.1. Acquisition of instrumentation (TÜV-SWD)
B.2. Choice and procurement of representative samples from the reactors MZFR, FR2, KNK and/or KWO (TÜV-SWD)
B.3. Laboratory activities, reference measurements, and correlation calculations for nuclear determinations on decommissioning wastes (all)
B.4. Evaluation and documentation of the results (TÜV-SWD)

C. Progress of work and obtained results

Summary of main issues

The aim of this research contract is the development of chemical separation methods to determine beta emitting radionuclides, especially electron capture nuclides and x-ray emitters in waste materials. During decommissioning of nuclear facilities large quantities of nuclides like Ca-41, Fe-55 and Ni-63 must be handled and may be released to the environment.

For this reason amounts of contaminated steel and concrete samples from nuclear power stations were gathered. At first these samples were analysed by gamma spectroscopy and the alpha- and beta activity was measured. The next step was the separation and chemical preparation of Ca-41, Fe-55 and Ni-63. For measuring the samples by a liquid scintillation counter, it is necessary to calibrate the liquid scintillation counter first.

After measuring with the liquid scintillation counter it is possible to compare the Co-60 activity with the Ca-41, Fe-55 and Ni-63 activity. The ratio Co-60, Fe-55 and Ni-63 corresponds to 1:1/3:1/50:1 or 100 % Co-60, 30 % Fe-55 and 2 % Ni-63. The measurements of Ca-41 have not been completed yet, hence it is not possible to give a statement of its ratio to Co-60 at the moment.

Progress and results

1. Acquisition of instrumentation (B.1.)

The analysis of the beta- and gamma emitting isotopes was performed with an end-window Geiger-Müller detector and a gas-flow proportional counter. For the determination of gamma emitting nuclides in the contaminated steel and concrete samples it was necessary to calibrate the applied Ge-detector. The calibration procedure was required to take into account the different sizes and volumes of the unformed samples.

For the determination of the beta- and electron capture activity it was also necessary to calibrate the liquid scintillation counter, because the counting efficiency of the solvent-solution system can be affected by many different factors which may reduce detection efficiency.

2. Choice and procurement of representative samples from the reactors KKP, KKN, WAGR (B.2.)

The samples for determination of radionuclides by quick measuring methods were obtained from nuclear power plants during decommissioning and refueling. Five samples were taken during refueling in 1989 and 1990. These samples were parts of the reactor cooling system contaminated by activated corrosion and fission products and pieces of concrete from the nuclear power plant of Philippsburg unit 1 (BWR).

Thirty samples were taken during dismantling of the Niederaichbach power plant. The concrete samples were parts of the biological shield and adjoining areas, containing neutron induced activation products.

We obtained three samples from Harwell Laboratory from the inner (active) edge of the WAGR bioshield.

3. Laboratory activities, reference measurements and correlation calculations for nuclear determinations on decommissioning wastes (B.3.)

The chemical processing of the concrete and steel samples were carried out by the radiochemical laboratory in the Fachhochschule Gießen. The samples were wetly decomposed, chemically processed and were run through an anion-exchanger. After separation of Ca-41, Fe-55 and Ni-63 the samples were prepared for liquid scintillation counting with the scintillation cocktail.

Before starting the measurement with the liquid scintillation counter it is necessary to look for quenching effects. Quenching affects the efficiency of the conversion process beta particle energy to photoelectron (see figure 1 to 4).

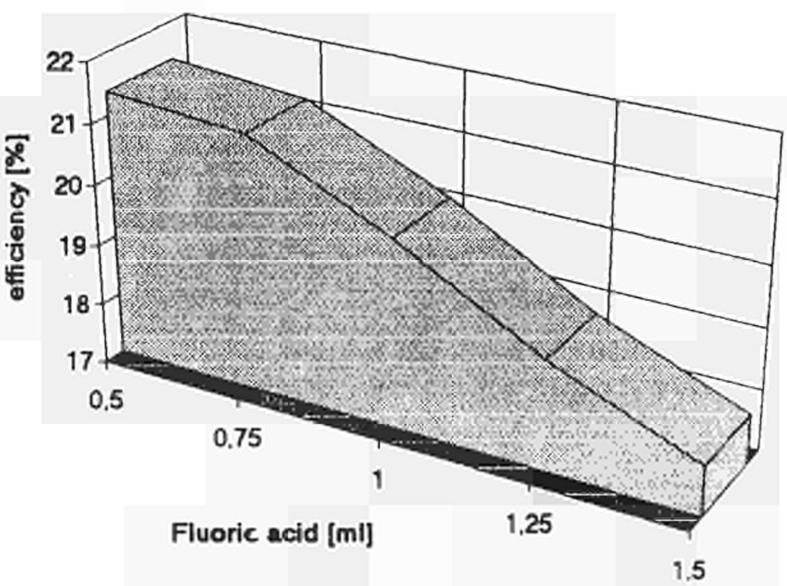


Figure 1: Dependency on the content of acid

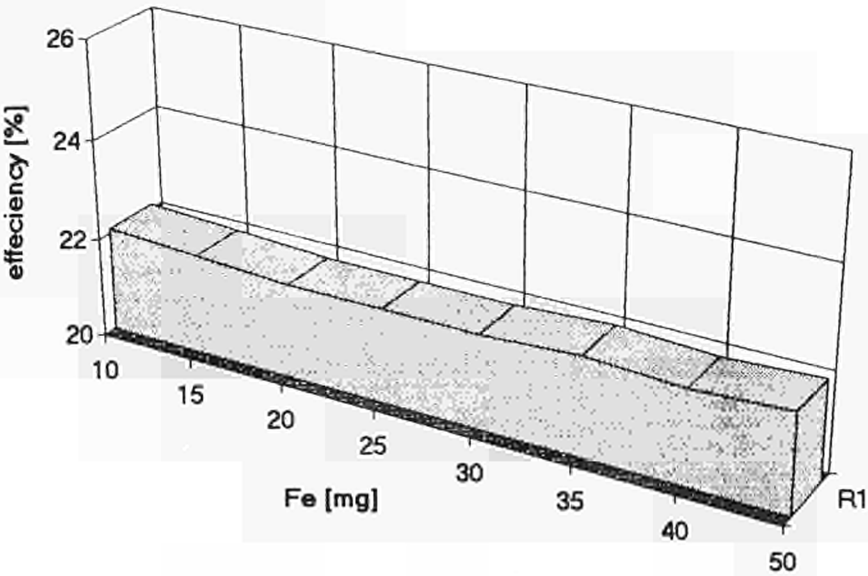


Figure 2: Influence of iron concentration to efficiency

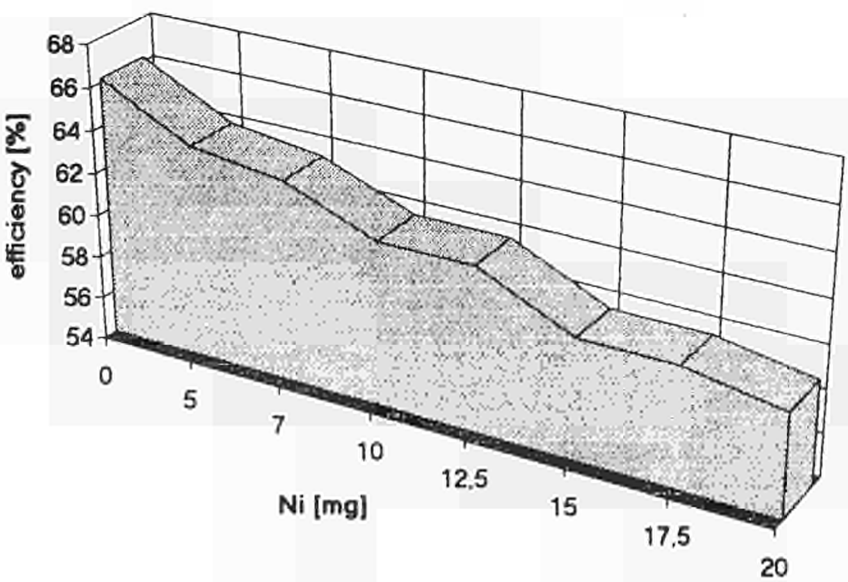


Figure 3: Influence of nickel concentration to efficiency

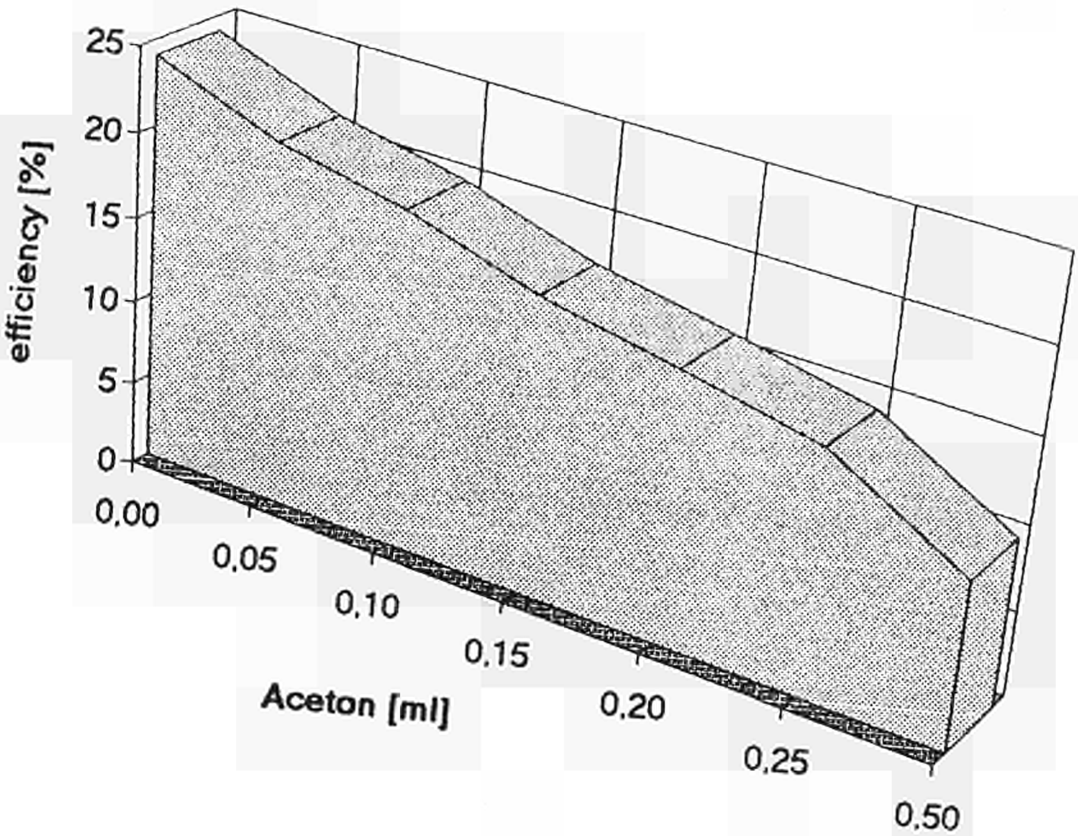


Figure 4: Influence of aceton concentration to efficiency

4. Evaluation and documentation of the results (B.4.)

The results obtained up to date are summed up in Tables I, II, III and Figures 5 and 6.

Table I: Dataset of Fe-55 and Ni-63 in steel sample

Sample No/Nuclide	Content		Quantity of samples [ml]	Yield		Activity	
	Fe [mg/sample]	Total [g]		[mg]	[%]	[Bq/sample]	[Bq/cm²]
a/ Fe	30.7	3.46	5	26.9	87.6	205	9.5
b/ Fe	30.7	3.46	5	21	68.4	197	9.1
c/ Fe	61.4	3.46	10	55.5	90.4	187	8.7
d/ Fe	30.7	3.46	5	25.8	84	214	9.9
e/ Fe	61.4	3.46	10	26	42.3	173	8
i/ Fe	30.7	3.46	5	45.2	147.2	229	10.6
a/ Ni	22.1		200	11.1	50.2	40.1	1.9
b/ Ni	22.1		200	9.98	45.1	38.8	1.8

Table II: Dataset of Fe-55 in ball-shaped samples

Sample No	Content		Quantity of samples [ml]	Yield		Activity	
	Fe [mg/sample]	Total [g]		[mg]	[%]	[Bq/sample]	[Bq/cm²]
I	111	2.19	30	79.7	71	17355	21
II	111	2.19	30	81.3	73	18519	23
III	111	2.19	30	83	74	17520	22

Table III: Dataset of Ca-41 in concrete samples

Sample No.	Content		Yield		Activity		Proof limit [Bq]
	Concrete [g]	Ca [Mg]	In 5 ml [mg]	[%]	[Bq]	[Bq/sample]	
A1	4	144	25	34.7	0	0	0
A2	4	144	49.2	68.3	0	0	0
B	4	327	152.5	93.2	0	0	0
C	4	369	141	76.4	0.4	1.04	1.3
D	4	327	96	58.7	0	0	0
E	4	246	99	80.5	0.03	0.07	1.1
F	4	247.5	70	56.8	0	0	0
G	4	329.1	159.5	96.9	1.95	5.91	0.9
H	4	370.2	171.5	92.7	0	0	0
J	4	411.6	98	47.5	0.06	0.17	1.8
I	4	288	151.5	105.2	0	0	0

Figure 5: Activity of Co-60, Fe-55 and Ni-63 in RA-samples

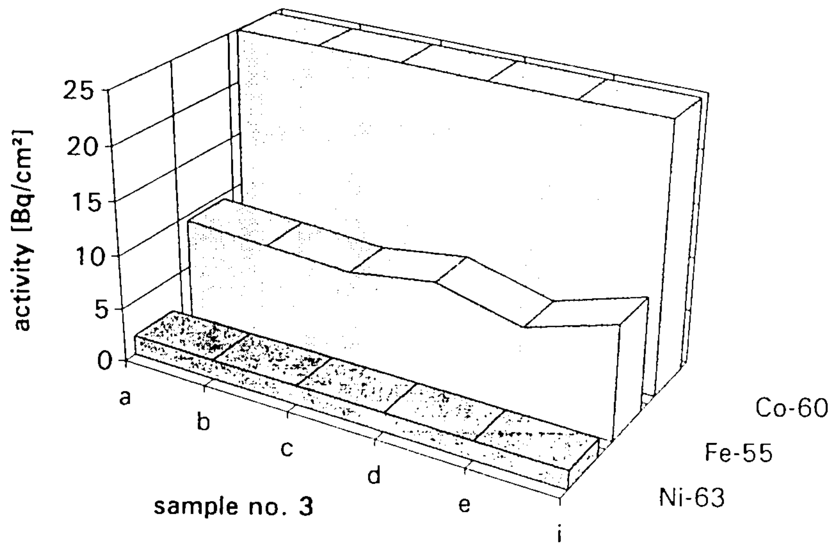
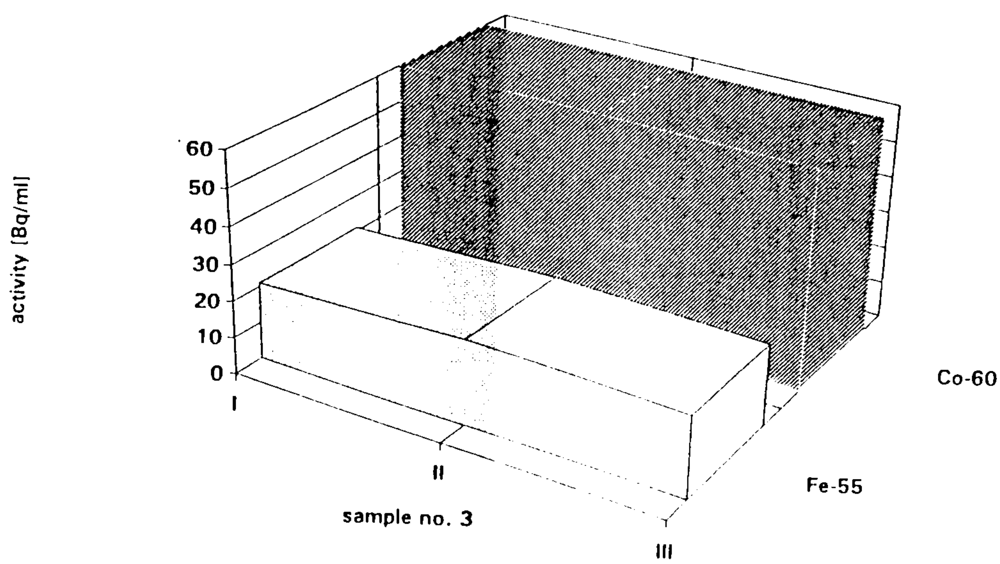


Figure 6: Activity of Co-60 and Fe-55 in ball shaped samples



6.4. RADIOLOGICAL ASPECTS OF RECYCLING CONCRETE DEBRIS FROM DISMANTLING OF NUCLEAR INSTALLATIONS

Contractors:	TÜV-Bay., RWE
Contract No.:	FI2D-0039
Work Period:	November 1990 - December 1993
Coordinator:	F J SCHMID, TÜV-Bay.
	Phone: 49/89/5791 1470 Fax: 49/89/5791 2606

A. OBJECTIVE AND SCOPE

Limiting values for the release of concrete with low-level residual radioactivity for the selective undangerous utilisation (e.g. for noise barriers, earth fill, earth bank or substitute for foundation material) are presently not defined. The research programme will examine whether it is possible to define limiting values for radioactively contaminated concrete in the range of the limiting values for steel. The effect of radioactively contaminated concrete on the soil (leach out of radionuclides) and on man (radiation exposure) will be determined.

The results of these studies will have an effect on the decommissioning activities as far as buildings of the controlled area and the kind and quantity of the radioactively contaminated concrete are concerned.

The advantage of the studies lies in an economic and safe recycling of large amounts of concrete with a low-level artificial residual radioactivity. Thereby, valuable ground storage space would be saved and natural gravel deposits would be preserved.

The research work will provide data concerning cost saving by recycling concrete from controlled areas, radiation exposure of the decommissioning workers and of the general public.

The research programme is performed in co-operation with CEA-IPSN, which has a research programme with a similar objective (see § 6.4.).

B. WORK PROGRAMME

B.1. Leach tests
B.1.1. Design of the test facility and determination of concrete test specimen. (all)
B.1.2. Construction and operation of the test facility. (TÜV-Bay.)
B.1.3. Literature survey on leaching out problems of radionuclides in concrete. (TÜV-Bay.)
B.1.4. Radiological measurements on concrete rubble before, during and after leach out tests. (TÜV-Bay.)
B.2. Natural radioactivity in concrete
B.2.1. Procurement of samples from recently produced and aged concrete. (RWE)
B.2.2. Measurement of alpha, beta and gamma radiation. (TÜV-Bay.)
B.2.3. Literature survey concerning the natural radioactivity of concrete.
B.3. Development of methods for recycling concrete.
B.3.1. Examination of concrete recycling possibilities by a literature study. (RWE)
B.4. Calculation of radiation exposure and determination of the artificial residual radioactivity
B.4.1. Determination of radiation exposure scenarios. (TÜV-Bay.)
B.4.2. Calculation of radiation exposure for man due to natural and artificial radioactivity. (TÜV-Bay.)
B.4.3. Derivation of criteria for the safe use of concrete with artificial radioactivity. (TÜV-Bay.)

C.Progress of work and obtained results

Summary of main issues

In the period under review we continued the wash-out-tests and the continuous measurement of the samples. The literature survey on the leaching problems and the natural activity of concrete has been finished.

Ten additional samples of aged concrete have been collected to determine their content of natural activity.

Progress and results

1.Wash-out-tests (B.1)

In 1992 we ran about 100 sprinkling-cycles (a cycle is a 24 hour sprinkling period), so that up to now we have already finished 138 cycles. The principles of sprinkling two times a week for 24 hours, filtration of the sediment of the taken samples and seperate measurement of water and sediment have been kept. As planned, we have started to sum up four sprinkling cycles to one measuring unit, beginning with the 45th cycle.

Figure 1 and 2 show the results of the measured activity in the filtrated water of Cs-137 and Co-60 respectively, Figure 3 and 4 show the results of the measured activity in the sediment (in relation to a litre of water), up to cycle 100. The shown values are not yet corrected by the activity of the used sprinkling water or other influences that might have falsificated the results slightly. So some unexpected high rates in single measurements will be explained and if necessary corrected in the final report.

Some complications arose from the use of natural rain water for the sprinkling of the samples. It led to the growth of algues in the supplying hoses and to deposits on the sprinkling equipment which influenced the sprinkling intensity by a lower water flow. To avoid these effects in the future the test facility will be improved in the beginning of 1993 to guarantee a constant high water flow until the end of the test period. The variing water flow in the past, which can be seen in the variing hight of the water in the collecting tubs after a sprinkling cycle, is shown in Figure 5. These variing water quantities per cycle will have no influence on the results of the experiment. The overall washed-out activity can be calculated with the measured water quantities in the collecting tubs. In 1993 we will rise the precipitation amount per cycle to reach the intented 20-year-precipitation-amount of 20 000 mm.

The literature survey on leaching out problems has been finished. In the report of the literature survey the results of 34 literature references are evaluated in detail. Besides this the abstracts of further 15 references are given in the report.

2.Natural radioactivity in concrete (B.2)

In addition to the samples of aged and freshly mixed concrete that had been collected and measured in 1991, 10 further samples of aged concrete from conventional buildings of the region of Wuerttemberg have been collected. These samples are now due to be measured. Although the results of the measurements are not yet present, no significant deviation of their activity content to the already obtained results is expected.

The results of the literature survey, concerning the natural radioactivity of concrete, were submitted. The results of eight reports which provide detailed data on the content of natural radioactive nuclides in concrete or its parts are shown in tables. The obtained results show a wide span of the radioactivity content in dependence on the used materials but also on the method of measurements (measuring the single components of the concrete and calculating the activity of the concrete or measuring the produced concrete itself). A detailed comparison of the data from the literature survey with the data of our own measurements will be given in the final report.

3.Development of methods for recycling concrete (B.3)

The literature study on recycling methods has been continued but is not yet finished. Its results will be given later.

4.Calculation of radiation exposure and determination of the artificial residual activity (B.4)

Besides the radiation exposure of the common population due to the washed-out-activity we intend to make an estimation of the radiation exposure of the workers who are involved in the recycling and the reuse of the concrete (so as workers at concrete crushers or construction workers). This estimation will be done as soon as the results of the literature study on the recycling possibilities are available.

The calculations of the radiation exposure due to the washed-out activity are starting at the beginning of 1993 with the preliminary results of the wash-out-tests.

Legend for the figures

ts 1: test specimen: cubes of concrete rubble, fixed with cement.
ts 2: test specimen: cubes of concrete rubble, fixed with bitumen.
ts 3: test specimen: cubes of concrete rubble, fixed with cement, covered with soil.
ts 4: test specimen: loosely piled concrete rubble, covered with soil.
ts 5: test specimen: loosely piled concrete rubble.

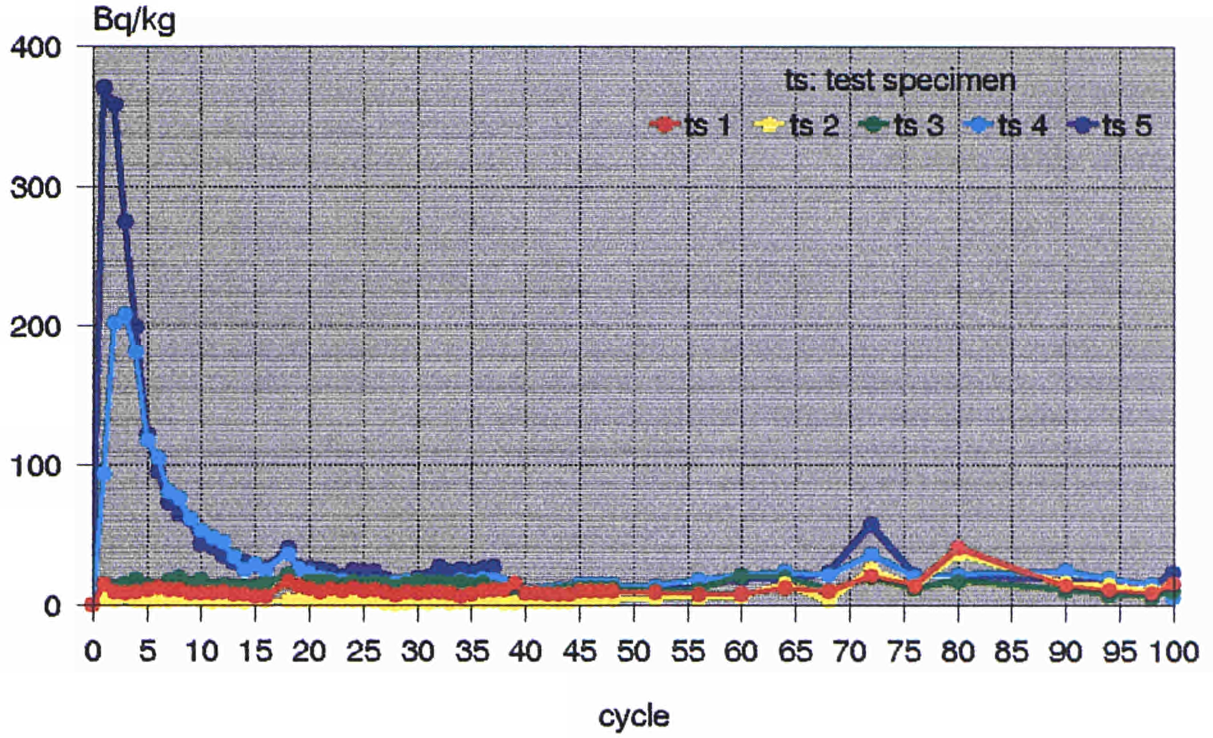


Figure 1: Cs-137 in wash-out-water

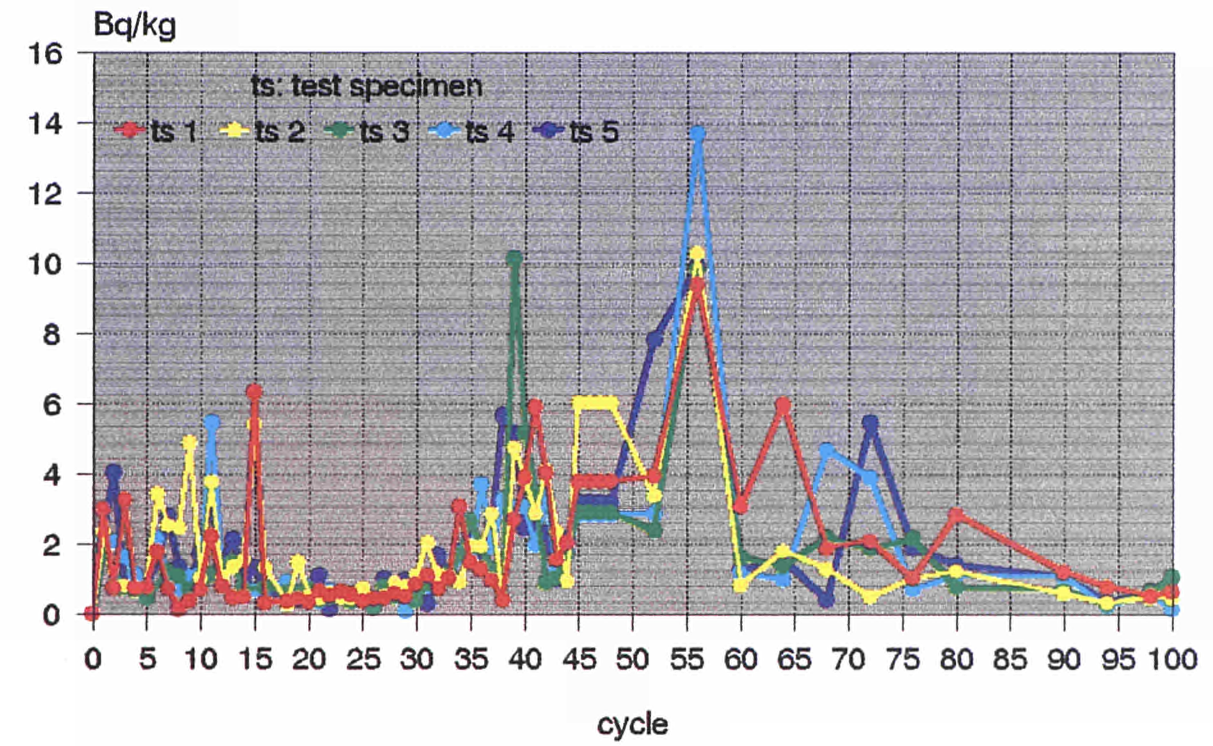


Figure 2: Co-60 in wash-out-water

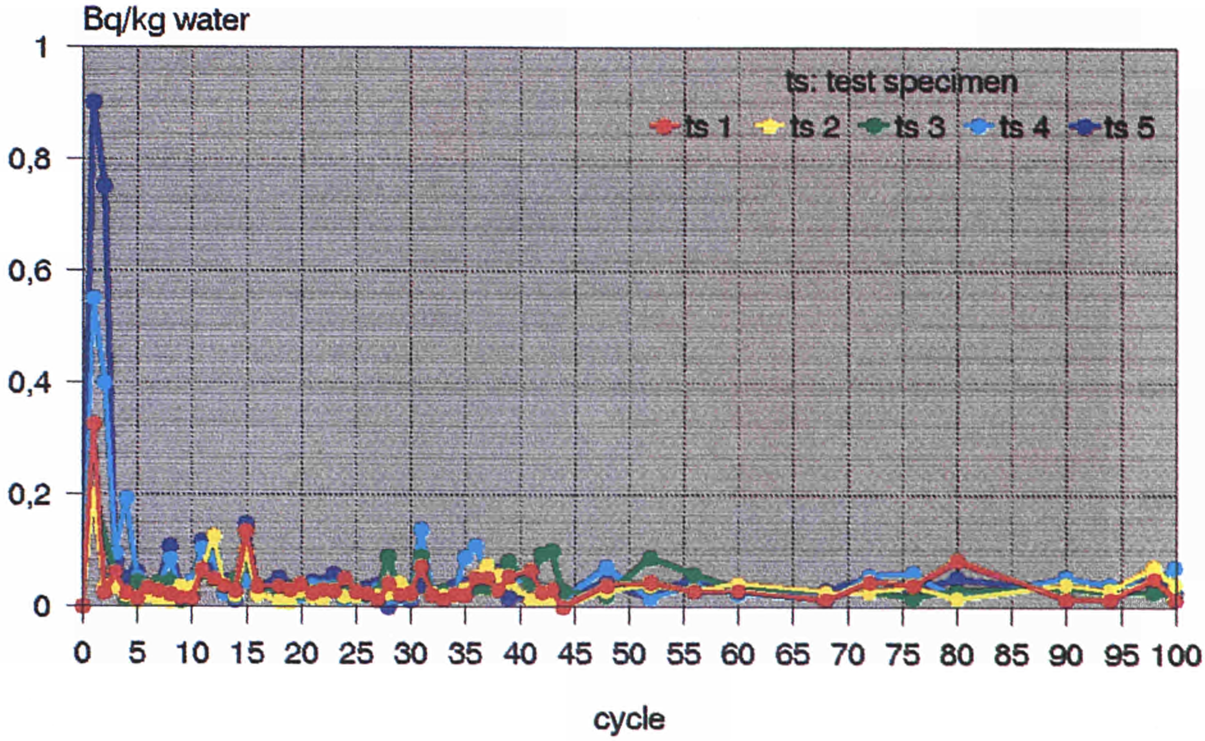


Figure 3: Cs-137 in sediment

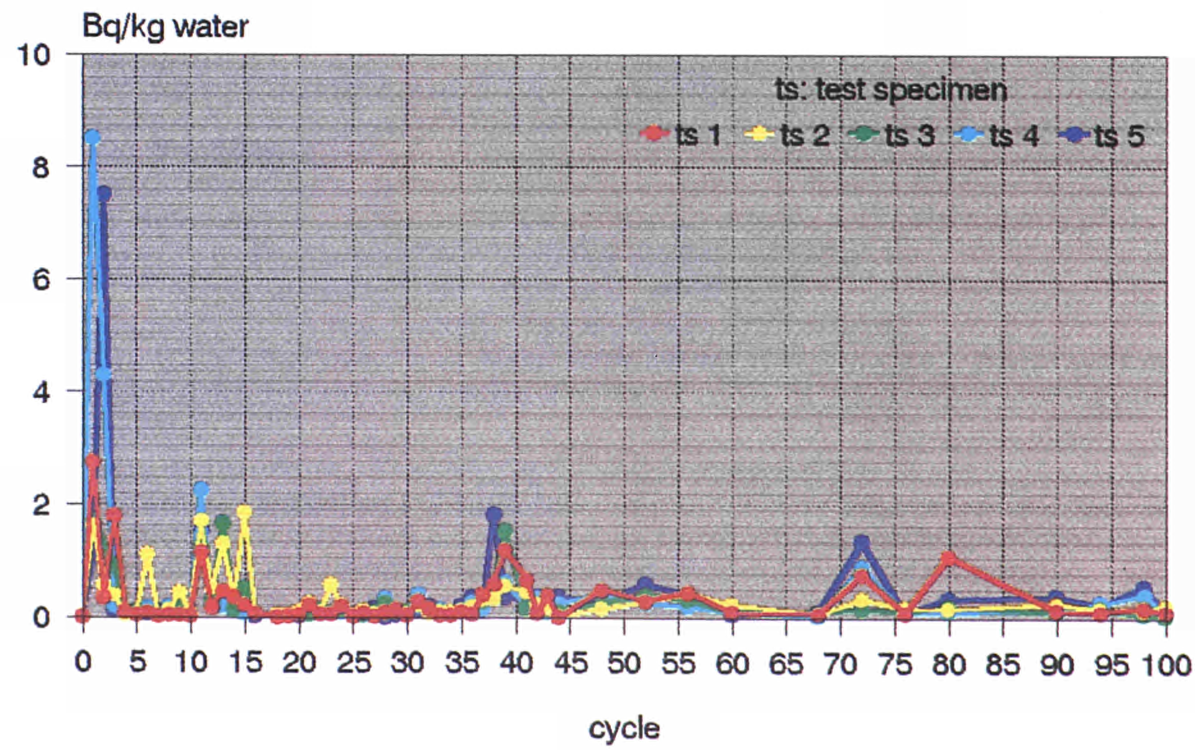


Figure 4: Co-60 in sediment

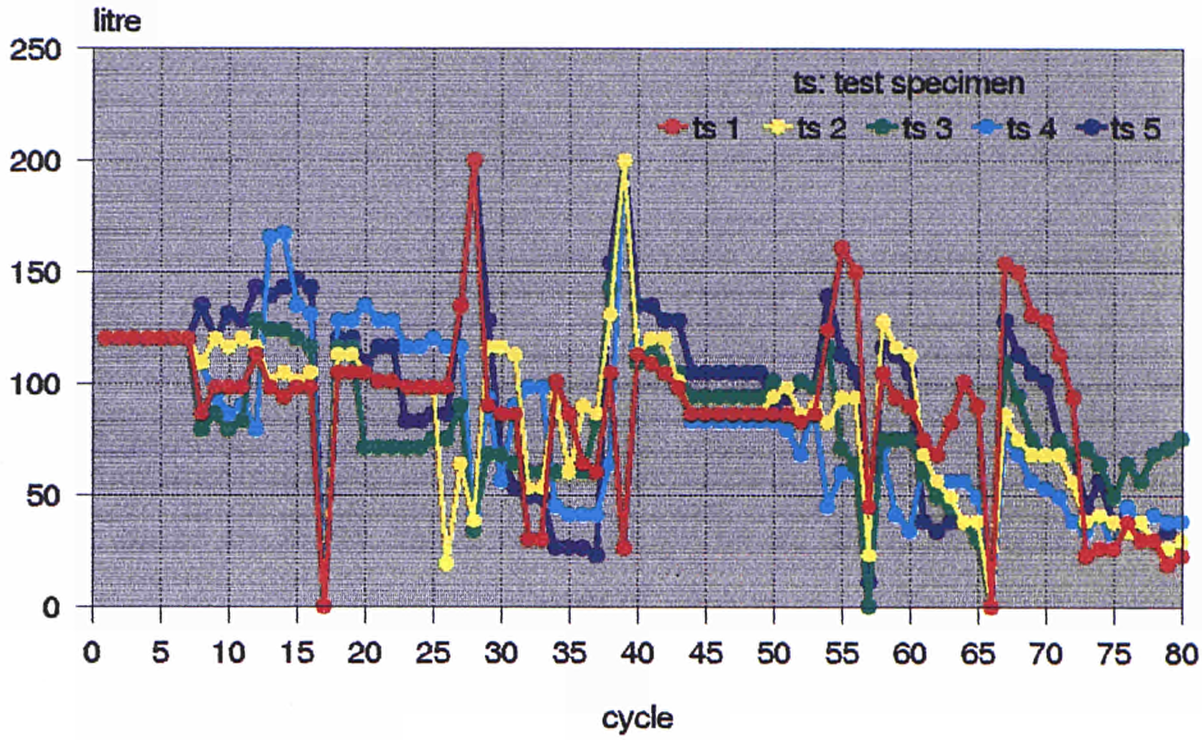


Figure 5: Sprinkling rate per cycle

6.5. DEFINITION OF REFERENCE LEVELS FOR EXEMPTION OF CONCRETE COMING FROM DISMANTLING

Contractors:	CEA-FAR
Contract No.:	FI2D-0040
Work Period:	October 1990 - September 1992
Coordinator:	Mr D. HARISTOY, CEA/IPSN/DPEI/SERGD, Fontenay-aux-Roses Phone: 33/1/46 54 71 56 Fax: 33/1/47 35 14 23

A. OBJECTIVE AND SCOPE

The objective of the study is to propose activity limits below which very slightly radioactive concrete arising from nuclear facility dismantling could be treated in conventional industry, or slightly contaminated buildings could be reused or decommissioned.

The study is based on the evaluation of concerned concrete quantities and allows the identification of groups of people exposed to radiation hazards. From the evaluation of individual radiological risk, the derived limits for exemption of concrete will be deduced. The study is also meant to participate in the harmonisation of criteria and rules between countries of the European Community. Potential benefits in determining such limits are:

- limitation of the decontamination time and operations, i.e. decontamination and disposal costs;
- recycling of valuable material to preserve natural resources;

The research programme is performed in co-operation with TÜV Bayern and RWE (contract No. FI2D-0039), into which CEA-IPSN will bring in the following information: natural radioactivity in concrete; work programme and results of each period; results of other French experiments connected with the subject.

B. WORK PROGRAMME

B.1. Data collection
B.1.1. Estimation of contaminated and activated concrete quantities.
B.1.2. Identification of radionuclides spectra and activity levels.
B.1.3. Estimation of concrete quantity which is recycled or disposed off.
B.1.4. Determination of the state of the art to identify the critical group of workers and public.
B.1.5. Investigations on the possibilities of exposure of the public to different concrete by-products.
B.1.6. Dust measurements and analysis in different crushing stations.
B.2. Treatment of information and modelling.
B.2.1. Collection and evaluation of parameters for the different by-product pathways.
B.2.2. Modelling of the different realistic exposure situations for critical groups.
B.3. Calculation of the radiological impact using the collected parameters during the inquiries
B.4. Determination of the activity limits for each radionuclide and for classes of them.

C .PROGRESS OF WORK AND OBTAINED RESULTS

1) Calculation of the radiological impact using the collected parameters during the inquiries and determination of the activity limits for each radionuclide (B.3.) (B.4.)

The whole system of recycle or disposal of the concrete has been studied.

Crushing or disposal are the first steps of the management of concrete from the dismantling of installation.

Many destination do exist for the aggegates coming out of the crusher. The most important are road basement construction and building blocks production.

Activity levels are proposed for different practices :

- crushing-transport-road basement construction
- crushing-transport-building blocks production-factory building
- crushing-transport-building blocks production-home building
- crushing-transport-aggregate (nursery lane or playing area)
- landfill disposal

Building transformation and reuse as warehouse or office are also considered.

The dose reference level used to calculated the proposed activity levels are :

- 10 μSv/y for the public
- 50 μSv/y for workers or for the public concerned with low probability scenario
- 1 mSv for accidental scenario (very low probability)

Table I gives, for selected radionuclides, the proposed activity levels.

Results, classified in an increasing serie, are presented in figure 1.

As usual, alpha emitters and high energy-high emission probability gamma emitters have the lowest values.

The highest values correspond to beta emitters with low energy and/or low level emission probability.

Landfill disposal has the highest activity level except for ^{3}H, ^{14}C, ^{41}Ca, ^{79}Se, ^{90}Sr and ^{99}Tc as a result of the vegetable pathway for the residential scenario.

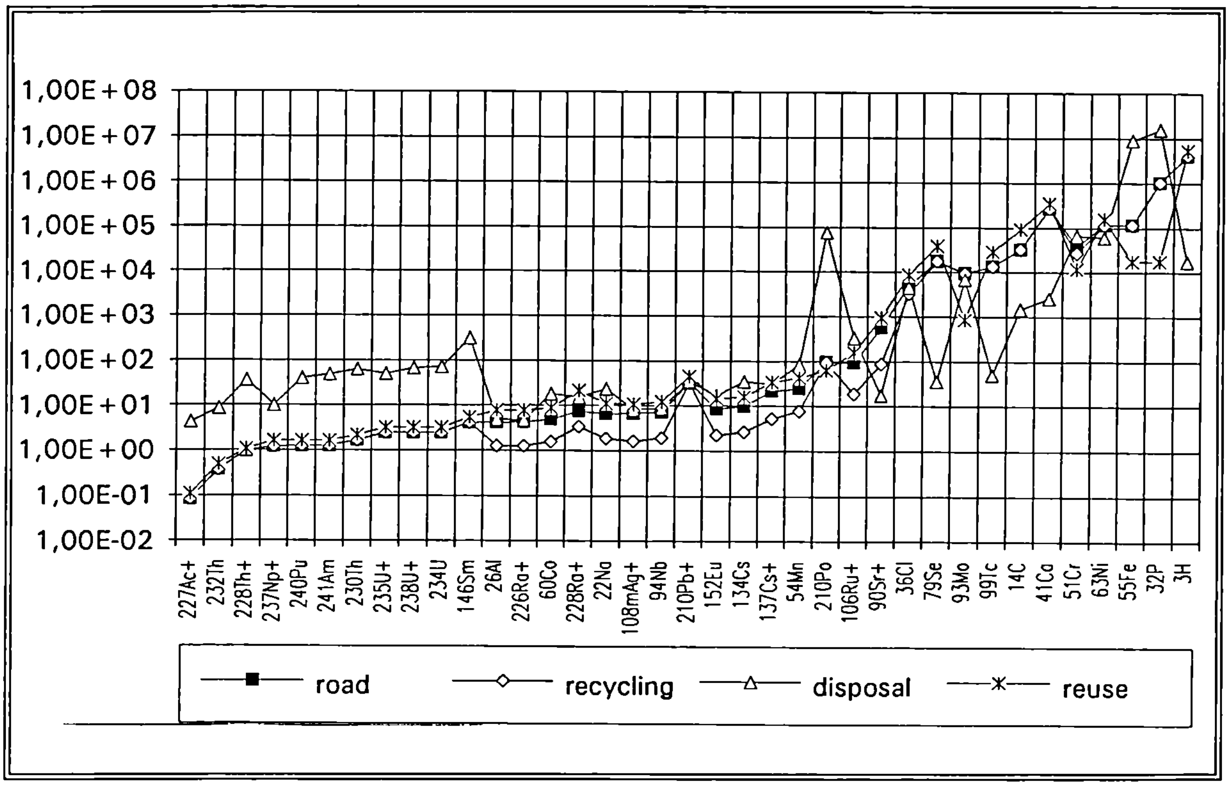


Figure 1 Comparison of the values for the different practices

	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/g	Bq/cm2
nuclides	crushing transport road	crushing transport blocks factory	crushing transport blocks house	crushing transport aggregate	minimum recycling	disposal	reuse
3H	4,13E+06	4,13E+06	4,13E+06	4,13E+06	4,13E+06	1,79E+04	5,35E+06
14C	3,27E+04	3,27E+04	3,27E+04	3,27E+04	3,27E+04	1,48E+03	9,38E+04
22Na	6,52E+00	6,52E+00	1,87E+00	4,63E+00	1,87E+00	2,37E+01	1,11E+01
26Al	4,20E+00	4,20E+00	1,26E+00	2,99E+00	1,26E+00	5,00E+00	7,92E+00
32P	1,05E+06	1,05E+06	1,05E+06	1,05E+06	1,05E+06	1,56E+07	1,75E+04
36Cl	4,95E+03	4,95E+03	3,23E+03	4,95E+03	3,23E+03	4,57E+03	8,68E+03
41Ca	2,75E+05	2,75E+05	2,75E+05	2,75E+05	2,75E+05	2,60E+03	3,60E+05
51Cr	3,85E+04	3,85E+04	3,85E+04	2,75E+04	2,75E+04	6,67E+04	1,21E+04
54Mn	2,45E+01	2,45E+01	7,87E+00	1,75E+01	7,87E+00	8,50E+01	4,20E+01
55Fe	1,19E+05	1,19E+05	1,19E+05	1,19E+05	1,19E+05	8,93E+06	1,76E+04
60Co	5,00E+00	5,00E+00	1,56E+00	3,55E+00	1,56E+00	1,82E+01	8,99E+00
63Ni	1,17E+05	1,17E+05	1,17E+05	1,17E+05	1,17E+05	6,02E+04	1,58E+05
79Se	1,80E+04	1,80E+04	1,80E+04	1,80E+04	1,80E+04	3,68E+01	3,82E+04
90Sr+	5,52E+02	5,52E+02	8,62E+01	5,52E+02	8,62E+01	1,79E+01	9,58E+02
93Mo	9,58E+03	9,58E+03	9,58E+03	9,58E+03	9,58E+03	6,94E+03	8,55E+02
94Nb	7,39E+00	7,39E+00	1,91E+00	5,26E+00	1,91E+00	8,85E+00	1,22E+01
99Tc	1,36E+04	1,36E+04	1,36E+04	1,36E+04	1,36E+04	4,95E+01	2,94E+04
106Ru+	9,35E+01	1,58E+02	9,35E+01	1,88E+01	1,88E+01	3,36E+02	6,66E+01
108mAg+	6,84E+00	1,08E+01	6,84E+00	1,63E+00	1,63E+00	8,33E+00	4,85E+00
134Cs	9,69E+00	9,69E+00	2,64E+00	6,90E+00	2,64E+00	3,50E+01	1,62E+01
137Cs+	2,17E+01	3,42E+01	2,17E+01	5,32E+00	5,32E+00	3,08E+01	1,55E+01
146Sm	4,17E+00	4,17E+00	4,17E+00	4,17E+00	4,17E+00	3,10E+02	5,39E+00
152Eu	8,64E+00	8,64E+00	2,27E+00	6,16E+00	2,27E+00	1,71E+01	1,45E+01
210Pb+	3,25E+01	4,55E+01	3,25E+01	3,25E+01	3,25E+01	3,55E+01	3,25E+01
210Po	9,54E+01	9,54E+01	9,54E+01	9,54E+01	9,54E+01	7,75E+04	6,27E+01
226Ra+	4,42E+00	7,86E+00	4,42E+00	1,27E+00	1,27E+00	4,85E+00	3,14E+00
228Ra+	7,58E+00	2,16E+01	7,58E+00	3,44E+00	3,44E+00	1,55E+01	7,58E+00
227Ac+	8,49E-02	1,08E-01	8,49E-02	8,49E-02	8,49E-02	4,35E+00	8,49E-02
228Th+	9,62E-01	1,08E+00	9,62E-01	9,62E-01	9,62E-01	3,60E+01	9,62E-01
230Th	1,67E+00	1,67E+00	1,67E+00	1,67E+00	1,67E+00	6,33E+01	2,16E+00
232Th	3,76E-01	3,76E-01	3,76E-01	3,76E-01	3,76E-01	8,62E+00	4,85E-01
234U	2,50E+00	2,50E+00	2,50E+00	2,50E+00	2,50E+00	7,25E+01	3,25E+00
238U+	2,49E+00	2,49E+00	2,49E+00	2,49E+00	2,49E+00	6,76E+01	3,25E+00
235U+	2,48E+00	2,48E+00	2,48E+00	2,48E+00	2,48E+00	5,00E+01	3,21E+00
237Np+	1,24E+00	1,61E+00	1,24E+00	1,24E+00	1,24E+00	1,00E+01	1,24E+00
240Pu	1,25E+00	1,25E+00	1,25E+00	1,25E+00	1,25E+00	4,03E+01	1,62E+00
241Am	1,25E+00	1,25E+00	1,25E+00	1,25E+00	1,25E+00	4,93E+01	1,62E+00

6.6. THE CHARACTERISATION AND DETERMINATION OF RADIOACTIVE WASTE FROM DECOMMISSIONING

Contractors: AEA-Harwell
Contract No.: FI2D-0042
Work Period: January 1991 - March 1993
Coordinator: J W McMILLAN, AEA Technology, Harwell
Phone: 44/235/43 48 53 Fax: 44/235/43 29 77

A. OBJECTIVE AND SCOPE

The objective of this laboratory-scale experimental investigation is to develop the "fingerprint" method for the characterisation of waste arising from decommissioning projects to the point where it could be used more extensively after its initial limited application.

The "fingerprint" method relies on the ability to carry out comprehensive analysis, for all of a specified range of radionuclides on a statistically justified set of samples. In order to achieve this, development of several aspects, related particularly to the difficulty of measuring some specific electron capture and low energy beta-emitting nuclides, is required.

It is expected that the establishment of accurate fingerprints, when coupled with simple measurement of the total activity, will enable correct sentencing of waste and thus minimise the cost of the disposal of the waste arising from the decommissioning of radiochemical laboratories or reactor facilities.

Collaboration is envisaged with TÜV Südwest, FRG (contract No. FI2D-0033), both on the development of radiochemical methods and on the assessment of measurement techniques.

B. WORK PROGRAMME

B.1. Acquisition of contaminated material and fabrication of simulants

B.2. Development of methods for the removal of radioactive contaminants to solution

B.2.1. Survey on existing methods for removal of radioactivity from contaminated materials.
B.2.2. Leaching experiments.
B.2.3. Investigation on microwave dissolution techniques.
B.2.4. Investigation on electrolysis for the recovery of tritium.

B.3. Development of preconcentration, separation and analysis methods

B.3.1. Selection and commissioning of slow injection analysis.
B.3.2. Development of methods for Fe, Ni and U.
B.3.3. Development of method for I.
B.3.4. Development of method for Ca.
B.3.5. Investigation on combustion techniques for the recovery of carbon and hydrogen

B.4. Development of counting methods for the difficult-to-measure nuclides

B.4.1. Development of liquid scintillation.
B.4.2. Development of gas proportional counting.
B.4.3. Development of x-ray counting

B.5. Statistical assessment to characterise the waste and satisfy the quality assurance standards

C. Progress of Work and Results Obtained

Summary of Main Issues

Further progress has been made towards assuring the validity of "fingerprinting" methods for sentencing radwaste from decommissioning programmes. The focus has been on correlating simple radiation measurements with complex characterisation methods.

A study of a WAGR bioshield core has shown that the "fingerprinting" method can be applied to predict activities with a reasonable degree of confidence, from a single comprehensive nuclide "fingerprint" for its most active point and widespread gross beta-counting. Caution is necessary where natural activities dominate activation products and in predicting mobile nuclide activities, notably for ^{3}H.

A second study on contaminated ventilation trunking is in progress.

Supportive studies on the separation of low energy beta-emitters from radwaste by flow injection analysis (FIA) methods, has reached a practical stage following a review and the design of separation schemes.

Progress and Results

1. Amendment of the Programme (B2.4, B4.2)

The completion date for the programme has been extended to March 1993, to accommodate delays in material and equipment acquisition. Two programme tasks have been omitted, as they have essentially been completed by alternative, superior means. They are;

B2.4 Investigation of electrolysis for the recovery of tritium: This is not required as combustion methods (B3.5) are entirely satisfactory.

B4.2 Development of gas proportional counting: This gas phase measurement method for ^{3}H and ^{14}C, is not required. These nuclides are being recovered as liquids and measured by liquid scintillation counting (B3.5 and B4.1).

2. Acquisition of Samples (B1)

The acquisition of samples has been confined to contaminated ventilation trunking from B220 Harwell Laboratory with associated dust samples.

No material has been received from TÜV Sudwest (Contract No. FI2D-0033). Insufficient quantities of decommissioning materials are available to allow their exchange. However, a sample of ^{41}Ca is expected to allow cross-calibration of the liquid scintillation counting equipment used in the two laboratories.

Three samples of concrete from the WAGR bioshield core have been sent from Harwell to TÜV Sudwest for examination.

3. Development of Methods for the Removal of Radioactive Contaminants to Solution (B2, B2.1, B2.2, B2.3)

To test the applicability of "fingerprinting" to the assessment of activity in contaminated waste the technique is currently being applied to contaminated ventilation trunking from B220, Harwell Laboratory. The application of the technique assumes that a comprehensive activity "fingerprint" for this decommissioning waste can be obtained by analysing dust samples from the trunking. By combining the "fingerprint" with surveys of trunking samples using gross beta- and gross alpha-counting methods, it is further assumed that the overall amount of contamination on the trunking can be derived successfully and accurately.

The test programme has progressed through 3 of its 5 planned stages:-

(i) Dust samples (4) have been analysed, principally by gamma- and alpha-spectrometry, to produce "fingerprints". The principal nuclides found include actinides, fission products and activation products. Variations have been observed in the "fingerprints" which are associated with the sampling locations.

(ii) The contaminated trunking samples (24) have differences in their measured ratio of gross beta- to gross alpha-count rates. This indicates some compositional variability.

(iii) The "fingerprints" of the 12 most active trunking samples have been measured by gamma-spectrometry to allow comparison with those of the dusts. Some differences have been observed.

The programme will continue with the 2 final stages:-

(iv) Removal of the contamination from the trunking prior to its further radiochemical analysis, using leach solutions identified through a survey (B2.1) by direct leaching (B2.2), and microwave enhanced leaching and sample dissolution (B2.3).

(v) Comprehensive radiochemical analysis of removed contamination and dusts, followed by an overall review of the value and limitations of "fingerprinting" for the activity assessment of contaminated trunking.

4. <u>The Introduction of Flow Injection Analysis Separation Methods for Radionuclides</u> (B3, B3.1, B3.2, B3.3 and B3.4)

The possible use of flow injection analysis (FIA) methods to automatically separate low energy beta-emitting nuclides from samples prior to radiometric analysis has been recently discussed in a review by J W A Tushingham (Ref [1]) (B3.1). As a consequence a number of FIA based separation schemes have been devised.

A FIA separation system has been devised to recover ^{55}Fe, ^{59}Ni, ^{63}Ni and ^{41}Ca by an automated version of the radiochemical separation scheme developed previously to recover these nuclides from irradiated concrete samples (Ref [2]) (B3.2 and B3.4). The operation of the automated FIA method is described elsewhere in detail (Ref [1]). While the scheme is theoretically applicable, it is complex, inflexible and probably impractical to operate successfully. Consequently it will not be implemented.

Simple schemes devised to recover and separate one or two nuclides from radwaste should be easier to implement, for example U, singly (B3.2), and ^{36}Cl and ^{129}I, sequentially (B3.3). The scheme devised for the latter nuclides is presented in Figure 1, and is based on the coprecipitation of silver halides, and their selective dissolution, prior to radiometric determination (Ref [1]). The testing of these simple schemes has been delayed by difficulties in acquiring equipment but is imminent.

5. <u>The Application of "Fingerprinting" to WAGR Concrete</u> (B3, B3.2, B3.4, B3.5, B4 and B5)

Work reported earlier concentrated on the development of methods needed to measure comprehensive radionuclide "fingerprints" for concrete and steel in a core from the WAGR bioshield (Ref [2]). Considerable effort was expended on measuring the low energy beta-emitting nuclides ^{3}H, ^{14}C, ^{41}Ca, ^{55}Fe, ^{59}Ni and ^{63}Ni. Assessment of the overall activity of the radwaste depends on coupling the comprehensive "fingerprint" with widespread, simple activity measurements. Two simple activity measurement methods have been introduced (B4) to determine the distribution of activity in the core concrete. The results have been used to assess core activities and the effectiveness of the "fingerprinting" method (B5).

The relationship of the results to the calculated neutron flux for the core has also been examined.

The first, simple activity method introduced to measure the distribution of radiation emitted by the core, comprised of a beta-probe type 1667A, collimated with lead bricks. While this produced results for upto 1000mm from the core inner face its sensitivity was too low to produce meaningful results for the remainder of the core. The activity of concrete throughout the core could be measured by a second system. This comprised a beta-probe type BP4, in a shielded enclosure, which detected radiation emitted from a 5g sample of ground concrete in a thin-based container placed in a high counting geometry. Its sensitivity, ca 0.01c s^{-1}, was 10^2 higher than the first. The results are compared in Figure 2. There is a reasonable degree of correlation between the two sets of results to ca 1000mm from the core inner face except for a peak of radiation observed with the 1667A monitor attributable to the detection of activated "Rebar" in the core. Beyond ca 1300mm from the core inner face the activity detected in the concrete by the BP4 detector reached a constant value of ca 0.1c s^{-1}. This residual activity is probably attributable to the presence of naturally radioactive nuclides, ^{40}K and those in the U and Th series. While the gross residual activity could be interpreted as a maximum potassium concentration of 1.1% (wt) in the concrete, determination of the content by inductively coupled plasma source optical emission spectroscopy produced a figure of 0.6% (wt). The latter is comparable with the potassium content of a variety of concretes from Germany (Ref [3]) where the ^{40}K activity dominates that from U and Th series radionuclides. The influence of these activities when applying the "fingerprinting" technique is discussed later.

The effectiveness of the "fingerprinting" method for predicting activities in WAGR concrete has been assessed by examining correlations between the activities of individual radionuclides, the gross beta-count rates, measured by the BP4 monitor and the calculated thermal neutron flux (see Table I, and Figure 3). The data on nuclide activities in Table I have been amended and extended from those presented previously (Ref [2]), in part to eliminate erroneous results caused by cross-contamination observed in a few samples.

The gross beta-count rates, for the BP4 monitor, shown in Figure 2, indicate that they fall logarithmically between ca 100mm and 1000mm from the core inner face. Comparison of logarithmic fits to ^{3}H, ^{152}Eu, ^{60}Co and ^{14}C activities in this region with those for the gross beta-count rate and thermal neutron flux, in Figure 3, suggests that they correlate reasonably well, and that "fingerprinting" will be effective in this region.

Predicted and measured activities in the WAGR concrete core are compared in Table I. The predicted values were obtained from the comprehensive "fingerprint" for the 110mm sample and normalisation using the gross beta-count rates, corrected for "natural activity". There is a strikingly good agreement between the experimental and predicted activities for ^{152}Eu. This is probably because the ^{152}Eu activity is the principal contributor to the gross beta-counts throughout much of the core. Few gross differences are apparent between predicted and experimental values. For activated materials differences may arise because of variations in the chemical composition of the material and differences in activation cross-sections for thermal, epi-thermal and fast neutrons. Post irradiation mobility can cause problems, notably for ^{3}H. Analytical errors can arise because of the loss of volatile species, or because of inadequate separation of difficult to measure nuclides such as ^{41}Ca, ^{55}Fe, ^{59}Ni and ^{63}Ni. The last may be revealed by intercomparison analyses at TÜV Sudwest. Some confidence in the analytical results is given by predictive calculations of activation products in the core by M T Cross, (Ref [4]) which usually were found to agree with experimental data for ^{3}H, ^{41}Ca, ^{55}Fe and ^{63}Ni within a factor of two.

Based on the "Drigg Conditions for Acceptance" (Ref [5]) concrete beyond 265mm from the core inner face could be sentenced as low level waste, <12K Bq g^{-1} (Beta/gamma activity). The major error in assessing this cut-off point is the rapid fall in radioactivity with distance along the core.

References

[1] TUSHINGHAM, J W A, AEA Technology, Fuel Services, Report, AEA-FS-0155(H) (1992).
[2] MCMILLAN, J W, CEC Report EUR 14498 EN (1991) pp185-191.
[3] SCHMID, F J, CEC Progress Report EUR 14498 EN (1991) pp176-181.
[4] CROSS, M T, Private communication to J W McMillan, 18/12/92.
[5] COYLE, A, "Conditions for Acceptance by British Nuclear Fuels plc of Radioactive Waste for Disposal at Drigg (CFA)" Issue 93/1, Dec, 1992.

Table I: Activities in WAGR Concrete; Measured and Predicted by "Fingerprinting", from Data at 110mm Sample and Gross Beta Counting

Distance from Core Inner Face (mm)	Type of Result	Gross ß-Count c s^{-1}		Nuclide Activities Bq g^{-1}									
		Measured	Corrected for Natural Activity	^{152}Eu	^{154}Eu	^{155}Eu	^{3}H	^{14}C	^{133}Ba	^{134}Cs	^{60}Co	^{55}Fe	^{41}Ca
10	Exptl	277	277	2151	257	9	28800	20	17	40	514	452	87
	Predicted	-	-	2209	164	4	30900	18	14	23	465	709	55
55	Exptl	268	268	2212	231	6	-	-	13	29	371	589	130
	Predicted	-	-	2137	159	4	-	-	14	22	449	686	53
110	Exptl	443	443	3533	262	7	49400	28	23	36	743	1134	88
<u>Ref. Point</u>	Predicted	-	-	3533	262	7	49400	28	23	36	743	1134	88
290	Exptl	111	111	927	67	2	9790	6	4	6	170	-	-
	Predicted	-	-	885	66	2	12380	7	6	9	186	-	-
530	Exptl	18.2	18.1	107	3	-	508	0.5	0.2	0.3	9	-	-
	Predicted	-	-	144	11	-	2020	1.1	0.9	1.5	30	-	-
590	Exptl	4.2	4.1	40	-	-	-	-	-	-	-	-	-
	Predicted	-	-	33	-	-	-	-	-	-	-	-	-
1020	Exptl	0.14	0.02	0.78	-	-	5.6	-	-	-	-	-	-
	Predicted	-	-	0.16	-	-	2.2	-	-	-	-	-	-
1320	Exptl	0.13	0.01	<0.03	-	-	9.8	-	-	-	-	-	-
	Predicted	-	-	0.08	-	-	1.1	-	-	-	-	-	-
1690	Exptl	0.11	(-)0.01	-	-	-	2.3	-	-	-	-	-	-
	Predicted	-	-	-	-	-	0.0	-	-	-	-	-	-
2170	Exptl	0.13	0.01	-	-	-	2.5	-	-	-	-	-	-
	Predicted	-	-	-	-	-	1.1	-	-	-	-	-	-
2660	Exptl	0.11	(-)0.01	-	-	-	1.5	-	-	-	-	-	-
	Predicted	-	-	-	-	-	0.0	-	-	-	-	-	-

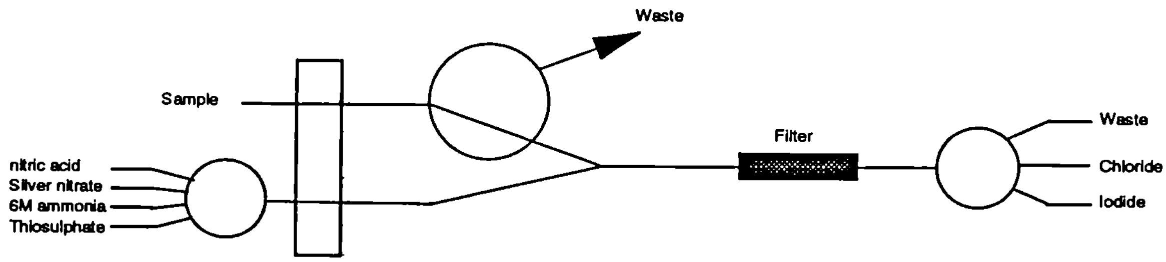


Figure 1: A flow-injection scheme for the separation of chloride and iodide using silver halide precipitation

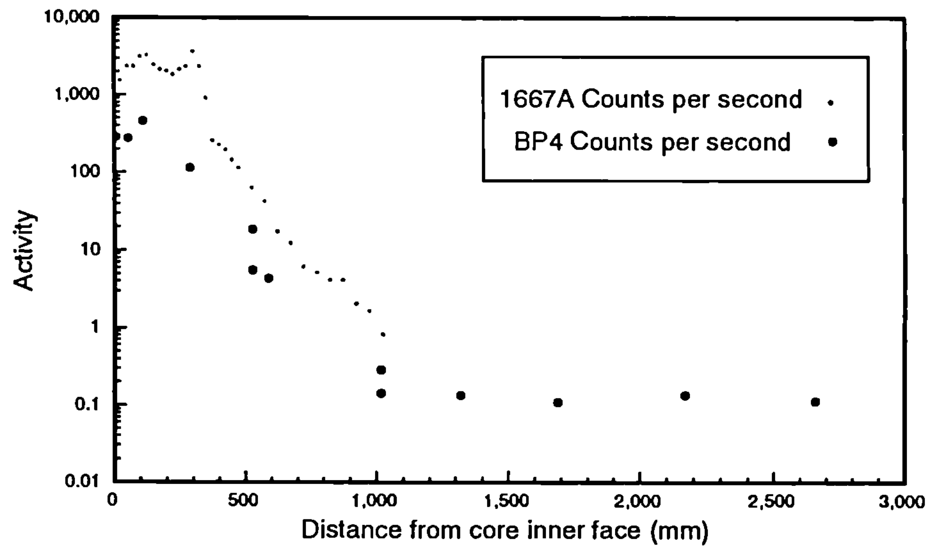


Figure 2: WAGR core radiation measurements using two different gross beta-counting systems

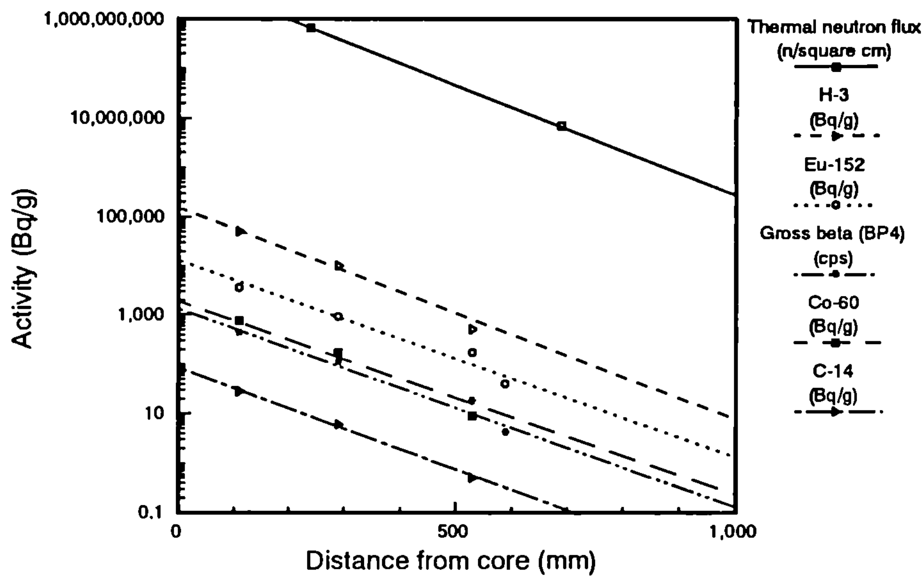


Figure 3: Correlation of the gross beta-count rate, the thermal neutron flux and the activation product distributions for H-3, Eu-152, Co-60 and C-14 in a WAGR concrete core in the high activity region ca 100-1000mm from the bioshield inner face

6.7. QUANTIFICATION OF ACTIVITY LEVELS AND OPTIMISATION OF DOSE RATE MANAGEMENT TO PREPARE STAGE 3 DECOMMISSIONING OF GAS-COOLED REACTORS

Contractors: CEA-VALRHÔ, Radia
Contract No.: FI2D-0044
Work Period: October 1990 - June 1994
Coordinator: J R COSTES, CEA/DCC/UDIN, Bagnols-sur-Cèze
Phone: 33/66 79 13 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

As part of the preparatory work for Stage 3 decommissioning of the G2/G3 gas-cooled reactors at Marcoule, the project involves:

- quantifying the activity levels of complex core structures based on theoretical analysis and on a large number of dose rate measurements;
- design of a software package to optimise the dismantling and related operations and best minimising of the dose rates incurred by the personnel.

It is important to determine the dose rates and time necessary on each manual dismantling operation, and to assess the material activity levels for optimum waste conditioning and disposal.

The development of suitable software tools and thorough examination of all the possible scenarios are very time-consuming undertakings requiring aid beyond national boundaries.

B. WORK PROGRAMME

B.1. Dose rate measurements and analyses of core samples after a literature review (CEA).
B.2. Analyses of geometric, physical and radiological data (Radia)
B.3. Development of a computer programme to calculate gamma-activity levels in the entire core (Radia)
B.4. Development of a computer programme to minimise dose rates during human interventions (CEA)
B.5. Comparison of calculated and measured results (CEA).
B.6. Examination of dismantling scenarios (CEA).
B.7. Revision of expert software considering decommissioning time and doses (CEA).
B.8. Evaluation of costs, radioactive job doses, working time and secondary waste arisings (All).

C. **PROGRESS OF WORK AND OBTAINED RESULTS**

Summary of main issues

Our objective is to develop a model to minimize occupational doses during manual decommissioning or nuclear waste handling operations. The ultimate goal is to determine the optimum task sequence so that operators will receive the lowest reasonably achievable radiation dose on completion of the G2 dismantling.

Progress and results

B4. Development of a computer program to minimize dose rates during human intervention : MINODDIN

In a zone encumbered with equipment and containing several radioactive sources of different intensities, it is reasonable to assume that the order in which the radioactive items are dismantled is a significant factor : the final occupational dose will vary according to the sequence of operations. The optimum sequence is not intuitively obvious, and can only be determined by computer-aided means. This involves defining the work area in simple geometric terms, defining the duration of personnel exposure, and minimizing the doses by selecting the best set of options for a given context.

In the MINODDIN code, the most detailed representation at the present time corresponds to two levels of 16 x 16 rectangular cells of known dimensions. The second level is provided to account for nuclear cells containing mezzanines or high-placed objects accessible from platforms, balconies or ladders. The operator may designate up to six routes between the first and second levels ; only a single exit is provided from the first level.

Calculating the Dose Rate at Any Point and the Absorbed Dose

The dose D absorbed by an operator is determined from the following relation: D = d.t.k.

where : D : absorbed dose (10^{-2} mSv)
d : ambient dose rate (10^{-2} mSv.h^{-1})
t : exposure time (hours)
k : adjustment or weighting coefficient

Radiation sources are characterized by their position (x, y), the necessary working time (tw), their type (point or extended) and their dose rate at 1 meter (D_0). The dose rate attributable to a source S at a given point (x, y, z) is expressed as follows :

$$d_{s(x,\ y,z)} = \frac{d_0}{r^2{}_s} \qquad \text{if S is a point source}$$

$$d_{s(x,\ y,z)} = \frac{d_0}{r_s} \qquad \text{if S is an extended source}$$

where r_S is the distance between source S and point (x, y, z).

The dose rate of all the sources S at a point S_m, which is also a source, is expressed by the following relation :

$d_{Sm} = d_{0Sm} + \Sigma ds$ A dose rate is thus assigned to each cell.

The exposure time is the sum of the unit working times t_w (i.e. dismantling, waste handling and removal) and the time required by the operator to enter and leave the zone. The absorbed dose for all sources S_m is then :

$$D_{Sm} = t_{wm} \cdot C_p \cdot \sum_1^n d_n + t_{wj} \cdot d_{Sm}$$

where d_n represents the dose rates in the *n* cells on the in/out route, and where the transit time is assumed equal to a fraction C_p of the working time t_{wm} for source S_m.

Obstacles are handled by the code as cells through which it is impossible to transit. Low level irradiating objects are considered as sources with a zero dose rate. It may be necessary to remove them before the sources in order to gain access or to save time.

Dose Minimization Algorithm

Determining an optimum dismantling sequence for *n* sources involves *n* factorial (n!) possible combinations. For each combination, all the minimal paths in the cell must be constructed in terms of occupational dose between the sources and the zone access door. No more than one source is dismantled at a time, and it is removed via the access door. The path must be recalculated after each operation, since the dose rate map has been modified.

The algorithm uses an iterative process of minimal dose propagation from the "exit" point p using 2 arrays. An array T containing 256 x 256 cells is created to enter all the corresponding doses T_{ij}. A second array T* containing 256 x 257 cells is then created to enter all the doses sustained in moving from point i to point p via another (unique) point j.

By generalizing this process to a 532 x 532 array, representing two 16 x 16 levels with the routes between them (e.g. stairs), it is possible to calculate the routes along which operators will sustain the lowest doses as they dismantle a source. The process is enhanced by simultaneously following the shortest possible geometric paths.

After examining all n! source permutations, the results are sorted according to the dose incurred. The code also displays the status of both cell levels after each source is dismantled, together with the recommended in and out routes from the cell exit door.

Artificial Intelligence Module

Processing n! permutations is time-consuming on a microcomputer. An artificial intelligence layer is used to limit the number of possible solutions to a maximum of 7! (i.e 5040) whenever possible.

It is therefore advisable to conduct several consecutive simulations, including preliminary tasks (e.g. setting up biological shielding, prior decontamination) in order to ensure that the final scenario does in fact represent the minimum occupational dose. This requires a specific "preparation" code version.

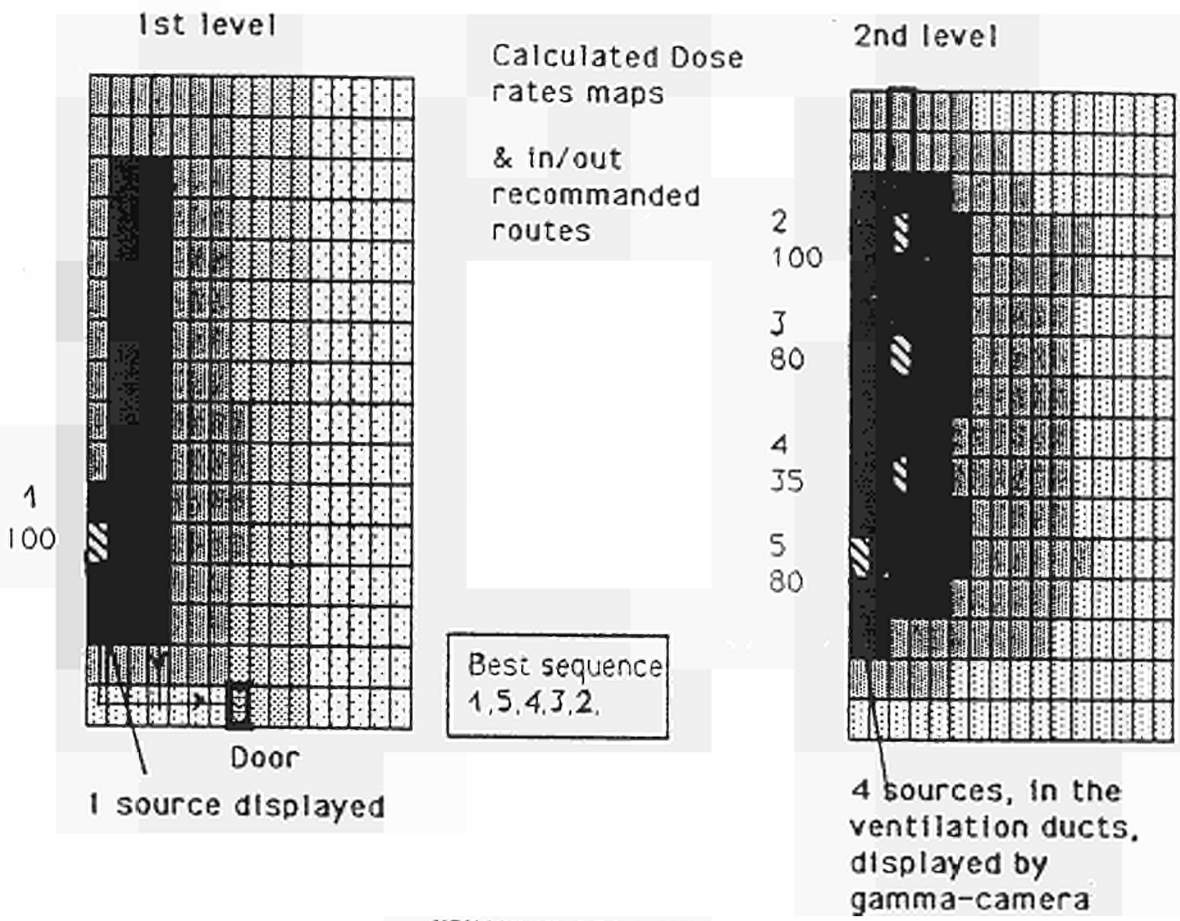
Some permutations are automatically excluded. These include sources hidden behind other sources, very low-level remote sources, and highly contaminating sources eliminated last. Allowance is also made for certain procedural requirements, such as the maximum exposure time, or the maximum work duration while wearing a mask ; these time limits imply additional round trips, working time and irradiation. This approach provides for a more realistic simulation.

The program runs on an i486 microcomputer with extended memory. It has been tested, for example, by simulating the human dismantling operations for a utility room adjacent to cesium source conditioning cells. Several contaminated ventilation ducts crossed through the room at a height of 2.5 meters. Figure 1 shows the two arrays corresponding to the two working levels : floor level and duct level (+2.5 m). Gamma camera measurements identified five sources which are marked in the arrays.

MINODDIN allowed several scenarios (setting up biological shielding or prior decontamination) to be simulated. In the recommended scenario, the ducts were opened and the contamination immobilized with varnish before cutting them into sections 50 cm long and placing them in waste drums. MINODDIN defined a task sequence involving 30 % lower occupational doses than the least favorable solution.

Next step

A more advanced version will be applied to the G2 Core dismantling.



MINODDIN DISPLAYS

Figure 1: Two arrays corresponding to the two working levels: floor level and duct level

6.8. DECOMMISSIONING COSTS FOR NUCLEAR INSTALLATIONS

Contractors:	NIS Ingenieurgesellschaft mbH
Contract No.:	FI2D-0051
Work Period:	July 1991 - December 1991
Coordinator:	P PETRASCH, NIS
	Phone: 49/6181/10 94 58 Fax: 49/6181/12 00 33

A. OBJECTIVE AND SCOPE

In 1977, the Commission of the European Communities initiated a study to calculate the decommissioning cost for nuclear power plants (EUR 5728d).
The main objective of this contract is to update the study on the state-of-the-art, taking into account the technical advances occurred since 1977 in the decommissioning of nuclear power plants as well as in the conditions and means to calculate the decommissioning costs. The study will focus on representative commercial German LWRs. Nevertheless, the calculation method is made in a form allowing comparison/extrapolation to the decommissioning costs of other EC nuclear installations.

B. WORK PROGRAMME

B.1. **Description of the boundary conditions for the decommissioning with particular view to nuclear power plants in Germany**
B.2. **Detailed technical description of decommissioning concepts for a BWR (referenced by nuclear power plant Biblis)**
B.3. **Calculation of the decommissioning costs for the concepts given in para B.2.**
B.4. **Comparability with costs of other EC decommissioning projects, either originating from real projects or estimated.**

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1991. Final Report is available as EUR report n° 14687 DE.

6.9. DEVELOPMENT OF A PROTOTYPE APPARATUS VISUALISING ON A SCREEN THE GAMMA SOURCES SUPERIMPOSED ON THE IMAGE OF THE VISION FIELD

Contractors:	CEA Valrhô, CEA Saclay
Contract No.:	FI2D-0055
Work Period:	October 1991 - May 1993
Coordinator:	G. IMBARD, CEA Valrhô
	Phone: 33/66 79 63 10 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

The project consists in further developing a measuring device composed of a video camera, a gamma detector, an image processor and a monitor on which the radioactive radiation intensities will be superimposed on the related visual field.
The instrument (diameter <200 mm, length <400 mm and weight around 50 kg) will be handled by a specific remote-controlled support.

The scope of the programme is to produce a prototype gamma-camera that can be used in hot cells of decommissioning projects.
The development of the R&D programme will entirely be performed by the two CEA research centres, CEN-Valrhô and CEN-Saclay, with CEN-Valrhô as coordinator.

B. WORK PROGRAMME

B.1. Measuring performance optimisation of the demonstration device
B.2. Testing the apparatus under real conditions
B.3. Integration of the measuring chain in a biological protection shield
B.4. Development of the image processing software
B.5. Calibration operations
B.6. Prototype application on a decommissioning site.

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

1992 has been, as expected, a crucial period for the development of the prototype gamma-camera. After an examination of the detection chain, and a serie of trials in laboratory and on site under real conditions, we were able to add a new intensification tube (20XX) which improves by 50 % the detection sensitivity.

A set consists of a signal acquisition system in real time (25 images/s), and a treatment system allowing to visualize on a video screen gamma and visible scenes and to improve their quality with a large variety of imaging functions (substracting the background noise, filtration, integration ...).

The major part of the acquisition software and signal treatment has been developed on a DEC work-station.

Finally, the integration of the whole detection set into a "case" has been studied. It will provide protection against both contamination and irradiation (Tungsten alloy shielding) and facilitate its subsequent connection to a remote handling and swivelling system for measurements inside hot cells.

Progress and results

1. Measuring performance optimization of the demonstration device (B.1.)

The electronic detection set, used at the end of 1991, was examined. That is the CCD intensified camera (LH 5038) and the sub-assembly consisting of the intensification tube TH 9473 associated with the optical fiber taper. This work carried out by the LEP (Philips Electronics Laboratory) indicated that although individual characteristics of the camera and the tube (gain and spectral response) are good, there is room for large improvement of global sensitivity by using a 20XX tube for electronic image reduction.

The fabrication of such a tube was then ordered and it was delivered at the end of September 1992.

Laboratory trials conducted on this tube from October underlined a significant gain regarding sensitivity (more than 50 % whatever the conditions). This fell however largely below the theoretical estimate calculated by the experts (according to the LEP, the estimated gain could have reached 1000 %). The deviation between estimation and reality is due to the higher noise generated by the new tube which has not been taken into account. This situation can be explained by the innovating characteristics of our implementation field.

The gain in sensitivity is nevertheless sufficient to maintain this new tube and to integrate it into the detection chain (between the scintillator and the camera : see figure 1).

Note : This gain in sensitivity comes however with a slight degradation of the angular resolution, but substantial improvement will be studied in 1993.

2. Testing the apparatus under real conditions (B.2.)

A series of measurements was performed in 1992 in a shielded cell of the Radiometallurgical Laboratory (RM2) in the process of being dismantled at the Fontenay-aux-Roses Nuclear Center (France).

Specific equipment was developed for this purpose to protect the detection material against irradiation and contamination (see Figure 2, 3 and 4), and also to allow its penetration through a trap door located in the upper roofing of the active cell (figure 5).

The radiological situation was of interest, and photographs were taken of the stainless steel floor, waste dustbins and highly irradiating fuel element parts. Figure 6 shows a highly irradiating zone corresponding to a basket filled with fuel element parts.

The global acquisition time of the gamma image was about 5 seconds with an equivalent dose rate of about 1 mSv/h at the detector location.

Two major points can be drawn from this experience:

a) the systems performing both acquisition and treatment (PERICOLOR and SAPPHIRE) lent to us for these trials have not enough potentialities for our expected development.

b) particular attention should be given to protect and maintain the detection equipment against contamination.

3. Integration of the measuring chain in a biological shield (B.3.)

Due to the advantages of both tubes TH 9473 and 20XX having respectively a better resolution and a better sensitivity, we adopted for the interchangeability of the tubes.

Two essential specifications of the case shielding were required :

a) a maximal external diameter of less than 170 mm (support included) allowing its cell introduction through the existing opening.

b) minimal attenuation of the shielding : a tenth value layer (attenuation by a factor of ten) for Co60 gamma rays.

The mass optimization study lead to the development of a preliminary project. The shielding will be of tungsten alloy (thickness >20 mm around sensitive components) with the following physical characteristics : diameter 120 mm, length 430 mm, mass 40 kg.

While developing the protection of the measurement chain, important points of view were taken into account, that is for maintenance (dismountability, interchangeability of collimators and scintillators, intensification tubes and camera) and for intervention (reservations for handling system, care of decontamination ...). The detection set will be integrated into the shielding at the beginning of 1993.

4. Development of the image processing software (B.4.)

After examination of the existing equipment, we selected the DVS 3000 system from Hamamatsu (acquisition and video signal treatment in real time), coupled with a DEC work station with UNIX.

This configuration combines the advantages of DVS 3000 (autonomy, compactness, limited cost, compatibility with IEEE 488 standard) and those of the work station (powerful and multi-task mode computer, graphic convivial interfaces on a high resolution screen, open system allowing illimited further developments). Since we got the equipment, we developed a specific software from the DEC work station. The acquisition part was done at the end of 1992 (running the DVS 3000 system from the station, image encrustation on the screen in real time, image transfer and filling) as well as part of the analysis operation (histogram, profil plotting, ...) and image treatment (spatial filters, look up tables).

5. Calibration operations (B.5.)

A series of measurements have been carried out with radioactive point sources of Co60 and Cs137 delivering absorbed dose rate levels between 0.025 mGy/h and 3 mGy/h at the camera location. A good detection linearity was found in the range between 0.2 to 3 mGy/h. These trials will be pursued in 1993.

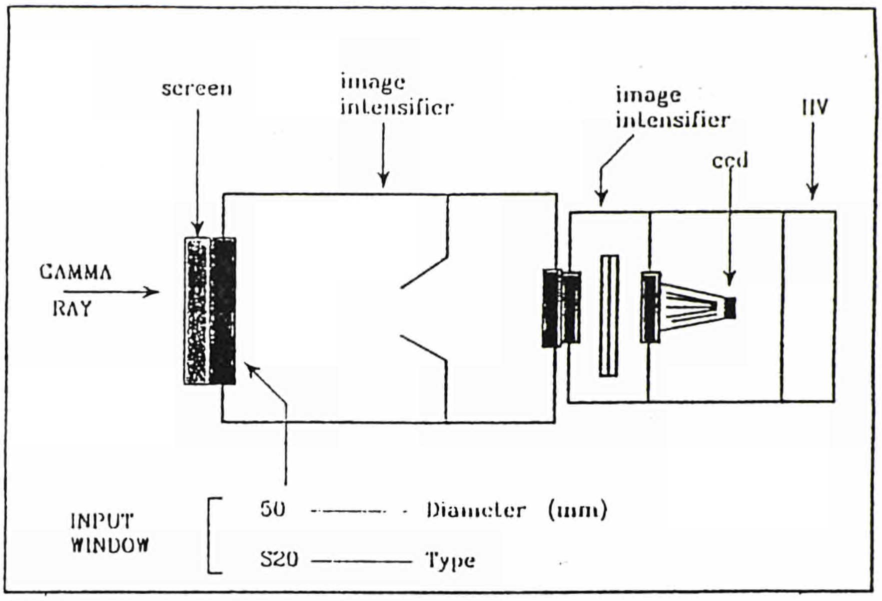
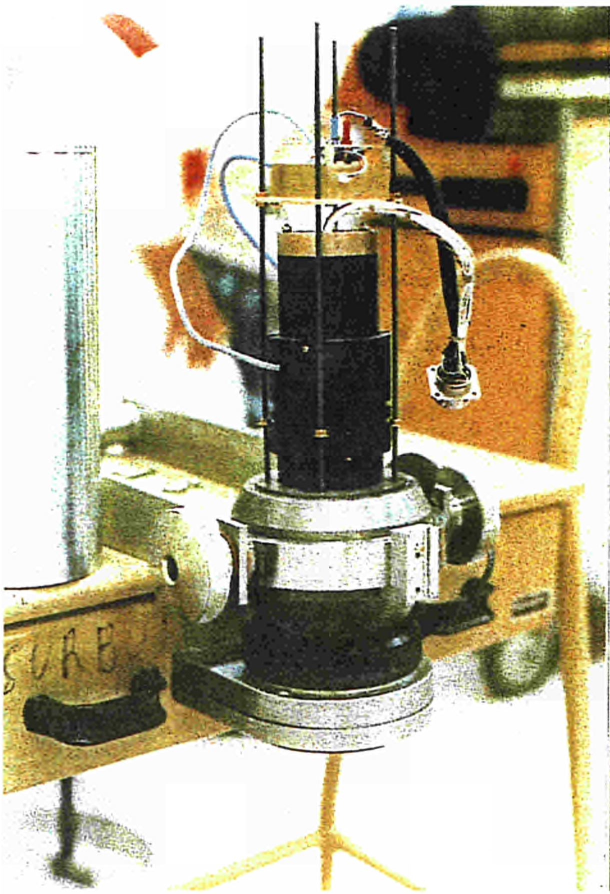


Figure 1 : Detection chain with 20XX intensification tube



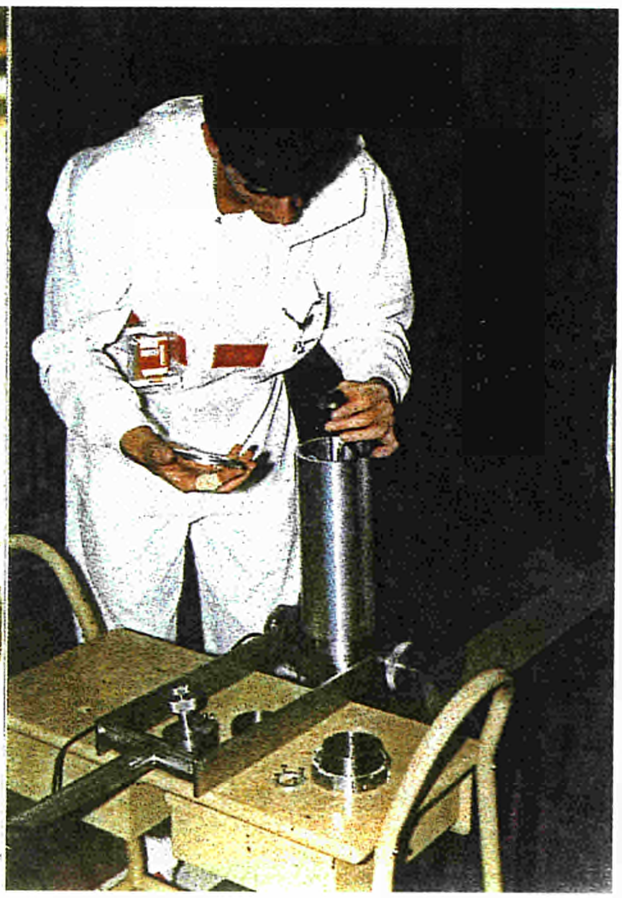


Figure 2 : Gamma camera with shielding

Figure 3 : Idem 2 + protection case against contamination

Figure 4 : Gamma camera placement on the introduction pole

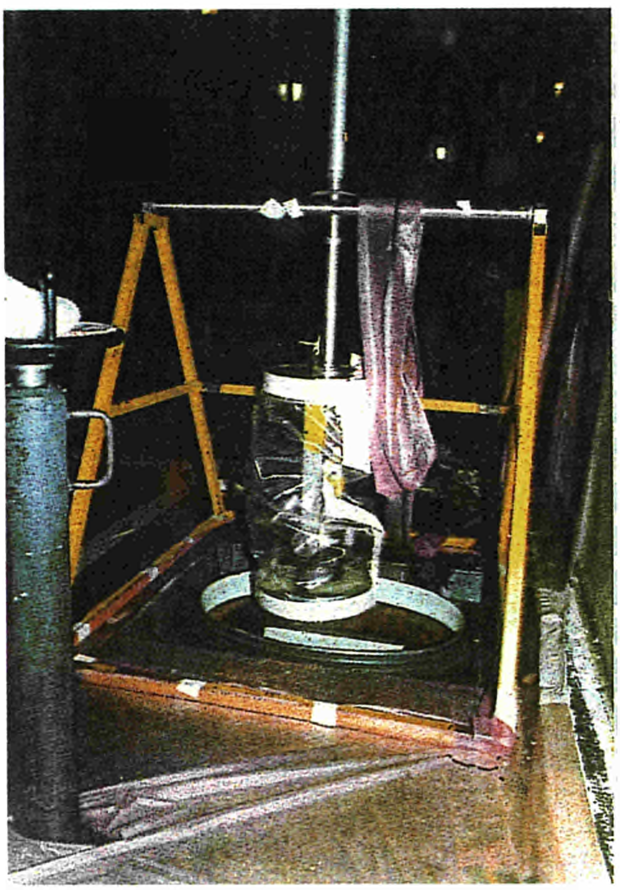


Figure 5 : Gamma camera ready to be introduced

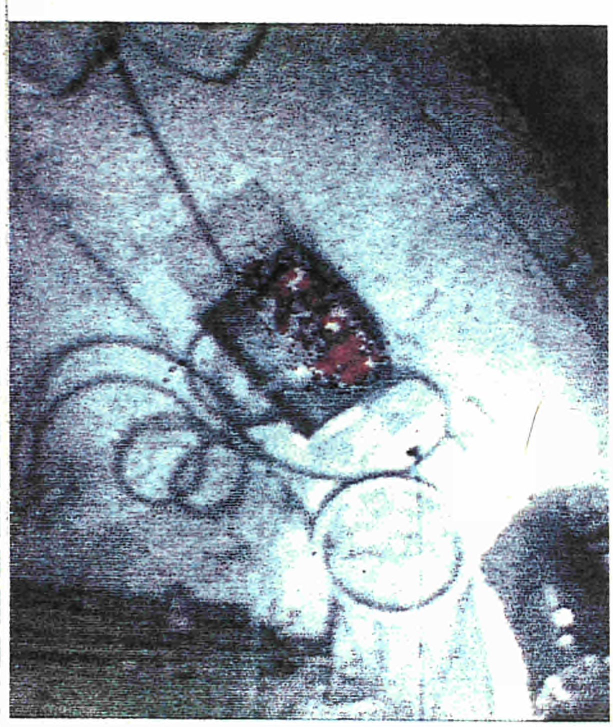


Figure 6 : Picture of radioactive fuel parts

SECTION B: IDENTIFICATION OF GUIDING PRINCIPLES

Section B of the programme is concerned with the identification of guiding principles relating to:

- the design and operation of nuclear installations with a view to simplifying their subsequent decommissioning;
- the decommissioning operations with a view to making occupational radiation exposures as low as reasonably achievable;
- the technical elements of a Community policy in this field.

7.1. POLICIES, REGULATIONS AND RECOMMENDATIONS FOR DECOMMISSIONING

A study is being performed by the Commission together with a group of experts, with the objective of assembling and discussing policies, regulations and recommendations for decommissioning, and of recommending Community actions in this field.

The study is structured as follows:

1. Introduction
2. General principles and international recommendations relevant for decommissioning
3. European Community requirements and recommendations
4. Policies, regulations and recommendations in Member States
5. Present decommissioning practice
6. Conclusions and recommendations of the Working Group

Annex 1 - National policies, regulations and recommendations
Annex 2 - Selected decommissioning cases

The Working Group met in September 1991, in January 1992 and in October 1992. By the end of 1992, a preliminary draft including all Chapters had been prepared and discussed.

7.2. PREPARATION OF A DECOMMISSIONING HANDBOOK

Contractor:	AEA-Wind., CEA/IPSN, ENEL, M. Lasch, F.W. Bach, CEA/UDIN, GNS, ONDRAF/NIRAS, FRAMATOME
Contract No.:	FI2D-0073 to FI2D-0081
Work Period:	April 1992 - August 1993

A. OBJECTIVE AND SCOPE

A handbook of the technology for decommissioning of nuclear installations will be prepared. The main subject of the handbook will be the detailed description of the state-of-the art techniques. For each of these techniques, the following information should be provided (as applicable and available):

- range of application conditions for which the technique is considered first choice,
- performance characteristics,
- by-product characteristics,
- specific radioprotection and safety aspects,
- cost data, including employment of labour,
- existing specific equipment,
- necessary auxiliary equipment,
- an example of past practical application,
- any other relevant information.

Techniques that are not state-of-the art, e.g. obsolete techniques or techniques which are being developed but not yet proven, should only be mentioned and qualified briefly.

The handbook should indiscriminately include available relevant information of any origin.

B. WORK PROGRAMME

B.1. Editorial assistance (AEA-Wind.)
B.2. Characterisation of radioactivity (CEA/IPSN)
B.3. Surface decontamination (ENEL, M. Lasch)
B.4. Dismantling and segmenting (F.W. Bach, CEA/UDIN)
B.5. Management of materials from dismantling (GNS, AEA-Wind.)
B.6. Radiation protection and safety techniques (AEA-Wind., ONDRAF/NIRAS)
B.7. Installation design and operating features (FRAMATOME)

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

One of the major aspects of producing the Handbook is the exchange of information between authors/Member States. To facilitate this, a meeting was held in Brussels on 8 December 1992 for all authors.

Progress and results

1. Editorial assistance (AEA Windscale)

Editorial assistance is divided into three main areas: the collation of a table of contents; the review of draft contributions to assure that the English is clear and correct and the type setting of the Handbook to assure conformity with the editorial requirements.

A detailed Table of Contents for the Handbook has been produced, based on information received from all authors. This is being used as the basis for the production of individual chapters, and for identifying areas where an exchange of information between authors is required.

Sufficient contributions from authors have now been received to allow the full editorial procedure to commence: although significant changes to the text are still possible.

2. Characterisation of radioactivity (CEA/IPSN)

The complete chapter is being written by CEA/IPSN. Sections will be included in the chapter on: the need for radioactivity characterisation, basic radiation measurement techniques, in-situ measurements, laboratory measurements, radioactivity inventory estimation, measurement strategies and clearance of buildings and sites for public use.

The draft contributions from CEA is expected to be received by the end of February 1993.

3. Surface decontamination (ENEL, M Lasch)

This chapter of the Handbook is concerned with decontamination. ENEL are providing sections on the general approach to decontamination, decontamination for segmented parts and decontamination for building surfaces. M Lasch is providing sections on basic decontamination methods and decontamination for large volume closed systems. This basic format was subject to some discussion at the contractor's meeting with the result that ENEL and M Lasch will be collaborating to ensure duplication within the chapter is kept to a minimum.

Draft contributions have been received from both ENEL and M Lasch. Some harmonisation of the technical content of these sections is required and was discussed at the contractors meeting. This is expected to be complete by the end of February 1993. Detailed editing of these sections is expected to begin shortly.

4. Dismantling and segmentation (F W Bach, CEA/UDIN)

Draft contributions have been received for section 4.2. (thermal cutting) from the University of Hannover and for sections 4.4, 4.5 and 4.6 (mechanical dismantling techniques, microwave spalling techniques and explosive techniques respectively) from CEA. The remaining sections (section 4.1 - introduction [F W Bach and CEA], section 4.3 - hydraulic cutting techniques [F W Bach], section 4.7 - cutting under water [CEA], and section 4.8 - guidelines for selection of appropriate dismantling and segmentation techniques [F W Bach and CEA]) are expected to be complete by the end of February 1993.

5. Management of materials from dismantling (GNS, AEA)

In this chapter, GNS are providing sections on conditioning techniques, conditioning equipment (for liquid and solid materials) and on recycling techniques. AEA are providing sections on materials arising during decommissioning, special conditioning techniques and waste transport and disposal packages.

The draft contribution from GNS has been received, and the AEA section is complete and will be distributed shortly.

6. Radiation protection and safety techniques (AEA, ONDRAF/NIRAS)

In this chapter, AEA are providing sections on radiation dose control measures, remote operations and filtration techniques. ONDRAF are providing the sections on manual operations, shielding and containment techniques and safe storage.

ONDRAF have supplied the draft sections, AEA's contribution on radiation dose control is expected to be complete by the end of February 1993, AEA's section on remote handling has been circulated and a number of comments have been received from authors.

7. Installation design and operating features (FRAMATOME)

The complete chapter is being provided by FRAMATOME and includes sections on the evaluation of activity and radwaste volumes, design features to facilitate decommissioning, replacement of damaged components, improvements of plant layout design, and on documentation and updating.

The draft chapter has been received from FRAMATOME and has been edited. Detailed comments will be sent to the author shortly.

SECTION C: TESTING OF NEW TECHNIQUES IN PRACTICE

A. Objective

The projects and studies in this section aim at testing, demonstrating and assessing new decommissioning techniques under real conditions of radioactivity configuration, size, accessibility and the state of the plant. The four large Pilot Dismantling Projects (WAGR/Windscale, BR3/Mol, KRB-A/Gundremmingen and AT-1/La Hague) are the focal point of this section. Large-scale active testing of new techniques is also performed in a number of other projects ("alternative tests").
In order to obtain more generally applicable knowledge, the costs, occupational doses, working hours and waste arisings determined for unit operation in the course of the above projects are compiled in a data base.

B. Subjects of the research performed under the previous programmes (1979-88)

Large-scale investigations on various decommissioning techniques (such as decontamination, cutting, activity measurements) were performed in the 1984-88 decommissioning programme. These investigations concerned the dismantling of five reactors, three fuel fabrication facilities and one high-level waste vitrification facility.

C. Programme 1989 to 1993

Section C includes:
- the execution of four pilot dismantling projects
- alternative large-scale tests to be performed in nuclear installations other than the pilot dismantling projects:
- secondment of scientific staff from Member States to the pilot dismantling projects: The operators of pilot projects receive staff from organizations in the other Member States for active cooperation within the framework of the project.
- Establishment of a data base for costs, operational doses, working times and waste arisings.
- Establishment of a data base for the performance of cutting/segmenting techniques.

D. Programme implementation

In the pilot dismantling projects, important milestones have been reached: at AT-1, after the dismantling and removal of an equipment, the final phase of decontamination, waste removal and site cleanup has started, whereas in BR-3, after the removal and segmentation of the thermal shield, work is now concentrating on the dismantling of the other RPV internals. At KRB-A and WAGR, the commissioning of equipment for remote disassembly of RPV internals is under way. Among the large-scale tests, the projects on remelting steel and other metal scraps made the most significant progress. The collection of specific data for the two data bases was pursued with particular priority in 1992.
Twenty-five research contracts relating to Section C were in progress during 1992, two were completed in 1990 and 1991.

8.1. PILOT DISMANTLING OF THE WAGR. DISMANTLING OF TOP BIOSHIELD REFUELLING STANDPIPES, VESSEL TOP DOME, HEAT EXCHANGER, REMOTE DISMANTLING OF HOT BOX, REMOTE WASTE PACKAGING

Contractors: AEA-Wind.
Contract No.: FI2D-0001
Work Period: October 1989 - December 1993
Coordinator: J H LENG, AEA
Phone: 44/9467/72430 Fax: 44/9467/72409

A. OBJECTIVE AND SCOPE

The Windscale Advanced Gas-cooled Reactor (WAGR) had a capacity of 33 MWe and was operated from 1962 to 1981. Dismantling of the plant has started and is planned to be completed in 1996.

Considering that the experience to be gained from the dismantling of the first large-scale nuclear installations in the Community should be made available to all Member States, the Commission selected WAGR as a pilot dismantling project for the 1989-93 R&D programme on the decommissioning of nuclear installations. The Commission, through shared-cost participation in specific parts of the project, is promoting the use of advanced techniques and the performance of collateral investigations, in order to enhance the production of useful knowledge and experience to serve in subsequent decommissioning tasks. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is an important objective of this project.

As a gas-cooled reactor, WAGR provides opportunities for testing decommissioning techniques against the specific requirements of such reactors, which represent the majority of the first-generation nuclear power reactors to be decommissioned in the Community in the near future. The first phase of the contract involves in particular the dismantling of the top biological shield, of refuelling standpipes and of the reactor pressure vessel top dome as well as inactive trials of the remote dismantling machine. The second phase covers dismantling of the hot box, one steam generator and the remote packaging of waste.

The estimated radioactive inventory is in the order of 10^5 Ci; estimated dose rates are in the range of 0,1 to 1,5 mSv/h.

B. WORK PROGRAMME

B.1. Dismantling of the top biological shield (TBS), a 60 t disc-shaped steel and concrete structure, by thermic lancing after its moving into a ventilated containment placed on the refuelling floor.

B.2. Cutting and handling of the refuelling standpipes, i.e. 253 pipes of 6.3 m length penetrating the upper part of the reactor block, by four cuts, with an internally rotating plasma arc torch.

B.3. Cutting and dismantling of the pressure vessel top dome, a complex steel structure of 6.5 m diameter and 98 mm maximum thickness, by in-situ segmentation in two parts using a semi-remote operated oxy-gas cutter placed on a tractor followed by post-segmenting in a temporary containment placed on the refuelling floor.

B.4. Inactive trials of the remote dismantling system, comprising a rotating floor shield, an extendable mast carrying a telemanipulator arm, and a remotely operated conveying system, in a test facility representing a 30° sector of the reactor pressure vessel.

B.5. <u>Generation of specific data</u> on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1., B.2., B.3., B.6., B.7. and B.8..

B.6. <u>Dismantling of a WAGR heat exchanger</u>, beginning with the top outer shell to gain access to the tube banks. Protected by temporary shielding, the tube bank and the stainless steel thermal insulation packs of the economiser, evaporator and superheater sections will be decontaminated and removed successively.

B.7. <u>Remote dismantling of the hot box</u>, using the remote dismantling system (see B.4.). This operation requires severing the refuelling tubes and the 130 small diameter stay tubes remotely.

B.8. <u>Remote packaging of intermediate level waste</u>, consisting of the transfer, monitoring, size reduction and cement encapsulation of activated pressure loop components, hot box sections and operational waste.

C. PROGRESS OF WORK AND RESULTS

Summary of Main Issues

WAGR has 253 standpipes, 247 normal refuelling standpipes and 6 pressure loop standpipes. The cutting and removal of the 247 normal standpipes down to cut number 5 (200 mm above the hot box) and 6 pressure loop tubes has been reported previously. However, during this reporting period dose and operational data from all completed phases of standpipe cutting have been assessed, the results of which are summarised in the following section.

The remote dismantling machine is currently installed in the purpose built Machine Test Facility. The first phase of operator training has been completed with the second phase commencing in January 1993.

The work associated with dismantling Heat Exchanger 'A' falls logically into three areas: those aspects concerned with the decontamination and subsequent treatment of the effluent, the methods and tooling required to dismantle and the facilities required to enable decommissioning. Methodologies for these tasks have been identified and task specifications written.

A series of cutting trials have been performed under remote conditions for the testing of tools for dismantling the hot box structure. Development work has been undertaken in association with cutting techniques, handling techniques and remote viewing. In addition, results from stay tube cutting operations have been assessed and recorded.

Progress and Results

1. Cutting and Handling of the Refuelling Standpipes (B.2.)

Standpipe Cut 1 was performed by personnel working on top of the reactor Top Biological Shield. The main sources of radiation to which the operators were exposed were from contamination on the internal surfaces of the standpipes and from operational waste stored in the reactor. Radiation surveys were taken whenever a standpipe shield plug was removed to allow access for the cutting machine. Typical readings were 6–10 $\mu Sv.h^{-1}$ beta gamma at contact with the Top Biological Shield.

Following the removal of the Top Biological Shield standpipe Cut 3 operators saw an increase in radiation dose rate to approximately 20 $\mu Sv.h^{-1}$ beta gamma at contact with the open ended standpipe.

Standpipe Cut 4 was performed in similar radiological conditions to that of Cut 3 by working from the temporary floor some 5 metres above the top dome. The average dose rate at contact with the temporary floor was 20 $\mu Sv.h^{-1}$ beta gamma.

At the start of standpipe Cut 5, the average radiation dose rate at contact with the temporary floor had risen to 78 $\mu Sv.h^{-1}$ beta gamma following the removal of the RPV Top Dome. Calculations indicated that this value could increase to 450 $\mu Sv.h^{-1}$ beta gamma due to loss of shielding as the standpipes were cut and removed. Upon completion of the Cut 5 operation the average radiation dose rate had in fact risen to 225 $\mu Sv.h^{-1}$ beta gamma at contact with the temporary floor.

Semi–remote cutting of four loop tubes was performed using the same method and following standpipe Cut 5. Hence the operators were exposed to a dose rate on the temporary floor of 225 $\mu Sv.h^{-1}$ beta gamma.

Following the decision to abandon semi–remote cutting of the two small loop tubes operators performed the task manually using portable disc grinders. Skilled operators working on top of the hot box were exposed to a dose rate of 4 $mSv.h^{-1}$ beta gamma.

The results from these operations are summarised in the following Tables.

TABLE 1
SUMMARY OF DOSE UPTAKE FOR STANDPIPE CUTTING

Cut Number	Total dose (man.mSv)	Radiation Levels ($\mu Sv.h^{-1}$)
1	1.5	6–10
2	Omitted	20
3	7.4	20
4	7.2	20
5	7.3	78 increasing to 225

TABLE 2
RADIATION UPTAKE IN LOOP TUBE CUTTING

Method	No of Tubes Cut	Estimated Dose Uptake (mSv)		Actual Dose Uptake (mSv)	
		Total	Highest Individual	Total	Highest Individual
Modified Standpipe Cutter	4	4.1	1.0	1.5	0.5
Bugo Equipment	0	2.1	0.4	3.0	0.7
Manual Grinding	2	3.7	0.7	1.1	0.5

TABLE 3
TOTAL WEIGHT OF STANDPIPES

Operation	Weight of Standpipes (tonnes)
Cut 1	5.5
Cut 3	18.5
Cut 4	No pipes retrieved from reactor
Cut 5	10.75

2. Inactive Trials of the Remote Dismantling Machine (B.4.)

The decommissioning of the WAGR reactor requires a remotely operated dismantling machine to cut and, in conjunction with a handling system, transfer the material to a Waste Packaging building.

Prior to installation of the machine there is a requirement to assemble the Mast Module in a purpose–built Machine Test Facility (MTF) and perform operator training. The facility consists of a steel structure at refuelling floor level onto which the module can be erected. Holes cut through the floor levels below the facility allow installation of the mast structure. At each floor level the mast is contained within a lightweight containment cladding structure of sufficient size to allow installation of test pieces for trail cutting.

Machine build on the MTF was scheduled to be completed mid July 1992 after which trials and operator training could commence. Figure 1 illustrates the construction of the Mast Module in the MTF.

Due to several problems with the machine build, the handover of the facility for trials and training did not take place until January 1993. Following handover of the system from the installation contractor to WAGR Operations Team on 4 January 1993, the following commissioning tasks have been undertaken:

(i) 3te Hoist and Slewing Mechanism
(ii) Grab System
(iii) Stereo Viewing System
(iv) Stereo Camera Boom
(v) Manipulator as illustrated in Figure 2
(vi) Thermal Cutting System

In conjunction with the tasks on the MTF work has continued with the erection of the RDM containment over the reactor vault as shown in Figure 3. Following this phase of operator training with the machine in the MTF, the machine will be transferred to its active position over the reactor.

3. Dismantling of a WAGR Heat Exchanger (B.6.)

Heat Exchanger A is one of four heat exchangers within WAGR designed for the production of high pressure steam using the reactor cooling gas.

The exchanger consists of a steel pressure vessel, 3.35 m in diameter and 20.6 m long, standing vertically within a concrete bioshield. Operation of the reactor has resulted in the internals of the heat exchanger being contaminated with caesium and cobalt up to 1.5 x 10^{12}Bq. The heat exchanger is to be partially decontaminated and dismantled for disposal as Low Level Waste (LLW).

The work associated with dismantling Heat Exchanger A has been sub-divided into 3 separate tasks; decontamination, operational methodology and the facilities required to enable decommissioning. These tasks have in themselves been assessed to form a work breakdown structure consisting of:

Decontamination
(i) Flushing
(ii) Liquor Treatment
(iii) Waste Packaging

Operational Methodology
(i) Tooling
(ii) Cutting
(iii) Viewing
(iv) Dismantling Strategy

Facilities
(i) Access
(ii) Containment
(iii) Waste Arisings
(iv) Ventilation

The programme has now reached the stage where scheme designs for all three areas are available and specifications have been compiled in line with the projects contract strategy. Cost information is currently being sought for design/build/install or build/ install packages of work.

At a recent mid–project review it was noted that if the Heat Exchangers could be removed and sent as Low Level Waste without size reduction this would greatly reduce the dose budget to be associated with the operation. As a result the practicality of this alternative proposal is being evaluated.

4. Remote Dismantling of the Hot Box (B.7.)

Significant progress has been made in the development and testing of remotely deployed tools for dismantling the hot box structure under full remote conditions. Development work has been undertaken in several key areas, namely cutting techniques, handling techniques and remote viewing. Work has also commenced on optimising the dismantling methodology in terms of waste packaging and remote handling.

The main shell of the hot box is fabricated from mild steel, however, the entire inner surface is lined with stainless steel insulation foils up to 35 mm thick and consisting of up to 30 separate sheets closely packed together.

One of the direct consequences of the insulation is the need to use a high temperature cutting process to penetrate both the stainless steel insulation and the mild steel shell. The cutting process developed for use on WAGR consists of an oxypropane gas mixture with a powder injection of iron particles. Development trials were also performed on the dismantling of the side wall structure and the co–axial ducts. During dismantling operations the co–axial ducts will require cutting to allow the complete removal of the side walls of the hot box. Again the inside surface of the co–axial duct is lined with insulation, and due to the geometry of the duct, specially designed torches have been developed. During all the development trials considerable use was made of remote viewing systems, in particular the 3–D stereo TRV system. Confirmation of the benefits of 3–D TV were again established during the remote cutting simulations. Development work is also in hand on a range of flame viewing techniques to enable operators to view the flame during the cutting process. Experience during the cutting trials has indicated the benefits of being able to view the flame during the cutting process. A range of filter units have been tested together with a device for automatically selecting a flame viewing mode for the 3–D stereo camera. In addition, development trials have been performed on a range of remote handling devices for removing various sections of the hot box including top plate, side wall and bottom plates. It is important that the interaction between the remote manipulator and the handling device is established and that operational procedures can be established and verified during the development phase. The relocation of waste items is also being tested, this will enable the waste packaging to be optimised and thus potentially reduce the ultimate number of waste boxes for disposal.

During this reporting period the results obtained from stay tube cutting operations were assessed and summarised as detailed below:

TABLE 4
PROGRESSION OF RADIATION UPTAKE ESTIMATES FOR STAY TUBE CUTTING

	Total Dose mSv	Specific Operations mSv	General Operations mSv
Original Estimate (450 μSv/hr)	54.88	23.87	31.01
Revised Estimate (225 μSv/hr)	24.35	11.94	12.41
Scheduled Cuts (60% time)	14.84	7.27	7.56
Actual Results	7.42	4.65	2.77

TABLE 5
LOG OF DAILY RADIATION UPTAKE FOR STAY TUBE CUTTING

Day	Date	Hours Worked	Stay Tubes Cut	Fitter μSv	Rigger μSv	PW 1 μSv	PW 2 μSv	Sup 1 μSv	Sup 2 μSv	Max dose Rate μSv/hr	Ave. dose per cut μSv/cut
1	31/7/91	2.5	2	0	0	0	0	50	0		
2	1/8/91	2.0	5	13	23	31	23	37	23	18.50	30.00
3	2/8/91	6.0	5	36	36	53	20	50	37	8.83	46.40
4	4/8/91	6.0	3	70	106	74	20	137	13	17.67	140.00
5	6/8/91	5.25	6	36	10	125	38	70	17	23.81	49.33
6	7/8/91	5.75	9	157	65	25	39	60	23	27.30	41.00
7	8/8/91	4.0	0	100	72	143	78	78	0	35.75	0.00
8	9/8/91	2.0	0	0	0	0	0	0	0	0.00	0.00
9	11/8/91	8.5	16	84	91	95	122	103	63	14.35	34.88
10	12/8/91	8.5	19	118	110	101	103	151	41	17.76	32.84
11	13/8/91	8.0	13	84	101	152	76	121	52	19.00	45.08
12	14/8/91	9.0	11	131	142	85	22	80	64	15.78	47.64
13	15/8/91	8.5	10	99	115	144	42	85	47	16.94	53.20
14	16/8/91	6.5	10	116	105	95	50	66	44	17.85	47.60
15	18/8/91	6.0	4	123	156	77	61	71	98	26.00	146.50
16	19/8/91	7.0	8	132	185	42	116	108	67	26.43	81.25
17	20/8/91	7.5	5	147	150	63	85	81	60	20.00	117.20
18	21/8/91	3.5	4	69	105	14	48	49	24	30.00	77.25
		106.5	130	1515	1572	1319	943	1397	673		

The information presented in Table 4 shows the original dose budget, how it reduced with the actual radiation values, and how it was further reduced by the preparation of a cutting schedule. The actual dose uptake incurred by the operations team was well within the revised estimates.

The information presented in Table 5 shows the daily log of dose uptake and cutting progress. The increasing magnitude of the average dose per cut during the last four days operation reflects the increased amount of work that was necessary to gain access to the remaining stay tubes which were scattered over the reactor.

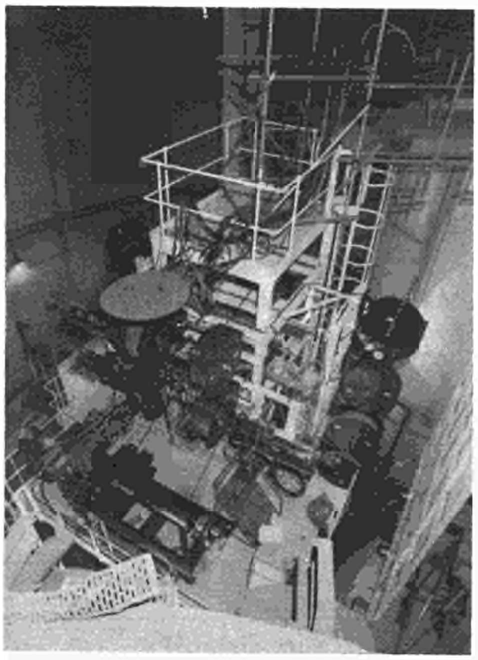


Figure 1: Construction of the Mast Module in the MTF

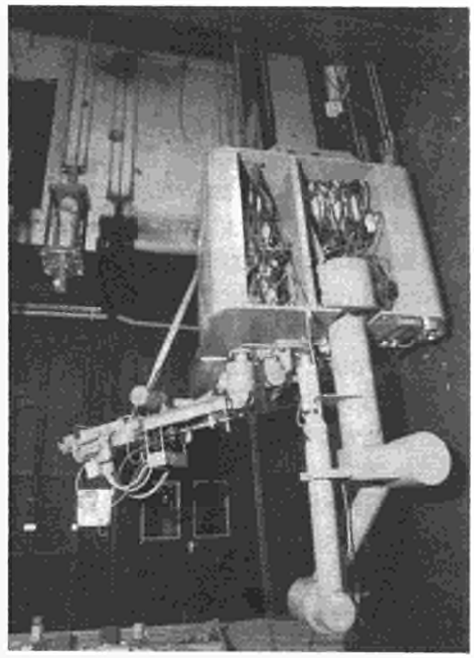


Figure 2: Deployment of the Manipulator during Operator Training

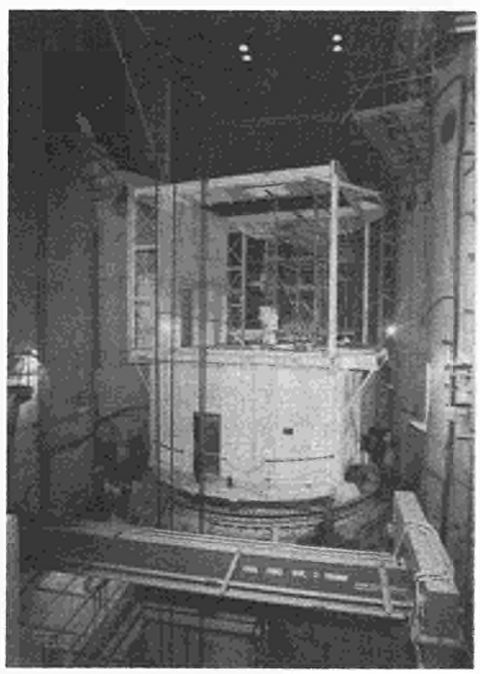


Figure 3: Construction of the RDM Containment

8.2. COMPARATIVE ASSESSMENT OF ALTERNATIVE UNDERWATER REMOTE OPERATION AND SEGMENTING TECHNIQUES FOR REACTOR VESSEL INTERNALS OF KRB-A

Contractors: KRB
Contract No.: FI2D-0002
Work Period: October 1989 - September 1990
Coordinator: H STEINER, KRB
Phone: 49/8224/783 730 Fax: 49/8224/782 900

A. OBJECTIVE AND SCOPE

The Boiling Water Reactor plant Gundremmingen A (KRB-A) is one of the four pilot dismantling projects of the EC programme (see also § 8.5.).

The above contract provided a preliminary design and assessment study of alternative remote operation and segmenting techniques for underwater dismantling of the pressure vessel internals of KRB-A. Occupational radiation exposure, costs, and the conditioning and minimisation of the radioactive waste were considered in particular.

In the course of its implementation this study was extended to the case of the VAK BWR of Kahl.

B. WORK PROGRAMME

B.1. Inventory of KRB-A conditions, e.g. materials and geometries of components, local dose rates and radioactivities, accessibility.
B.2. Literature study on the state-of-the-art.
B.3. Analysis of underwater segmenting techniques including thermal, mechanical, electrical and chemical techniques.
B.4. Investigation of remote-operation techniques considering alternative manipulator designs and various degrees of automatisation.
B.5. Comparative evaluation of the alternative techniques investigated, considering all relevant aspects; selection of the optimum technique(s); identification of experimental investigations needed, if any.

C. PROGRESS OF WORK AND RESULTS OBTAINED

This study was completed in 1990. The final report is being prepared for publication.

8.3. PILOT DISMANTLING OF THE BR-3 PWR. DECONTAMINATION OF A PRIMARY CIRCUIT, REALISATION OF CUTTING EQUIPMENT, SEGMENTATION OF ALL REACTOR INTERNALS

Contractors: SCK/CEN, Siemens-KWU, Framatome
Contract No.: FI2D-0003
Work Period: October 1989 - July 1994
Coordinator: F MOTTE, SCK/CEN
Phone: 32/14/33 21 11 Fax: 32/14/31 19 93

A. OBJECTIVE AND SCOPE

The BR-3 Pressurised Water Reactor had a capacity of 11 MWe and had been operated from 1962 to 1987. CEN/SCK has started the dismantling and decontamination of certain parts of the plant and is examining the possibility of its complete dismantling.

Considering that the experience to be gained from the dismantling of the first representative nuclear installations in the Community should be made available to all Member States, the Commission selected BR-3 as a pilot dismantling project for the 1989-93 R&D programme on the decommissioning of nuclear installations. The Commission, through shared-cost participation in specific parts of the project, intends promoting the use of advanced techniques and the performance of collateral investigations, in order to enhance the generation of useful knowledge and experience to serve in subsequent decommissioning tasks. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

The assessment of techniques and procedures will be performed in collaboration with Kernkraftwerk RWE-Bayernwerk GmbH (Gundremmingen), which is decommissioning the Boiling Water Reactor KRB-A (see § 8.5) and with VAK GmbH which is decommissioning the VAK BWR (see § 8.10).

As a Pressurised Water Reactor, the BR-3 is representative of the reactor type most frequently used in the Community in Phase 1. The contract involves the decontamination of the primary circuit of the reactor and the dismantling of the thermal shield, a highly radiating steel component (specific activity 10^8 - 10^9 Bq/g, estimated contact dose rates 10^2 - 10^3 Sv/h, estimated radioactive inventory 10^4 - 10^5 Ci at plant shut-down) and in Phase 2 the dismantling of the lower and upper core support assembly and of the reactor collar with the instrumentation basket.

The contract is implemented in close cooperation between SCK/CEN as main contractor and Siemens-KWU (FRG) and Framatome (F) as associated contractors, with an agreement for cooperation with Belgatom.

B. WORK PROGRAMME

B.1. Chemical decontamination of the primary loop
B.1.1. Cost benefit analysis and selection of a procedure
B.1.2. Decontamination operation
B.1.3. Treatment and removal of decontamination waste
B.2. Segmenting of the reactor internals
B.2.1. Concept and design of the segmenting and remote operation equipment
B.2.2. Manufacturing and procurement of the segmenting and remote-operating equipment
B.2.3. Inactive testing and commissioning of the segmenting and remote operating equipment
B.2.4. Segmenting of activated components
B.2.5. Waste treatment and packaging
B.3. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1., B.2., B.4., B.5. and B.6.
B.4. Underwater dismantling of the lower core support assembly
B.5. Underwater dismantling of the upper core support assembly
B.6. Underwater dismantling of the reactor collar with the instrumentation basket

C. Progress of work and obtained results

Summary of main issues

1. Phase 1 work (B.1., B.2.)

The segmentation work has been completely achieved in 1991 and was previously reported. Only the points B.1.3. (treatment and removal of decontamination waste) and B.2.5. (waste treatment and packaging) were performed partially in 1992. The removal of the thermal shield cut segments and their transportation from the reactor pool to the deactivation pool were carried out without any problem by means of a specially designed transfer container, able to carry and shield up to 900 Ci ($3.33.10^{13}$ Bq) of Co-60 emitters.

2. Phase 2 works (B.4. to B.6.)

The main objectives of the 1992 activities were first to elaborate a scenario for the dismantling of the different reactor internals presenting a large variety of shape and geometry (Figure 1). At the same time the selection of the segmenting and dismantling techniques had to cope with this dismantling scenario and to use the cutting experience gained during the phase 1 of this pilot project. The selection of cutting and supporting equipment for the different pieces to be cut as well as the design and procurement of these materials was also one of the major achievements of the 1992 activities.

For the first operations to be done *in-situ*, cold testing of the cutting and dismantling equipments were performed in order to assess their feasability and to demonstrate their reliability in a situation close to the one met in the reactor pool.

Progress and results

1. Phase 1 work (B.1., B.2.)

The transfer of the thermal shield cut segments and of the different filters used for collecting chips and particles produced by the different cutting methods was performed in a quite short time duration : 15 working days for the primary waste and 5 working days for the secondary waste. The total dose distributed during this process can be summarized as follows : 4.14 man-mSv for the primary wastes and 0.48 man-mSv for the secondary waste.

One problem has been encountered for the evacuation of the plasma arc torch prefiltration strainer : due to its high specific activity content and to its geometrical shape, it cannot be transferred as a whole. A special equipment has then be designed to cut it into two parts while recovering the highly active slags contained in the filter.

Grinding has been selected as cutting process. Cold testing of hydraulic driven grinding was carried out with two types of grinder disks : a common abrasive grinding disk (thickness : 3.5 mm; O.D. : 300 mm), and a tungsten carbide saw-like disk (thickness : 2.2 mm; O.D. : 305 mm); this last one showed the most promising results for what concerns the produced waste (chips instead of fine particles) and the cutting force and vibrations. This disk can be used with a rotating speed of 1800 r.p.m. and produces much less water drag than grinding disks.

2. Phase 2 work (B.4. to B.6.)

The complete scenario with the selection of dismantling tools leads to the choice of the following strategy for the dismantling of all the reactor internals (thermal shield excluded), see also figure 1:

- the best dismantling location is situated above the reactor pressure vessel for what concerns the available space, the fastening possibilities, the horizontality and the collection of falling chips;
- the different internals will be placed and fastened on a so-called turn-table located on the reactor upper flange (Figure 2). This turn-table allows to present the pieces to the dismantling and cutting tools which are clamped on extensions of the turn-table support structure;
- as far as experience gained during phase 1 is concerned, the mechanical cutting techniques are promoted everywhere it is possible : indeed, it produces a low amount of waste which is easy to collect and the tool contamination is very limited. The different main techniques selected are summarized in Table I. Electro Discharge Machining has also been selected for difficult "surgery" operations where mechanical techniques do not seem to be reliable.

For complex dismantling operations involving complicated pieces geometry, the help of a remote controlled telemanipulator is necessary. This will be associated to specific dismantling and cutting tools like hydraulic jaws, alternative saw, etc.

Cold tests, on mock-ups of the internal pieces (Figure 4) were also carried out, mainly for what concerns the disassembly of the reactor vessel collar and its instrumentation basket. Underwater EDM (Electro Discharge Machining) has been tested for the removal of stainless steel bolts placed with their head beneath the piece to be dismantled. Tube shaped copper electrodes were used for this purpose. Such a bolt can be removed or cut in about 1.5 hour for a 40 mm plate thickness.

Other dismantling and disassembling techniques were also successfully tested, like remote impact unbolting of secured bolts (Figure 3).

All the dismantling operations being carried out remotely under water, for radiation shielding reasons, common commercial cutting and dismantling equipments have to be more or less deeply adapted to the specific needs of this particular environment.

Table 1 : Selection of dismantling and cutting equipment - Summary table

Subassembly or operation \ Cutting tool	Hydraulic shears	EDM cylindr. electrode	Core Drilling	Alter-native saw	Impact unbolter	Circular saw	EDM/MDM	Band saw	Telemani-pulator	Turn-table
Disassembly of vessel collar from instrumentation basket	X	X	X	X	X					
UCSA + instrumentation basket (plates excluded)	X			X					X	X
LCSA horizontal cutting	(X)*					X	X		(X)*	X
LCSA vertical segmentation							X	X		X
Vessel collar	(X)**			(X)**				X		X
Plates and grids							X	X		X

<u>Notes</u> : (*) : to be used eventually for cutting the pipes running along the LCSA
(**) : to be used to cut the penetration pipes as close as possible of the collar surface.

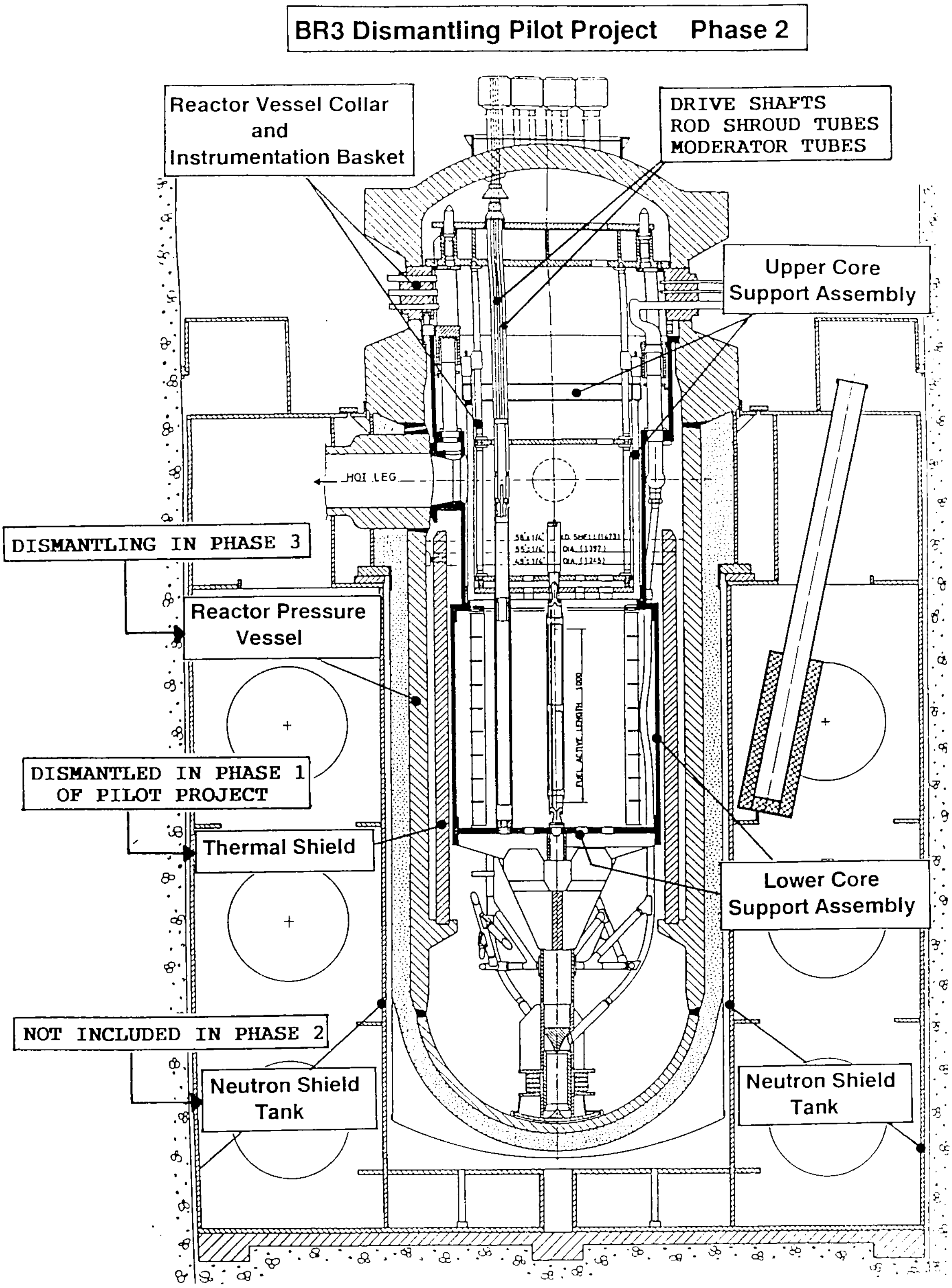

BR3 Dismantling Pilot Project Phase 2
Reactor Vessel Collar and Instrumentation Basket
DRIVE SHAFTS
ROD SHROUD TUBES
MODERATOR TUBES
Upper Core Support Assembly
HOT LEG
DISMANTLING IN PHASE 3
Reactor Pressure Vessel
DISMANTLED IN PHASE 1 OF PILOT PROJECT
Thermal Shield
Lower Core Support Assembly
NOT INCLUDED IN PHASE 2
Neutron Shield Tank
Neutron Shield Tank

Figure 1

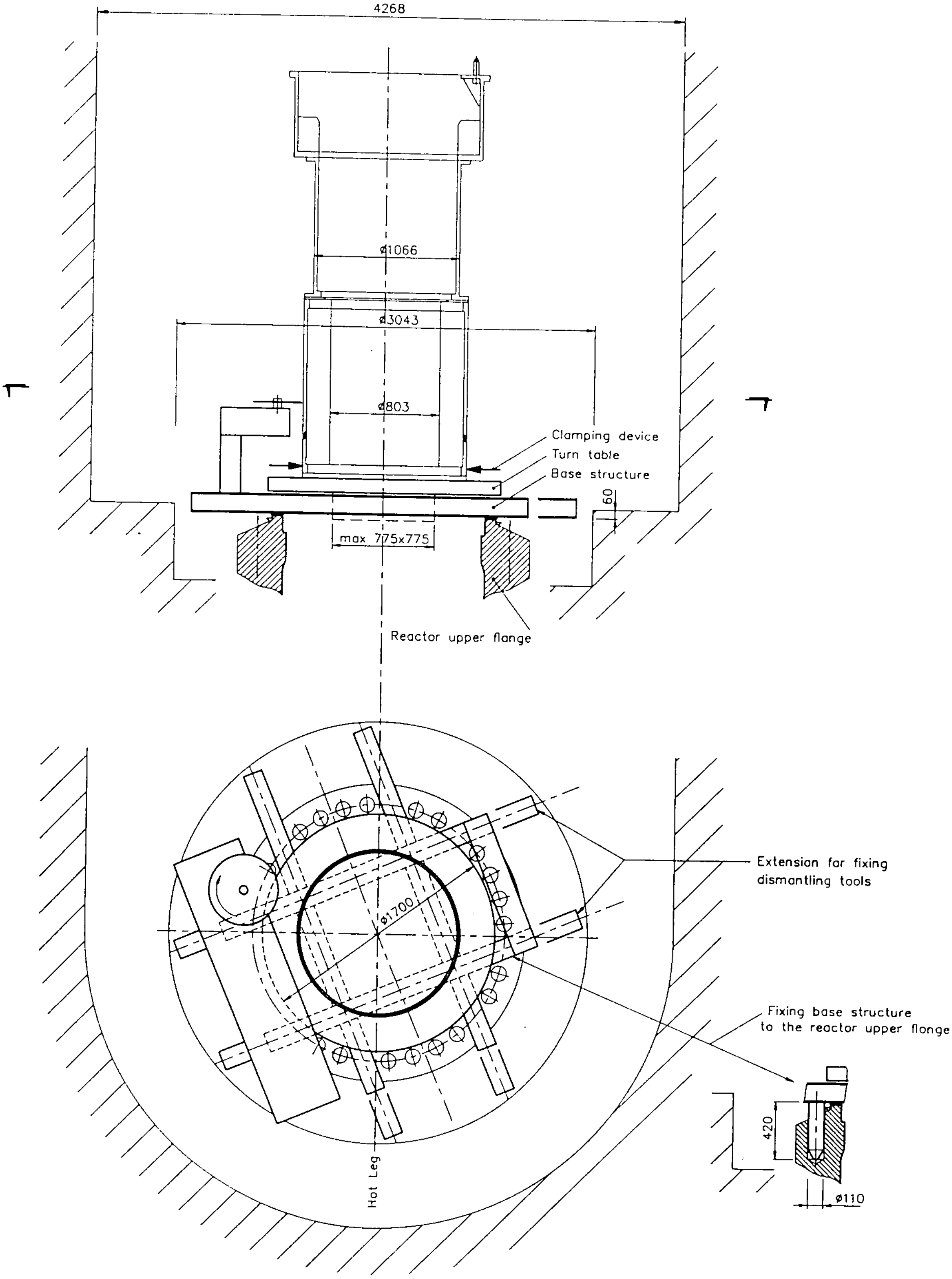


Figure 2 : Schematic view of the supporting turntable located above the RPV upper flange

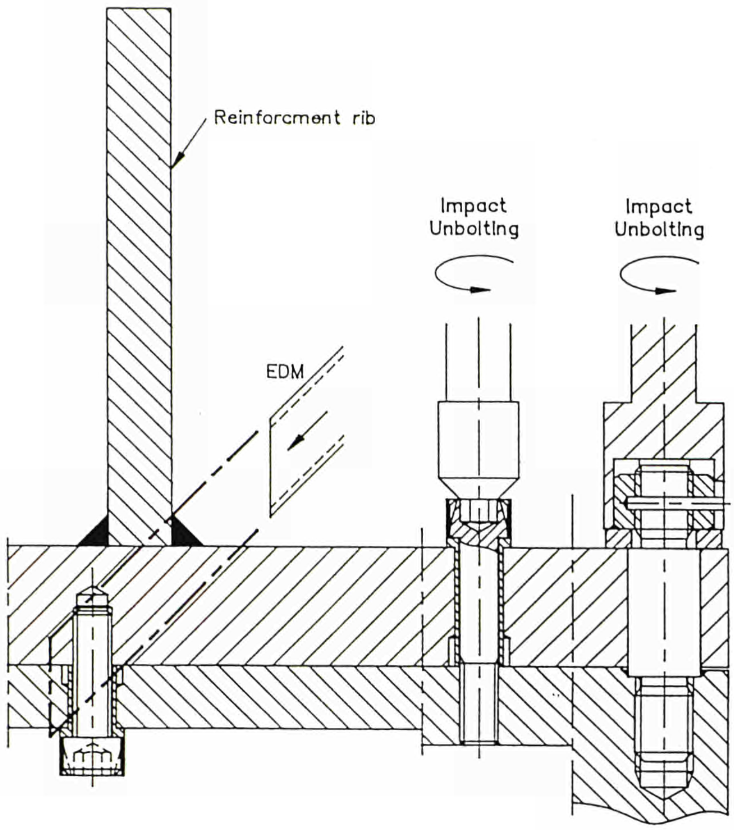


Figure 3 : Summary of the different techniques tested for disassembling the "Rod shroud support plate"

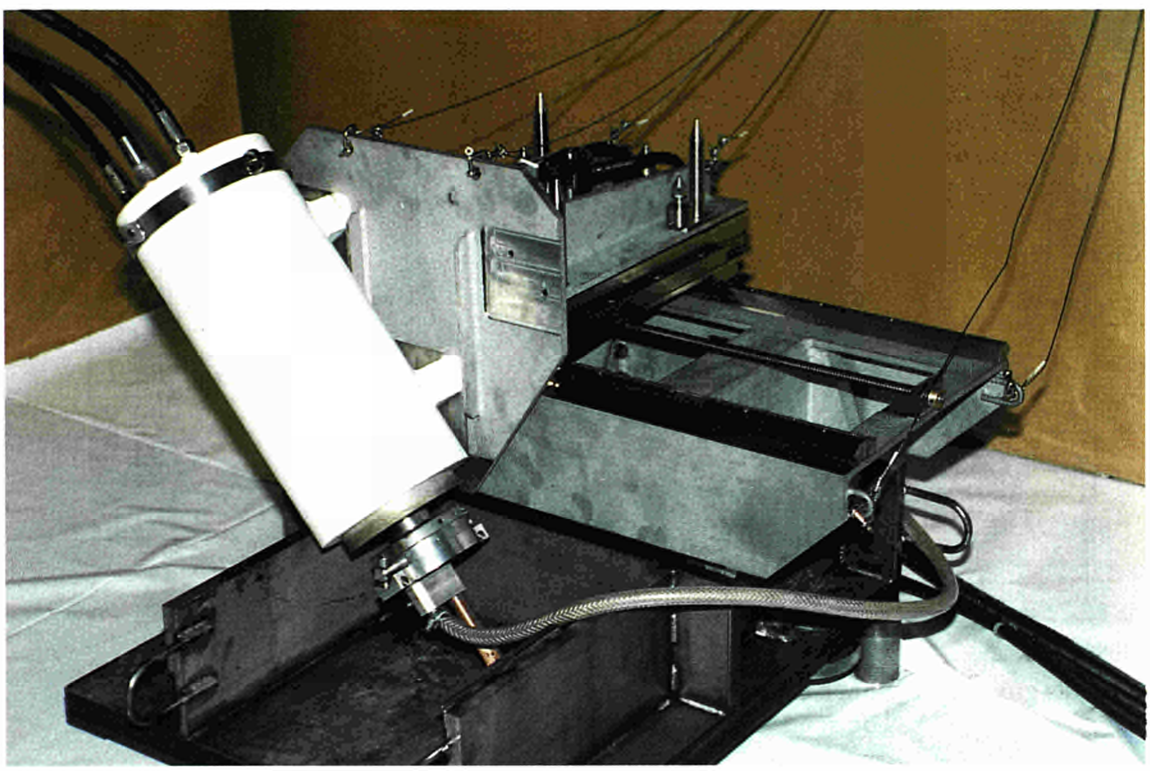


Figure 4 : Cold testing of Electro Discharge Machining on a mock-up of the uppermost internal plate

8.4. PILOT DISMANTLING OF THE FBR-FUEL REPROCESSING FACILITY AT-1. DISMANTLING OF DISSOLUTION AND EXTRACTION SYSTEMS AND OF FISSION PRODUCT STORAGE TANKS; DECONTAMINATION AND REMOTE DISMANTLING OF CONCRETE WALLS

Contractors:	CEA-Valrhô
Contract No.:	FI2D-0004
Work Period:	October 1989 - December 1993
Coordinator:	F CORNU, CEA/Ets. COGEMA, La Hague
	Phone: 33/33 03 66 71 Fax: 33/33 03 60 14

A. OBJECTIVE AND SCOPE

The pilot facility AT-1 for the reprocessing of FBR-fuel had a capacity of 2 kg/day and had been operated from 1969 to 1979.

Considering that the experience to be gained from the dismantling of the first representative nuclear installations in the Community should be made available to all Member States, the Commission selected AT-1 as a pilot dismantling project for the 1989-93 R&D programme on the decommissioning of nuclear installations. The Commission, through shared-cost participation in specific parts of the project, intends to promote the use of advanced techniques and the performance of collateral investigations. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

The dismantling of the AT-1 facility is concerned by specific problems associated to the reprocessing of irradiated fuel, namely the presence of a mixture of alpha, beta and gamma emitters. This necessitates the use of remotely operated and controlled equipment for the dismantling and decontamination, partly due to the specific conception of the cells, without direct viewing. For this, the carrier ATENA is used (telescope + polyarticulated arm) supporting the telemanipulators MA 23 or RD 500.

Specific problems are also encountered with radioactive measurements needed for the sorting and preconditioning of the arising dismantling waste.

The contract started with Phase 1 work involving the dismantling and waste assaying and conditioning of cells 903, 904, 905 and the dismantling of fission product storage cells. The subsequent Phase 2-work is devoted to the remote dismantling of a concrete wall and to the decontamination of the concrete walls and floors in the dismantled cells.
Estimated maximal values for the specific contamination and for dose rates are in the order of 10,000 Bq/cm^2 and 1 Gy/h, respectively.

B. WORK PROGRAMME

B.1. Remote-operated dismantling of equipment out of the strongly contaminated cell 903 (used for dissolution), and of cells 904 and 905 (used for extraction).
B.2. Measurement of the radioactivity and conditioning of the waste arising from B.1.
B.3. Dismantling of tanks for the storage of fission products.
B.4. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1., B.2., B.3., B.5. and B.6.
B.5. Remote dismantling of a reinforced concrete wall
B.6. Semi-automatic decontamination and contamination measurements of concrete walls and floors.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The dismantling of the high activity cells has been totally completed using the ATENA machine (B.1.).

Waste removal from cells 904 and 905 (1st extraction cycle) (dissolution cell) has been totally completed and has been conditioned using the "workshop cell" (B.2.).

After three years of operation, the running order of the remote equipment, used to dismantle the high activity cells of AT1 and to remove waste, was investigated (B.4.).

Two semi-automatic carriers have been studied for decontamination and contamination measurements of concrete walls and floors: one will be used for walls (vertical surface) the other for floors (horizontal surface). The two carriers have been ordered.

Work programmes B.3. (Dismantling of tanks for the storage of fission products) and B.5. (Remote dismantling of a reinforced concrete wall) have been totally completed in 1991.

Progress and results

1. Remote operating dismantling of equipment out of the strongly contaminated cells (B.1.)

No dismantling operation was performed in cells 903 and 904, dismantling work being almost over. Waste removal has been pursued and completed at the end of June for cell 904 and at the end of September for cell 903.

2. Measurement of radioactivity and conditioning of waste (B.2.)

The "workshop cell" (cell used for conditioning high activity waste) has been used normally during 1992.

The teleoperation tools include 2 mechanical M8 remote manipulators and a remote electrical manipulator MA 23 (which replaced the RD 500 robot in September 1991).

Waste conditioning has progressed according to the following steps:

- Filling of bin with waste with the ATENA machine fitted with the manipulator MA 23 arm.

Nota: For waste removal from cell 903, the manipulator MA 23 and the polyarticulated arm were introduced into cell 903 in order to fill a waste carrier. This waste carrier was removed from cell 903 into cell 904 with polyarticulated arm and cleared into a bin.

- Removal of bin from cell 904 to workshop cell with the twin-beam carriage and the 5 KN hoist.
- Cutting of pipes into workshop cell with shears fixed on a workbench. Pipes are handled with the MA 23 or the M8 remote manipulators.
- Radiological mapping in the workshop to sort out waste before drumming.
- Drumming (either 120 l drums or half drums with or without shielding) with the MA 23 and the M8 manipulators.
- Removal and checking of drums:
  - . $\beta\gamma$ activity
  - . α counting
  - . weight
  - . non contamination
- Filling the 5 m^3 standard containers and checking.

At the end of 1992, the quantities of conditioned waste are represented in Table I.

3. Dismantling of tanks for the storage of fission products (B.3.)

- Cell 920 (extension building storage)
- The cell was totally dismantled (including the recovery pan) at the end of 1990.
- Cells 908/909 (main building storage).

Linear-shaped explosive charges were used to dismantle the tanks in December 1990. Cutting operations of the recovery pan and waste removal were carried out during the 1st half of 1991.
We consider this programme as completely finished.

4. Generation of specific date (B.4.)

The efficiency balance of the teleoperation material, ATENA machine and MA 23 remote manipulator was drawn up for the years 1990, 1991 and 1992.

During 1990, ATENA was mainly used for cutting in cells 904 and 905 with a MA 23 manipulator and an abrasive disc-saw.

During the first half of 1991, the ATENA machine was used to:
- cut the separation wall between cells 903 and 904 with a diamond disc-saw;
- dismantle cells 903 and 904 with an abrasive disc-saw.

During the second half of 1991 and the first half of 1992, the ATENA machine was mainly used to remove the waste from cells 903 and 904.

One can record the high reliability of the ATENA carrier around the average of 0.8/0.9; MA 23 telemanipulators reliability was around 0.4/0.5 for each arm in 1991. During 1992, the reliability reaches 0.8/0.9 for each arm: working conditions are better for waste removal than for dismantling for "tapes" system manipulators.

5. Remote dismantling of a reinforced wall (B.5.)

A concrete wall separated cell 903 and cell 904.

To introduce the ATENA machine for the dismantling of cell 903, the ATENA machine made an opening of 4.5 m high, 1.2 m long, and 0.2 m thick in the separating wall.

For this operation, the ATENA machine used a diamond disc-saw.

We consider this programme as completely finished.

6. Semi-automatic decontamination and contamination measurements of concrete walls and floors (B.6.)

Two carriers are ordered (studies and implementation): one for vertical surfaces (walls), the other for horizontal surfaces (floors).

These carriers will support either a decontamination tool, or contamination measurement equipment and they will allow semi-automatic movement of these pieces of equipment.

Table I: Conditioned waste of cells 903 and 904 at the end of 1992

	Cell 903	Cell 904			
	1992	1991	1992	Total Cell 904	Total Cell 903 + Cell 904
Number of drums	23	33	39	72	95
Weight (kg)	2100	1800	4659	6459	8559
Activity $\beta\gamma$ (GBq)	21	48	180	228	249
Activity α (GBq)	12.2	33	101	134	146

Nota: the main results of this table are provisional.

8.5. PILOT DISMANTLING OF THE KRB-A BWR. DISMANTLING OF CONTAMINATED COMPONENTS OF THE REACTOR BUILDING AND OF ACTIVATED INTERNALS OF THE REACTOR PRESSURE VESSEL; DEVELOPMENT AND APPLICATION OF CONCRETE SAWING AND MELT ENCAPSULATION (ONION PACKAGE)

Contractors:	KRB	
Contract No.:	FI2D-0005	
Work Period:	May 1990 - December 1993	
Coordinator:	H STEINER, KRB	
	Phone: 49/8224/783 730	Fax: 49/8224/782 900

A. OBJECTIVE AND SCOPE

The prototype Boiling Water Reactor Gundremmingen A (KRB-A BWR) of the Kernkraftwerk RWE-Bayernwerk GmbH (KRB) had a capacity of 250 MWe and was operated from 1966 to 1977. Dismantling work has been started for some time (especially the turbine hall has been dismantled), and complete removal of the power station is foreseen to be completed by 2000. The two foregoing EC programmes have been involved by four R&D contracts in the past dismantling work on KRB-A. KRB-A dismantling is a European undertaking according to the definition of the Euratom Treaty.

Considering that the experience to be gained from the dismantling of the first representative nuclear installations in the Community should be made available to all Member States, the Commission selected KRB-A as a pilot dismantling project for the 1989-93 R&D programme on the decommissioning of nuclear installations. The Commission, through shared-cost participation in specific parts of the project, intends promoting the use of advanced techniques and the performance of collateral investigations, in order to enhance the generation of useful knowledge and experience to serve in subsequent decommissioning tasks. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

The assessment of techniques and procedures will be performed in collaboration with CEN/SCK Mol and VAK-GmbH, which are decommissioning the Pressurised Water Reactor BR-3 and the VAK BWR, respectively (see § 8.3. and 8.10). The results and conclusions of the assessment work undertaken in contract FI2D-0002 are taken into account for the implementation of work in this contract.

As a BWR, KRB-A is representative for such reactors, existing elsewhere in the Community. The first phase of the contract involves the dismantling and segmenting of contaminated components of the reactor building in air (partly with subsequent decontamination), and of activated internals of the reactor pressure vessel (RPV) in remotely controlled underwater operation. Estimations of maximal values for specific contamination or activation are in the order of 4.10^4 and 10^6 Bq/cm^2, respectively. The second phase contains the development of specific tools and the segmentation of further steel components and concrete structures as well as the development of procedures for the conditioning of molten steel (onion package) and of decontamination waste.

B. WORK PROGRAMME

B.1. Dismantling in air of contaminated and low-activated components **of the reactor building, partly with subsequent decontaminating/melting.**

B.1.1. Dismantling of a secondary steam generator with various tools (band saw, flame cutting)

B.1.2. Dismantling of a primary circulation pump by band saw.
B.1.3. Dismantling of a primary clean-up cooler with various tools (band saw, diamond-tipped wire saw)
B.1.4. Dismantling of a shutdown cooler with various tools (band saw, shears, flame cutting)
B.1.5. Dismantling of the RPV-cover by flame cutting
B.1.6. Decontamination of segmented components by dipping technique and meltling for recycling and disposal.
B.2. Underwater dismantling of activated and highly contaminated components of the RPV
B.2.1. Segmenting of the steam-dryer by various tools (shears, plasma-arc torch, consumable electric electrode torch)
B.2.2. Segmenting of the water-steam separator with the core head by various tools (saw, shears, plasma arc torch with special gripping system)
B.3. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1., B.2., B.4., B.5., B.6., B.7., B.8. and B.9.
B.4. Development and application of a carrier and handling system for automated segmenting of thick-walled pipes and pipe fittings in limited space,
B.5. Remote-operated underwater segmenting of internals of the reactor core,
B.6. Development and application of a facility for the conversion of iron-oxalate, generated during decontamination,
B.7. Qualification and large-scale testing of a wire saw device for the dismantling of reinforced concrete walls,
B.8. Development and qualification of a procedure for the pre-conditioning of metallic dismantling waste by melting (optimised "onion package");
B.9. Qualification and application of underwater segmenting of control rods by shearing.

C. Progress of work and obtained results

Summary of main issues

In 1992 the following main activities have been carried out in the framework of the CEC pilot project:

- completion of freezing down a secondary steam generator for ice-sawing,
- investigations on the activity status of the RPV-cover,
- underwater cutting tests at the shell of the steam dryer using a plasma torch,
- inactive tests for qualification of a wire cutting procedure for dismantling the water-steam separator.

Progress and results

1. Dismantling of a secondary steam generator (B.1.1.)

After clearing the access to the loop room by removing about 20 t of pipes and valves, the necessary equipment for cutting a secondary steam generator by a heavy band-saw with the ice-sawing technique was installed. Figure 1 shows a schematic drawing of the secondary steam generator prepared for ice-sawing. The principle of cooling down by air of the water-flooded component is also given in this figure.

In order to check the installed equipment the first upper cut to remove the top dome of the secondary steam generator was done without ice.

However, some problems occurred to complete the freezing of the upper part of the component above the heat exchanger bundle. By an additional installation of a third cooling aggregate and by blowing cold air also on the water surface inside the component, it was possible to freeze the secondary steam generator to -15 °C by mid of December. The implementation of ice-sawing of a secondary steam generator has been started in January 1993 and will be reported later on.

2. Dismantling of the RPV-cover (B.1.5.)

The cover of the reactor pressure vessel has already been pre-segmented into 5 single pieces on the reactor floor under a temporary plastic tent. This initial segmenting work of the RPV-cover was done using a special oxy-propane torch and resulted in a high consumption of HEPA filters. The post-segmenting and the comparison of different cutting techniques at this component will be executed next year in a new cutting cabine in the turbine house under constant conditions. During the reporting period the erection of this cabine was completed and a new and recleanable filtration system for this cabine was installed.

A detailed investigation on the status of contamination and activation at the cover of the reactor pressure vessel was achieved. The obtained activation profiles are of great interest:

A cobalt-activation of about 1.2 Bq/g has been detected in the austenitic cladding at the inside of the RPV-cover. This activation is probably caused by thermal neutrons during operation of the plant. Also the total ferritic base material is activated (0.4 Bq/g Co-60).

As an example, one typical activation profile for Co-60 is represented in Figure 2. Table 1 gives the nuclide vector for the cladding and the base material. The high percentage of Ni-63 in the stainless cladding corresponds to the high amount of nickel in this alloy.

3. Segmenting of the steam dryer (B.2.1.)

The RPV-steam dryer was moved from the flooding chamber to the refuelling storage pool for executing underwater segmenting tests. During this year, a cutting to position the steam dryer and the segmenting unit ODIN I for guiding the cutting torch was installed in the pool.

Preliminary, the dismantling of the steam dryer was focused on the removal of the outer shell. Several steel segments have been successfully cut out of the shell using the segmenting unit ODIN I together with an underwater plasma torch. The dimensions of these single austenitic segments were defined to be about 400 x 800 mm because of subsequent handling and packaging. Figure 3 shows the storage pool with the arrangement of all equipments applied for underwater plasma cutting.
These cutting tests at the steam dryer shell with 5 mm wall thickness have been performed by an amperage of 250 A and an average cutting velocity of 300 mm/min.

During cutting, the conditions of visibility in the water did not get worse; also the specific activity of water did not increase substantially. Most of the kerf material dropped to the ground of the water pool. Furthermore, an increase of the aerosol concentration in the atmosphere has been avoided by an additional installation of a suction device above the cutting torch. Plasma cutting and the segmenting unit have proved to be appropriate for this purpose and also for further segmenting of other reactor internals in future.
Because of high contact dose rates up to 6 mSv/h the handling of the segmented metal parts in air had to be done keeping at least 2 m distance to the pieces. Activity measurements by gamma-spectroscopy indicated a mass specific activity of about 3000 Bq/g, respectively 5800 Bq/cm^2 surface contamination. About 0,4 % of this activity is coming from Am-241.

For one shell segment, a decontamination by electropolishing has been performed. The decrease of the specific activity is shown in Table II. After 8 hours of decontamination no further decrease of the activity was possible due to activation. This residual Co-60 activity is approx. 30 Bq/g. A controlled recycling of the total shell after pre-decontamination is possible.
Large-scale segmenting of the outer shell and the removal of the inner structures of the steam dryer will be continued after completion of the current conditioning of about 500 fuel element channels.

4. Segmenting of the water-steam separator (B.2.2.)

Some inactive tests have been performed in order to prove the application of new cutting techniques for the dismantling of complicated RPV-internals, such as the water-steam separator.
Underwater tests have been done at different stainless steel sections by the wire-sawing technique using a conventional steel wire on one hand and a wire with diamond beads on the other hand. The pre-tests resulted in the decision to continue the tests with the diamond wire to prove the possibilities for an active application. Further inactive tests will be performed to test the wrap around and the straight cable cutting method to get assessment for a practical implementation.

5. Remote-operated underwater segmenting of internals of the reactor core (B.5.)

Dismantling, handling and packaging of RPV-internals must be done under water and by remote operations due to the high dose of these components. Not for executing the segmenting, but for tasks like handling, positioning and change of tools, picking-up segmented parts, inspection with cameras etc, a master-slave manipulator will be applied.
A comparative study on commercially available electric and hydraulic-driven master-slave manipulators is initiated. Six standard tests have been defined to show the advantages and disadvantages of each manipulator. Partially, these tests have already been done on the site of the proposers. After evaluation and practical assessment for a large-scale operation, the selected manipulator will be used for dismantling of RPV-internals.

6. Qualification and large-scale testing of a wire saw device for the dismantling of reinforced concrete walls (B.7.)

A thick-walled shielding wall of concrete in the turbine hall has been cut into single blocks with a special diamond wire saw.
The real cutting time for this application, however without installation, was determined to be 1.5 hour/m² of cutting surface. On an average, with 1 m diamond wire, it was possible to cut about 1.5 m². The results of this campaign will be used for the dismantling concept of the biologicial shield. Furthermore, an inactive 1:1- mock-up of the biologicial shield has been erected on KRB-site in order to test appropriate segmenting techniques.

7. Development and qualification of a procedure for the pre-conditioning of metallic dismantling waste by melting (optimised "onion-package") (B.8.)

The design of a new waste container, by applying the onion-casting procedure, for final storage of material from the dismantling of the RPV, has been completed. The recommended container layout was given by the target to realize an optimal volumetric efficiency with a maximum activity inventory. An inactive container has already been produced to carry out the necessary tests for getting a license for this new type of waste container.

Table I: Nuclide vector of the RPV-cover

Nuclide	Nuclide Percentage [%] Cladding	Base material
Co-60	24,7	12,6
Am-241	0,0	0,0
Fe-55	32,0	68,4
Ni-63	43,3	18,7
Others	0,0	0,3

Table II: Decontamination of a shell segment of the steam dryer

Duration of Decontamination [h]	Contact-Dose rate [μSv/h]	residual Gamma-Activity [Bq/g]	Nuclides
0	5800	3046	Co-60/Am-241
4	20	320	Co-60
6	10	131	Co-60
8	10	24	Co-60
10	10	32	Co-60
12	10	29	Co-60

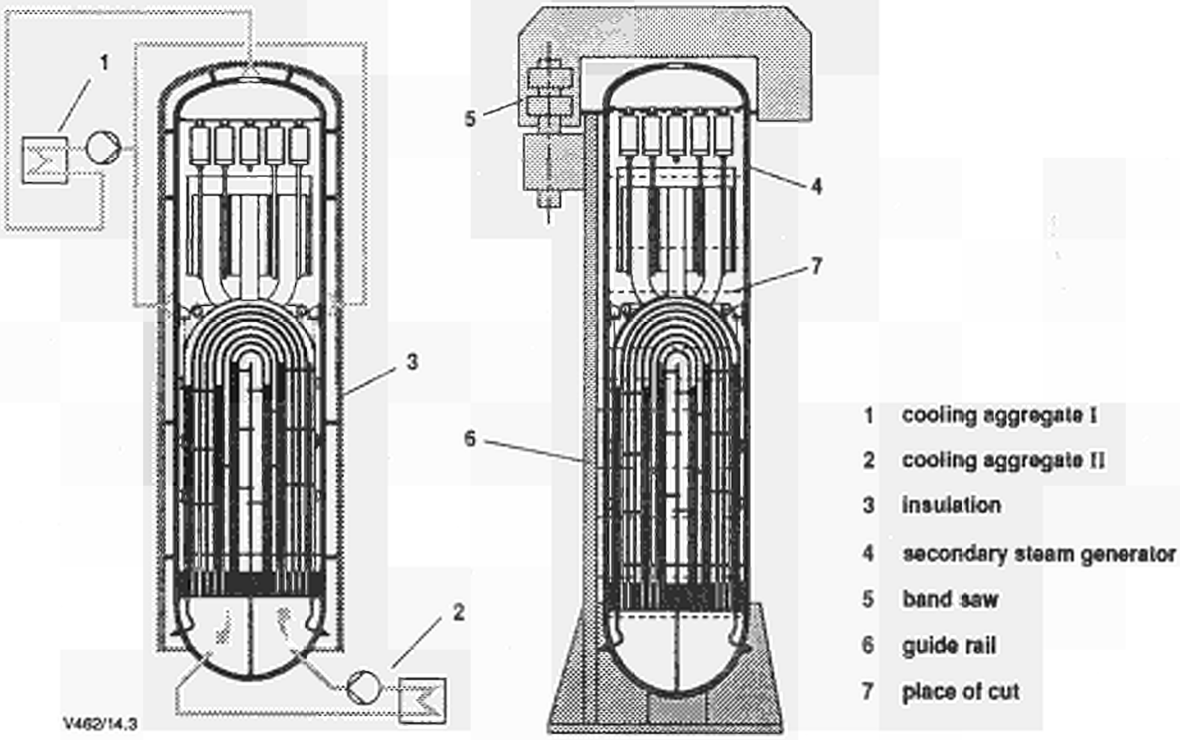


Figure 1: Equipment and cooling principle for ice-sawing of a secondary steam generator

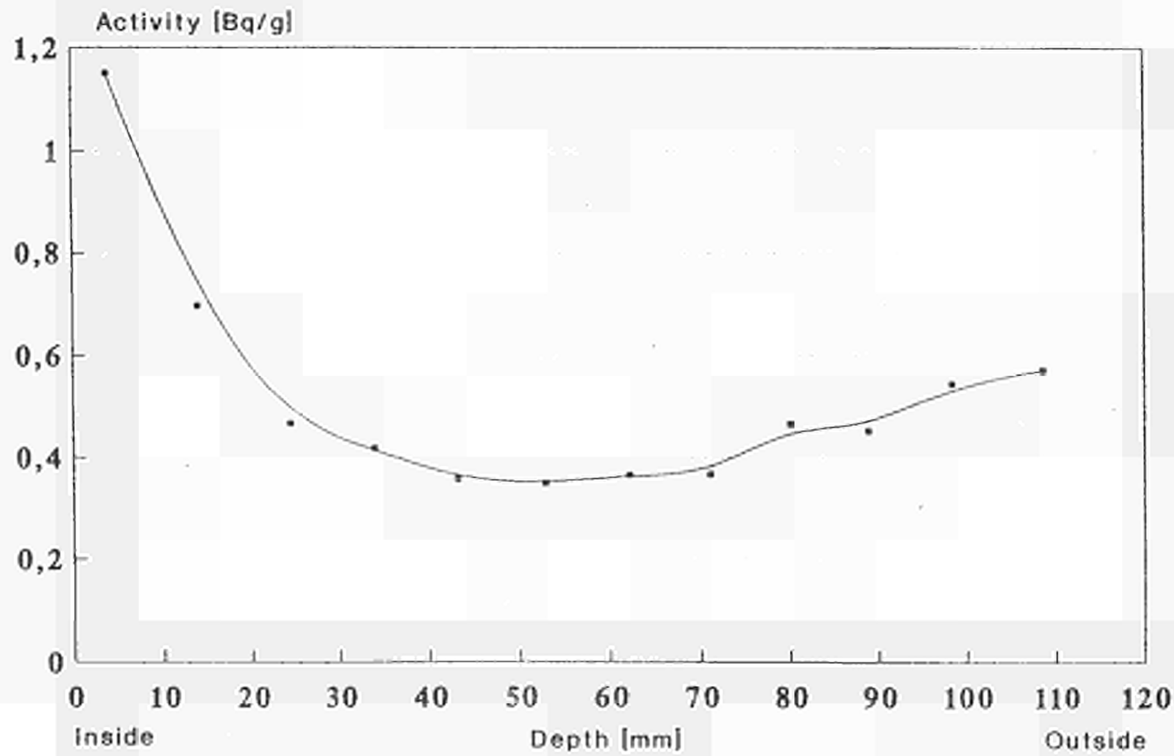


Figure 2: Co-60-activation profile of the RPV-cover

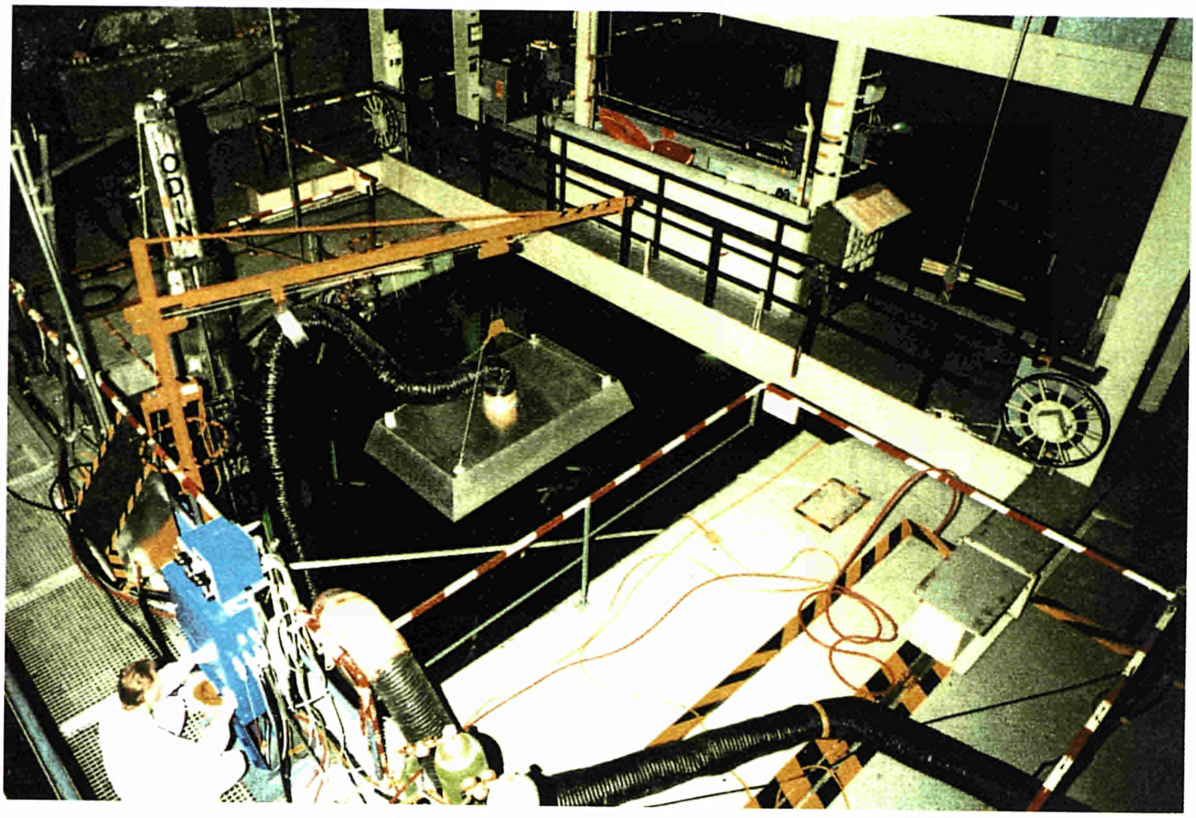


Figure 3: Test arrangement of equipment for underwater plasma cutting of the steam dryer

8.6. DECOMMISSIONING OF THE RISØ HOT CELL FACILITY

Contractors: Risø National Laboratory
Contract No.: FI2D-0011
Work Period: July 1990 - December 1993
Coordinator: H CARLSEN, Risø National Laboratory
Phone: 45/423/712 12 Fax: 45/423/511 73

A. OBJECTIVE AND SCOPE

The Risø Hot Cell Facility, which was in operation for 26 years (1964-1990), comprises six concrete cells, lead cells, glove boxes, a shielded unit for temporary storage of waste until shipment, a frogman area, decontamination areas, workshops, various installations of importance for safe operation of the plant, offices, etc. The facility presented was used for physical and chemical post-irradiation investigations of various types of fuel pins (LWR, HTGR), including Pu-enriched pins.

The general objective of the decommissioning programme for the Hot Cell facility is to obtain a safe condition for the whole building that does not require the special safety provisions which were necessary for operation of the hot cell plant. As a result, the Hot Cell building will be usable for the other purposes.

Work includes the removal of all irradiated fuel items, of other radioactive items and of contaminated equipment, and decontamination of all cells and rooms. The project is expected to produce specific data on manpower, waste arisings and radiation exposures for the decommissioning of a total hot cell line.

The contractual work will lead to the identification of an assessed procedure appropriate for the decontamination and the dismantling of equipment of a hot cell line used for post-irradiation tests on nuclear fuel pins of different types.

The contractor will execute the work programme in co-operation with BNFL plc, Sellafield (UK), which is decommissioning the B 205 Fuel Reprocessing Pilot Plant, by using, to any suitable extent, common techniques, procedures and instrumentation.

The latest dose rate measurements determined after a former partial decontamination of a concrete cell were in the order of magnitude of 1-2 mGy/h.

B. WORK PROGRAMME

B.1. Removal of fissile material in the form of uranium oxides and uranium/plutonium mixed oxides

B.2. Removal of large contaminated equipment, including the power manipulator, the cell crane and all experimental equipment.

B.3. Removal of large contaminated facilities, including all lead-shielded steel boxes and glove boxes, the shielded storage facility, the conveyer, the microscope cell.

B.4. Decontamination of concrete cells by various procedures, with preceding and subsequent radiation measurements

B.4.1. Initial mapping of radiation levels in remote operation

B.4.2. Coarse cleaning by vacuum cleaning, conventional washing and possibly by special techniques

B.4.3. Final cleaning with conventional methods

B.4.4. Hot spot removal by special techniques.

B.5. Decontamination and radiological measurements of cell ventilators and ventilation ducts

B.6. Decontamination of room surfaces

B.7. Removal of active drains from various facilities

B.8. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2. to B.7.

C. Progress of work and results obtained

Summary of main issues

The main results for 1992 were as follows:

- remaining fissile material was removed;
- four of six cells were emptied for contaminated material;
- a shielded storage facility was decontaminated and sealed;
- one room was classified at a lower level, and two rooms were declassified;
- two sets of carbon filters for iodine retention were removed;
- preliminary measurements of radiation and contamination were performed in three cells , now ready for final cleaning by frogman entrances.

Progress and results

1. Removal of fissile material (B.1.)

All fissile material has now been removed from the facility. Part of it was transferred to the Risø Waste Treatment facility for temporary storage, another part was exported.

2. Removal of large contaminated equipment (B.2.)

Several tons of scrap material including experimental equipment from the cells have up to now been cut, packed and transferred to the Waste Treatment facility. The cells number 1, 4, 5 and 6 have been emptied, while a minor amount of material in cells number 2 and 3 remains to be removed during 1993. A major part of the material was unforeseen and not included in the planning, because it was stored at another facility at Risø.

3. Removal of large contaminated facilities (B.3.)

The structural parts of a shielded storage facility, see Figure 1, was dismantled and decontaminated. The structure consists in the top end of some heavy steel plates acting as shielding. In the lower plate which can be rotated, are 18 holes, on which storage baskets of steel are welded. In these baskets could be placed containers with fissile and contaminated material. The walls of the cylindrical ring volume are painted concrete. The decontamination was performed by conventional cleaning using household detergents. The most difficult part of the structure to decontaminate was the sump, see Figure 1. A pneumatic hammer was used to remove part of the concrete, but a minor amount of contamination remained. The cover plates were again arranged on the top, and the whole facility was sealed by silicone.

A room housing miscellaneous only slightly contaminated equipment was emptied and decontaminated by washing to a lower classification limit.

In the concrete cell ventilation system, having two operation modes, two sets of carbon filters for iodine retention were removed as slightly contaminated waste.

4. Decontamination of concrete cells (B.4.)

4.1 Initial mapping of radiation levels in remote operation (B.4.1)

In order to give room for the in-cell crane and powermanipulator running through the whole cell line, a shutter is situated in a housing above each door between adjacent cells. All of these shutters and their housings have been examined by smear tests. The α-isotopes ^{239}Pu, ^{240}Pu, ^{241}Am and ^{244}Cm and the β/γ-isotopes ^{60}Co, ^{154}Eu, ^{134}Cs and ^{137}Cs were detected. The results are shown in Figure 2, assuming a 100% uptake of the contaminants in the smear test. The first result for each shutter (on the abscissae) is from the horizontal upper surface of the shutter; the remaining results are from vertical surfaces on the shutter and the housing. It is observed that the major contribution to the contamination is found on the horizontal surfaces as expected.

4.2 Coarse cleaning by vacuum cleaning, conventional washing and possibly by special techniques (B.4.2)

Concrete cells number 1, 4, 5 and 6 have been vacuum cleaned and the tables have been washed. Few hot spots on the tables in cells number 4, 5 and 6 were removed by spot grinding. Hot spot removal in cell number 1 has not been performed. The radiation levels in the cells after this coarse cleaning were measured preliminary by an ion chamber. The results are given in Table I.

Cell number 1 has been used for reception and dismantling of rigs with fuel elements and single fuel pins. As the levels are greater than 100 mSv/h γ more cleaning has to be performed remotely before frogman entrances.

Cell number 4 has through the years been kept as clean as possible to serve as an entrance cell for repair work on the in-cell crane and powermanipulator as well as on experimental equipment. Direct contaminating operations were never performed in this cell.

Cells number 5 and 6 have primarily been used for destructive post-irradiation measurements, such as piercing and preparation for ceramo- and metallography.

The observed hot spots consist of fission products as well as small active Co-particles, each weighing 7 mg, arising from fabrication of radiotherapy sources.

4.3 Final cleaning with conventional methods (B4.3.)

Concrete cells number 4, 5, 6 are now ready for cleaning by frogman entrances. We use cell number 6 as a model for the operations. We plan to clean the horizontal surface of the shutter between cell number 5 and 6 by wiping with wet cloths. Then we clean the lower part of the cell walls, the table and the floor by high pressure water jetting to an acceptable level.

4.4 Hot spot removal by special techniques (B4.4.)

A draft report on research work done at Sellafield on removal of old epoxy paint in the cells using paint strippers was received. The conclusion is that removal is possible provided polythene oversuits are used if personnel wearing protective clothing are to apply the paint stripper by hand, and that an applicator system for remote handling is developed. The technique is of importance now or at a later stage.

5. Decontamination of room surfaces (B.6.)

Two classified rooms, totally 36 m^2, containing offices and laboratories have been declassified after washing of the floors and thorough spot checking for contamination.

Table I: Measured levels of radiation in cells number 1, 4, 5 and 6.

	Pos.	In general [mSv/h γ]	Hot spots [mSv/h γ]
Cell 1	floor	(hot spot dominates)	>100 on floor (dominates)
	table	2-6	50 on table 20 on table
Cell 4	floor	0.8-1.5	4 on filterbox
	table	0.4-1	2 on conveyor cover
Cell 5	floor	2-5	40 on floor 25 on filterbox
	table	1-2	10 on conveyor cover 6 in conveyor duct 10 at several joinings
Cell 6	floor	2-6	15 on floor
	table	1-4	20 in conveyor duct

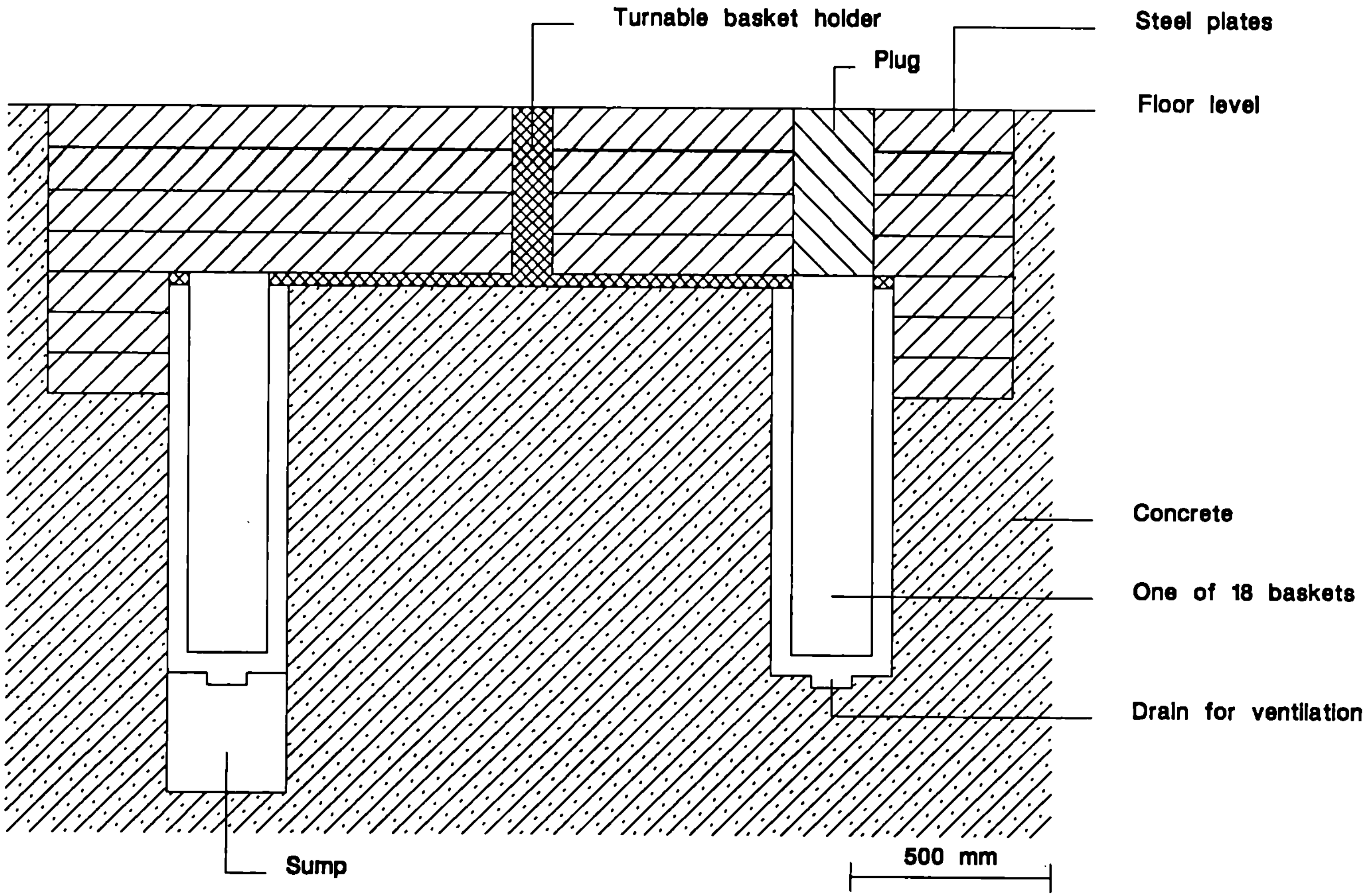


Figure 1: Shielded storage facility schematically.

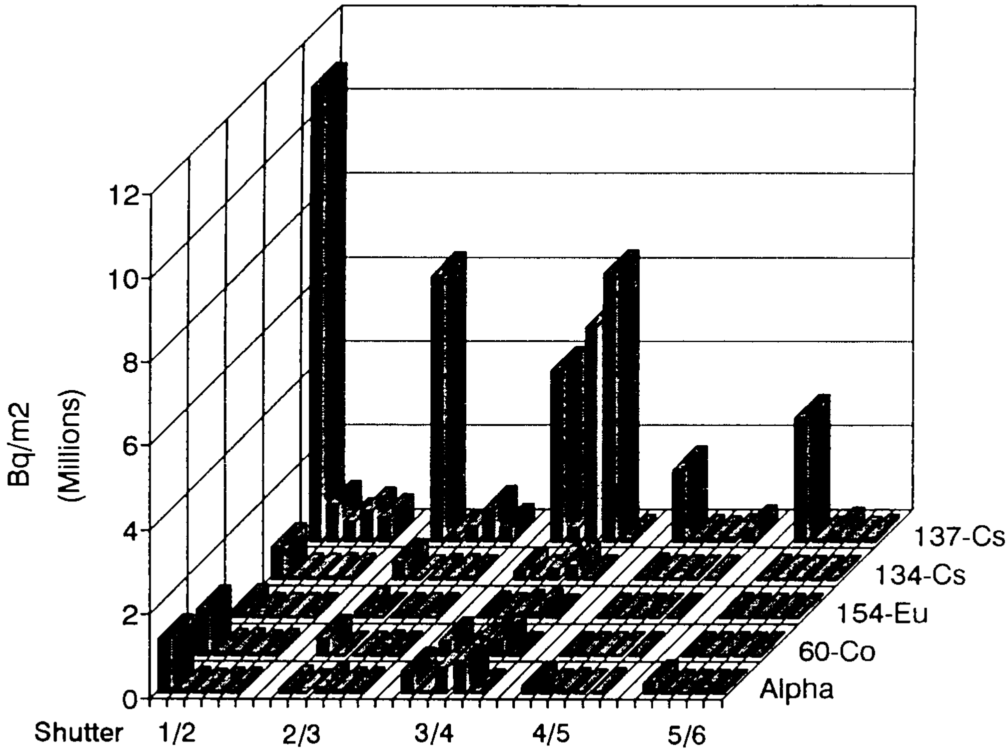


Figure 2: Contamination on shutters

8.7. FINAL CLEAN-UP OF THE PIVER PROTOTYPE VITRIFICATION FACILITY: DECONTAMINATION OF THE HOT CELL

Contractors:	CEA-Valrhô	
Contract No.:	FI2D-0018	
Work Period:	July 1990 - June 1991	
Coordinator:	A JOUAN, CEA-Valrhô.	
	Phone: 33/66 79 63 76	Fax: 33/66 79 66 03

A. OBJECTIVE AND SCOPE

The PIVER pilot vitrification facility at Marcoule was operated between 1969 and 1980, first using a batch process to vitrify Gas-Cooled Reactor fuel element reprocessing waste, and then to develop a continuous process to vitrify Fast Breeder Reactor (FBR) fuel reprocessing waste. A total of 12 t of glass was treated. It was then decided to remove the equipment and clean up the cell in order to install new equipment (PIVER II).

PIVER is the first HLLW vitrification cell to be decommissioned. Under a previous contract (FI1D-0057), all process equipment items of the main cell were removed, followed by preliminary decontamination carried out in remote operation. So, the internal radiation level was reduced from several Gy/h to less than 10 mGy/h. The remaining radioactivity inventory is estimated at about 1.1 x 10^{13} Bq (300 Ci). At this level, access to the cell is now possible for durations not exceeding about one minute; the cell remains highly contaminated and requires the use of ventilated protective clothing under severe working conditions.

This project is aimed at continuing decontamination and dismantling work enabling further dismantling of in-cell equipment with hands-on techniques and finally to reach a radiation level allowing the installation of new equipment with standard working conditions for controlled zones. The generation of specific data on costs, working hours, job doses and the amount of created secondary waste is an important objective of this project.

The contractual work will result in assessed decontamination procedures for highly contaminated cells.

B. WORK PROGRAMME

B.1. **Dismantling of the telemanipulators in the PIVER cell including two MT 200 master-slave manipulators, a robot manipulator (CAROLINE) and a pantograph manipulator (ANTOINE).**

B.2. **Further decontamination of the PIVER cell with various decontamination techniques (chemicals using liquids, foams and gels, electropolishing, and cryogenics), accompanied by radiological measurements.**

B.2.1. First stage decontamination by short time in-cell work, aimed at strongly reducing the dose rates.

B.2.2. Second stage decontamination by long time in-cell operators'work.

B.2.3. Final decontamination aimed at obtaining standard working conditions for controlled areas in the cell (dose rate $\leq$ 0.1 mGy/h).

B.3. **Dismantling of the remaining pipes not needed for the future use of the cells.**

B.4. **Identification of the remaining cell internals by photogrammetry for facilitating design work for the reuse of the cell.**

B.5. **Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1. to B.3.**

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1991. The final report is available as EUR report N° 14764.

8.8. DESTRUCTION OF CONTAMINATED SODIUM OF THE PRIMARY CIRCUIT OF EXPERIMENTAL RAPSODIE REACTOR

Contractors:	CEA-Cadarache
Contract No.:	FI2D-0022
Work Period:	July 1990 - June 1993
Coordinator:	J ROGER, CEA-Cadarache
	Phone: 33/42 25 76 45 Fax: 33/42 25 72 56

A. OBJECTIVE AND SCOPE

French regulations prohibit, for safety reasons, the disposal of sodium with other low-level radwaste in shallow land burial. The development of an industrial-scale procedure for the transformation of sodium into an acceptable product is thus a useful target generally for all LMFBRs.

The CEA has developed, at laboratory-scale, the so-called NOAH procedure transforming sodium by controlled addition of water into aqueous sodium hydroxide.

The objective of the present contract is to conceive and manufacture an industrial-scale facility (600 Kg/d), based on the NOAH process and its application to 13 t (out of a total of 37 t) of contaminated sodium (specific activity 4.1 KBq/g, mainly Cs-137) from the RAPSODIE pilot FBR. The facility will be conceived thus (mobile system, limited dimensions, easy adaptation), that it can be used on other FBR-sites.

The facility will be installed at the containment building of RAPSODIE (DESORA programme). Contractual work will be implemented in cooperation between two departments of the CEA-UDIN (Unité de Démantèlement des Installations Nucléaires) and LEPE (Laboratoire d'Etudes, de Procédés et d'Expertises).

In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

B. WORK PROGRAMME

B.1. Conceptional studies

B.1.1. Studies for the industrial application of NOAH (LEPE)

B.1.2. Studies for the installation of NOAH into the RAPSODIE containment building, including the needed auxiliary equipment (UDIN)

B.2. Manufacturing, installation and testing of equipment

B.2.1. Manufacturing and installation of equipment (UDIN)

B.2.2. Commissioning, testing of equipment and operator training with non-radioactive sodium (UDIN)

B.3. Main operation for the transformation of sodium (UDIN)

B.4. Conditioning and disposal of generated liquid waste

B.4.1. Investigations into possible ways for utilisation or treatment of waste including associated costs (LEPE)

B.4.2. Temporary storage of liquid waste (UDIN)

B.5. Technical and economical balance on the feasibility for an industrial application of NOAH (UDIN)

B.5.1. Preliminary balance before main operation

B.5.2. Final balance after main operation, including generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.3. and B.4.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The whole of the DESORA equipment was set up in the RAPSODIE containment during the first semester of 1992. Global tests in fluids (without sodium) were carried out and gave satisfactory results. Afterwards, the installation was filled with 1,500 l inactive sodium on 26 October 1992. It was not possible to operate the circulation pump in a reliable way, which did not allow us to perform the sodium injection tests in the reaction tank.

When modifications will be completed, the restart of the NOAH prototype is planned for March 1993. However, the humidification unit was with the tests.

Progress and results

1.1. Studies for the industrial application of NOAH (B.1.1.)

In order to allow the restart of NOAH, modifications of part of the sodium loop were proposed, designed and carried out. For budgetary reasons, the monitoring board is not completed. Due to this delay, the restart of NOAH prototype is planned for March 1993.
These modifications will allow:
- new means of investigations on the sodium loop
- improving the sodium loop operation
- priming with sodium the LEWA pump (high pressure pump) at the beginning of the operations.

In parallel, the dehumidification unit was completed and tested with helium and water. Experiments gave satisfactory results concerning the efficiency of this unit to separate water from gas. The results were compared to those obtained by a simulation code (called LOURYD). This code will allow the contractor to evaluate the efficiency of each apparatus (filter, condenser) in a soda water/hydrogen situation.
A methodology for the contamination balance of the DESORA installation was implemented. The measurements and analyses are divided into three parts:
- before the beginning of sodium destruction (*"zero point"*)
- during sodium destruction
- after the operation (measurements of residual activity).

The measurements will be carried out on sodium, soda, and hydrogen and will be located on various parts of the DESORA installation (tank, several pipes, hydrogen release, etc....). A schedule summarizes the measurement programme to control the contamination balance during the operation of DESORA.
Gamma scanning gaugings on severa radio-contaminants (Na-22, Cs-137, Eu-152, Ba-140), in various diameter pipes have already begun.

1.2. Studies for the installation of NOAH into the RAPSODIE containment building, including the required auxiliary equipment (B.1.2.)

The final adjustment studies were completed during the first term.

2.1. Manufacturing and installation of equipment (B.2.1.)

Concerning the sodium group, despite important delays in the delivery of specific equipment (LEWA pump, valves), the on-site assembly was ended in compliance with the schedule of mid-April 1992. All these works were carried out in Quality Assurance according to the calculation and mechanical construction rules of nuclear islands (the so-called RCCMR rules).

Concerning the soda group, more advanced than the sodium one, the schedule was respected and the acceptance report was established on March 30, 1992. These works were carried out within the framework of a Quality Assurance programme.
Concerning the instrumentation and control group, the delivery of equipment and connection cables on the site took place at the end of March 1992, while the connection of line heatings and the thermal insulation were carried out in June. These works were also performed within the framework of a Quality Assurance programme. As far as the specific equipment group is concerned, the electromagnetic pump of sodium recirculation was set up in March 1992.

2.2. Commissioning tests (B.2.2.)

1. **Suppliers tests:**
   These tests aimed at verifying that the whole of the equipment which makes up the installation was installed in conformity with the drawings, documents, safety standards, and that it corresponded to the operational specifications. They were then performed for every group: from March 20 to April 8 for the sodium group, from February 6 to March 6 for the soda group (ahead of the schedule), from April 20 to May 30, 1992 for the electricity, instrumentation and control group.
   The test reports and their conclusions were examined by the test commission on June 5, 1992. This examination did not give rise to major observations. However, the commission demanded that some test renewals be made during the global tests.
2. **Global tests** (without sodium)
   For this phase, a commissioning programme (PPE) was first studied and a test commission was appointed.
   Three important types of tests were performed by UDIN during that period, with the help of the contractors:
   - global tests in fluids with the control and instrumentation system
   - global tests in fluids for the Nitrogen and Hydrogen circuits
   - global tests of the sodium circuits with water, followed by the drying and the filling with Nitrogen.
3. **Staff training**
   The staff chosen for the operation attended a two-week training session focused on operational documents and safety instructions. It was followed by a practical training in the installation.
4. **Non-radioactive sodium tests**
   These tests began on October 28 in accordance with the schedule:
   - Filling test of the circuit with non-radioactive sodium: 1,500 liters of sodium were transferred into the installation as had been previously planned.
   - Test of the electro-magnetic pump circuit: it was observed that the pump flow-rate decreased with time, which will require a subsequent modification.
   - Test of the circulation pump (LEWA pump):
     After several satisfying circulation tests, it was impossible to restart the pump.

   A modification of the diaphragm of the pump added to a simplification of the circuit allowed the contractor to obtain a sodium circulation, but only in an unreliable way.
   After analysing the different problems, it was decided on December 18 to stop the tests and to order check-valves larger in diameter.

8.9. DECOMMISSIONING OF THE JEN-1 EXPERIMENTAL REACTOR

Contractors: CIEMAT, ENRESA, ENSA, LAINSA, UH-IW
Contract No.: FI2D-0023 / 0062
Work Period: July 1990 - June 1994
Coordinator: L MAÑAS, CIEMAT
Phone: 34/1/346 60 00 Fax: 34/1/346 60 05

A. OBJECTIVE AND SCOPE

JEN-1 is an experimental reactor of the swimming-pool type, moderated and cooled by light water, with a power of 3 MWt. It was operated from 1958 till 1984 with a total generated energy of 2,700 MWd. The radioactive inventory is estimated in the order of 3.5 x 10^{11} Bq (9.5 Ci), the dose rates are estimated in the range of 20 to 150 mGy/h.

The main aim of this project is the study and development of decontamination, cutting and melting techniques on contaminated or neutron-activated aluminium components of JEN-1.

Underwater segmenting of aluminium components still represents some problems to be solved, which consist especially in the limited visibility of the cutting environment, due to an important amount of very small suspended articles (10%, compared to steel 1%) and in the difficult filtration of these particles. An important aspect relevant to safety is the high H_2 generation rate due to a rather long lasting reaction of molten aluminium particles with the surrounding water.

Industrial-scale melting of aluminium components still needs development work for appropriate foundry techniques, especially concerning crucible material and slag formation.

Results obtained in this contract will be useful in the future for the dismantling of numerous research reactors with aluminium components.

The project is expected to produce specific data on costs, working time, waste arisings and radiation exposures to operators for the dismantling of the JEN-1 reactor.

After the second call for proposals in Section C a follow-up contract was concluded for the dismantling of further components: the primary circuit coolant collector and the support structures of the ionization chamber. Work in contract FI2D-0023 was limited to following components: core grid, support grid and control blade housing. Work in the second contract will need an adaptation of techniques and procedures already developed in the first work-programme for components having now larger wall thicknesses and different geometric configurations.

The work programme will be implemented in co-operation between following Spanish organisations: CIEMAT, ENRESA, ENSA and LAINSA, and with Institut für Werkstoffkunde of Universität Hannover (UH/IW), CIEMAT being the coordinator. A co-operation on aluminium melting will be installed with Siemens AG KWU Group and Siempelkamp Giesserei Krefeld (SG).

B. WORK PROGRAMME

B.1. Radiological characterisation of components to be dismantled, and of melting products (CIEMAT)

B.1.1. Radiological characterisation of the grid and support grid.
B.1.2. Radiological characterisation of the control blade housings.
B.1.3. Radiological characterisation of the primary coolant circuit collector.
B.1.4. Radiological characterisation of the support structure for ionization chambers.
B.1.5. Radiological characterisation of the melting products.

B.2. <u>Development, manufacturing, testing and subsequent installation in the JEN-1 reactor of an underwater cutting facility</u> by plasma arc and by consumable electrode techniques (UH-IW, CIEMAT).

B.2.1. Development and manufacturing of prototypes of plasma arc torch and consumable electrode torch (UH-IW)

B.2.2. Cutting tests with both tools on representative aluminium sheets, aiming at defining optimal working parameters, cutting effluents and appropriate air and water filters (UH-IW, CIEMAT)

B.2.3. Comparison of both tests with respect to cutting performance, generation and type of cutting effluents and tool handling abilities with subsequent selection of the most appropriate tool (CIEMAT + UH-IW)

B.2.4. Design and manufacturing of a cutting facility, including the selected cutting tool, handling and sensor systems and the cutting cell (UH-IW)

B.2.5. Testing at UH-IW and optimisation of the whole system in water depths of 5 m (UH-IW, CIEMAT)

B.2.6. Training of the CIEMAT staff at UH-IW (UH-IW + CIEMAT)

B.2.7. Transport and assistance for the installation of the cutting facility in the JEN-1 reactor (UH-IW)

B.3. <u>Underwater dismantling of reactor internals</u> after preceding dismounting work (CIEMAT + UH-IW)

B.3.1. Dismantling of the grid and grid support

B.3.2. Dismantling of the control blade housings

B.3.3. Dismantling of the primary circuit cooling collector

B.3.4. Dismantling of the support structure for ionization chambers

B.4. <u>Decontamination of reactor internals</u> (ENSA, LAINSA).

B.4.1. Selection of suitable procedures with respect to decontamination efficiency, amount and type of arising secondary wastes, reprocessing abilities and radiological impact

B.4.2. Decontamination of the grid and grid support

B.4.3. Decontamination of the control blade housings.

B.4.4. Decontamination of the primary circuit coolant collector

B.5. <u>Melting of aluminium waste</u> (CIEMAT, ENRESA)

B.5.1. Selection, manufacturing and adaptation of a melting furnace and implementation of cold melting tests.

B.5.2. Main melting programme, including generation of data on volume reduction and decontamination effects.

B.6. <u>Assessment of radiation protection</u> including both the personal and the ambient radiological impact (CIEMAT, UH-IW)

B.6.1. Assessment of radiological impact during dismantling operations (CIEMAT, UH-IW)

B.6.2. Assessment of radiological impact during decontamination operations (CIEMAT)

B.6.3. Assessment of radiological impact during melting operations (CIEMAT)

B.7. <u>Generation of specific data</u> on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.3., B.4., B.5. and B.6.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

During 1992, activities went on in the following tasks:

Dismounting of JEN-1 reactor internals, characterization of such internals concerning their radioactive content, plasma arc and consumable electrode cutting tests, designs relating to: modular plasma torch, sensors (ultrasonics and eddy current) to enable correct distance between tools and workpiece as well as the angular position, procurement and mounting of systems to undertake the dismantling operations of components involved in this present project, manufacturing of a basin cutting where the components are to be cut under water, decontamination tests on contaminated samples taken up from JEN-1 internals involved in the present project, mounting activities concerning the melting facility erection relating to aluminium scraps melting as a recycling material method, as well as melting laboratory tests on aluminium samples, surveillance of radiation protection during dismounting and characterization activities.

On the other hand, forty spent fuel elements, which as reported in the last annual report, were stored in shielded wells, have been moved out from them and again transferred to JEN-1 pool to be loaded in GOSLAR casks for their transport and storage in the United Kingdom.

Progress and results

1. Radiological characterization of components to be dismantled, decontaminated and melted (B.1.) (CIEMAT)

Through this period, a complete characterization from all JEN-1 reactors internals has been undertaken, not only by γ-spectrometry, but by means of α and ß-spectrometry on the many taken samples.

Special mention is made of the sampling on the grid frame, 150 mm thickness, in order to assess the radioactive content distribution as a function of depth. Samples were taken up on four points presenting the higher values relating to contact dose rates, 4, 5, 7 and 10 mGy/h. On each of these points, four samples were taken, coming at different depth ranges, 0-5, 5-25, 25-45 and 45-65 mm. From the analysis of these samples may be inferred that Co-60 radioactive content distribution keeps practically constant through the bulk material.

Table I shows specific activity (Bq/g) relating to main radionuclides concerning following internals: control blade housings, grid and ionization chambers support. γ-spectrometry relating to grid support and primary cooling collector is given in Table II.

2. Development, manufacturing, testing and subsequent installation of an underwater cutting facility (B.2.) (CIEMAT, IW)

2.1. Development and manufacturing of prototypes of a plasma torch and a consumable electrode torch (B.2.1.)

The development and manufacturing of the modular plasma torch are completed and first cutting tests were performed. The results of these cutting tests are comparable to those obtained with a commercial 600 A Ar/N_2 torch, with a tendency to the better, especially in terms of kerf geometry.

2.2. Cutting tests (B.2.2.)

Underwater plasma arc cutting tests went on at CIEMAT on aluminium sheets of 6, 24, 31 and 60 millimetres thickness. Cutting parameters and secondary emissions, concerning sedimented dross, suspended particles and aerosols were obtained as well as the percentage contribution of such emissions to the total work-piece mass loss. Size distribution analyses were done by sieving on sedimented dross obtained from the above cutting tests.

2.3. Design and manufacturing of a cutting facility (B.2.4.)

The cutting basin, 3 m diameter and 5 m high, is under construction. Its inside will be provided by a grid, 1 m above the bottom, to attach the components to be cut, supported in turn on a funnel to collect sedimented dross in a bucket which will be emptied periodically, lifting it up through a tilting door in the grid. A duct from its bottom will feed the water through a water purification system for filtration.

The purification system, in a closed circuit, is composed of a centrifugal multi-cell suction pump, two packing columns loaded with AG material able to remove suspended particles sized larger than 10 μm and two filtration units containing each seven cartridge polypropylene filters.

Gases and aerosols arising from cutting operations will be conducted through a ventilation system consisting of: a suction hood, slightly immersed in the water surface so that only gaseous products coming from cutting will be collected, a filter bank composed by a prefilter, HEPA filters and a centrifugal fan with a flow control valve.

The fine positioning of the tool against the work-piece will be done with the aid of a twin-sensor system which was developed at IW. This sensor device consists of two independent sensors. One is a commercial ultrasonic sensor, and the other one an eddy-current sensor.

3. Underwater dismantling of reactor internals (B.3.) (CIEMAT, IW)

Disassembling of all the JEN-1 reactor internals was achieved through the present year. They have been stored in the low power section of JEN-1 pool for its dismantling under water.

Such dismantling operations were achieved without lowering the water level of the pool and by means of long articulated poles, operated manually from one of the JEN-1 pool bridges. These operations have been undertaken at the great depth (more than 8 m); they had to be controlled and monitored by means of TV cameras.

4. Decontamination of reactor internals (B.4.) (ENSA, LAINSA)

4.1. Selection of suitable procedures with respect to decontamination efficiency, amount and type of arising secondary wastes (B.4.1.)

In 1992, cold decontamination tests on aluminium samples went on with liquid chemical reagents, the results of which are presented in the particular report concerning LAINSA.

Relating to hot decontamination, several samples of aluminium belonging to JEN-1 internals have been subjected to various tests using an alkaline-permanganate media enhanced with ultrasonic power.

From the results obtained can be established that the Co-60 contribution to the solution remains practically constant, indicating a homogeneous distribution of this radionuclide in all the JEN-1 internals. Concerning Cs-137, the higher contribution to the activity concerns the first 5 μm of sample with a slight increase to a depth of 50 μm.

5. Melting of aluminium waste (B.5.) (CIEMAT, ENRESA)

In 1992, the shop civil work was finished, as well as the surface treatment given to the walls and floor.

During this period, the auxiliary systems (electric power supply, light etc.) as well as the safety systems (alarms, fire protection, communication, etc.) reached the final stage of assembly.

Several aluminium ingots are being prepared with a 0.37 % Co for inactive melting tests. Using the technique of Differential Scanning Calorimetry (DSC), it is possible to detect and quantify the physio-chemical changes and phenomena produced in the melting process.

6. Assessment of radiation protection (B.6.) (CIEMAT)

During this period, the workers occupied with the dismantling, transfer to the low power section of the pool and sampling of JEN-1 internals, were provided with direct reading dosimeters which did not register any significant dose value.

Table I - JEN-1 Reactor Internals, Main Nuclides Specific Activity (Bq/g)

Internal Nuclide	Grid	Housing No. 1	Housing No. 2	Ionization chamber Support
Co-60	11,000	170,000	140,000	10,000
Ni-63	55	320	310	5
Ni-59	210	630	310	220
Nb-93	300	130	94	77
Nb-94	24	19	12	13
Eu-152	-	790	220	80
Eu-154	-	2100	2400	-
Eu-155	-	-	600	-
Cs-137	57	1,800	1,700	61
Sr-89	3.7	420	210	3
Sr-90	53	1,600	1600	7.3
Pu-238	0.8	39	31	-
Pu-239	2.6	66	58	-
Pu-241	-	2,600	2,300	10
Am-241	0.4	96	43	-
Cm-242	-	0.5	0.2	-
Cm-244	-	1	0.3	-

Table II - Specific activity (Bq/g)

	Co-60	Cs-137
Grid support	940	41
Collector	690	310

8.10. DEVELOPMENT OF SEGMENTING TOOLS AND REMOTE HANDLING SYSTEMS AND APPLICATION TO THE DISMANTLING OF VAK BWR REACTOR PRESSURE VESSEL INTERNALS

Contractors: VAK GmbH
Contract No.: FI2D-0029
Work Period: July 1990 - December 1993
Coordinator: H H KALWA, VAK
Phone: 49/6188/499136 Fax: 49/6188/499125

A. OBJECTIVE AND SCOPE

The experimental Boiling Water Reactor Kahl (VAK-BWR) of 16 MWe has been shut down after 25 years of operation. Dismantling has been started for some time. The present estimation of the radioactive inventory of the reactor is in the order of 5 x 10^{15} Bq.

The aim of the present contract is the development, qualification and practical application of different underwater (UW) segmenting and remote handling techniques on a series of internal components out of the reactor pressure vessel (RPV). Important targets are: minimisation of operators' dose uptake and of primary and secondary waste generation and economics of the procedure. Specific radioactivity of such components is in the order of magnitude of 10^5 - 10^8 Bq/g (activation) and of 10^4 - 10^5 Bq/cm^2 (contamination). Due to its long-term operation, VAK dismantling can be considered to a large extent (dose rates, activation, contamination, material ageing) as representative for the future decommissioning of LWRs. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

Work will be implemented in close co-operation with the pilot dismantling projects BR-3/Mol (§ 8.3.) and KRB-A (§ 8.5.). The results of the comparative assessment study made by KRB (§ 8.2.) will be considered in the implementation of the contract.

B. WORK PROGRAMME

B.1. Conceptual studies and construction of a 1:1 scale facility for UW testing of cutting tool and devices for remote operation

B.2. Preliminary tests on non-radioactive components, including devices for segmentation, remote operation techniques, definition of generated secondary waste and studies of dismantling scenarios

B.3. Qualification of dismantling procedures for an application to radioactive components

B.4. Dismantling of a series of RPV internals (upper grid plate, chimney above the core, control systems)

B.5. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2., B.3. and B.4.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

After evaluation of seven decommissioning studies (two for VAK and five for KRB-A) a number of questions were still not answered. Therefore, we performed the so-called "additional evaluation" with NIS and NUKEM for the choice of the principal cutting and handling technology for the VAK RPV dismantling which resulted in a dismantling concept based on so-called "clean" cutting technologies ([roll-]shearing, milling, sawing, grinding and electro discharge machining [EDM]).

The RPV wall and cylindrical internals such as the "core case", "thermal shield" and geometrical similar components shall be cut using a so-called "circumferential tool support". Therefore we have performed sawing and milling tests with a welding cladded RPV sample both under water and in air.

For dismantling of the internals of the RPV other "special tool supports" shall be used. We have therefore performed under water

- grinding cutter tests with typical internal material samples using a rack-and-pinion supported, hydraulically driven grinding tool /1/ (abrasives: diamond, corundum), (Figure 1) and
- sawing cutter tests using a rod guided, pneumatically driven hacksaw /2/, (Figure 2).

The importance of a "bi-arm electromechanical master slave manipulator" (bi-arm EMSM, /3/) in our dismantling concept is considerable. It will not only be used for such important "support tasks" as

- installation of a video system,
- placement of suction nozzles,
- changing of tools,
- assisting crane rigging,
- support of the container handling devices etc.

The presence of an EMSM is necessary to achieve high performance for all the decommissioning tasks.

Thus, important targets of our work is to

- simplify the tool support,
- designing reduction of the tool support force to a more practical level,
- minimization of aerosol and secondary waste generation.

Progress and Results

1. Conceptual studies and construction of a 1:1 scale facility for UW testing of cutting tools and devices for remote operation (B.1)

1.1 Conceptual studies (B.1)

After the "additional evaluation" /4/ of the above-mentioned seven decommissioning studies, it was established that the choice and development of a cutting technology without simultaneous consideration of adequate technologies for the handling of cut pieces, secondary waste, aerosol and water filters would lead to interim decommissioning equipment.

Only after performance of the above additional studies were we able to choose the principal dismantling components of our currently preferred decommissioning equipment.

If the cutting tool has a high tool reaction force, a tool support must be constructed that is sturdy and reliable. After performing the reactor decommissioning tasks it is very possible that all the cutting tools and heavy tool supports will become secondary waste as well.

In other words, after performence of the "planned" decommissioning works it would be additionally necessary to dismantle our contaminated decommissioning equipment.

1.2 Construction of a 1:1 scale facility for UW testing of cutting tools and devices for remote operation (B.1.)

After performence of the above-mentioned conceptual studies, we were able to formulate the main requirements of the VAK test installation. This led to an extension of the tasks and schedule.

The test installation will now allow

- simulation of the 1:1 scale space relations of the VAK reactor shaft in three dimensions, which is the main point,
- sampling of suction water (filter),
- variation of the water level,
- observation of the cutting and handling process from the outside of the test installation through windows, which are at different viewing levels.

After adoption of the upper part of the test installation for simulation of other space relations, it also will enable all of the international decommissioning teams, which are interested in UW cutting or handling, to perform parts of their experiments, in a 1:1 scaled in depth environment.

2. Preliminary tests on non-radioactive components (B.2.)

2.1. Cutting tool tests for internals (B.2.)

Underwater saw (and grinding) cutting is the state of the art in UW dismantling techniques for nuclear components /5/, /6/, /7/.

The main objectives of performing these UW cutting experiments was to find the optimum parameters for:

- abrasive type (corundum / diamond),
- bonding material type (hardeness),
- grinding tool type (thickness / diameter),
- perimeter or alternating speed,
- tool support force

to achieve performance criteria for:

- high tool reliability (low level of probability for breaks or other tool failures),
- low tool abrasion for a minimum of tool exchanges (job dose, secondary waste),
- minimizing tool support force for application of easily usable "special tool support" (no heavy, precision tool support),
- minimizing secondary waste generated into the reactor water for reduced filter loading and disposal volume.

For complex geometry internals dismantling ("upper grid plate", "sprinkler ring for emergency core cooling" and comparible internals), we have performed grinding and sawing cutting experiments.

For the above-mentioned necessity of reduction of the tool reaction force, we will use cutting technologies with so-called

- "indifferent cutting edges" (grinder) or tools with an
- "alternating" reaction force (reciprocating or hacksaw).

These types of tools are based on grinding or hacksawing which have the advantage of variable alternating speeds for using of tool inertness effects for a "low-reaction-force" cutting.

We have performed our UW grinding experiments using following equipment. A tool and sample support was mounted into a water pool (350 liters or demineralized water). The tool unit was hydraulically driven. The vertical tool motion was supported by a manual spindle gear, the horizontal tool motion by a hydraulic rack-and-pinion gear. The tool support direction was always contrarotating.

The chosen sample cross-section for grinding cutting (147x10mm) was conservative in comparison with our original "Upper grid plate" cross-section (147x8mm). The sample material is (X8CrNiTi1810) and approximately of the same quality for grinding cutting like the original (X7CrAl13).

The chosen sample cross-section for saw cutting (pipe 57mm x 3.5mm) was conservative in comparison to most of the applications to the SRN, which has pipes of 30mm x 2.5mm (X8CrNiTi1810) is a sample material of comparable for sawing parameters, especially the significant material strength (X10CrNiNb18.9).

2.2. Diamond cutting tools

At first we performed grinding cutting tests with six diamond cutting discs (D=230mm). They had two different thicknesses (2.5mm and 3.0mm) and three different types of diamonds in its bonding material. We carried out cutting experiments with diamond grinding (modified sawing) discs. They were modified by specific diamond density and type of bonding material.

Because of the especially chosen low tool support force (700N), all the diamonds were smeared with sample material. For a successful cut it was necessary to increase the support force up to for our tool support design impractical values of 3kN.

2.3. Corundum cutting tools

We performed tests with 10 corundum grinding discs. Not one of their producers have had any experience in UW grinding cutting of austentitic material at a low level of tool support force (<<700 N) and at a perimeter speed much less than the commonly recommended 80m/s.

The corundum grinding cutting results were good reproducable. For its demonstration we have made slices of 1mm thickness only through the cross-section of about 147x10mm which is possible to do with low-reaction-force tools only /8/.

The grinding swarf, generated by corundum grinding is also good collectable by a 100-µm-sieve.

We obtained the best results in steadiness with special grinding discs (D230x1.5mm) without fiber reinforcement. The final selection of a special type of grinding discs and the development of a cutting tool support is still underway.

2.4. Hacksaw tests (B.2.)

We performed cold cutting tests with a pneumatically driven hacksaw and successfully cut the cross-section (5.9cm²) of the middle pipe of our "sprinkler ring for emergency core cooling". The results were reproducable. This pipe consists of X8CrNiTi18.10 material. Cutting was performed in the air and without sawblade cooling (Figure 2). For results see section 5.

2.4.1. Construction of a tool support for underwater sawing (B.2)

For cutting the pipes of our "sprinkler ring for emergency core cooling" (SRN) we constructed a tool support, which corresponds to our above requirements:

- easy to handle,
- does not need same force manipulation on the operator level,
- after accurate tool placement on the pipe to be cut, the tool support automatically works,
- the tool support force depends on the strength of the spring only.

After successful performing of cutting tests, the spring shold be substitude by a pneumatic cylindre which can work in two directions.

3. Qualification of dismantling procedures for application to reactor components (B.3)

The parameter "cutting speed" was not optimized because of the emphasis on the reliability of the entire decommissioning process, which contains not only the cutting process but also the handling, decontamination, waste volume reduction and so on.

In spite of this, we always observed the parameter "cutting speed", but primarily to define the t e n d e n c y of the more important parameter of "cutting tool steadiness".

3.1 Cutting tool tests for internals. Dismantling of the "sprinkler ring for emergency core cooling"- (SRN-) model (B.3)

We are performing cutting and handling tests with a sample of our "sprinkler ring for emergency core cooling", which would be the first internal to be dismantled because of the location in the RPV. The cutting sample was placed in the VAK test installation at the same distance from the tool operator. We used a pneumatically driven hacksaw (VAK abbreviation or symbol: PSS1), manufactored by the "Fein" company which was modified as follows:

- redirected exhaust air location,
- increase alternating gear inner-pressure,
- redesigned tool support.

To raise the gear inner-pressure against water infiltration we modified the tightness of the bearing between the gear and the pressure part of the pneumatic tool.

We have chosen a so-called "self-supporting" tool support system. "Self-supporting", because the tool would support itself after accurate placement to the component to be cut, independent of the user. Therefore it is not necessary to have a defined supporting force at the tool operator level (at + 8m or 3m) which is an important objectiv of our work.

4. Dismantling of a series of RPV internals (B.4.)

We have not yet received any results.

5. Generation of specific data (B.5.)

Using a hacksaw, the average specific waste generation (gramms per one cm^2 of cut cross-section) is in the range of $1.0g/cm^2$. Using diamond cutting we generated $2.0g/cm^2$ and for corundum grinding cutting $0.5g/cm^2$ of a good collectable coarse swarf.

For comparison, the plasma melt cutting technique with consumable electrode generates $14g/cm^2$ of suspended particles /9/.

6. Lessons learned

The use of grinding discs for under water grinding cutting is possible and meaningful in cases of cutting relatively thinwalled (up to 10mm) internals. Althougth the number of tool exchanges is more frequent, the special corundum grinding discs generate less secondary waste in summary because the discs are two times thinner than diamond discs.

The power dissipation of the tool drive by UW working is of a practical level. Thus the application of UW-grinding cutting depends more on the breaking strain of the disc than on the technical possibilities for power supply for the tool drive.

The hacksaw cutting is a promising technology for complex internal dismantling up to the hereby investigated 10mm material thickness especially in view of the

- constant water visibility without cleaning,
- large swarf particles generated (> 1mm),
- practical tool steadiness.

7. Outlook

Application of EDM to thin-walled internals seems to be a possible method for their dismantling. Application of EDM to relatively thick-walled internals, such as the "Thermal shield", is the subject of other decommissioning projects /5/.

In the meantime the application of laser cutting (percussion laser drilling) was of more interest, even considering the aerosol generation, because of the thin cutting kerf (0.5mm) and the corresponding low secondary waste generation. The UW cutting depth in austentitic material is presently up to 8mm but not reproducable.

To gain a better understanding of this cutting technology under NPP dismantling conditions, we performed preliminary tests at the LASAG (CH) experimental site at Thun and received the following results:

- the laser power is presently not sufficient,
- the glass-fibre cable dissipates too much power,
- it cannot work without additional process gases, which are not advantageous, because of the aerosol spreading.

Nevertheless we are maintaing the connections to developers and users of the laser cutting technique.

As near time test we have planned extended cutting experiments using sawing, grinding and EDM for complex geometry internals dismantling.

References

/1/ VAK "Development of segmenting tools and remote handling systems and application to the dismantling of VAK pressure vessel and internals", Contract FI2D-0029, 4th Progress Report

/2/ as above, 5th Progress Report

/3/ ANSA Assistance Nucléaire SA., Bouzonville, "Description of the EMSM2B"

/4/ NIS/NUKEM, "Weiterführende Bewertung von Stillegungsstudien", 02/91

/5/ S.C.K./C.E.N., BELGATOM, "The BR-3 Pressurized Water Reactor pilot dismantling project", Contract FI2D-0003-B(TT) av. 1, 1st Progress Report

/6/ WCL Inc., Manchester, ENC 90 Conference, Lyon, Sept. 1990, Text of Presentation, "Isolation Of B5 And B15 Water Ducts At BNFL Sellafield"

/7/ Decommissioning Handbook, US Department of Energy, National Technical Information Service, DOE/EV/10128-1

/8/ GROF, H.E.; "Beitrag zur Klärung des Trennvorganges beim Schleifen von Metallen", Dissertation, Technische Universität München, 1977

/9/ CIEMAT ENRESA ENSA LAINSA IW(UH) "Decommissioning of the JEN-1 experimental reactor", Contract FI2D-CT90-0023, 3rd Progress Report

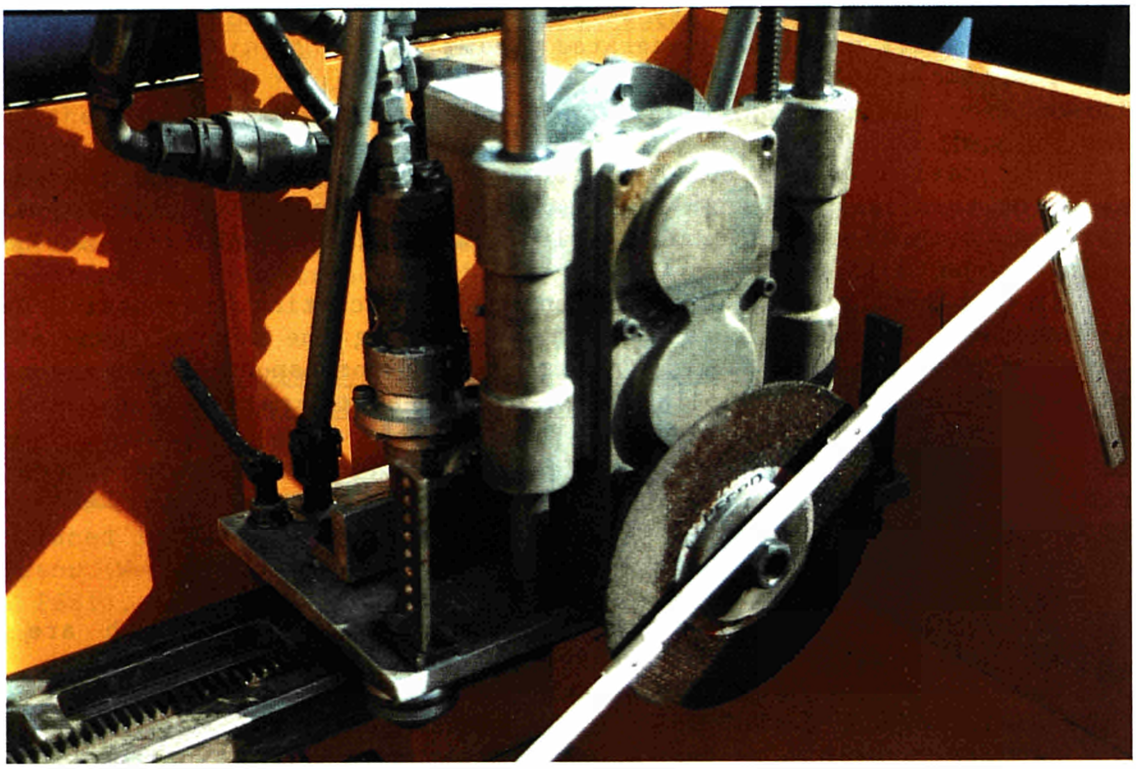


Figure 1

Figure 2

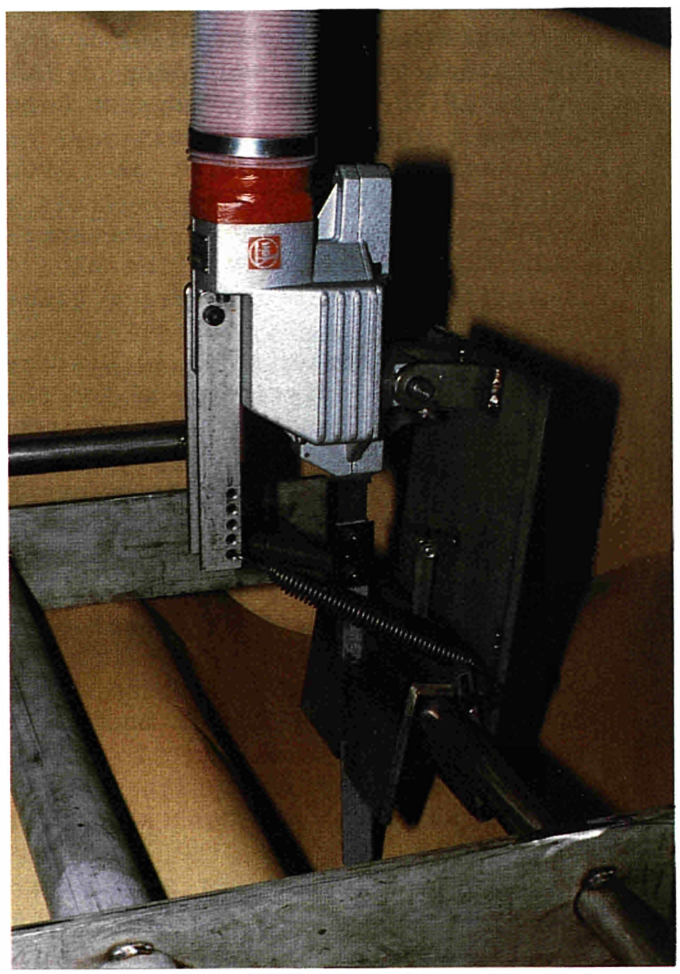


Figure 1 Grinding cutting

Figure 2 Hacksaw cutting

8.11. MELTING OF FERRITIC STEEL ARISING FROM THE DISMANTLING OF THE G2/G3 REACTORS AT MARCOULE IN A FURNACE INSTALLED AT THE DISMANTLING SITE

Contractors: CEA-Valrhô
Contract No.: FI2D-0034
Work Period: September 1990 - September 1993
Coordinator: J S FEAUGAS, CEA-Valrhô
Phone: 33/66 79 63 03 Fax: 33/66 79 64 32

A. OBJECTIVE AND SCOPE

In two foregoing EC R&D decommissioning programmes, a series of research contracts had been devoted to melting of metallic radwaste, mainly steel, going from laboratory scale to applications in adapted foundries, treating waste transported from the dismantling site to the melting facility.

The objective of the present contract is to conceive, manufacture and install a 15 t electric arc heated melting furnace on the dismantling site of the G2/G3 graphite/gas reactors at Marcoule, and to condition by melting 700 t (out of a total of 4,000 t) of ferritic steel having a specific contamination in the order of 20 - 40 Bq/cm^2.

The innovation lies mainly in the on-site installation of the furnace, avoiding packaging and transportation on public roads and in the large dimensions of the furnace (2 m), enabling feeding of pieces up to 1.7 m and reducing segmenting work. This should lead to economics by reducing the number of operations and by an optimised management of waste streams, enabling to a large extent unlimited recycling of steel.

In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

B. WORK PROGRAMME

B.1. Conceptual studies and categorisation of waste
B.1.1. Studies for the installation of the melting furnace in the reactor building.
B.1.2. Investigation into the management of waste streams before and after melting.
B.2. Manufacturing, installation and testing of equipment
B.2.1. Manufacturing and installation of equipment including auxiliary and control systems.
B.2.2. Commissioning testing of the melting facility.
B.3. Main operation for the melting of 700 t of dismantled steel
B.4. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1., B.2. and B.3.

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

After conceptional works carried out during 1991, the melting process with a furnace of 15 tons capacity was installed inside the G3 reactor building.

Final inactive tests concerning the starting of melting facilities were implemented from March 9 to March 30, 1992 and active tests started on April 3rd, 1992. They were carried out on 128 t of scrap resulting from dismantling the reactors G2 (34%) and G3 (66%).

The good working of the facility was checked, that is to say ventilation, confining, cooling as well as mass and radiological follow-up of melting. Electrical conditions for fupply of melting were decided.

The recovery of waste, of dust on filters, and of slags at the top of the bath, was carried out in good condition. A first series of operational melting concerning the 700 t from the initial contract, was implemented from April 27 to July 30, 1992. From August to the end of the year, 960 additional tons were melted.

At the entrance, the radiological characterization is based on the following distribution: 5% of Cs, 95% of Co. The cast steel and the slag contain no cesium which is mainly collected in the dust filters.

Production is mainly ingots (25 kg) when melting activity is less than 1 Bq/g and blocks (4,000 kg) of pyramidal shape with squared base, in the other cases (activity more than 1 Bq/g). Ingots and blocks are stocked in rooms which are emptied by dismantling on the G2 and G3 site because the French authority did not decide the limit of activity for recycling the melting products without restriction ("de minimis"). Final dust and slags will be disposed in a waste management site.

Progress and results

1. Operating conditions tests (B.2.2.)

From January to July 1992, two main actions were carried out:

- the end of inactive tests to define the normal conditions for the furnace,
- the execution of active tests with about 120 tons of iron, in order to obtain the authorization for an industrial operation. During this period, the estimated level of irradiation, the dispersion of radioelements, the efficiency of filtration devices were checked.

2. The melting of 700 tons (B.3.)

This melting carried out from April 27 to July 30, 1992, with the recycling of ingots, which activity was superior to 1 Bq/g, and also with the recycling of dusts. The mass of scrap was 699,182 kg before melting from which 671,275 kg of cast iron and 36,838 kg of total waste were obtained.

The radioactivity of scrap was at the beginning 1.9 Bq/g on average, and this is distributed in 0.72 Bq/g in the ingots, 6.50 Bq/g in the blocks and 19.30 Bq/g in the dusts.

3. Radiological characterization

Before melting, it is very easy to know the radioactivity of G2-G3 circuits and its distribution between the "Cesium" and the "Cobalt" nuclides.

A measure of gamma dosis flow, outside the container, is made and the total activity is then calculated.

During melting, samples of some grammes are taken and are then measured.

After melting, countings on some samples are made.

4. Radioprotection

Radioprotection is the work of the CEA health physics specialists. During the meltings realized in 1992, specific problems did not appear either for the people working in this field, or for the buildings.

5. Present storage

All the ingots are stored in buildings near the furnace, because the activity levels for release have not yet been defined.

TABLE I - RESULTS FOR THE YEAR

Melting with recycling	95 t
Number of melting	134
Material to be melted in 1992	1,700 t
Graphite	86 t
Ferrosilicon	38.4 t
Melting	1,653 t
Ingots <1 Bq/g >1 Bq/g (mean)	0.65 Bq/g 9.71 Bq/g 1.15 Bq/g (1,174 t)
Blocks <1 Bq/g >1 Bq/g (mean)	0.64 Bq/g 6.89 Bq/g 6.65 Bq/g (479 t)
Remaining dust	14.3 t (33.9 Bq/g 87% Cs 13% Co)
Slags	49.3 t
Electric energy	8,283 kWh per melting

TABLE II - WORKING CONDITIONS

Atmospheric contamination	melting zone: 0.08 to 3 Bq/m3 maxi preparation zone: 0.02 to 1 Bq/m3 maxi
Surface contamination	melting zone: 5 to 25 Bq/cm2 preparation zone: 2 to 160 Bq/cm2
Surrounding flow rate	10 to 20 MGy/h

Reference: report 1991 [EUR 14498]

8.12. MELTING OF ALPHA-CONTAMINATED STEEL SCRAP AT INDUSTRIAL SCALE

Contractors:	Siemens-KWU, SG
Contract No.:	FI2D-0038
Work Period:	October 1990 - December 1993
Coordinator:	K H GRÄBENER, Siemens-KWU
	Phone: 49/69/807 36 45 Fax: 49/69/807 20 66

A. OBJECTIVE AND SCOPE

The underlying large-scale investigation into melting of alpha-contaminated steel from nuclear facilities aims at demonstrating the feasibility of the unrestricted reuse of such radwaste within legal limits.

The work programme will be based on the results and experience obtained on melting of radwaste in former research contracts within the second EC programme on Decommissioning (1984-88), especially contract FI1D-0044 with Siemens AG and contract FI1D-0016 with Siempelkamp Giesserei GmbH.

Starting with laboratory-scale melts aimed at identifying the most suitable crucible material and slag former will be followed by large-scale melts with subsequent detailed analysis of the prevailing alpha-distribution in and between steel, slag and filter dust.

Based on the foregoing results, large-scale melts with about 100 t of uranium and Pu-contaminated material from Siemens fuel fabrication will be carried out and finally, by two large-scale melts of Pu- and Th-contaminated steel waste (5 t), will be assessed how these alpha-emitters will behave.

It is anticipated that extensive testing and radiological measurements will enable the assessment that alpha-contaminated steel can be conditioned by melting for safe unrestricted reuse and that the melting plant can be operated safely also with respect to radiation protection of workers and the environment of the foundry, with special consideration of the arising slag and filter dust. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

The specific contaimination of the treated radwaste is estimated to be in the range of $\leq$ 200 Bq/g (alpha/beta) and the anticipated fission product inventory for large-scale melting is estimated at about 200 g of U-235 and 1 g of Pu. Expected dose rates in the controlled melting area are in the order of magnitude of < 0.1 mGy/h.

Work will be executed in close co-operation between Siemens AG, KWU Erlangen (Siemens) acting as coordinator and Siempelkamp Giesserei (SG).

B. WORK PROGRAMME

B.1. Identification of appropriate materials for crucible and slag formers and procurement of U, Th and Pu containing radwaste samples (SG)

B.2. Installation of an induction-heated laboratory furnace and execution of reference tests with non-radioactive materials (Siemens)

B.3. Laboratory-scale melting tests with U, Th and Pu-contaminated steel (selection of materials for crucible lining and slag formers) (Siemens)

B.4. Procurement of U and Pu-contaminated material (Siemens) and Th-contaminated material (KEMA)

B.5. Pilot melting tests aimed at determining the U (alpha)-content in ingot, slag and filter system (SG, Siemens)

B.6. Main melting programme of about 100 t of U and Pu-contaminated radwaste with subsequent alpha-content determination in each ingot, slag and filter dust (SG)
B.7. Execution of two large-scale melts with Pu and Th-contaminated steel (SG)
B.8. Determination of the alpha-distribution in the crucible material (Siemens)
B.9. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2. and B.6.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

Tasks 1 to 5 are now completed. Task 6 has also progressed: about 58 tons of alpha-scrap have been melted. During the melting campaign, samples were taken from metallic melt, slag, filter system and crucible.

From these samples, gamma spectra were taken and the total alpha- and beta-activities were estimated by a low-level counter. Finally, the U/Pu ratio of each sample was estimated by chemical separation, electrolytic deposition and alpha-spectrometric determination of the U and Pu content. The results spread very wide because the Pu content of the samples was extremely low. The maximum U content found in a melt sample was 0.3 Bq/g, the maximum Pu content 0.02 Bq/g.

Progress and results

Task 6: Main melting programme of 100 t of U and Pu contaminated radioactive waste

From the first melting campaigns at Siempelkamp, samples were obtained from the metallic melt, slag, filter dust and from the crucible. The sampling, during melting and activity measurements, was coordinated between Siemens and Siempelkamp. During a melting process, 3 phases of material arise. The major part is the metallic fraction with more than 98% in weight, followed by the slag with more than 1% and the filter dust with less than 1%.

With each sample, gamma-spectroscopy and alpha, beta low-level measurements were performed. From earlier investigations, the behaviour of some radionuclides is already known. The alpha-emitters like Uranium and Americium are transferred to the slag. For Plutonium, the same chemical behaviour as for U and Am is expected. That means it should be found enriched in the slag together with Uranium. To check whether this assumption is right, the ratios between U and Pu in the samples were estimated by alpha-spectrometric examinations. For this method, it is necessary to separate the alpha-emitting nuclides from the matrix and to prepare thin and uniform layers by electro-deposition. The radionuclides being measured and their corresponding yield-tracers (U232, Pu236) were determined simultaneously by the same detector on the same prepared sample. In this way, chemical yield and counting efficiency were corrected automatically. To measure the nuclide-specific alpha activities, an alpha spectrometer was used, a 900 mm^2 surface barrier detector with an energy resolution of $\leq$ 20 KeV (FWFM).

Before measuring, it was necessary to prepare the samples. Drillings were taken from the metallic ingots. The slag samples were grund to a size smaller than 0.1 mm. The powder-like samples were then prepared for dissolution and specific separation. The required sample amount was estimated by the Plutonium content of the samples. The minimum of Pu needed should be 20 mBq.

A schematic view of the chemical separation method is given on Figure 1. The separation of U and Pu from the matrix is done by precipitation of the hydroxides, dissolving again and extraction of U and Pu with organic solvents. After backextraction, U and Pu were collected on a stainless steel plate by electrodeposition.

An internal standard of 200 mBq Pu236 was added to each sample. For Uranium, the known content of U235 was taken as the internal standard.

The results of our examinations showed that gamma measurements are sufficient to estimate the uranium content of the samples, provided the composition of the U-isotopes is well known. For the results of the gamma-measurements, see Table II.

To decide whether the activity of a steel sample is within the limiting values for unrestricted reuse or not, a low-level alpha and beta-measuring device is sufficient. The results of low-level α-measurements are also shown on Table II.

Alpha-spectrometry is only needed for estimating Uranium and Plutonium isotopes. As can be seen from Table III, the results of the alpha-spectrometric measurements spread very wide.

This is caused by the extremely low Pu and in parts U-content of the samples. The maximum U content, found in the melt steel samples (samples KOXS; X = 1,2,3,4) was < 0.3 Bq/g. This is the reason for the experimental difficulties in separating especially the extremely small Pu content from the iron matrix, and why the results of estimating the isotopic ratios of U and Pu spread so wide on Table IV.

Another parameter which estimated the accuracy of the experimental results was the varying pH value during the electrolytic deposition of U and Pu. To clear up this point, more investigations would be necessary.

At Siempelkamp foundry, 58 Mg of low α-contaminated scrap were melted. The uranium nuclides were transferred into the slag, so that the slag is enriched with fissionable material. The limit in the licence for the melting campaign is 3 g fissionable material per 100 kg slag. For future melting of the rest of scrap (42 Mg), depleted uranium should be added so that in total a renaturalized content should be achieved.

Future work:

The wide spread of the values for the U:Pu ratio is a consequence of the very low U and Pu contents.

To clear up the question whether there is any enrichment of Pu during the melting, the contractor proposes to perform a laboratory melting experiment with higher values of Pu.

Table 1: Nuclide Vector of Steel Scrap
Normalized to U238 = 1

Nuclides	Bq/g	Activity
U232	0.028	Alpha
U233	0.150	Alpha
U234	8.250	Alpha
U235	0.281	Alpha
U236	0.083	Alpha
U238	1.000	Alpha
Summe Uran	9.81	
FP U232 FP U235 FP U238 Transurane without Pu239/241 Spaltprodukte	2.33	Alpha + Beta
Pu239	0.01	Alpha
Pu241	0.01	
	12.16	

Specific activity of Uranium = 1.17 E5 Bq/g

values from first annual report

Table 2: Results of gamma and low level alpha measurements

low level measurements: with infinitely thickness, 1αcpm/cm^2 =~50 Bq/g
Gamma Measurements: U235 from 144 KeV and 186 KeV, 1 Bq U235 = 35 Bq U_{total}

No.	Sample	Gamma Measurement		low level measurement	
		U235 Bq/g	U_{total} Bq/g	α-cpm/cm^2	α-Bq/g
1	K01T	< 0.25	< 8.7	0.088	4.4
2	K01S	< 0.004	< 0.14	0.004	0.24
3	K01R	0.71	25	0.67	33
4	K01Z	4.3	150	2.2	109
5	K02T	< 0.15	< 5.3	0.025	1.3
6	K02S	< 0.0035	< 0.12	0.003	0.15
7	K02R2.2	1.65	58	0.89	45
8	K02R2.4	2.4	84	1.6	81
9	K027	0.3	11	0.19	10.3
10	K03S	< 0.0035	< 0.12	< 0.0019	< 0.10
11	K03R	0.37	13	0.34	18
12	K04T	< 0.19	6.7	0.04	2.2
13	K04S	< 0.0035	< 0.12	0.0022	0.11
14	K04R4.2	0.56	20	0.42	21
15	K04R4.3	0.69	24	0.65	32
16	K07R	< 0.013	< 0.46	0.006	0.33
17	K12 2-R Scherben	0.48	17	0.72	36
18	K13 Hinter-futter	< 0.015	< 0.53	0.004	0.19

Table 3: RESULTS OF ALPHA-SPECTROMETRIC MEASUREMENTS

Nr.	sample	weight g	theoret values Bq/g			measured values Bq/g		
			Pu 239	U 238	U 235	Pu 239	error ± %	U 238
1	K01T	5.00	0.0060	0.600	0.170	0.0020	± 15	0.70
2	K01S	20.30	0.0002	0.020	0.006	sample lost		
3	K01R	2.20	0.0250	2.500	0.700	0.0200	±100	4.70
4	K01Z	0.60	0.1800	18.000	5.000	0.0120	± 16	22.00
5	K02T	4.10	0.0090	0.900	0.260	0.0070	± 14	0.20
6	K02S	20.00	0.0002	0.020	0.006	0.0004	± 33	0.023
7	K02R2.2	2.00	0.0430	4.200	1.200	0.0020	± 50	7.30
8	K02R2.4	2.00	0.0720	7.100	2.000	sample lost		
9	K02Z	4.00	0.0110	1.100	0.300	0.0150	± 12	2.80
10	K03S	20.00	0.0001	0.007	0.002	<0.0001	± 50	0.002
11	K03R	3.00	0.0180	1.800	0.500	0.0004	± 38	3.03
12	K04T	5.00	0.0070	0.700	0.200	0.0003	± 16	0.95
13	K04S	20.00	0.0002	0.020	0.006	0.0100	± 23	0.017
14	K04R4.2	4.00	0.0110	1.100	0.300	0.0005	± 28	1.70
15	K04R4.3	2.00	0.0430	4.200	1.200	<0.0001	± 23	7.10
16	K07R	20.30	0.0004	0.040	0.010	<0.0001	±100	0.057
17	K12 2-R	3.00	0.0200	1.900	0.550	0.0220	± 18	3.20
18	K13	20.00	-	-	-	<0.0001	±100	<0.01

R= slag ; S= metallic melt ; T and Z = filter dust
error = 1σ error of counting statistics

Table 4: U:Pu-Ratios [Bq/Bq]

No.	Sample	U:Pu-Ratio U235/Pu239	U:Pu-Ratio U238/Pu239
1	K01T	70:1	290:1
2	K01S	sample lost	sample lost
3	K01R	35:1	240:1
4	K01Z	420:1	1 850:1
5	K02T	37:1	29:1
6	K02S	15:1*	58:1*
7	K02R2.2	60:1	3650:1
8	K02R2.4	sample lost	sample lost
9	K02Z	20:1	186:1
10	K03S	*	*
11	K03R3.6	1 250:1	7 600:1
12	K04T	670:1	3 200:1
13	K04S	0.14:1*	2:1*
14	K04R4.2	600:1	3 400:1
15	K04R4.3	*	*
16	K07R	*	*
17	K12-2R Scherben	25:1	145:1
18	K13 Hinter-futter	*	*
theoretical values (from table 1)		28:1	100:1

*Pu content is in principle to low for analysing in this matrix

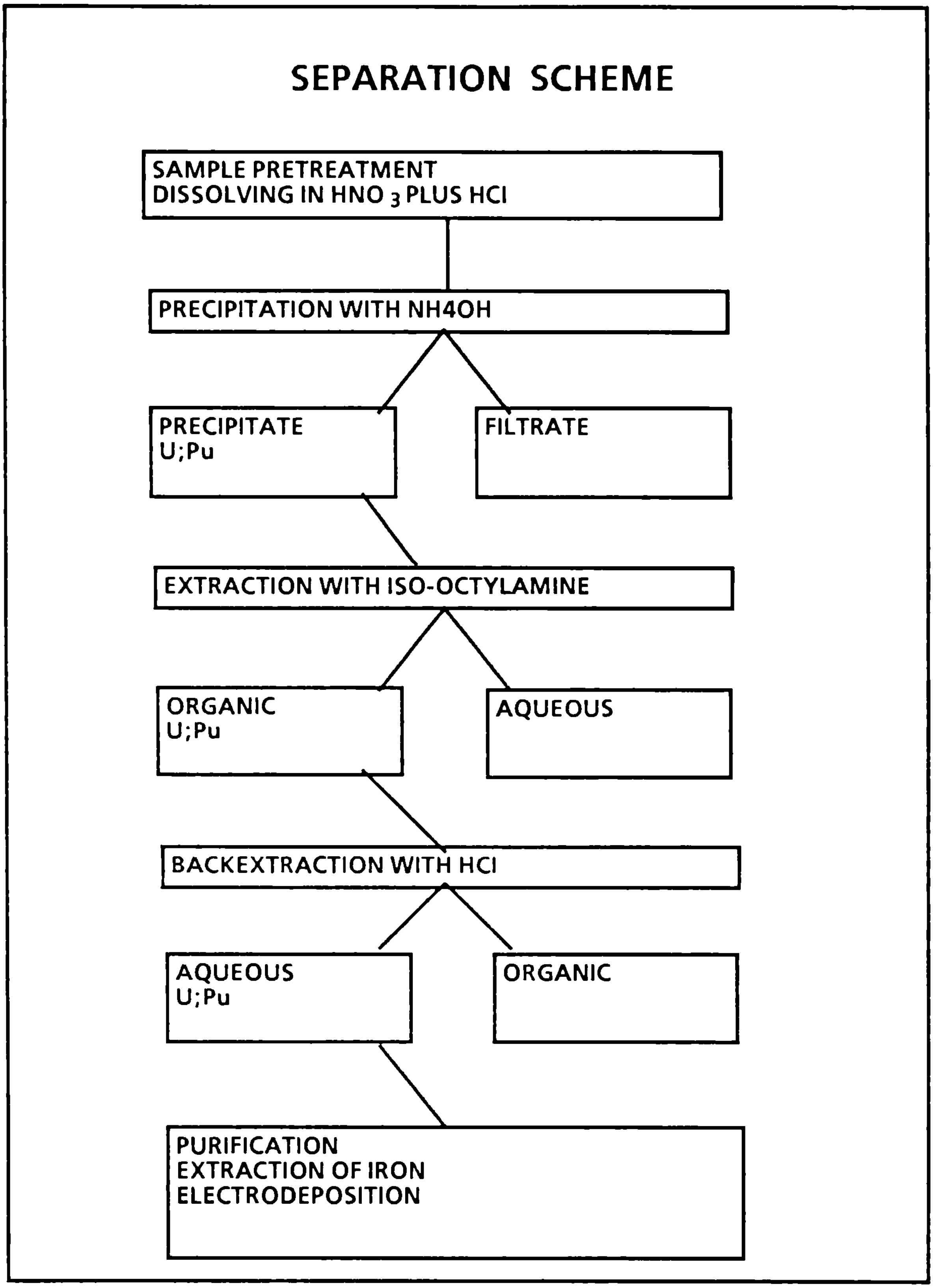


Fig. 1 Schematic overview of the separation scheme

8.13. DEMONSTRATION OF EXPLOSIVE DISMANTLING TECHNIQUES OF THE BIOLOGICAL SHIELD OF THE NIEDERAICHBACH NUCLEAR POWER PLANT (KKN)

Contractors: BE, Noell, Siemens-KWU
Contract No.: FI2D-0046
Work Period: November 1990 - June 1994
Coordinator: U FREUND, BE
Phone: 49/69/79 08 23 46 Fax: 49/69/790 880

A. OBJECTIVE AND SCOPE

This project aims at demonstrating explosive dismantling techniques on the biological shield of the nuclear power plant Niederaichbach (KKN), which was operated from 1972 to 1974 and is foreseen to be completely removed. The radioactive inventory of the shield is estimated in the order of 3.7 x 10^9 Bq (0.1 Ci). The level of activation is estimated to be in the order of 10 Bq/g, and the associated dose rates in the order of 10 μSv/h. Within this contract, blast peeling of the activated concrete from a 30° sector of the biological shield will be performed.

This technique will be applied as one of two main techniques (hydraulic hammer besides blast peeling) for the dismantling of the whole biological shield of KKN; for this, the licensing authorities have already given their agreement. This demonstration project will be conducted according to the guidelines of the ongoing total dismantling of KKN.
In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project. This will facilitate the application of this technology and acceptance from the safety point of view in future large-scale decommissioning operations.

The project is a follow-up of small-scale work on inactive samples performed jointly under contracts FI1D-0011 and FI1D-0012.

The work programme will be implemented jointly by three main contractors: Battelle Europe e.V./Frankfurt (BE), acting as coordinator, Noell/Würzburg (Noell) and Siemens/KWU (Siemens), as well as Stangenberg, Schnellenbach & Partner (SSP) as sub-contractor.

Further cooperation is foreseen with TÜV Bayern for the assessment of air filter systems.

B. WORK PROGRAMME

B.1. **Preparatory planning and design work for on-site equipment and regulatory requirements (BE, Noell)**
B.1.1. Layout of blasting patterns and of bore holes charging, according to the area of application (BE)
B.1.2. Design of blasting schemes according to the area of application (BE)
B.1.3. Definition of blasting area subcontainments for the retention of dust, including associated filter systems (Noell, BE)
B.2. **Demonstration blasting on the KKN shield by manual handling (BE, Noell)**
B.2.1. Site preparation for the installation of tools and measuring devices (BE, Noell)
B.2.2. Assessment and implementation of auxiliary techniques such as bore hole drilling, cutting of the reinforcement by hydraulic shears, use of a hydraulic ram (Noell)
B.2.3. Main operation and concrete removal, consisting of a sequence of about 10 individual blasts, including pre- and post-blast working (BE, Noell)

B.2.4. Assessment of blasting performance, with respect to predetermined criteria such as concrete removal rate, safety aspects, integrated doses and generation of secondary waste (BE, Noell)

B.3. Assessment of dust retention by industrial filter systems with respect to efficiency and safety of handling (Noell, BE)

B.4. Assessment of structural safety (BE, Noell)

B.4.1. Modelling of shield response to the blast transient loading (BE)

B.4.2. Modelling of building response by simple models and comparison to pre-evaluations at selected safety-relevant locations (BE)

B.4.3. Safety control for compliance with limiting values by test accompanying measurements (BE, Noell)

B.5. Final assessment of the blasting procedure (BE, Noell)

B.5.1. Technical feasibility and reliability (BE, Noell)

B.5.2. Compliance with safety regulations concerning radiation protection, radioactivity release, contamination/decontamination and structural safety (BE, Noell)

B.5.3. Comparison with other concrete dismantling techniques, such as sawing by diamond or wire saw, core drilling, possibly combined with sawing, high pressure water jet with abrasives (Noell, BE)

B.5.4. Setting up of guidelines and rules for general application of the bore hole blasting technique to other shield structure, and of cost estimates (BE, Noell)

B.6. Related investigations of general applicability to various types of nuclear power plants (BE, Siemens-KWU, SSP).

B.6.1. Building response by advanced modelling for the reactor building (BE, SSP)

B.6.2. Local damage, prediction of cracks and material failure (BE, Siemens)

B.6.3. Blast loading limits with regard to the integrity of light structures in close vicinity to the charge location (BE)

B.7. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2. and B.6.

C. PROGRESS OF WORK AND RESULTS

1. Preparatory planning and design (B.1.)

Due to the reduction of the free release limit to 0.37 Bq/g and on the basis of additional drill-core sample analysis the low level activated concrete region had to be significantly extended, see Fig. 1. The concrete dismantling efforts will therefore be larger than originally planned, extending into the distributor room above the shield level.

A new height-adjustable modular platform was designed, see Fig. 2. The working level height is 2 m, the maximum specific load is 2 to/m^2. This load bearing capacity allows heavy hydraulic equipment to be placed on the platform if necessary to remove cracked concrete from the wall.

2. State of licensing (B.1.)

Approval for the application of explosive concrete dismantling had been granted in 1986 in accordance with the German Atomic Energy Law. The detailed procedural steps were only recently approved by the county construction office. Any deviations from approved techniques need the agreement of the state office of technical licensing TÜV.

3. Assessment of non-explosive techniques and auxiliary tools (B.2.2)

The following techniques and tools have been tested with respect to their performance in partly high reinforced concrete and baryte concrete of the biological shield:

- core drilling
- full volume drilling
- wire saw
- hydraulic hammer
- diamond saw.

The results show that several tools should be readily available as auxiliary tools for the blasting method. High power full volume drilling will be used as main tool for bore hole preparation, core drilling will be applied only when strictly necessary. Wire saw cutting is omitted due to high wear in baryte and the problems associated with water rinsing. Diamond saw cutting is regarded of limited use since thick reinforcement bars are not cut and due to the high aerosol release.

4. Advanced modelling of the reactor building (B.6.1)

The general characteristics of the 3-dimensional finite element model were formulated. This model is non-rotational due to the off-axis position of the reactor in the containment building. Symmetry criteria are used where applicable. The element discretization is adjusted to an upper limit of 80 Hz for building vibrations.

Damping of oscillations will depend on overall construction status and on local material properties at critical points. Material damping is assumed to be 2 % for reinforced concrete. A modal damping matrix is set up using this value. High damping values at frequencies below about 5 Hz are predicted. This is in favour to meet the limits of the german standard for building excitation DIN 4150.

5. Local damage: prediction of concrete deformation (B.6.2)

The heavily reinforced concrete in blasting section 2 is modelled for calculation using the finite element code NONDYN. The loading in and around the bore hole is modelled using a finite difference HEMP code. For the concrete properties the dependence of deformation velocity and stress state are taken into account. Slip between concrete and rebar is regarded zero, i.e. ideal bonding is assumed up to a critical deformation. Various realistic properties as known from impact simulation calculations are included into the stress strain curve of the concrete. The present status of calculations is documented in Fig. 3. The deformation as shown is subjected to various crack and fragmentation criteria. These hypotheses will be subjected to selective proof by the real blast contour and fragment size distribution.

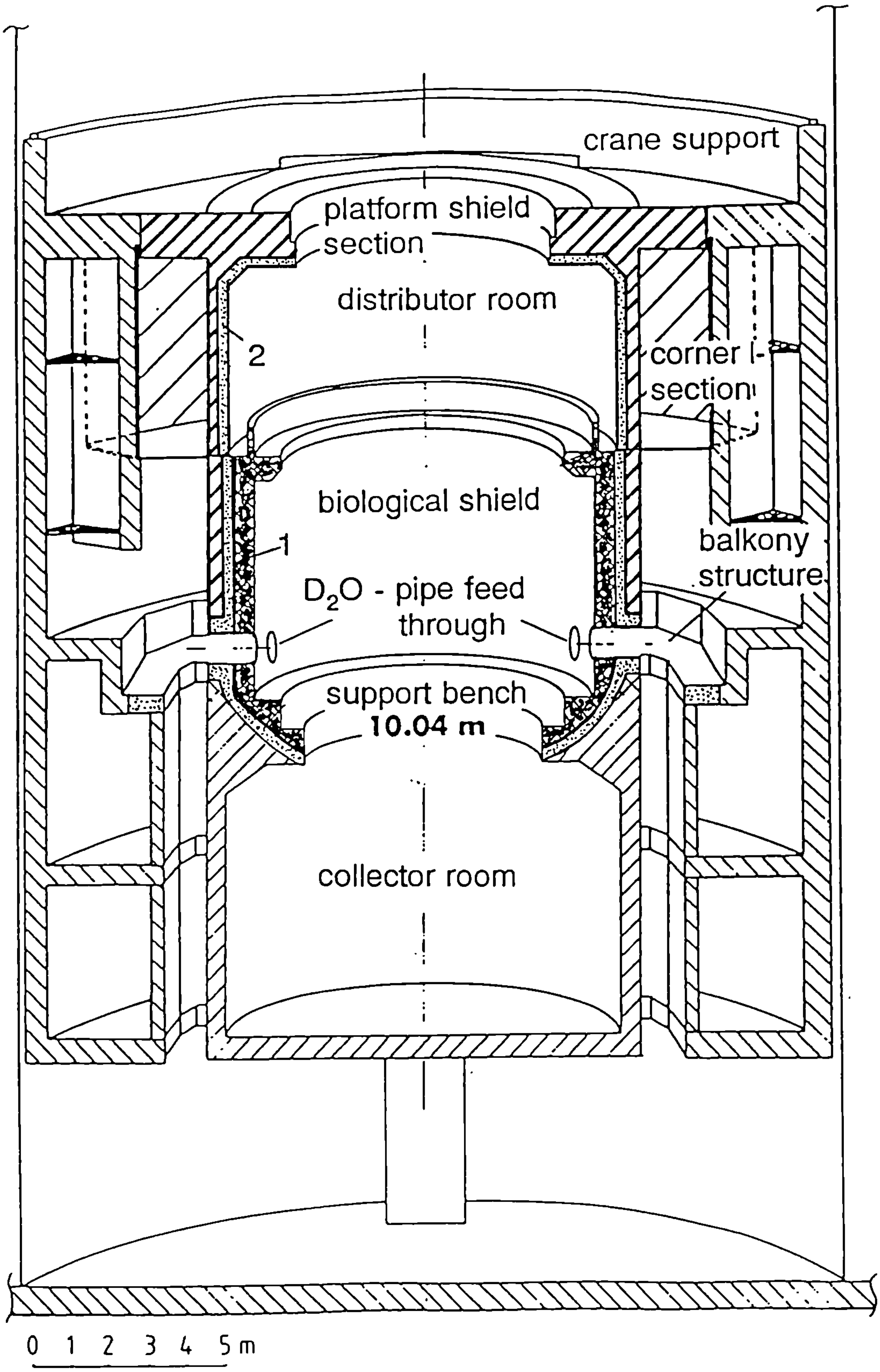


Figure 1: (1) activated material, original limit for free release

(2) additional activated material after reduction of limit for free release

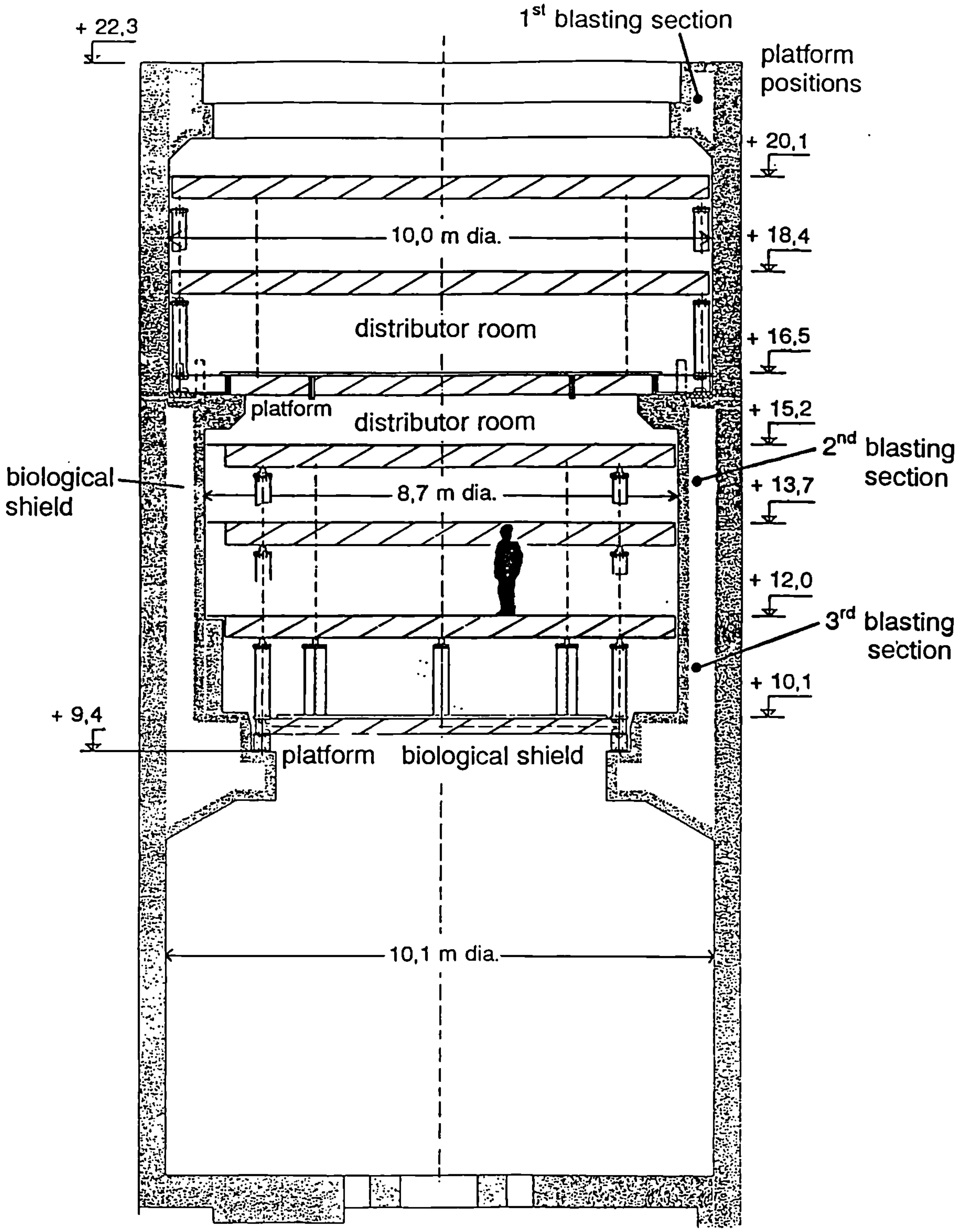


Fig. 2: Modular working platform positions and blasting sections

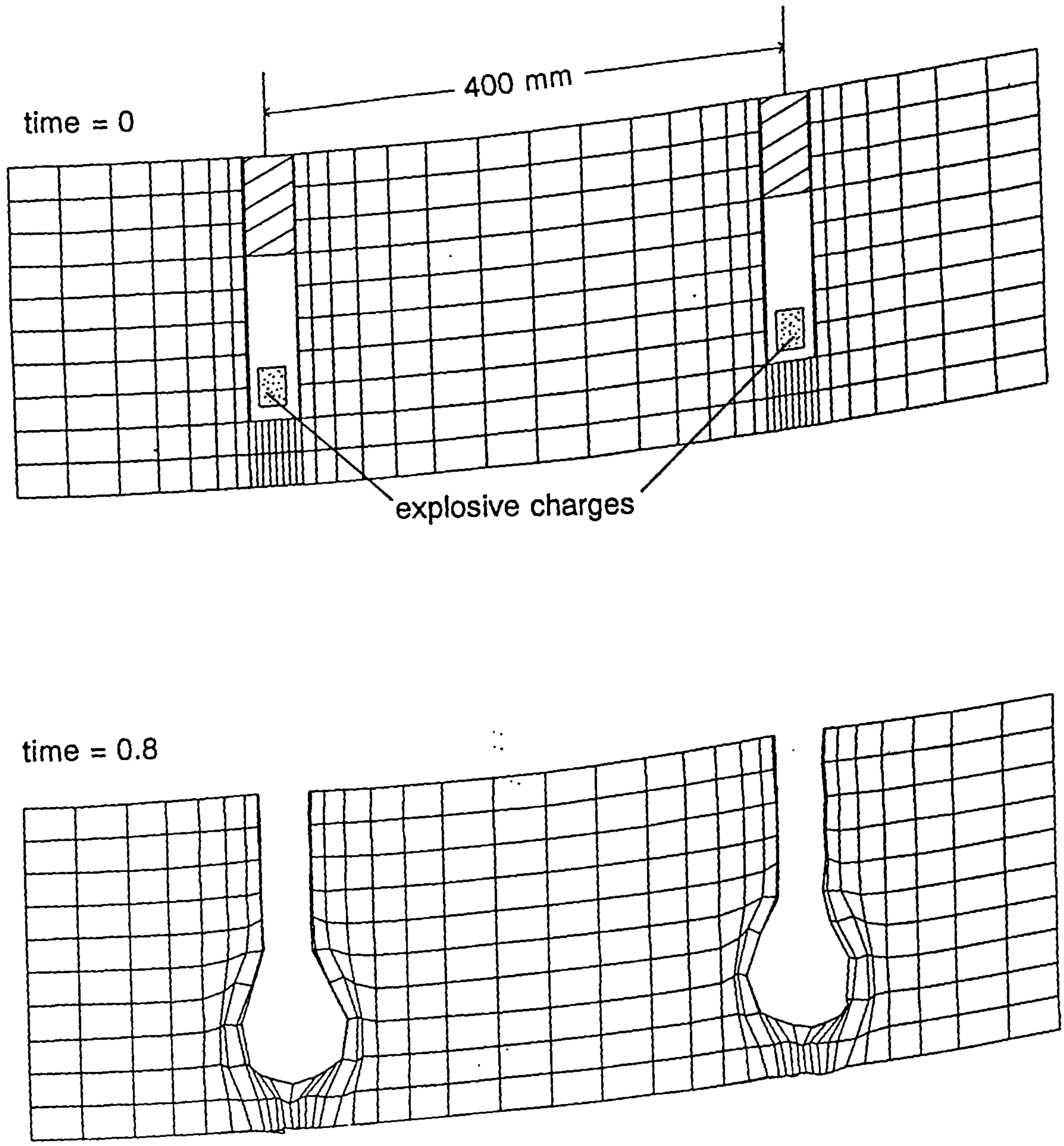


Fig. 3: Computer simulation of bore hole blasting in section 2
Concrete deformation without fracture criteria

8.14. DECOMMISSIONING OF THE B205 FUEL REPROCESSING PILOT PLANT

Contractors: BNF plc Sellafield
Contracts Nos.: FI2D-0050
Work Period: May 1991 - October 1993
Coordinator: S F CHALLINOR
Phone: 44-9467-75081 Fax: 44-9467-74070

A. OBJECTIVE AND SCOPE

The object of the underlying work programme is the dismantling of the B205 Fuel Reprocessing Pilot Plant. The plant was operated from 1957 to 1965. It processed uranium metal, totalling several hundred kilograms of fuel (typically 4,000 MWD/te). The product purification, solvent wash and sampling facilities were dismantled in the early 1970's leaving the MA cell nearly empty and the suite of HA cells untouched.

The aim of the project is to remove all of the contaminated structures from the laboratory (including the plinths that the cells are standing upon). It is ultimately intended to reuse the laboratory. This project will be used as a development project to demonstrate the techniques for dismantling this type of facility.

The contract covers dismantling work of the New Dissolver Cell, of the original HA Cell and of the Metal Cutting Cell. Information on the dismantling of the MA Cell (working period: August 1991 - March 1992) and of the New Primary Separation Cell (working period: July 1994 - September 1995) will be made available to the CEC.

This decommissioning project provides scope for testing a full range of techniques - for visual/radiometric inspection; remote handling; containment and shielding; decontamination of stainless steel, lead, concrete/brickwork; waste categorisation, segregation, monitoring and size reduction. Residual metal fuel, fuel cladding and historic dissolver liquor spillages provide authentic decontamination problems.

Estimated mean dose rates vary between $< 10^{-2}$ mSv/h for the laboratory and 200 mSv/h for the cutting cell with hot spots in the latter area of up to 470 mSv/h.

Lessons to be learnt include operational effectiveness, reliability, "user-friendliness", secondary waste arisings, manpower needs, dose-uptake, etc. The data sought is fundamental to the evaluation of future large-scale decommissioning projects and invaluable feedback into technique development programmes. It is planned to effect industrial-scale evaluation of decommissioning techniques, thereby providing data to assist planning, cost estimation and implementation of subsequent major projects.

The work programme will complement, and involve co-operation with the parallel Danish project at Risø National Laboratory.

B. WORK PROGRAMME

B.1. Preparatory work including assessment and/or backfitting and installation of auxiliary equipment and access routes.

B.1.1. Removal and decontamination of redundant service lines, shielded liquor transfer line, fume hoods, internal wall etc.
B.1.2. Refurbishment of the cell ventilation/filtration system,
B.1.3. Establishment of waste decontamination facilities,
B.1.4. Establishment of waste handling/export facilities for LLW, ILW and PCM waste,
B.1.5. Installation of new lifting beams to support manipulators and of other new equipment where required.

B.2. Dismantling of the New Dissolver Cell

B.2.1. Installation and commissioning of the manipulator

B.2.2. Removal of all supplementary shielding as far as possible and construction of a modular containment for working area and manipulator maintenance area.

B.2.3. Removal of concrete panels and dismantling of the inner stainless steel skin by using the manipulator

B.2.4. Removal of all process plant equipment and the remainder of the cladding,

B.2.5. Clean-up/scabbling of all inside faces of the cell using the manipulator,

B.2.6. Removal of the remaining structure.

B.3. Dismantling of the Original High Active Cell

B.3.1. Installation and commissioning of the manipulator

B.3.2. Removal of all supplementary shielding as far as possible and construction of reusable modular containment, backed by lead brick as necessary, to form a working area and manipulator

B.3.3. Installation of waste handling arrangements,

B.3.4. Use of the manipulator to breach the lead brick wall and gain access,

B.3.5. Retrieval of the existing hoist from the cell to the maintenance area to be removed manually,

B.3.6. Dismantling and removal of all process plant using the manipulator,

B.3.7. Clean-up/scabbling of inside faces using the manipulator,

B.3.8. Removal of remaining walls and plinth.

B.4. Dismantling of the Metal Cutting Cell

B.5. Generation of specific data

C PROGRESS OF WORK AND OBTAINED RESULTS

Summary of Main Issues

a This progress report summarises the work carried out from 1 January 1992 to 31 December 1992. The work carried out during this period covers work package B1; Preparatory Work.

b The demolition of the Medium Active Cell has been completed in preparation for the installation of remote handling equipment for the decommissioning of the Highly Active Cell.

c The demolition of the fumehoods has been completed in preparation for the installation of remote handling equipment for the decommissioning of the New Dissolver Cell.

d Decontamination techniques for lead and concrete have been assessed. Only 5% of lead arisings have been assigned as waste the remaining material has been sent for reuse within Sellafield.

e Waste volumes generated and dose uptake levels during demolition operations have fallen well within predicted values due to improved semi-remote demolition techniques.

f Dose uptake levels continued to fall well within predicted values.

Progress and results

The progress of this project was presented at the CEC seminar on Decommissioning Nuclear Facilities, Gundremmingen, June 1992.

A schematic diagram of the facility is shown in Figure 1.

B1.1 Removal and decontamination of redundant service lines, shielded liquor transfer line, fume hoods.

a Redundant service lines to the facility have been isolated and removed.

b The shielded liquor transfer line from the New Dissolver Cell to the New Primary Separation Cell has been isolated and removed. Lead shielding has been decontaminated, monitored and despatched for reuse at Sellafield.

c The fumehoods located in laboratory 190B at the rear of the New Dissolver Cell have been removed in order to provide access for the remote handling equipment required to dismantle and decontaminate the cell. The three double fumehoods and plinths were substantial structures comprising 6 inch thick concrete plinths and dividing walls. The fumehood bases and walls were lined with lead and required decontamination prior to demolition. Previous plinth demolition had utilised concrete crushing techniques and a waste volume of 46.4 M3 of rubble had been predicted from the demolition of the fumehoods. In order to reduce waste and dose uptake a concrete saw was used resulting in a reduction of waste of 55 %, all waste was consigned as Low Level Waste (LLW). Dose uptake of 255 microSv per man was received for semi remote demolition operations compared to predicted doses of 1.1 mSv per man using hands on concrete crushing techniques. The cost of fumehood demolition averaged £1770 per M3.

d The Medium Active Cell, Figure 1 Area C, demolition has been completed. This work was carried out in a low dose - high contamination area, alpha contamination was above 3000 CPS, work being carried out in pressurised suits. Dose uptake was in the region of 1 mSv per man per month for hands on work.

Areas of high residual fixed alpha contamination are being decontaminated in one of two ways:

1) Residual contamination on the plinth extending from the Highly Active Cell and a small area of adjacent floor was decontaminated during trials of the DECOHA process. This process employs a mixture of fluoroboric and hydrofluorosilicic acid at room temperature. The acids react with the concrete surface to produce calcium fluoride, silica and calcium metaborate while carbon dioxide is evolved. The surface of the material is destroyed but only to a depth of 0.1 to 1.0 mm producing a solid waste form ostensibly neutral. Mechanical abrasion assists in removing this layer.

   Residues collected by vacuuming during the trial were mixed with calcium hydroxide to adjust the pH to 11, followed by 50% by volume cement powder to produce a solid waste form.
   Monitoring the progress of the decontamination process during the trial was difficult due to the low gamma and beta dose and the wet nature of the process affecting alpha detection. Initial results indicate a DF of 9 during the trial but it is proposed to identify another site with higher beta

gamma dose rates on which to trial this process during early 1993. Monitoring of the solid residue showed a contact dose of 150 cps alpha.

2) Comparison tests are being carried out using conventional scabbling techniques in conjunction with decontamination agents (DSS 30). This work is proving to be time consuming due to the extremely hard nature of the concrete screed. Results from this work are due in early 1993.

Project data has been submitted to the CEC Section C Database through the BNFL project representative.

B1.2 Refurbishment of the cell ventilation/filtration system.

a Ventilation equipment serving the Medium Active Cell has been isolated and removed. Ventilation ducts serving the New Dissolver Cell are being rerouted to facilitate the installation of remote handling equipment.

B1.3 Establishment of waste decontamination facilities.

a An adjacent laboratory will be made available for waste decontamination operations and trials during decommissioning of the major project facilities.

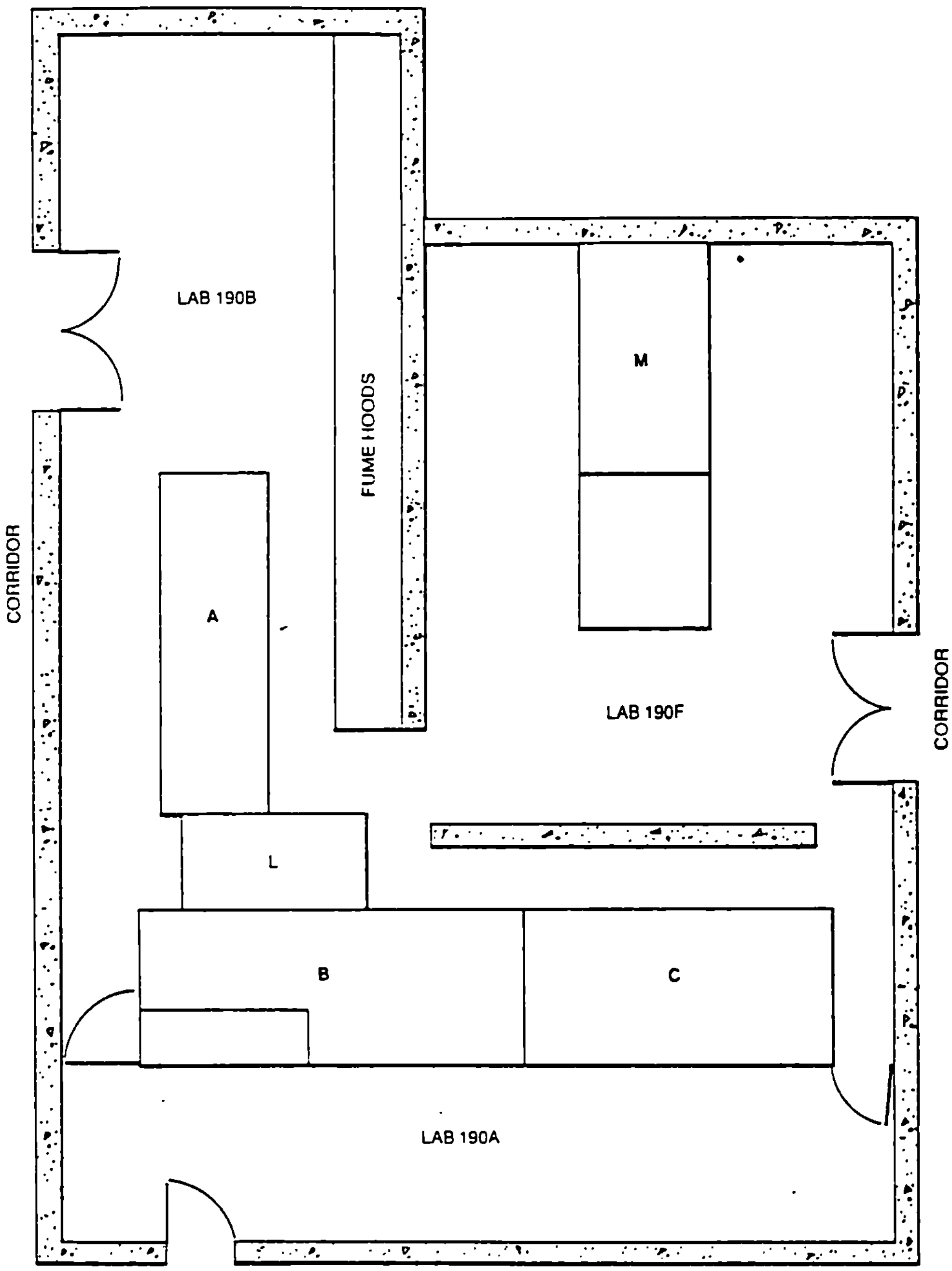
B1.4 Establishment of waste handling/export facilities.

a The detailed design, manufacture and installation of the waste export facilities are being carried out by external contractor. Additional preparation work has been identified to enable the construction of this facility which results in the requirement to remove a shielded storage compartment and its contents. This work is being programmed into work area B1.1 and will be completed prior to the existing equipment installation programme date.

B1.5 Installation of new lifting beams.

a Inspection of the installed lifting beams has indicated that this work package will not be required.

The future programme for phases B2-B5 is shown in Table 1. The project programme extends beyond the current CEC programme end date currently 1994, project data covered in the CEC contract generated for the remainder of the project will be made available.



KEY:

A	NEW DISSOLVER CELL
B	ORIGINAL HIGHLY ACTIVE CELL
C	MEDIUM ACTIVE CELL
L	METAL CUTTING CELL
M	NEW PRIMARY SEPARATION CELL

FIGURE 1 B205 Pilot Plant, Lab 190 - B229

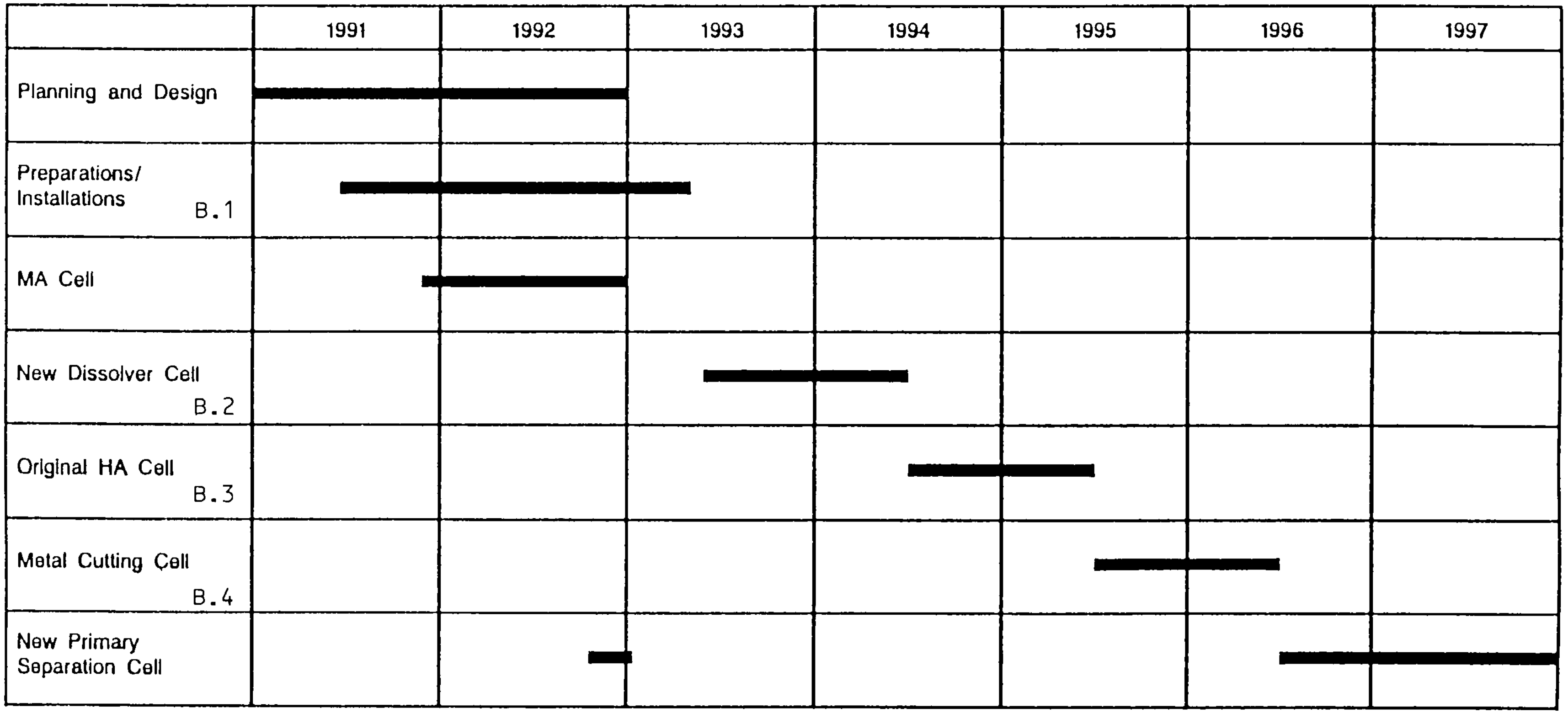


TABLE 1 B205 Pilot Plant Decommissioning Programme

8.15. LARGE-SCALE DEMONSTRATION OF DISMANTLING TECHNIQUES UNDER REALISTIC CONDITIONS ON THE LIDO BIOLOGICAL SHIELD

Contractors:	Taywood, AEA Winfrith	
Contracts Nos.:	FI2D-0052	
Work Period:	December 1991 - December 1993	
Coordinator:	C C FLEISCHER	
Phone:	44-81-575 45 82	Fax: 44-81-575 40 44

A. OBJECTIVE AND SCOPE

The aim of the project is to demonstrate/investigate the efficiencies of the explosive cutting and microwave techniques developed under preceding CEC programmes, on the full scale biological shield structure of the decommissioned LIDO enriched uranium thermal swimming pool reactor based at Harwell in U.K.

The reactor was operated from 1956 to 1972 and was used for shielding and nuclear physics experiments. The reactor core was made up from uranium/aluminium plates clad in aluminium. The core was movable through the water into any position on the centre plane of three large aluminium windows or beam holes for heavy shielding experiments to be set up outside. Initial analysis of the LIDO biological shield indicates surface dose rates of up to 10 $\mu Sv.h^{-1}$ and contamination levels of up to 50 $Bq.g^{-1}$. It is anticipated at this stage, that the main source of activity is Eu-152, Co-60 and Cs-137, but further radiological assessment will be carried out during the programme. It is proposed that up to 10 m^2 of surface will be removed by the dismantling techniques under consideration.

The present work will concentrate on the use of the two dismantling techniques to achieve concrete removal to meet both decontamination and dismantling requirements. This will offer direct comparisons between the techniques and lead to the assessment of their respective economic efficiencies. Effort will be directed at regulatory and safety aspects involved in the large-scale application of the techniques. Practical problems associated with the full-scale use of the techniques will be identified and realistic solutions obtained.

This work programme will be harmonised and carried out in close technical collaboration with work being co-ordinated by Battelle Institute on the application of explosive techniques for the decommissioning of the biological shield of the HWR KKN at Niederaichbach in Germany and the work being carried out by ENEA-CRE on the application of the microwave technique. A separate programme is also being carried out by KfK, aimed at developing remote manipulator deployment systems for the microwave unit.

B. WORK PROGRAMME

B.1. Preparatory work (AEA, Taywood)
B.1.1. Establishment of operational control and safety documentation (AEA and Taywood)
B.1.2. Preparation of documentation for the control of operations on the LIDO structure (AEA)
B.1.3. Development and construction of a sealed sub-containment on the biological shield including associated ventilation and filtration systems (AEA)
B.1.4. Pre-test structural assessment (Taywood)
B.1.5. Activity assessment and coring for material samples (AEA and Taywood)
B.2. Large-scale technique demonstrations on non-active zones (AEA, Taywood)
B.2.1. Implementation of explosive techniques (Taywood)
B.2.2. Implementation of Microwave technique (AEA)

B.3. <u>Large-scale technique demonstrations on active zones</u> (AEA, Taywood)
B.3.1. Implementation of explosive techniques (Taywood)
B.3.2. Implementation of microwave techniques (AEA)
B.3.3. Recontamination assessments (AEA, Taywood)
B.4. <u>Aerosol characterisation</u> (AEA)
B.5. <u>Structural assessment</u> (Taywood)
B.6. <u>Theoretical assessment</u> (Taywood)
B.7. <u>Techniques assessment and conclusions</u> (AEA and Taywood)
B.8. <u>Collaboration with related programmes</u>
B.9. <u>Generation of specific data</u>

C. Progress of work and results obtained

Summary of main issues

To comply with Site Licence conditions an operational control and full safety case for the dismantling trials has been prepared. Activity assessment on the biological shield walls has been completed. A combination of simple in-situ gamma-ray detection methods and core sampling were used. A sealed containment has been designed and constructed over the LIDO biological shield to withstand shock and overpressure resulting from explosive blast, make provision for particulate containment and to provide microwave shielding. A mobile filtration unit (MFU) has been installed (See Plate 1). The microwave generator has been refurbished and a wave guide and microwave head have been manufactured and the complete system tested to full power of 25KW.

Progress and results

1. Operational control and safety documentation(B.1.1 and B.1.2)

A full safety case has been prepared to comply with the station's Nuclear Site licence conditions. The following sections were included in the safety report:-

* preliminary safety analysis
* safety classification
* management of the project
* hazard assessment
* quality assurance
* calculations to support the containment design
* specification of the ventilation system
* method statement on installation and commissioning of the ventilation plant
* statements and instructions on the implementation of the trials
* waste management

2. Development and construction of a sealed sub-containment (B.1.3)

2.1 Sealed sub-containment

A specification for the design of a sealed containment has been prepared to withstand the shock waves and overpressure resulting from explosive blast, make provision for particulate containment, microwave shielding, waste posting and door interlocking. For the maximum proposed charge mass of 0.1kg and confined volume of $127m^3$, a loading ratio of $7.9x10^{-4}kg/m^3$ has been calculated, giving a steady state overpressure of 1.0KPa. This quasi-static pressure has been used to design the containment roof and access door apertures. The design was prepared using glass reinforced polyester (GRP) modular panels to cover the roof and access door. The complete containment structure installation has now been completed.

2.2 Ventilation and filtration system

The experimental programme aims to demonstrate the two techniques at full scale by removing sections of the bioshield surface. LIDO operation has led to activation products within the concrete (mainly Co 60, Eu 152 and Cs 137) and radiological data suggests surface dose rates up to 10μSv/h and contamination levels up to 50Bq/g. The proposed demonstration will generate potentially radioactive aerosols, and the provision of a filtered ventilated containment system will minimise the risk of a release to the local and general environment.

The ventilation flow and the inlet arrangements were designed to address two operating conditions:

* man entries into containment contaminated with radioactive concrete debris and dust.
* dismantling trials that will generate momentary over-pressures within the containment.

A ventilation rate of approximately 30 air changes was selected to maintain an average velocity of 1.0m/s through the entry port and thereby prevent any back diffusion of particles during man entry operations. The MFU to be used has a capacity of $6800m^3/h$ with two stages of HEPA filtration.

3. <u>Activity Assessment</u> (B.1.5)

3.1 Measurement philosophy

The essential requirement of this study was to devise a method whereby the gamma-ray activity of the bioshield could be determined by simple in-situ measurements. In order to simplify the investigation the approach taken was to monitor the gamma flux at selected points on the bioshield walls and subsequently to take cores at these points. These cores were then analysed for the activities of the identifiable gamma-ray emitting radioisotopes by conventional laboratory spectrometry techniques.

The result of this exercise would be a data set relating the activity of sections of the bioshield walls, as determined by the laboratory measurements on borehole cores, to the in-situ gamma-ray detector. The essential part of the work provides the data which forms the link between in-situ gamma flux measurements and how these relate to the specific activity of the corresponding regions of the bioshield.

3.2 In-situ measurements

The whole of the interior and floor of the bioshield (See Plate 2) was marked with a 50cm grid. Each line was denoted either with a letter or number to clearly and unambiguously identify each node of the grid.

A portable gamma-ray scintillator unit was used to monitor the whole of the inside of the bioshield. Above a height of 2.5 metres the readings were indistinguishable from background and detailed measurements of every node point were not deemed necessary. Below this height every node point was monitored and in addition, in the regions where 'hot spots' occurred additional measurement points -

every 25cm - were used. The minimum measured gamma-ray count-rate was 4 cps while the maximum was 1200 cps.

Spectra were taken at the sites. The main contributions were found to be due to Co60 and Eu152. Both are generated from the material forming the concrete with additional Co60 being formed in steel reinforcing bars etc.

3.3 Core sampling

Suitable sites for coring were identified on the basis of the gamma flux measurements. As wide a range as possible was chosen with surface gamma count-rates ranging between 4 and 1200 cps at the sites chosen. The cores were cut into 50mm lengths and alternate core lengths were supplied for activity analysis by gamma-ray spectrometry. The total activity results are tabulated on Table 1 together with the corresponding surface in-situ gamma count rate measurements at the selected sites. There is a degree of correlation between the surface gamma count rate measured on the walls of the bioshield and the specific activity of the surface cores.

4. Microwave Generator

The microwave generator to be used in the trials has been refurbished and a new wave guide, control system and microwave head have been manufactured and the complete system has been tested to full power (25KW). The microwave unit has now been delivered to Harwell site where it is being installed with the associated services and is subsequently to be tested in-situ.

5. Generation of Specific Data

Work to date has concentrated on the establishment of operational control and safety documentation for the dismantling trials. The times taken and manpower requirements for the specific tasks undertaken to date have been made available to be included in the data base being prepared for specific data. Given below is a summary of current estimates for the preparatory work carried out to date.

Title of Working Step	Duration (Days)	Manpower (man hours)
Operational Control and Safety Documentation	155	3500
Development and Design of Sealed Sub-containment	60	1350

TABLE 1
Total Activity

Core	Surface Flux	Core Depth mm 0-50	50-100	100-150	150-200	200-250	250-300	300-350	350-400	400-450
		Activity Bq/kg								
c	700	39695	11302	7864	1317	733	453			
d	100	3135	6167	1268	964	732	537	245		
e	250	9700	6315	4852	1960	1102	539	264		
f	450	47320	25273	13700	6446	1405	1071	604	326	273
g	1100	99581	27237	11448	5610	1893	594	201	30	
h	60	1181	674	521	353	77	44			
i	45	1190	653	642	175	400	75			
j	12	31	39	52	54	47	54	38		
l	70	0	0	0	0	0	0	0		
m	25	0	0	0	0	0	0	0		
n	125	2781	3976	2762	2518	2145				

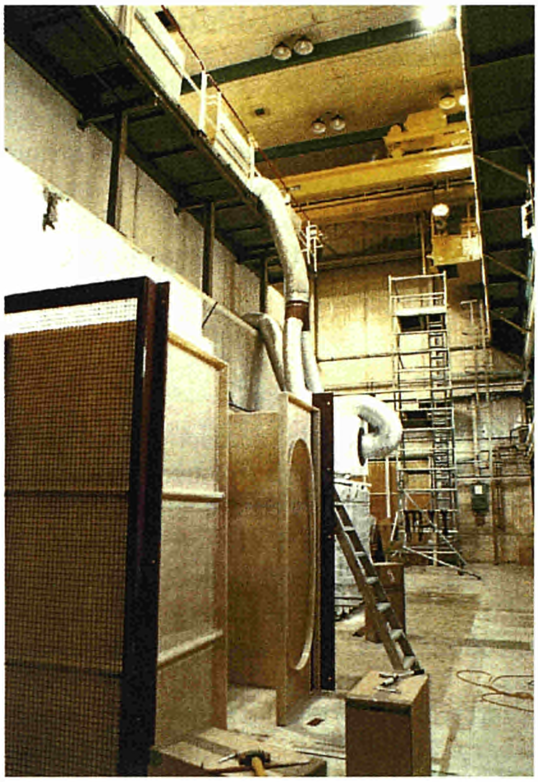


Plate 1. Installation of sub-containment and ventilation and filtration system.

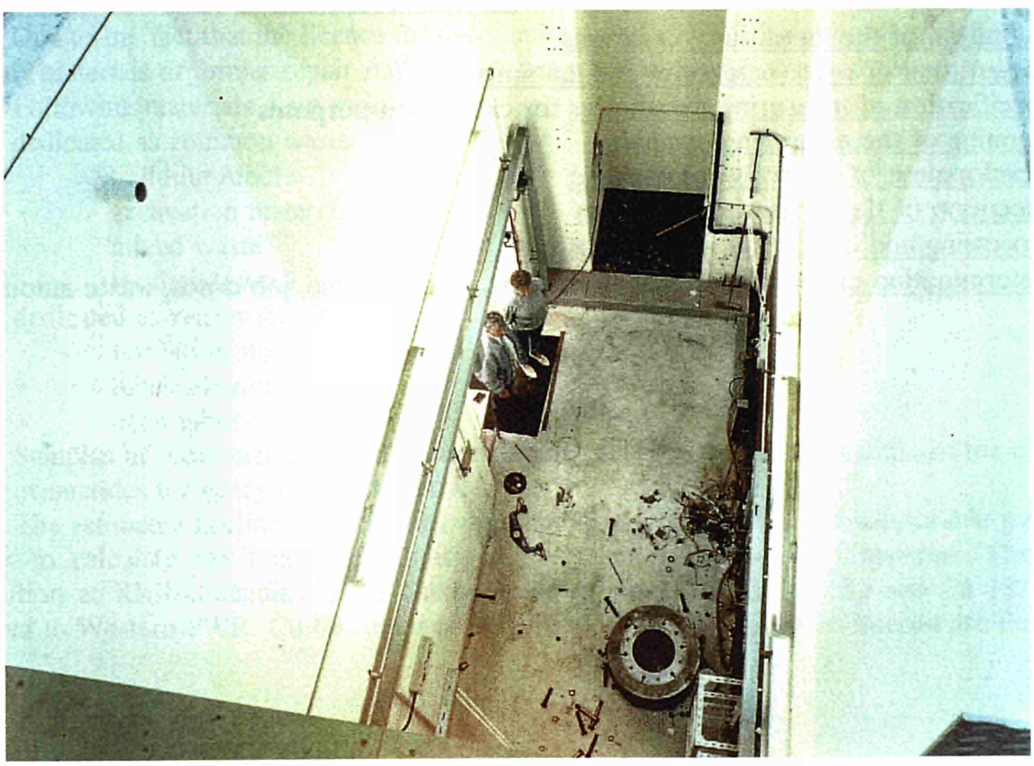


Plate 2. Interior of the Lido bioshield

8.16. FURTHER DEVELOPMENT AND OPERATION OF AN AUTOMATED LARGE-SCALE RADIOACTIVITY MEASUREMENT FACILITY FOR LOW-LEVEL DECOMMISSIONING WASTE

Contractors:	NIS, KKWR
Contract No.:	FI2D-0063
Work Period:	November 1991 - December 1992
Coordinator:	I AULER, NIS, Hanau
	Phone: 49/6181/1094 73 Fax: 49/6181/12 00 33

A. OBJECTIVE AND SCOPE

For the application of an existing mobile radioactivity measurement facility (RMF) for the decommissioning of WWER nuclear power plants (former GDR) new measuring - and calibration procedures as well as new software, to simplify data evaluation, should be developed.

In extended tests, the success of the RMF modifications will be demonstrated for a relatively large spectrum of radioactive wastes produced during operation or occurring during decommissioning of the 75 MW WWER of Rheinsberg. Investigations and trial measurements should indicate to what extent money could be saved, avoiding waste disposal by means of the RMF (classification of wastes).

NIS Ingenieurgesellschaft will elaborate the facility requirements in cooperation with NIS Rheinsberg and KKW Rheinsberg. The project is a continuation of the research work performed under the previous CEC contract No. FI1D-0062.

B. WORK PROGRAMME

B.1. Definition of the measuring conditions
B.2. Registration of waste categories and amounts
B.3. Specification of measuring parameters for calibration purposes
B.4. Planning of the measuring campaign
B.5. Development of the software package
B.6. Execution of the measuring campaign
B.7. Processing and evaluation of the measuring results
B.8. Determination of specific data such as: costs, working time, job doses, waste amounts etc.

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

The nuclear power plant Rheinsberg (KKR) was closed down in 1990. Until that time there did not exist any procedures for the unrestricted release of materials out of the controlled area. The type of KKR wastes and its nuclide composition are different compared to Western light water reactors. For further development of an existing Release Measurement Facility (RMF) different types of materials out of the controlled area were measured with the RMF. The measuring at KKR in autumn 1992 lasted about two monthes. Afterwards 34 % of about 100 Mg measured material in total could be declared as releasable, but some types of wastes could be released up to 90 %. The analysis of the measuring data will be continued.

Progress and results

1. Definition of the Measuring Conditions (B.1)

Relevant data of NPP Rheinsberg are presented in Table I. During operation no procedure for the release of materials out of the controlled area of KKR for unrestricted reuse or disposal existed. Most of the generated waste was transported directly to the final repository at Morsleben. During the temporary close down of the repository at Morsleben all newly generated waste was stored at KKR-site.

The kind of waste and its nuclide composition is different compared to Western light water reactors due to:

- layout of systems
- material composition
- water chemistry
- frequent decontamination of primary system
- handling, documentation and classification of waste.

2. Registration of Waste Categories and Amounts (B.2)

Due to the fact that the licence for decommissioning of KKR is expected not before summer 1993 only materials of former repair and maintenance work were available for release measurements.

Following materials were measured:

- dedicated as common waste
    - building rubble
    - excavation material (soil)
    - mixed waste
    - insulation wool
- dedicated as retrievable materials
    - insulation steel sheets
    - lubricating oil
    - neon tubes

Samples of these materials had to premeasured nuclid specifically to estimate the different portion of nuclides for every type of material.

The estimated nuclide compositions were applied on the basis of the measurable gamma nuclides to calculate the total radioactivity of every measured batch of material. The nuclide composition at KKR-contamination is characterized of more Fe 55 (42 %) and Cs 137 (17 %) compared to Western PWR. Co 60 comes to only 30 %. Further nuclides of interest are Eu 152 and Pu 241.

3. Specification measuring parameters for calibration purposes (B.3)

The release measurements were made with the NIS-Release Measurement Facility (RMF) /1/ which was reconstructed into a 20' shipping container for measuring campaigns at different sites (Fig. 1).

The calibration of the RMF is made related to type and amount of mass of the materials to be measured. For that purpose dummies definitely free of artificial radioactivity were used and furnished with reference nuclide sources. The calibration factors are related to the measuring efficiency and the nuclide composition estimated in earlier investigations. Due to this calibration factors there could be established a clear relation between gamma nuclides directly measured with the RMF and the radioactivity of all nuclides.

4. Planning of the measuring campaign (B.4)

For the execution of the measuring campaign procedures for commission, calibration and execution of measurements were developed and specially suited for the different type of materials. All procedures were presented and clarified with the inspection agency in detail.

Due to the requisition of the inspection agency separate limit values were established for waste and for retrievable materials (Table II).

Materials to be measured were subjected to a first rough contamination check during their transfer out of the controlled area, to prevent higher contaminated materials from entering the RMF.

After this check the materials were made ready in size and mass for the execution of the release measurements.

5. Development of the software package (B.5)

New software for data collection, processing and display has been developed for:

- easier and faster calibration procedures for different materials
- consideration of natural radioactivity of the material to be measured
- calculation of the Co 60-equivalent to consider other gamma-nuclides and especially taking into account their different effects of selfshielding
- calculation of the total radioactivity

  The RMF measures directly the gros gamma radioactivity only. Radionuclides which do not emit gamma radiation (Fe 55, Ni 63, Pu 241) or emit gamma radiation with very low energy only (Am 241), are considered by an extrapolation factor. The extrapolation factor is determined on representative samples of different materials in a radiolaboratory prior to the release measurements.
- coloured graphical display for local activity concentrations of the material measured.

6. Execution of the measuring campaign (B.6)

After the inspection agency has given green light for the execution of the release measurements with the RMF the measuring campaign was started on October 20th and finished on December 11th, 1992. During this period about 106 Mg of different materials were measured in about 560 cycles. Mass and result of the release measurements are shown in Table III on detail.

7. Processing and evaluation of the measuring results (B.7)

The measuring data was collected for further data processing in dBase datafiles. 36 Mg or 34 % of the materials measured could be classified for common disposal or as retrievable material without relevant restrictions. During measuring campains at other nuclear facilities the portion of releasable material had been essentially higher (about 90 %). The lower portion of releasable material, especially for the mixed waste (4 %), resulting from conservative assumption taken due to the inhomogenity and uncertain determination of Pu 241-contents in this type of wastes. The content of Pu 241 in contamination with traces of irradiated fuel is generally high, unfortunatly this beta emitting nuclide is difficult to measure due to its very low beta energy. With lower limit of detection for Pu 241 in laboratory measurements (< < 140 Bq/g) the uncertainties in it's determination could be decreased and the amount of releasable mass of mixed waste could be increased. The analysis and assessment of data will be continued.

References

/1/ I. Auler, F. Helk, E. Neukäter, U. Zimmermann
Meßverfahren zum Nachweis der Unterschreitung niedriger Grenzwerte für große freizugebende Massen aus dem Kontrollbereich.
EUR 13438 DE, 1991

Table I: Data of NPP Rheinsberg

Typ	VVER
Commission	1966
Close down	1990
Application for licence of decommissioning	1992
Thermal power	265 MW
Electrical power (projected)	70 MW
Primary pressure	10 MPa
Reactor outlet temperature	265°C
Uranium content	18 t
Averaged enrichment	2 %
Number of fuel elements	132
Number of fuel rods per element	90
Averaged burn-up	10 MWd/t
Net power generation	8127 GWh
Number of fuel cycles	22
Availability	61,1 %

Table II: KKR Limit values for release of materials out of the controlled area [4])

Limit values	Wastes	Reusable Materials	
		unrestricted	common melting
mass specific	$10^{-4} * \frac{FG^{1)}\ [Bq]}{g}$	0,1 Bq/g	1,0 Bq/g
averaged mass	≤ 100 kg		
surface specific[3)]	0,05 Bq/cm²: α-nuclides[2)] 0,5 Bq/cm²: other radionuclides 5,0 Bq/cm²: Fe 55, Ni 63, H 3		
averaged area	≤ 1 m²		

1) FG: absolute activity limit value from German Protection Ordinance in Bq: i.e. for 100 % Co 60: $5 \cdot 10^4$ Bq hence it follows: 5 Bq/g Co 60
2) for radionuclides with FG ≤ $5 * 10^3$ Bq
3) no concern for materials with undefined surface (i.e.: sand, concrete, liquids)
4) for defined materials and defined procedures (preliminary)

Table III: Measured materials

No.	Material	Mass measured	Mass releasable %
1	building rubble	7,9	86
2	excavation material (soil)	27,7	94
3	mixed waste	66,3	4
4	insulation wool	0,7	0
5	insulation steel sheets	0,9	81[1)]
6	lubrication oil	2,0	17[2)]
7	neontubes	0,1	0
	Σ	105,6	34

1) releasable for melting: < 1 Bq/g
2) releasable for combustion: < 1 Bq/g

Fig. 1: Radioactivity Measuring Container

8.17. DEVELOPMENT, MANUFACTURING, COMMISSIONING AND TESTING OF AN AUTOMATED DEVICE FOR THE PROJECTION OF CHEMICAL GELS AND ITS APPLICATION TO G2/G3 REACTOR PIPES

Contractors: CEA-Valrhô
Contract No.: FI2D-0064
Work Period: December 1991 - June 1994
Coordinator: J R COSTES, CEA-Valrhô
Phone: 33/66 79 63 13 Fax: 33/66 79 64 32

A. OBJECTIVE AND SCOPE

This project is aimed at industrial-scale testing of an automatic machine designed to spray gel or foam compounds inside pipes with a contamination level of up to 200 Bq/g due mainly to Co-60. The envisaged decontamination performance is about 8 metric tons a week to less than 1 Bq/g; the scrap metal will then be melted down on the site for unrestricted release.

The innovative nature of this project lies in the automation of all steps of the decontamination process.

This contract will demonstrate that 450 tons of steel can be released for unrestricted use after automatic decontamination and provide a cost-effective alternative to any manual decontamination process.

The contractual work involves the internal decontamination of 450 tons of ordinary carbon steel pipes with diameters ranging from 0.5 m to 1.6 m, using an automatic decontaminant gel spraying machine. The dose rate is about 0.3 mGy/h at the pipe surface contact.
These pipes belong to the graphite-gas reactors G2 and G3, which are presently under stage 2 decommissioning.

The contractor is coordinating all decommissioning operations concerning the G2 and G3 gas-cooled reactors at Marcoule. The present state of knowledge is based primarily on CEC research contract FI1D-0003, covering small-scale manual gel spraying. The process will begin by spraying a highly alkaline gel or foam, followed by pressurized water rinsing. This will eliminate all greasy or oily deposits accumulated during 22 years of reactor operation.

In some cases, it may be necessary to follow this procedure by spraying with strongly acid gels or foams, followed by rinsing; this step may be repeated in exceptional circumstances.

B. WORK PROGRAMME

B.1. Supply and adaptation of equipment
B.2. Implementation and commissioning testing of the device
B.3. Preparatory work and preliminary active decontamination tests
B.4. Industrial decontamination of 450 tons of contaminated pipes
B.5. Analysis of results
B.6. Generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The construction of SAMIT ("*Système d'Application de Mousses à l'Intérieur des Tuyauteries*"), has been completed and the automatic cycle, as well as its various sequences, are operational. The matter is now to test the process before proceeding to the stage of industrial application in March 1993.

Progress and results

1. Supply and adaptation of the equipment (B.1.)

The spray machine which has been designed for surface treatment inside pipework comprises the following (s. Fig. 1):

- a chassis
- a beam supported by the chassis and with a front bearing mounted at the end
- a mobile carriage running along the length of the beam between the front end and the other end
- a cylindrical axle supported by front and rear bearings
- means of rotating the axle and fixing it to the carriage
- means for moving the carriage along the beam
- a decontamination head mounted on the end of the axle
- a containment bonnet mounted on the end of the beam so as to close off and seal the pipe to be decontaminated.

The intended decontamination will be achieved by carrying out the following operations in the following order:

- spraying a uniform coat of basic foam over the whole interior surface of the pipework, four times (once every ten minutes)
- rinsing with water at 150 bar after 40 minutes
- spraying a coat of acidic foam three times (once every 10 minutes)
- rinsing with water at 150 bar after 30 minutes.

2. Implementation and commissioning testing of the device (B.2.)

All the movements defined in Chapter B.1. are implemented from a control panel connected to a Télémécanique TSX 47-20 process micro-computer. Some manual orders can be carried out through the computer by functional groups, and were tested:

- up and down movements of the chassis
- forward and backward beam movements
- 360° beam rotation to the left and to the right
- basic foam projection by opening EV1 electrovalve and switching on high pressure pump 1 and the effluent pump
- acid foam projection by opening the EV2 and EV3 electrovalves and switching on the acid high pressure and the effluent pump
- water rinsing by switching on high pressure pump 2 and the effluent pump.

It should be noted that the effluent pump is running as soon as one of the two high pressure pumps are on, and that it stops a little while after these pumps are switched off.

Before going through an automatic cycle, the machine has to be aligned in front of the pipe to be decontaminated. Then the computer asks the operating questions:

How many basic cycles?

How many rinsing cycles? etc.

In case of emergency, a push-button trips all running functions.

The reagents are all liquids, and bottled in 50 l plastic containers:

- 50% soda solution
- 85% phosphoric acid
- 92% sulphuric acid
- detergent.

By the mean of measuring pumps, they are pumped directly from their container into the main water line. Initially, the priming of these pumps was difficult, but now the entire system is operational.

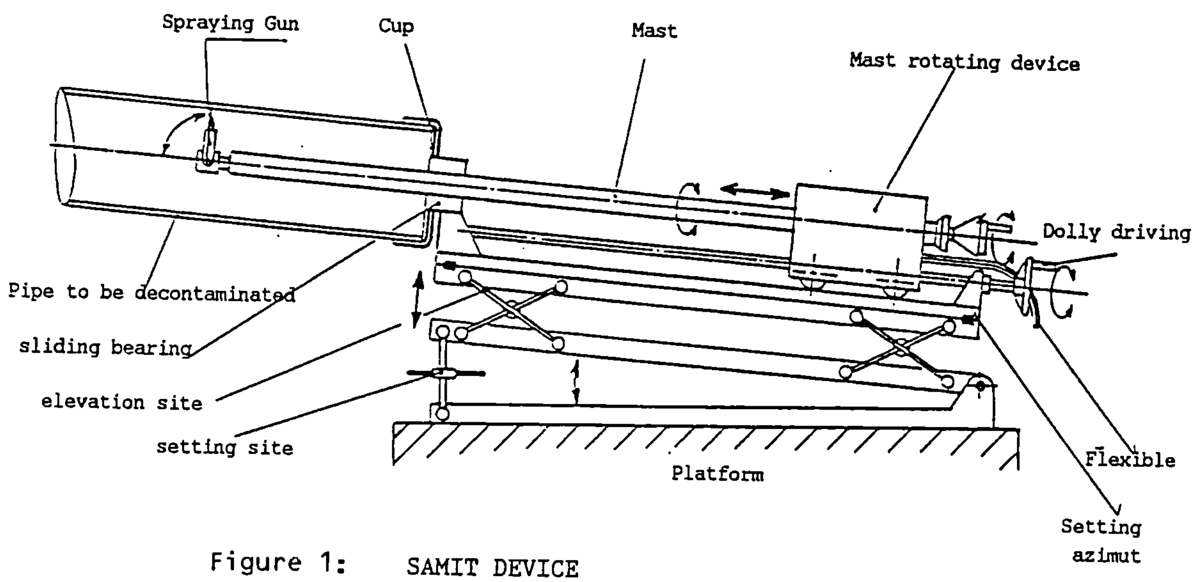


Figure 1: SAMIT DEVICE

8.18. DEVELOPMENT AND MANUFACTURING OF A FACILITY FOR THE DECONTAMINATION BY ELECTRO-ETCHING AND APPLICATION TO ALPHA RADIOACTIVE WASTE FROM THE "RM2" INSTALLATION

Contractors: CEA-Valrhô
Contract No.: FI2D-0065
Work Period: December 1991 - June 1994
Coordinator: J R COSTES, CEA-Valrhô
Phone: 33/66 79 63 13 Fax: 33/66 79 64 32

A. OBJECTIVES AND SCOPE

The "RM2" installation is a disaffected laboratory for post-irradiation fuel examination operated from 1967 to 1982 at Fontenay-aux-Roses. This project is aimed at the industrial-scale testing of a new drum-type nitric acid electro-etching process to be used for RM2 waste decontamination.

The innovative nature of this project lies in the decategorization of Pu-bearing waste. At present in France, only surface storage sites are available for nuclear waste. These sites can only accept beta-gamma waste with little or no alpha contamination.

The prevailing radiological conditions are as follows:
- specific alpha contamination: 3.7×10^3 to 3.7×10^5 Bq/g;
- dose rates: 10 mGy/h to 1 Gy/h.

This industrial decategorization process for alpha-bearing waste can be applied wherever necessary in the European Community (reactor decommissioning to Stage 3, decommissioning of research laboratories or fuel reprocessing plants). The end result will be a net reduction of waste volumes for underground waste storage.

The radioactive waste will be removed from the RM2 installation at a rate of 700 kg per month beginning in December 1991 and conditioned in special 200 l stainless steel drums which will be placed in waste drums and transferred to the Saclay waste treatment centre in type B casks for decontamination. Before the transfer to Saclay, the drum activity will be carefully measured by an automatic neutron counter. After decontamination, the waste will be reconditioned for transfer to the ANDRA facility.

The potential benefit is that, by thorough decontamination, 7 tons of metal waste provided for underground storage could be accepted for definitive surface storage. This would considerably reduce the total cost of this operation.

B. WORK PROGRAMME

B.1. Waste characterisation and conditioning
B.2. Adaptation of the electro-etching decontamination process
B.2.1. Design and manufacture of the decontamination device
B.2.2. Implementation and commissioning testing of the device
B.3. Preliminary decontamination tests on radioactive components
B.4. Industrial decontamination of 7 tons of radioactive waste
B.5. Analysis of results
B.6. Generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

When taken over by UDIN, there was no fuel in the cells. The various process equipment used during the operation of the laboratory were still in place: remote manipulators, microscopes, radiography benches, etc.

The beta and gamma radiation levels in the cells can reach several grays per hour. There is a large amount of surface contamination:

- by alpha emitter: plutonium
- by beta and gamma emitter: FP (Cs-137) or AP (Co-60).

Radiation and contamination levels vary significantly from one cell to another.

Initially, operations were carried out by remote manipulation controlled from in front of the cells. Subsequently, the reduced ambient radiation levels allowed direct entry by the operatives.

Cells 8, 9, 10 and 11, which make up the "small line", are now empty and have been partially cleaned up (there remain some points of stable surface contamination). After cleansing, the ambient radiation in the "small line" cells amounts to about 0.5 mGy/h^{-1} (apart from some hot spots - sumps ... which reach several hundred $mGy/^{-1}$).

Residual contamination of surfaces does not exceed 3,000 Bq/cm^2 for beta and gamma emitters and 1,200 Bq/cm^2 for alpha emitters (IPAB check 7.1/SIA and S1B2 sensors).

Progress and results

Waste characterization and conditioning (B.1.)

1. Remote operation

   Each cell is equipped with two remote manipulators which can be controlled by the operators from in front of the cell. The remote manipulators are used for cutting up, packaging and removal of the various laboratory equipment (welding bench, press, etc.) as well as the internal dismantling of furnishing (work surfaces). They will also be used in decontamination operations such as with dry ice.

2. Work inside cells

   Once the ambient radiation level in the cell is low enough (determined by IFF 103 measurement from the front face of the cell), the operatives can enter directly into the cell (equipped with adequate individual protection: ventilated suits with air supply, masks etc.).

3. Wastes

3.1. Determination of activity in wastes

- **Gamma spectrometry**

  This consists of a Ge-HP (Germanium Hyper-Pure) of 40% efficiency associated with a multi-channel analyser (4096) - in Gaussin mode, release of stack of coils. The cryostat/detector arrangement is mounted on a mobile trolley on rails, which enables measurements to be taken in various places (taking calibrations into account). The detector is protected from ambient radiation by the presence of a low activity lead envelope of 10 cm thickness.

  Collimation is used for the measurement of relatively irradiant wastes (2 mGy/h^{-1}). The package for measurement is rotated on the stand facing the detector. Counting lasts for 900 seconds.

  The nuclides identified are shown on a screen in the form of an energy spectrum.

- **Determination of alpha activity**

  This is calculated from the activity measured by gamma spectrometry of the Europium 154. At the outset, a mean spectrum was defined after carrying out a series of smear tests in the various cells of the installation. This was done in order to identify the radioelement present and determine their percentages. Eu-154, present in the spectrum, was chosen as the "tracer" due to its constancy and to its beta/gamma activity relationship with Pu-239.

- **Neutron measurement**
  Neutron emitted from the package being measured are slowed down by a cadmium shield surrounding the detectors, then thermalized by the polyetylene before being counted. There is no discrimination in this measurement (alpha -> n). The measurement enables the spontaneous fissions emitted by the Cm-244 to be shown.

3.2. Classification of wastes

Wastes resulting from the cleansing and dismantling operations are split up into two categories:

- **Wastes which already comply with the "ANDRA" storage specifications** (< 3.7 GBq/t for alpha emitters - this being the most restrictive value);
- **Bulky and exclusively metallic wastes which do not comply with the "ANDRA" specifications** are sent to the CEA Saclay where they are taken over by the SPR/SIDS department (Radiation Protection Service/Storage Decontamination Section) in order to be made acceptable for the "ANDRA" storage norms.

The initial agreement between UDIN and SPR/SIDS put the number of containers to be processed at 76. The mass of wastes was estimated at 13.5 tons. Up to now, the actual shipments to Saclay amount to 12 stainless steel containers with a total mass of 2 tons (taken in 3 tips - each comprising 4 containers).

The low volume of waste shipment compared with the provisional planning (about one shipment of 4 containers per month, is due to various technical problems).

4. Design and manufacture of the decontamination device (B.2.1.)

The device was made in Villeurbanne, near Lyon. It is a copy of industrial electro-deposit techniques for bolts, for instance. Some minor improvements were implemented:

- most of copper electric current bars were replaced by stainless steel bars,
- the main drum metal is titanium,
- some control devices were added,
- of course, the electric current direction was reversed.

The device has treated 3 x 200 kg of waste to-date.

8.19. DEVELOPMENT OF A ROBOTIC SYSTEM (TRT) FOR THE REMOVAL OF TUBES FROM A LATINA STEAM GENERATOR, WITH SUBSEQUENT MELTING AND RADIOLOGICAL CHARACTERIZATION

Contractors:	Ansaldo Spa, Siempelkamp
Contract No.:	FI2D-0066
Work Period:	November 1991 - December 1993
Coordinator:	Ing. M. Ciaravolo, Ansaldo/Genova
	Phone: 39/10/655 8705 Fax: 39/10/655 8799

A. OBJECTIVES AND SCOPE

The work is related to the design, manufacturing and testing of a Robotic system to Retrieve Tubes (TRT) from a steam generator of Latina Magnox Power Plant owned by ENEL. The retrieved material will be characterized radiochemically, melted and characterized again with a view to reuse.

The use of TRT will allow the reduction from 40 mSv to about 2 mSv of the radiation dose to workforce and a significant cost reduction mainly due to the reuse of obtained material.

The objective of the proposed project is the detailed study of a dismantling technique based on a robotic system and characterisation of the material obtained after melting. The Magnox steam generators are the largest contaminated plant items although activity levels are low.

The results of the work should strengthen the cost-effectiveness of reusing material from decommissioning. Moreover, it will allow to get detailed information about the radiological aspects of this technique and be applicable to other Magnox or gas/graphite type reactors.

B. WORK PROGRAMME

B.1. System requirements definition and studies (ANSALDO with ENEL support)
B.1.1. Analysis of the robotic system (TRT) layout and of the environmental and radiological conditions (in cooperation with ENEL)
B.1.2. Functional requirements definition of the TRT
B.1.3. TRT design at system level (Two TRT designs will be investigated)
B.2. Mock-up design and manufacturing (ANSALDO)
B.2.1. Design of a SG tube nest mock-up based on real geometric requirements
B.2.2. Mock-up manufacturing and installation in the Ansaldo testing facility
B.3. TRT design and manufacturing (ANSALDO)
B.3.1. TRT detailed design based on B.1. and control system design (Hardware and Software)
B.3.2. Mechanical part manufacturing, commercial part purchasing and assembly and integration of all parts.
B.4. Testing of the TRT on the mock-up (ANSALDO)
B.5. Site preparation (ANSALDO with ENEL support)
B.6. TRT operation and material transportation (ANSALDO + ENEL & SIEMP. support)
B.6.1. TRT installation on the SG made available by ENEL
B.6.2. Tube cutting and removal; charging of cut material in the transport containers
B.6.3. Chemical and radiological characterization - Transportation to Siempelkamp, Krefeld
B.7. Restoration of the site
B.8. Radiochemical analysis (SIEMPELKAMP)
B.9. Melting (SIEMPELKAMP)
B.10. Radiochemical analysis (SIEMPELKAMP)
B.11. Generation of specific data

C. PROGRESS OF THE WORK AND OBTAINED RESULTS

Summary of main issues

The activities performed in this period have first led to the definition of functional requirements of the system for both lay-out, technological and safety related aspects, allowing the development of the conceptual design and the progress of design activities up to the constructive assembly and sub-assembly drawings.

A preliminary activity of qualification tests, in particular to define the essential parameters of the cutting operation, has been carried out, providing useful informations for detailed design and elements for a more accurate time estimate.

The system is basicly composed of two parts:

- the tube cutting and retrieval device
- the contamination containment system.

The cutting and retrieval device operates inside the steam generator and the contaminated area is isolated from the external environment by means of a confined working cabine; the cutting system is remotely controlled by a console located outside the confined cabine.

A visual and instrumental monitoring system allows the operator to survey the various working phases which are foreseen to be executed in a semi-automatic mode.

The architecture of the control system has been defined on the basis of the operative sequence enabling software development and hardware definition

Progress ans Results

1. System requirements definition and studies (B.1)

A preliminary analysis to define the best way of cutting and handling the amount of tubes needed for the melting campaign, taking into account lay-out and transportation constraints, has been performed.

The main results of this activity are the following:

- the optimal tube cutting length is about 1.5 m varying in function of tube row elevation
- considering the above length the number of tubes to be cut to obtain about 5/6 tons of metal is approximately 300
- the tubes will then be manually cut in two halves in ENEL hot shop to be put inside 200 l drums whose weight will not exceed 500 kg.

The drums will be classified as containing IAEA Low Specific Activity LSA-II type material and characterized according to Siempelkamp specifications to determine the concentration of the main radionuclides.

The maximum exposure level allowed for each drum will be:

- 2.0 μSv/h in contact
- 0.1 μSv/h at 1 m distance

and the maximum surface contamination level:

- 37 kBq/m^2 for β-gamma emitters
- 3.7 kBq/m^2 for α emitters

The conceptual design of the dismantling system has been completed, Figures 1 and 2 show a schematic view of the equipment with reference to the steam generator lay-out and defines the two main sub-systems:

- the tube cutting and retrieval device
- the contamination containment system.

The tube cutting and retrieval device consists of a trolley which moves on a guiding rail structure which has to be previously manually installed inside the steam generator under the tube bundle to be cut.

This guiding rail is supported at one end by an external scaffolding, penetrated through the S.G. manhole being supported and centered to the S.G. by-pass at the other end.

The trolley can swivel of 90° to align with the tubes and a pantograph allows the cutting heads to raise to get in contact with them.

The contamination containment system basically consists of a box providing a dust tight environment during tube cutting, a working controlled area where personnel performs cut tubes handling and cutting device routine checks and maintenance, and a control cabine from which the operations are controlled.

2.Mock-up design and manufacturing (B.2)

A test equipment (see Figure 3), aiming at validating the design choices related to the cutting process, has been designed and ordered; it is composed of a simplified mock-up of the tube bundle and a cutting unit mounted on a slide with all the necessary instrumentation to detect:

- cutting forces
- feeding speed
- disk revolution speed

Tests have helped in defining the optimal parameters of the cutting operation, showing also how much the disk wears out, thus revealing the frequency of disk replacement.

At the same time the design of the SG tube nest mock-up, based on real geometric requirements, to be used for shop testing of the cutting and retrieval device has been carried out, allowing the issuing of the construction order.

3.TRR design and manufacturing (B.3)

3.1. Electro-mechanical part

Detail design of the cutting and retrieval device, comprising commercial components choice, has been completed, while definition of the contamination containment system has been performed.

The above mentioned cutting heads consist of ad hoc adapted commercial grinding units provided with 250 mm diameter cutting disk; they are mounted on the pantograph on two slides for manual adjustment of the cutting length and can pivot to assume both the cutting configuration, with disk perpendicular to the tube bundle, and the input/output one, with disk tilted to allow passage through the manhole.

Two pivoting holding devices hinged to the pantograph come in abutment with the tube bundle at the end of the raising phase, similar devices on the trolley provide an analogous function with respect to the tube bundle below the guiding rail, they all realize a sort of docking to improve stiffness, thus reducing vibrations, during cutting operations.

Cutting heads tilting and holding devices pivoting are obtained by means of pneumatic cylinders, while all the other movements are driven by DC electric motors and reductors.

Figure 4 shows the various components of the cutting and retrieval device, which is now being manufactured and will be ready in February 1993 for shop testing.

3.2. Control system

On the basis of the operating sequences the philosophy of the control system and the instrumentation requirements have been defined; a semi-automatic approach has been chosen, meaning that an automatic sequence, such as positioning, cutting or tube retrieval, can start only after an operator consent.

Moreover the operator can manually intervene on the operation if needed, in particular this can be of great interest for positioning of the cutting trolley under the tube to be cut; in fact, due to manufacturing tolerances, the correct position can be shifted of several millimeters from the planned one and a fine repositioning is therefore needed.

This is obtained by checking alignment by means of two CCTV cameras mounted on the pantograph.

The control system hardware, which is based on a Siemens PLC, has been defined and software development is in progress.

4. Site preparation (B.5)

On site inspections of the steam generators have been made, these have led to the definition of the SG object of the intervention and of the activities to prepare the working area around and inside the chosen SG, which have begun and will be finished in February allowing the assembly of the contamination containment system.

5. Chemical and radiological characterization (B.6.3)

A preliminary radiological characterization of cut down tubes has been performed by α, β and gamma spectroscopy to determine the concentration of the main radionuclides; the results are summarized in Table I.

Table I: Radiological characterization of tube samples

Nuclides	Specific Activity [Bq/g]	Nuclides	Specific Activity [Bq/g]
Co – 60	11.5±0.6	U – 235	0.0011±0.0007
J – 129	0.3±0.1	U – 238	0.0042±0.0011
Cs – 137	7.8±0.6	Pu – 238	0.039 ± 0.002
Ni – 63	13 ±1.0	Pu–239/240	0.114 ± 0.003
Fe – 55	2.4±0.2	Pu – 241	2.8 ± 0.3
U – 234	0.015±0.002	Am – 241	0.39 ± 0.02

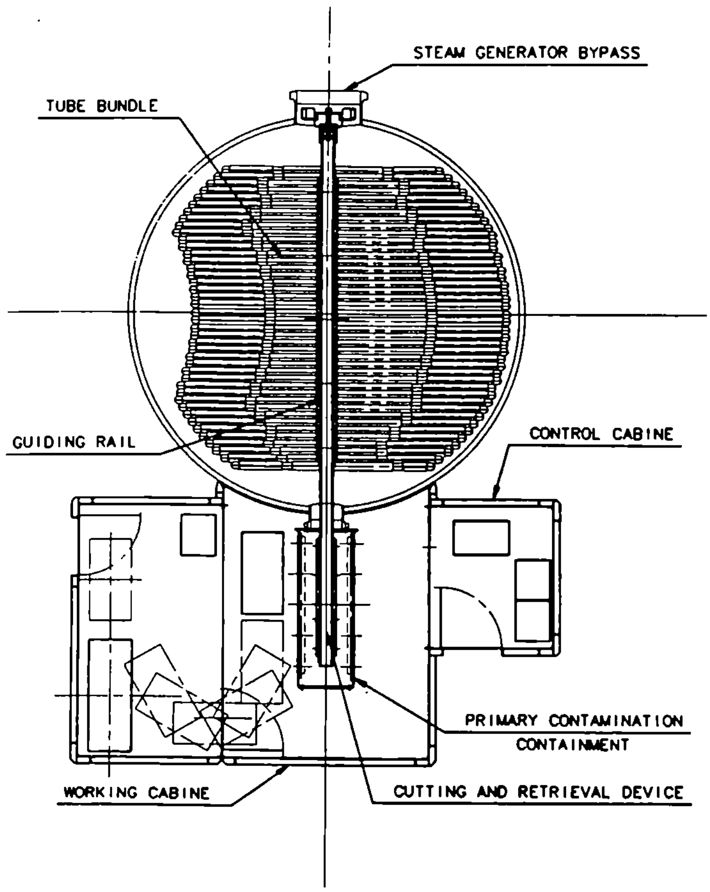


Figure 1: Dismantling System - Plan View

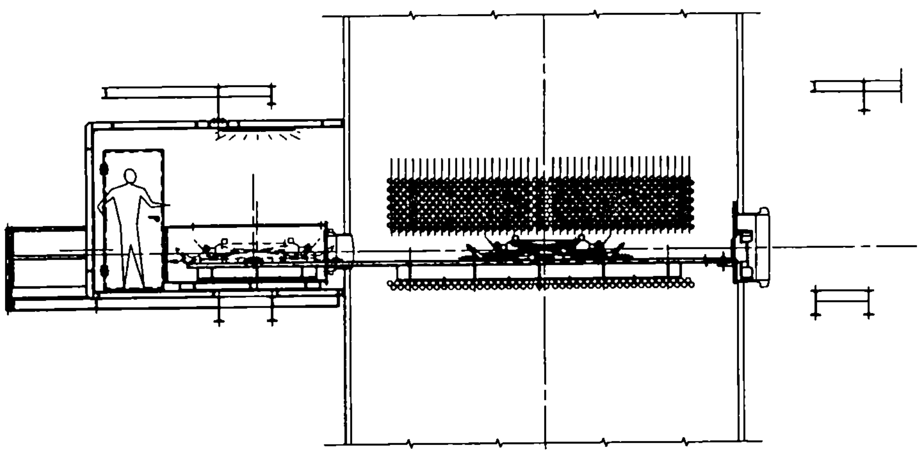


Figure 2: Dismantling System - Elevation

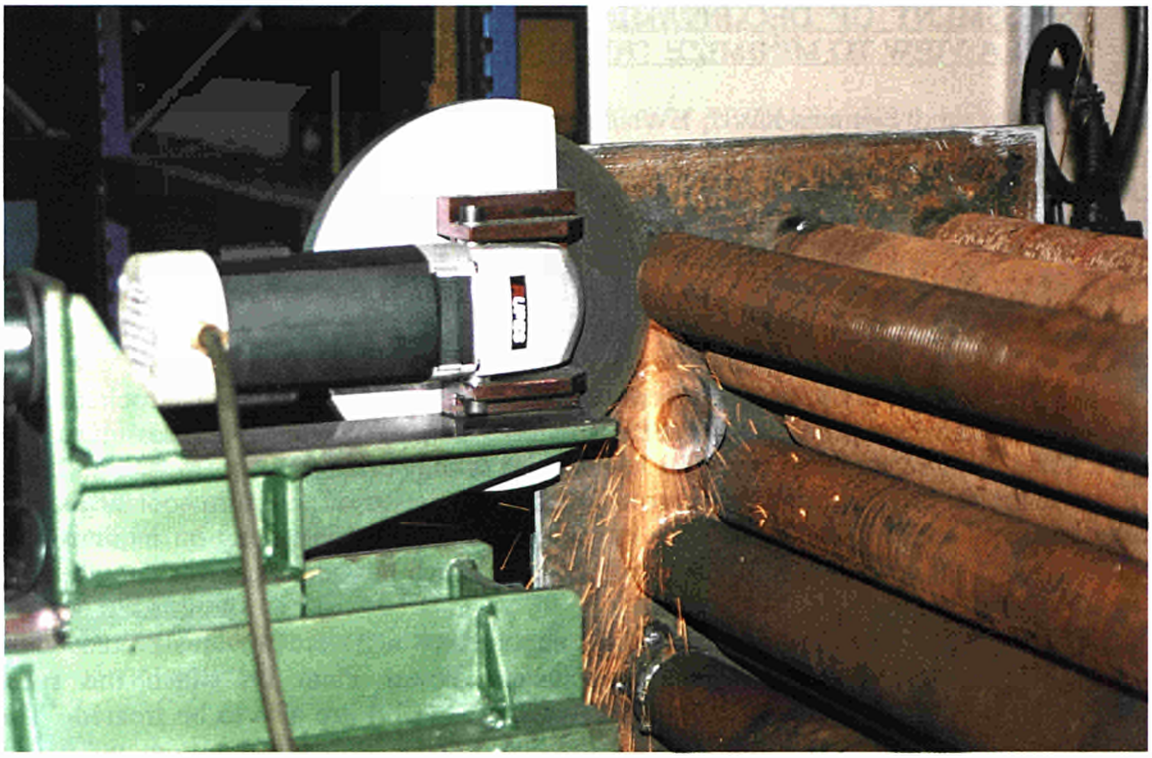
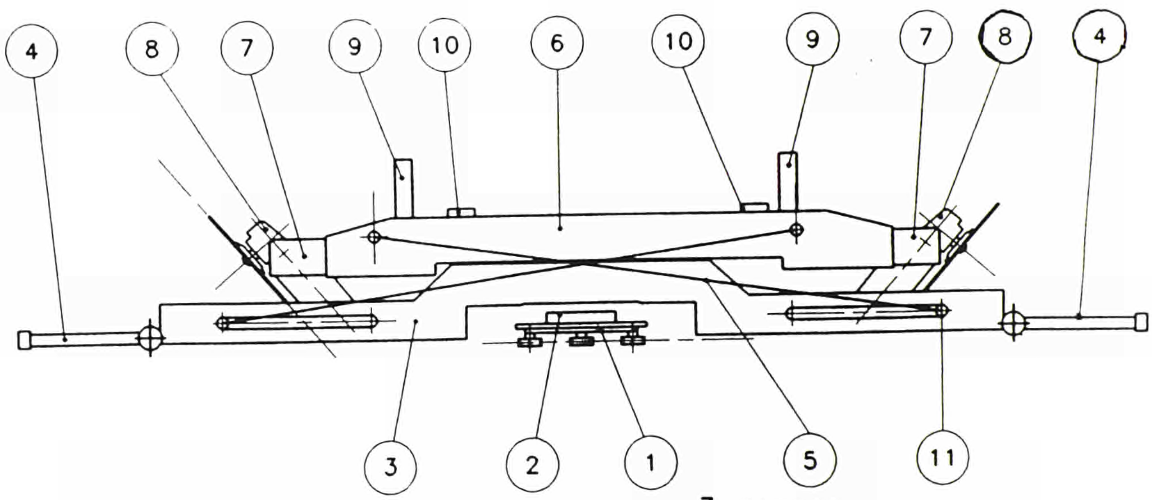


Figure 3: Cutting test equipment



1- TROLLEY
2- SWIVELLING SYSTEM
3- LOWER GIRDER
4- LOWER HOLDING DEVICES
5- PANTOGRAPH
6- UPPER GIRDER
7- SLIDES
8- CUTTING HEADS
9- TUBE CARRIERS
10- UPPER HOLDING DEVICES
11- PANTOGRAPH DRIVING SYSTEM

Figure 4: Cutting and retrieval device

8.20. ASSESSMENT OF DECONTAMINATION PROCEDURES FOR WWER-PWRs WITH A VIEW TO MINIMIZE THE GENERATION OF WASTE

Contractors: Siemens-KWU, EWN/KKW Rheinsberg
Contract No.: FI2D-0067
Work Period: December 1991 - December 1993
Coordinator: H WILLE, Siemens KWU Erlangen
Phone: 49/9131/18 33 39 Fax: 49/9131/18 28 21

A. OBJECTIVE AND SCOPE

The objective of this project is to verify the efficiency of the chemical, electrochemical and physical decontamination processes for the decommissioning of WWER-type pressurized water reactors. The testing of chemical processes has priority, as only these could be applied in the most important technical task of the programme, the decontamination of an entire coolant loop with a present activity content of 35 Ci. The concept places special emphasis on minimizing the amount of waste, the principle being that only material which has been removed should have to be stored in a repository.

The investigations will be carried out on components which were previously removed from the reactor coolant system of Rheinsberg Nuclear Power Plant on which the specific contamination is 10^4 Bq/cm^2. Selected components of the BR-3 are also to be treated. These parts, among others, are to be decontaminated in accordance with the requirements to levels permitting unrestricted release or reaching at least remelting conditions.

The most promising processes shall be applied for the decontamination of one of the reactor coolant loops at Rheinsberg Nuclear Power Plant. This treatment will aim at reducing contamination to a release (melting) level without dismantling the loop components. On the basis of the investigation results, a concept is to be established for the decontamination of WWER-type PWR reactors, illustrated with reference to Rheinsberg Nuclear Power Plant.

B. WORK PROGRAMME

B.1. Process evaluation of decontamination processes permitting subsequent unrestricted release (Siemens): selection of the most effective process for the decontamination of component surfaces.

B.2. Decontamination of removed components: the process selected in para B.1. shall be tested on laboratory samples and full-sized components.

B.3. Decontamination on one primary loop of the Rheinsberg NPP (Rheinsberg and Siemens) The in-situ decontamination of one of the three primary loops of the reactor will consist of:
- pre-treatment with APCE/CORD to remove oxide layer;
- succession of oxidation and pickling cycles as determined in B.1. and B.2.
- determination of residual activities.
- treatment and conditioning of process waste.

B.4. Overall concept for the decommissioning of a WWER nuclear power plant (Siemens and Rheinsberg). Based on the experience and the results of the preceding work programme, an optimized concept for the decommissioning of reactor components for WWER NPPs shall be developed.

B.5. Generation of specific data: Specific data on costs, worker exposure, working time and waste arisings will be derived from the execution of items B.2. and B.3.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

During the work period, work was performed in parallel for the three main objectives:

- large-scale test of a waste minimization process during a full system decontamination of a BWR;
- preparation and performance of laboratory-scale and full-size original component decontamination tests;
- preparations for the decontamination of one loop of the Rheinsberg WWER-type nuclear power plant.

For the execution of the planned investigation, a plan for further actions was discussed and agreed upon. On Figure 1, the detailed tests and the aims or subjects to be investigated are mentioned. A comparison between Figure 1 and the work programme shows that step B.1. of the work programme represents the process evaluation; B.2. is represented by the tests mentioned under steps 1 and 2, while B.3. is step 3. Also mentioned in the plan, the locations of investigations and the working group involved.

Progress and results

The main purpose of the project is to reach the following boundary conditions for the free release of the material:

< 200 Bq/g:	for melting and nuclear reuse;
< 1 Bq/g:	for free release after melting;
< 0.1 Bq/g:	free release.

Within the project, the major aim is to decontaminate the components to be below 200 Bq/g if possible without dismantling.

A special treatment method was tested for the selection of a decontamination process suited for an application aiming at the minimization of waste. The CORD process was selected for application of a decontamination process to the complete BWR primary loop. In a special treatment step at the end of each decontamination cycle, the deconchemical was decomposed in passing through a newly developed organics destruction module where organic acids were decomposed to CO_2 by intensive UV-light. The result of the test was the complete decomposition of the organic acid to CO_2 and H_2O. By this treatment, a large amount of resin capacity necessary for waste removal was saved as the resins were mainly required for the removal of the oxides and activity removed from the contaminated surfaces of the NPP systems.

For the laboratory tests and the decontamination tests on dismantled components, a decontamination vessel with a volume of 13 m^3 is available in Rheinsberg. The dimensions of the tank are sufficient to receive a complete primary loop isolation valve. Additionally, a chemical injection system is required.

For the laboratory tests performed in Rheinsberg and Mol, the conditions were:

1. Decontamination tests in Rheinsberg

1.1. Laboratory tests (B.1.)

Application of CORD decontamination cycles

- 2 hours pre-oxidation
- Study of the influence of oxalic acid concentration on decontamination effect
- Decontamination tests with various acid mixtures
- Test material: samples removed from the primary circuit.

Depending on the acid concentration, a decontamination factor of 1.3 - 61 was obtained. If ultrasonic treatment is additionally applied, the decontamination factor rises to 1.4 - 1155. The tests showed that higher acid concentration did not lead to a noticeable increase of the decontamination factor. The final activity contamination on the surface of steam generator tubing was $\pm$ 1 Bq of Co-60 g/steel.

1.2. Tests in decontamination vessel Rheinsberg (B.1.2.)

With the results of the above-mentioned tests, large components such as isolation valves and primary loop piping were or will be decontaminated in the 13 m^3 decontamination vessel. The results of these decontaminations will be evaluated in the next quarter.

2. Decontamination tests in Mol (B.2.)

2.1. Laboratory tests

For the laboratory test in Mol, a laboratory decontamination vessel with ultrasonic transducers will be used to test the CORD process with and without the use of ultrasonic transducers and with various pipe geometries.
For the tests, non-decontaminated pipes (3" SCH160) of the primary circuit are available. These pipes were not decontaminated during the full system decontamination of April 1991 and show a dose rate value between 0.2 and 0.5 mSv/h. The loop will be operational in the beginning of 1993.

2.2. Decontamination tests on a component (B.2.)

2.2.1. Selection of the component

The Regenerative Heat Exchanger (RHX) has been selected for decontamination tests on a component; this equipment has been decontaminated, as part of the purification loop of the BR-3, during the full system decontamination of April 1991. The RHX is a GRISCOM coiled tube heat exchanger with primary water flowing both through the shell and tube side; it is thus contaminated on both sides. The material is stainless steel 304 and the total surface to decontaminate is 22.5 m^2.
The RHX was disconnected from the primary circuit, transported outside the reactor building, and radiologically characterized. The exchanger first shows a homogeneous contamination leading to a dose rate between 0.3 and 0.7 mSv/h. However, some hot points of 2 to 3 mSv/h are localized at the bottom of the exchanger at the position of the baffles inside the RHX.

2.2.2. Decontamination loop for the component

The decontamination loop which will be built for the pipes decontamination tests will also be used for the decontamination test for the component; the RHX exchanger will simply be connected to the decontamination loop after the heater; the decontamination vessel acts as buffer tank.

2.3. Preliminary decontamination test on small samples of the RHX inlet pipe (B.2.)

Decontamination was performed at laboratory scale on a small sample of the inlet pipe of the RHX; this pipe is a stainless steel 304 1"SCH160. A small 10 cm long sample was used for the decontamination test. The piece was submitted to one oxidation step and to two successive decontamination steps. An extension of the decontamination during 6 additional hours resulted only in a slight improvement of the decontamination. A final decontamination factor of about 109 was reached, leading to a residual activity of 77 Bq/cm^2, which for such a pipe corresponds to a mass activity of about 12 Bq/g, i.e. lower than the regulatory melting limit (200 Bq/g).

3. Loop decontamination (B.3.)

As regards the decontamination planned for one loop of the Rheinsberg NPP, a full system decontamination (No. 19) was performed at the beginning of the last outage. Leaving the loop with the following calculated contamination levels (only Co-60):

Hot leg	195 Bq/g
Reactor coolant pump	50 Bq/g
Lower section:	
Steam generator	510 Bq/g (total)
Steam generator	2700 Bq/g (tubing)

About 18% of all tubes are closed by plugs and the majority of the tubes have sleeves in the area of the tube plate. The reactor coolant pump of the Rheinsberg NPP can no longer be operated due to requirements of the authorities and therefore, pumps and a heater of the Siemens AMDA will be used for the decontamination.

The positioning of the AMDA components and the connections to one loop of the Rheinsberg NPP was discussed. The preliminary flow diagram is shown in Figure 2. An electricity supply via provisional connections will be realized. The steam generator for the test is:

SG No. 2 in Loop 1 (about 7.5 % of tubing closed).

As connections to the AMDA, flanges in DN 80 are foreseen.

For the removal of the activity, one MOSAIK II shielding tank with an internal ion exchange vessel will be used.

The decontamination test of the loop is scheduled for the second quarter of 1993.

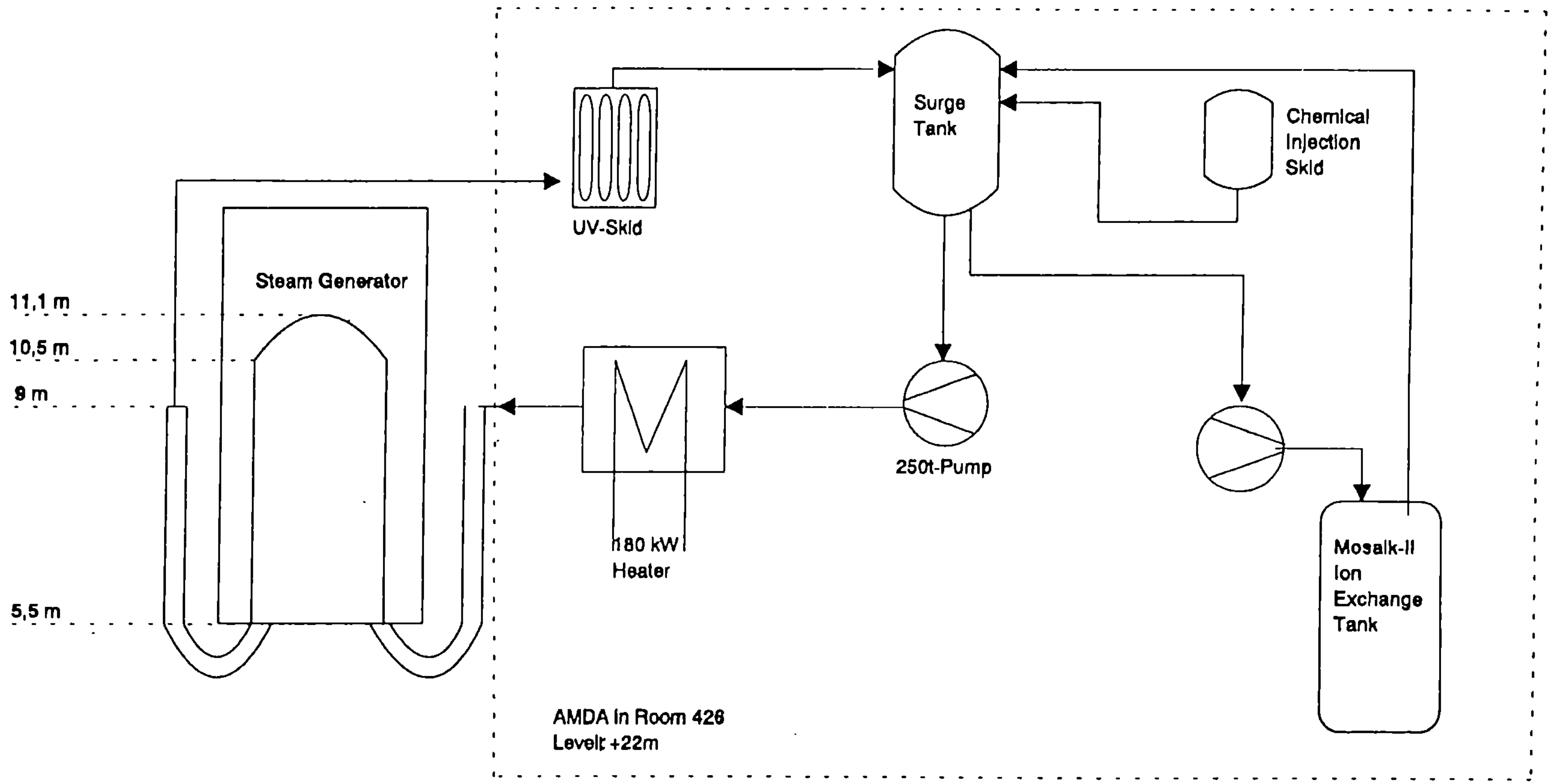
The tests foreseen in step 2 (B.2.) will be finished in the first half of 1993.

DECO - WWER / PWR

Assumption: FSD ---> Segmenting Components to Parts ---> Parts Decon
(Loop)

			Step 1	Step 2		Step 3
	Process Evaluation		Laboratory Test	Rheinsberg Decon Vessel		Loop Decon
	Processes	Aspects		Small parts < 200 l	component	part of a loop
Test Condition	---	---	samples			
Full System Decon	CORD + UV AP - CE etc.	o DF o Waste o Feasibility o Time requirement	Aspects: o mat.-attack o chem. concentr. o temperature o efficiency	---	o MOL o Rheinsb.	Rheinsberg (?)
Part decon	Chemical and electrochem. processes			o Rheinsb. o MOL	---	---
Location of Performance	Erlangen		o Rheinsberg o MOL	E-chem. ---> MOL (Siemens) Rheinsberg Chem. US ---> Rheinsb. (Siemens)		Rheinsberg
1. Step Evaluation	Siemens		x Rheinsberg x MOL	x Rheinsberg o MOL		Rheinsberg
2. Step Performance of Lab. Tests	---		x MOL x Rheinsberg	---		---
3. Step Performance	---		---	o MOL o Rheinsberg		Rheinsberg

Figure 1 Assessment of Decontamination Procedures for WWER-PWR's



Flow Diagram AMDA Decontamination Steam Generator NPP Rheinsberg

Figure 2

8.21. DESIGN, MANUFACTURING AND APPLICATION OF A TELEOPERATED DEPLOYMENT SYSTEM BASED ON THE "NEATER" ROBOT

Contractors:	AEA Technology
Contract No.:	FI2D-0068
Work Period:	November 1991 - December 1993
Coordinator:	G V COLE, AEA Technology Harwell
	Phone: 44/235/43 47 64 Fax: 44/235/43 61 38

A. OBJECTIVE AND SCOPE

The work relates to the construction and operation of a full-scale teleoperated cutting, monitoring and decontamination system to remove highly contaminated equipment. The project includes the proving of the system in an inactive mock-up followed by size reduction of contaminated equipment in the DIDO High Activity Handling Cell, removal of items, monitoring and decontamination.

Advantage will be taken of techniques and equipment developed under earlier work in the Programme for Decommissioning of Nuclear Installations (e.g. FI2D-0012), as well as of other initiatives (e.g. Teleman).

The range of applications covers all types of nuclear facilities. Specific benefits of the project will be:

- a reduction in occupational exposure to radiation, due to automatic and remote manipulation;
- a reduction in background radiation levels in the vicinity of the work areas;
- an improvement of the Quality Assurance of operations and waste accountancy;
- improved safety and protection of operators;
- a reduction in the cost of decommissioning activities due to e.g. more rapid work completion;
- improvement in the awareness of remote technology for decommissioning.

The system will be based on already available components such as the Nuclear Engineered Robot System (NEATER), the Telerobotic Control System (HTC) and the Stereoscopic Television System (TV3).

B. WORK PROGRAMME

B.1. Design, specification and commissioning of the robot and auxiliary equipment support frame
B.2. Design, specification and commissioning of the decontamination equipment
B.3. Design, specification and commissioning of the tools and tool change system
B.4. Design, specification and commissioning of the control room
B.5. Service requirements and interface connection will be specified and survey services provided.
B.6. Design, specification and commissioning of the viewing system, including pan/tilt unit and controls, cables, lighting.
B.7. Design, specification, construction and commissioning of the mock-up area
B.8. Preparation of safety case and obtention of approval for robotic safety, active area safety and of operational method statements and procedures.
B.9. Site-specific activities including delivery of equipment, installation of control room, etc.
B.10 Removal, monitoring and decontamination of in-cell equipment, waste management, removal of robot system and cleaning up work area.
B.11 Specific data on costs, worker exposure, working time, waste arisings and fissile material recovery will be derived from the execution of work.

C. Progress of work and results obtained

Summary of Main Issues

Within the reporting period the application for the telerobotic dismantling system has been changed, from the active drain line of B14 Windscale to the DIDO High Activity Handling Cell (HAHC) at Harwell. The latter having much higher dose rates and thereby offering a better justification for remote dismantling. The equipment required being similar in both cases.

The work carried out during the year has involved the production of estimates for cost, time-scale, and dose for different methods together with the design and procurement of equipment and the construction of a mock-up.

Progress and Results

1. Design, specification and commissioning of the robot and auxiliary equipment support frame (B.1.)

A computer simulation was carried out to optimise the equipment configuration and the equipment capability with particular reference to cell coverage. The simulation results gave the basic design data enabling system design layouts to begin.

The cable deployment proved to be a difficult problem that was resolved with a link type cable carrier. Suitable equipment has been supplied and provisional assembly made, see Figure 1. Design Notes have been produced for cable deployment design and arrangement.

The robot trolley has been designed and assembled in a non-motorised form, see Figure 1. The trolley design includes the tool change rack and the anchoring of the cable carrier.

Work has begun on updating the NEATER robot to the latest standard. Equipment has been ordered. The robot has been assembled on to the trolley for initial stability testing.

2. Design, specification and commissioning of the decontamination equipment (B.2.)

The decontamination equipment arrangement was reviewed using information from a similar decontamination. This revealed that the higher decontamination efficiency using the foam technique, will produce higher level effluent. The system design has now been altered to reflect this and manufacture is under way.

3. Design, specification and commissioning of the tools and tool change system (B.3.)

A radiation monitoring tool has been identified and its integration into the system is underway. Close liaison was required with the facility management to ensure that health physics requirements were met.

A set of standard cutting and handling tools has been manufactured. These are 2 saws, one drill and two jaws. A saw was modified for right angle mounting to take into account information resulting from the computer simulation work.

A device for changing cutting tool blades remotely using master slave manipulators (MSM's) has been designed, made and tested.

4. Design, specification and commissioning of the control room (B.4.)

The specification for the control station has been written and approved. Purchase and manufacture of the control equipment has begun.

5. Service requirements and interface connection specification (B.5.)
   Provisional work started.

6. Design, specification and commissioning of the viewing equipment (B.6.)
   Viewing equipment comprising two sets of: cameras, lenses, lights, and pan and tilt units; have been acquired and successfully bench tested. Final assembly has now started.

7. Design, specification and commissioning of the mock-up area (B.7.)
   The non-active mock-up assembly has been design and constructed, see Figure 2. This included the positioning of partition walls, MSM frames, and installation of emergency stop and other safety equipment. A railway has been laid for the robot trolley to run on. A bench has been constructed together with simulated bench equipment. This is ready for the robot to carry out size reduction tasks.

8. Preparation of safety case and obtaining approval for robotic safety, active area safety, and of operational method statements and procedures (B.8.)
   Draft documentation for the safety case has been written and sent for 'peer' review. This documentation will have to satisfy the facility licensing bodies.

9. Site-specific activities (B.9.)
   Site surveys have been carried out. These included an active entry to gain vital dimensional information for design purposes.

10. Removal, monitoring and decontamination of in-cell equipment (B.10.)
    Not Started

11. Specific data on costs, worker exposure, working time, waste arisings and fissile material recovery (B.11.)
    A project definition paper was produced which included costs, time, man dose uptake, feasibility and risk comparisons between traditional and telerobotic methods, see figures 3 and 4. The costs and time to produce this was presented for inclusion in the costs data base. The information generated resulted in the robot being mounted on its own trolley, rather than using the existing cell trolley, to reduce overall cost and time.

Figure 1 - Transport Trolley and Cable Deployment.

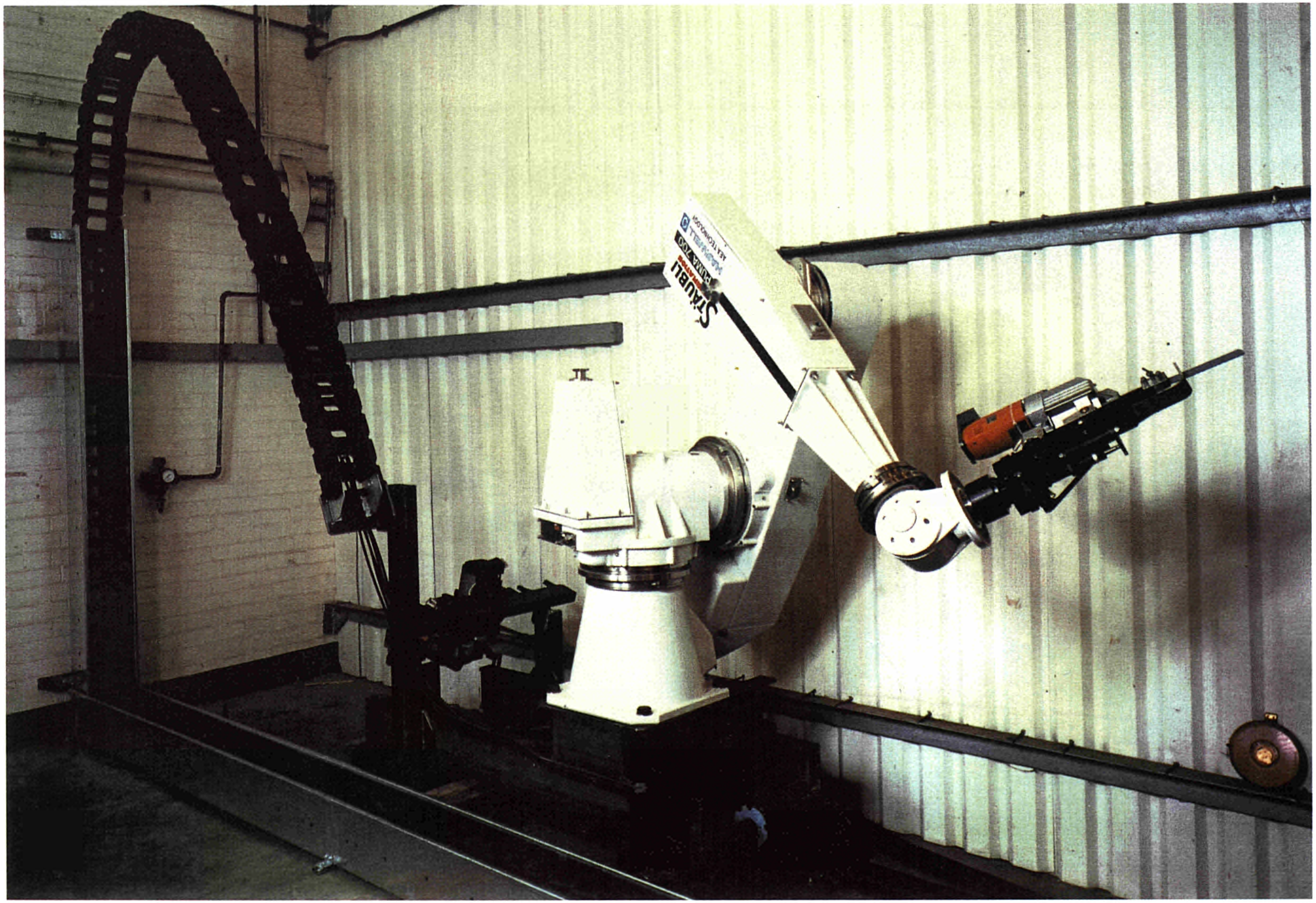
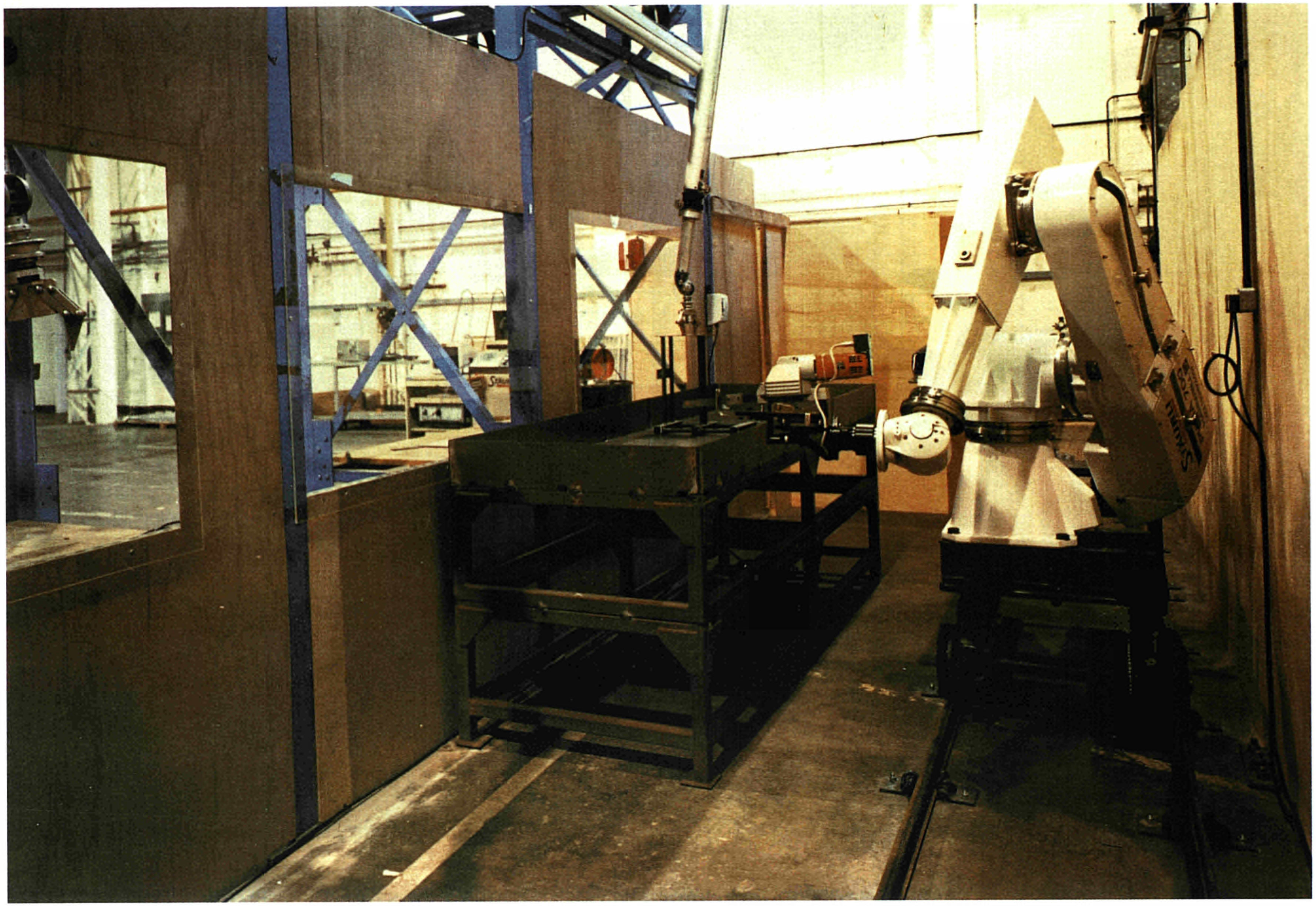


Figure 2 - DIDO High Activity Handling Cell Mock-up Facility



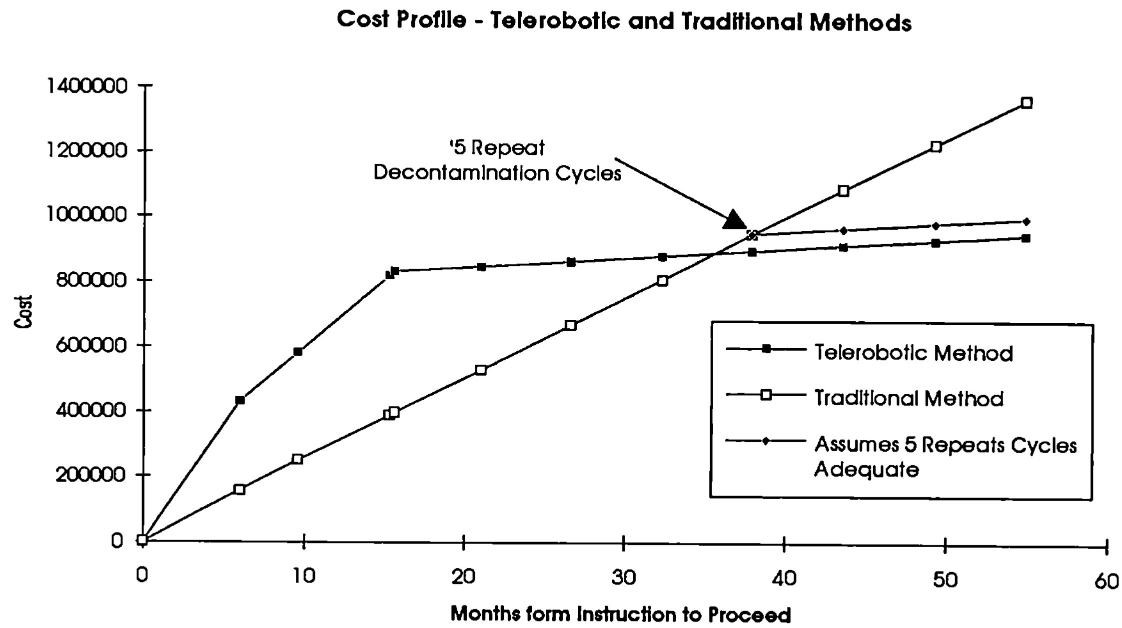


Figure 3

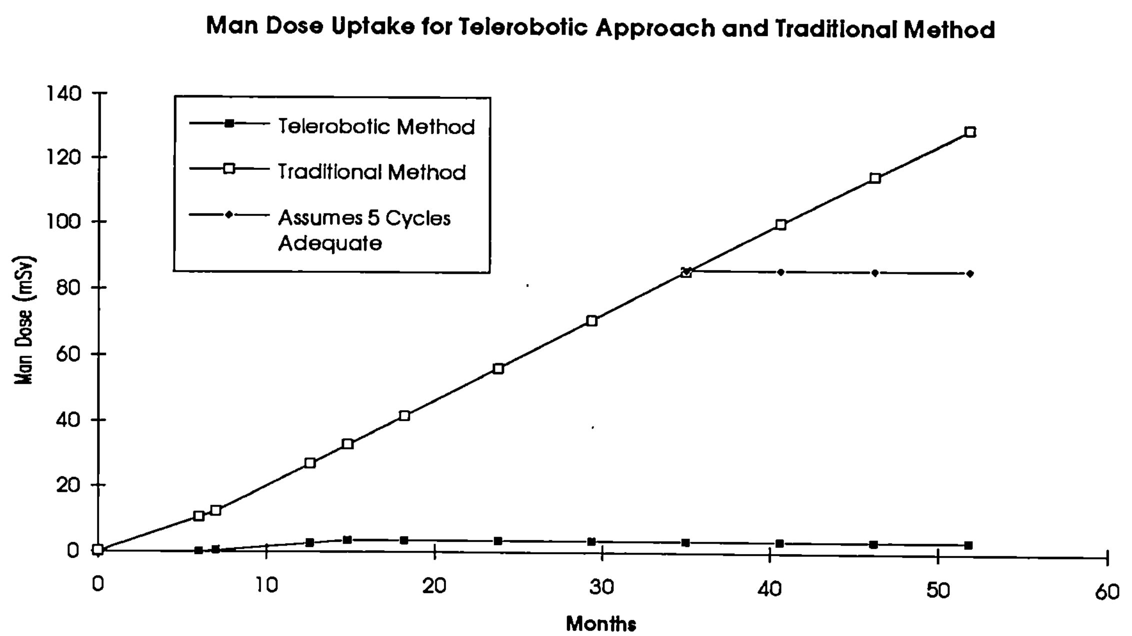


Figure 4

8.22. IN-SITU DECONTAMINATION OF THE TUBE BUNDLE FROM A STEAM GENERATOR OF THE DAMPIERRE PWR AND SUBSEQUENT WASTE TREATMENT

Contractors: Framatome, EdF, CPN Dampierre
Contract No.: FI2D-0069
Work Period: January 1992 - December 1993
Coordinator: G DORIMINI, Framatome, Chalon s/S.
Phone: 33/85 96 30 76 Fax: 33/85 96 35 54

A. OBJECTIVE AND SCOPE

The present work concerns the "hard decontamination" of a tube bundle installed in a steam generator (SG) removed earlier from the DAMPIERRE 900 MWe PWR, in order to reduce significantly the radioactive dose rate before the final dismantling of the steam generator.

The SG in question is one of the original SGs designed and built by Framatome. They were removed from service in 1990 because of the large number of plugged tubes (approximately 10%) and the numerous primary-to-secondary leaks apparently due to stress corrosion cracking attack (SCCA).

The SGs have been stored since 1990 in a building designed for this purpose and the radioactivity present in each SG, at this time, is estimated to be about 100 Ci (essentially Cobalt 60). The dose rate on the outer surface of the dry SG varies from 0.05 mSv/h to 0.5 mSv/h.

The proposed decontamination process provides using a combination of nitric acid and Cerium nitrate with regeneration of the Cerium (Cerium 3+, Cerium 4+) by injecting ozone during the decontamination operation. The decontamination operation is performed at low pH and at ambient temperature.

After neutralization and precipitation of the decontamination solution containing the removed activity, the residue will be dryed in order to fix it in solid form for storage and disposal.

The objectives of the method selected to decontaminate the SG bundle made of Inconel 600 material are:

1. obtaining a high decontamination factor DF equal to approximately 1000;
2. decontamination at atmospheric pressure and at a temperature of less than 60°C because of doubts concerning tube integrity and the risk of leakage into the secondary side;
3. minimize the volume of generated secondary wastes;
4. filtration of sludge and residues generated by the decontamination process, if possible, using a standard process.

The programme will be implemented jointly between: Framatome (FRA), Electricité de France (EdF) as partners with the assistance of the plant owner Centrale Nucléaire de Dampierre (D) and KWO, owner of Obrigheim NPP which will provide some participation.

B. WORK PROGRAMME

B.1. Definition of the decontamination and of the waste treatment process

B.1.1. Definition of the decontamination process based on the existing oxide layer thickness and the needed time to remove this oxide layer to obtain a DF of approximately 1000 (FRA).

B.1.2. Implementation of corrosion tests and laboratory studies for the assessment of the corrosion resistance of the equipment in the decontamination loop (EdF).

B.1.3. Establishing of the procedure for the in-situ decontamination (FRA and EdF).

B.1.4. Definition of the conditioning of effluents and waste treatment arising from the decontamination of the SG bundle and of shielding and protection requirements (EdF and D).

B.2. Procurement and adaptation of equipment

B.2.1. Adaptation of existing equipment formerly used with an AP Citrox process (EdF)

B.2.2. Fabrication of the additional decontamination equipment, taking into account the characteristics of the provided decontamination unit, followed by commissioning testing (EdF, FRA)

B.3. Decontamination operations and liquid waste treatment

B.3.1. Preparatory operations on site, including preparation and qualification of operating specifications and procedures (FRA, D)

B.3.2. Implementation of the decontamination operation (FRA)

B.3.3. Liquid waste treatment and conditioning (FRA, EdF)

B.4. Analysis of results including the estimation of the residual activity and the definition of an industrial-scale decontamination procedure (all)

B.5. Generation of specific data: Specific data on costs, worker exposure, working time and waste arisings will be derived from the execution of items B.2. and B.3. (all).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

All the work is going forward in accordance with the planning (s. programme B). The first half of the year was devoted to Phase I: definition of the decontamination methods and treatment of the effluents.

The work carried out during the second half of the year essentially concerned the changes to bring to a loop performing soft chemical decontamination, to render it compatible with the strong ozone decontamination process.

All modifications brought to the loop were validated during a test run in Sweden to decontaminate a small steam generator (SG) that had been removed from the Agesta nuclear power plant (NPP). The results obtained at Agesta confirm that the target value for the decontamination factor (on the order of 1000) can be reached.

The remainder of the work performed during the second half of 1992 resulted in the development of a suction cup that would enable the decontamination of about 350 SG tubes in a single pass. The dosimetric evaluation of the decontamination operation planned for 1993 at Dampierre (about 40 rem), however, casts doubt on the use of such a suction cup.

The request for an authorization to perform the Dampierre SG decontamination in 1993 will be submitted to the French safety authorities during the first months of next year.

An alternative solution that would enable treating the entire SG tube bundle in a single operation is presently under study.

Progress and results

The report covering the work performed during the first half of 1992 presented the results of the decontamination test of an auxiliary heat exchanger, on the Tricastin NPP site.

The decontamination process was applied to a regenerative heat exchanger removed from a nuclear power unit at Dampierre.

The main parameters for the decontamination operation were as follows:

- nitric acid: Ph = 0.6 - 0.25 to 0.50 mol/l,
- cesium nitrate: 8 to 12 g/l,
- ozone: 7 to 8 ppm,
- fluid-circulation speed: approximately from 2.5 $cm.s^{-1}$ to 40 $cm.s^{-1}$.

The results obtained after about 36 hours of decontamination enabled observing:

- residual activity inside the SG tubes equal to 1 to 3 Bq/cm^2, and
- residual activity outside the SG tubes equal to 3 to 5 Bq/cm^2, with a hot spot at 20 Bq/cm^2.

The aspect of the stainless steel tubes after decontamination was perfectly clean. The volume of the effluents produced, including the rinsing effluent, was 3.5 m^3. In addition, work was done covering project Phase B: "Definition of the Decontamination Methods and Effluent Treatment", as described below.

1. Evaluation of the decontamination factor (B.1.1)

Depending on the nature of the material used to make the steam generator (stainless steel or an Inconel alloy), the decontamination factor is function of the erosion velocity of the decontaminating solution, which is estimated at 2 μm/24 h for type 316 stainless steel and at 3 to 4 μm/24h for Inconel 600.

A decontamination factor from 500 to 10,000 can be obtained on stainless steel SG tubes over a period of 70 hours (Agesta results). The same result can be obtained for Inconel 600 SG tubes in less than half that amount of time.

During the decontamination of the first small SG at the Agesta NPP in Sweden, a sample representative of a contaminated SG tube from the Dampierre NPP was also decontaminated as a test. After 70 hours of decontamination, a residual activity of 0.55 Bq/cm^2 was measured.

2. Thickness of the oxide layer to be removed (B.1.2)

On the internal surface of 116 tubes pulled, the thickness of the deposited oxide layer varied from 0 to 5 μm. The most frequent value was 1 μm, and the average was 1.18 μm.

The oxides were sampled by dissolving the base metal in a bromine-methanol solution, followed by filtering of this solution. The mass of oxide recovered per unit of surface area was highly variable, with an average of 4 g/m^2.

On the average, the oxide deposited on the internal surface of the SG tubes is composed of 50% chromium oxide, 30% iron oxide and 20% nickel oxide.

3. Decontamination method (B.1.3)

The establishment of the in-situ decontamination method depends on the modifications of the decontamination loop accomplished during the last half of 1992 (see below). Nevertheless, a document package comparable to a Preliminary Safety Analysis Report (PSAR) was drawn up and submitted to the safety authorities for a preliminary authorization.

4. Effluent conditioning and treatment (B.1.4)

The definition of effluent conditioning was the subject of an analysis given in the progress report for the first half of 1992. The conclusions of that analysis can be summarized as follows:

Treatment of the solutions by caustic soda coprecipitation of the metallic hydroxides is the solution presently envisaged. The work performed during the second half of the year mainly concerned the procurement of additional equipment and adaptation of the existing equipment.

Tests to check the capacity of PALL filters and predrummed filters to retain the precipitates were carried out during the last half of 1992, to confirm the validity of the solution adopted to treat the decontamination effluents.

5. Equipment adaptation (B.2.1)

The modifications carried out on the existing loop were as follows:

- installation of two 40 m^3 pumps operating in tandem,
- modification of the instrumentation and control system to take into account both pumps,
- modification of the nozzles diameter from 50 to 80 mm,
- replacement of the 50 mm diameter piping by the 80 mm diameter piping,
- installation of a 4 Kw cooler in the 4.6 m^3 tank (added in series to the existing 3 Kw cooler)
- making the tank ozone-tight,
- adding a 30 Kw cooler to the 1 m^3 tank,
- adding a fluid flow reverser, fitted with a bypass,
- adding sleeves for connecting the pre-embedded filtering system,
- adding a peristatic pump to transfer the effluents,
- adding a test cell containing a sample of Dampierre SG tube, equipped with a delta P indicator, and
- adapting the counter 2 to measure the volumes of air and liquid passing in both directions through the piping.

A cold hydraulic validation test of the loop was performed in November 1992 at the Tricastin NPP.

All the modifications were validated in a test run in Sweden at the end of 1992 on a small steam generator that had been removed from a nuclear power unit at the Agesta NPP (1600 stainless steel U-tubes representing a total surface area of 300 m^2). The loop operated satisfactorily, and the decontamination factors were in conformity with the objectives of the present project (500 to 10,000 for stainless steel, during the first decontamination operation carried out in December 1992).

A decontamination operation on two small SGs removed from service at Agesta is presently in progress in Sweden, using this mobile decontamination loop (see figures 1 and 2).

6. Manufacture of additional equipment (B.2.2)

Framatome has made and tested a suction cup capable of treating 336 SG tubes at the same time, representing about 4 m^3 of decontaminating solution for a surface to be decontaminated of about 400 m^2.

However, the dosimetric estimation made for use of this suction cup along with recycling of the decontaminating solution, for three lots of 350 SG tubes each leads, in a first analysis, to a radiation dose for the entire operation (about 40 rem or 0.4 Sv) that is excessively high. An alternative solution that would enable treating the entire SG tube bundle in a single pass is therefore presently being studied, to reduce the overall dose.

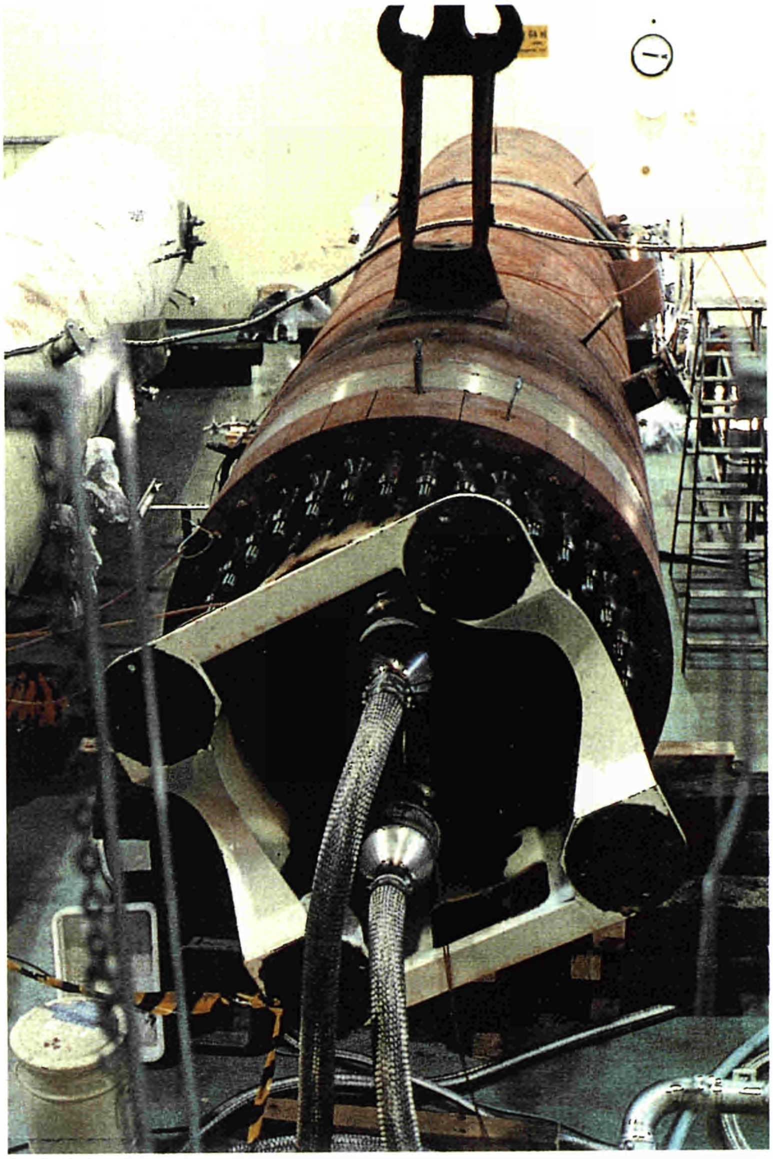


Figure 1: *AGESTA STEAM Generator - Nb of tubes : 2000*

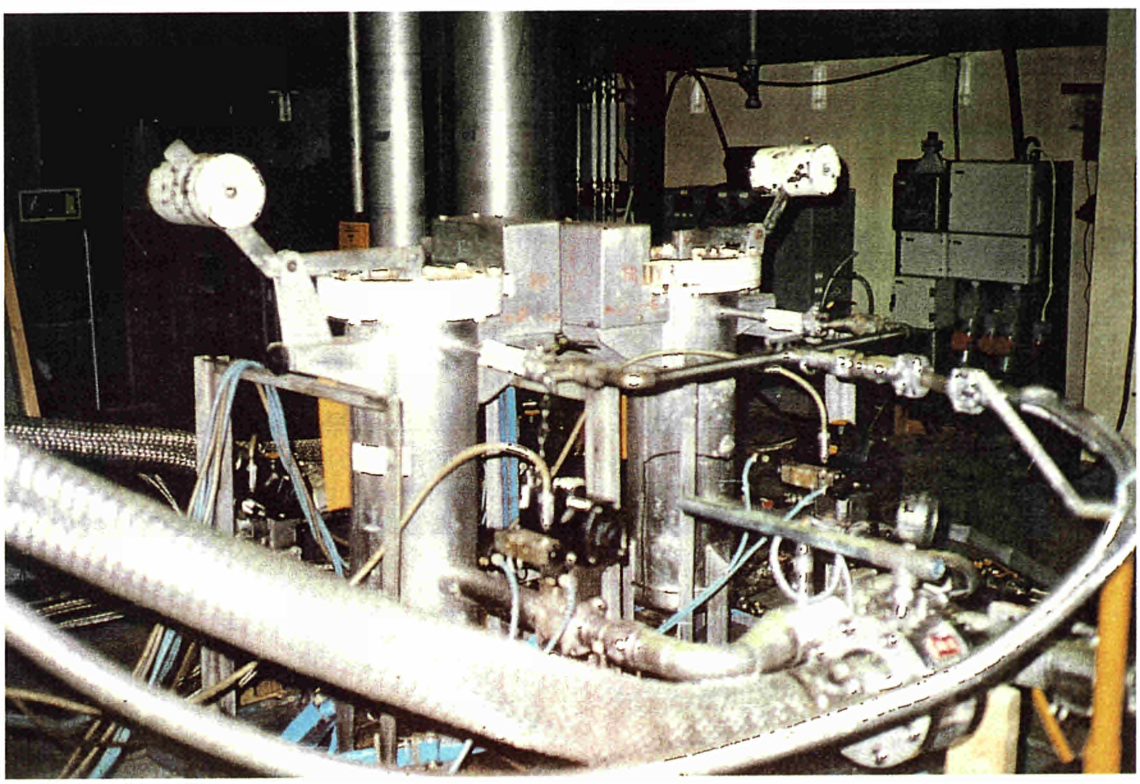


Figure 2: *View of the decontamination loop*

8.23. DECOMMISSIONING OF THE DRY GRANULATION PLANT USING MACHINE ASSISTANCE

Contractors:	BNF plc Sellafield
Contract No.:	FI2D-0071
Work Period:	December 1991 - December 1993
Coordinator:	S BUCK, BNFL Decommissioning
	Phone: 44/9467/75300 Fax: 44/9467/74040

A. OBJECTIVE AND SCOPE

In the course of decommissioning a mixed oxide fuel fabrication plant, the aim of this contract is to select, test and evaluate a range of equipment to minimise operator radiation uptake without incurring the costs or development delays of fully remote operation. The operating philosophy will therefore be for COntact Deployment, Remote Operation (CODRO). It will provide data and experience for the planning and implementation of subsequent plutonium plant decommissioning work in the United Kingdom and elsewhere in the Community.

Particular aspects of the project which will be used to explore plutonium facility decommissioning technologies will include:

- alternative inspection and remote surveillance techniques,
- investigation of size reduction and dismantling tools and methods,
- the development of the remote handling equipment required to carry out a wide range of tasks including dismantling, size reduction, handling of radiometric equipment and cleaning equipment such as vacuum cleaners and transferal of removed components to a packaging facility,
- keeping detailed records of machine operations and reliability and of manual intervention for maintenance, redeployment or inadequacies of the remote equipment;
- manpower, personal radiation uptake and waste arisings data will also be recorded for comparison with the fully manual operations carried out in the Co-precipitation Plant and other projects.

The plant in question was commissioned in 1975 as a development pilot plant to make mixed oxide (MO_2) granules for vibro-compacted fuel manufacture. It was subsequently modified, with additional shielding and a separate control cubicle, as the main production facility for the supply of MO_2 feed to the pellet presses of the Fast Reactor fuel element plant. It operated in this form until April 1988.

B. WORK PROGRAMME

B.1. Study of the overall decommissioning scheme
B.2. Detail design, planning and safety studies
B.3. Delivery and commissioning of remote equipment
B.4. Deployment and (active) operation on all appropriate tasks of a range of cutting and dismantling tools
B.5. Generation of specific data
B.6. Evaluation of the effectiveness of the CODRO philosophy and of the selected equipment.

Progress of Work and Obtained Results

Summary of Main Issues (B1,B2)

Initial investigations confirmed the plant status, followed by a Decommissioning Option Study which determined the selection of techniques and equipment for the dismantling of plant and equipment. High radiation, due to the ingrowth of Americium 241, high plutonium inventory and the statutory requirement to reduce radiation uptake to operators resulted in the study recommending the use of a mixture of manual and remote operations in line with the Contact Deployment Remote Operations (CODRO) philosophy.

A detailed Design and Engineering Study outlined the methodology and procedures for the removal of the plant and equipment in order to maximise the use of the remote handling machine. Parallel planning work included the approval of a Preliminary Safety Report, identification of remote tooling equipment and tool changing requirements and development of plasma arc cutting techniques. Tender Specifications have been prepared for the remote handling machine, tool changing equipment and plutonium measurement equipment for cut off pieces and for waste drums. During preparation of the detailed remote equipment engineering specification for tender purposes it has become apparent that the initial work programme had not allowed for an adequate development input for the electrical and instrument components. Consequently a revised programme has been included, lengthening the detailed design and manufacture phase by several months with equivalent delays to the practical phases. (Figure 1) However this does allow a completely integrated engineering and control system to be manufactured and tested which will result in cost and programme savings.

Progress and Results (B1,B2)

A Post Operational Clean Out (POCO) was completed by the plant operators prior to shutdown in 1988, and loose material and equipment removed. All services to the plant remain operational and radiation surveys were carried out to determine dose levels to operators and working times. Using the above information an Option Study examined the various approaches to dismantling and size reduction, concluding that a mixture of manual and remote handling methods should be deployed. (See Radiometric Status and Plutonium Inventory).

The detailed Design and Engineering Study outlined the methodology and procedures for the removal of the plant and equipment in order to maximise the use of the remote handling machine. The study produced drawings, process flowsheets and operating sequence diagrams for the removal of plant and equipment from both the ground and first floor area. (Figure 2) From arrangement drawings and a brief specification produced by the Design and Engineering Study, detailed design, engineering and electrical specifications have been produced for tender/manufacture purposes. Invitations to tender for the work will be issued to suitable manufacturers in early January 1993. (Figures 3 & 4)

Several months delay have been experienced due to the need to produce a complete tender specification for engineering design, viewing and control systems including identification of a suitable manipulator deployment arm and tool changing system compatible with the use of mechanical and/or plasma arc cutting equipment and the manipulator arm.

Detailed investigations of drum handling, waste posting locations, area ventilation and fume collection systems are well advanced in conjunction with plasma arc and mechanical cutting requirements. Tenders were issued for the manufacture and testing of a piece monitor to assess the plutonium content of size reduced items so that the 200 litre drum may be filled to near its maximum fissile material content and a drum monitor to produce a more accurate plutonium content prior to interim storage. A suitable manufacturer has been identified and pre-let discussions are underway.

A Preliminary Safety Report has been approved which identifies areas of operations where more detailed safety assessments will be required before practical operations can commence, such as use of plasma arc cutting equipment and installation and commissioning of the remote handling machine.

In parallel, work is progressing in developing the use of plasma arc cutting techniques with remote equipment in highly contaminated plutonium environments. A localised fume extraction and cleaning system was used in conjunction with plasma arc trials in a non-active "mock-up" of the first floor containment. The results were encouraging, the use of a proprietary bag filter and extract system has successfully removed all fume and particulate matter leaving the workplace free from fume during cutting operations. Active trials are due to commence in February 1993 using the same system but in a plutonium active environment, cutting pieces of a size-reduced glovebox contaminated with similar MO_2 mixtures and contamination levels expected during dismantling of DGPP, in order to confirm the fume collection data gathered during the non-active trials.

Radiometric Status and Plutonium Inventory

Glove box γ (gamma) rates average 1.5 to 2.0 mSv/hr with localised areas on the ground floor at 8.0 mSv/hr. Shielding had been added in the early 80's in an attempt to reduce dose rates to operators to acceptable levels due to the increase in radiation from the ingrowth of Am^{241} from Pu^{241}.

The 6 mm thick mild steel shielding proved to be very effective in absorbing the low energy of the Am^{241} γ (gamma) radiation (60Kev), reducing surface radiation levels of 1.5 mSv/hr unshielded to less than 50μSv/hr when shielded. The background level at 1 metre from an unshielded ground floor window panel is approximately 350μ Sv/hr.

From such readings it is obvious that manual dismantling would result in high radiation uptake to operators, especially when internal equipment is exposed during dismantling operations. This would not be in line with the ALARP principle and with the reduced annual radiation uptake now permitted to operators (currently Company Standard is 30 mSv with a rolling maximum of 75 mSv in 5 years with a divisional action level of 15 mSv) would require large numbers of people or a very long time scale to complete the project.

Recent measurements of the glove boxes for Plutonium content using the passive neutron coincident counting techniques developed as part of the previous CEC nuclear decommissioning programme Development Contract F11D 0027 has shown the first floor glove boxes to contain approximately 1.7 Kg Pu (5 Kg MOX) and the large grould floor glove box 7.5 Kg Pu (22 Kg MOX), a total of 9.2 Kg Pu (27 Kg MOX) even after the Post Operational Clean Out (POCO).

Because of problems of radiation uptake to operators exceeding the divisional action level of 15 mSv/yr annual limit using manual means, the study carried out recommended that a mixture of manual and remote operations should be used in line with the Contact Deployment Remote Operations (CODRO) philosophy.

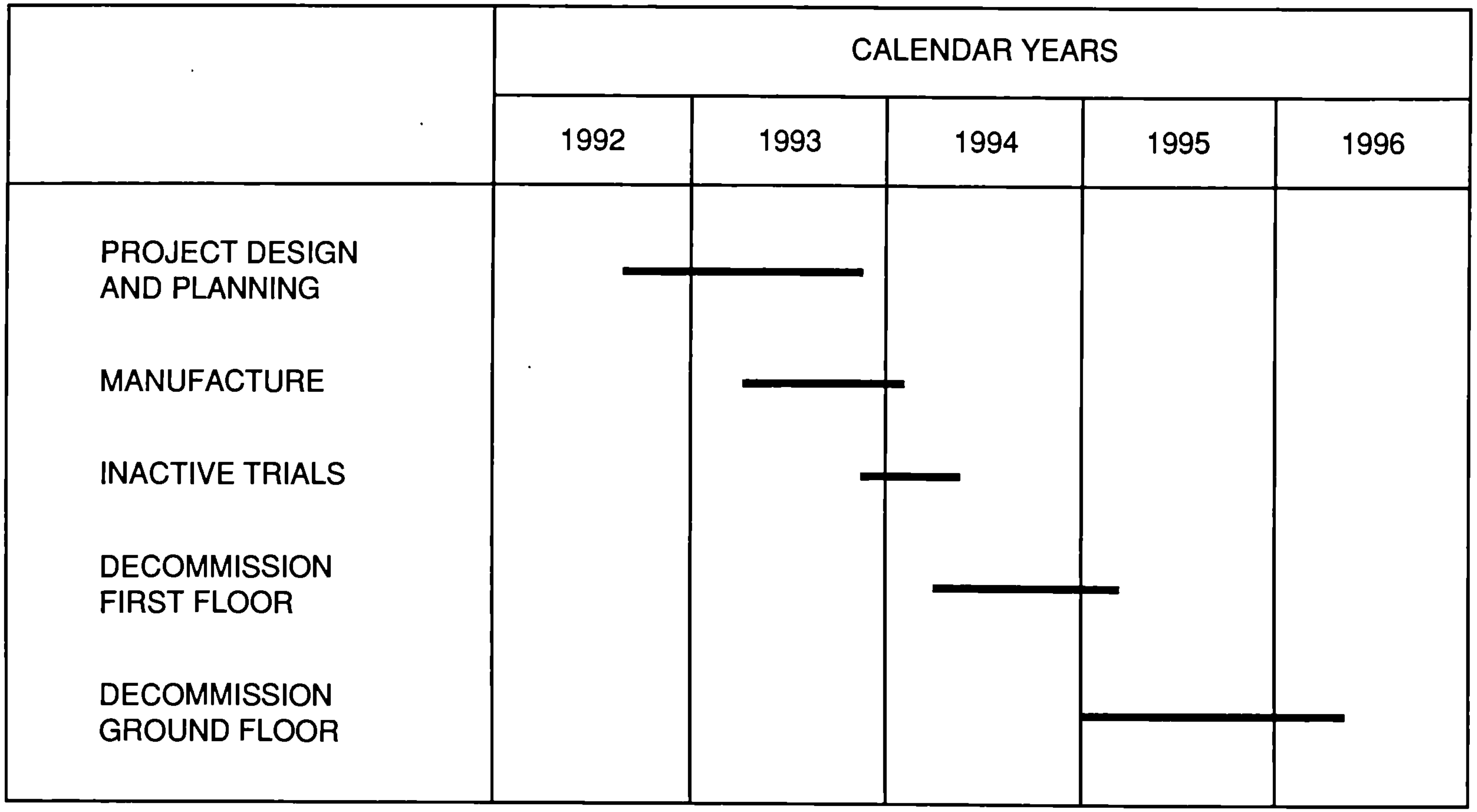


Fig. 1 DGPP Decommissioning Programme

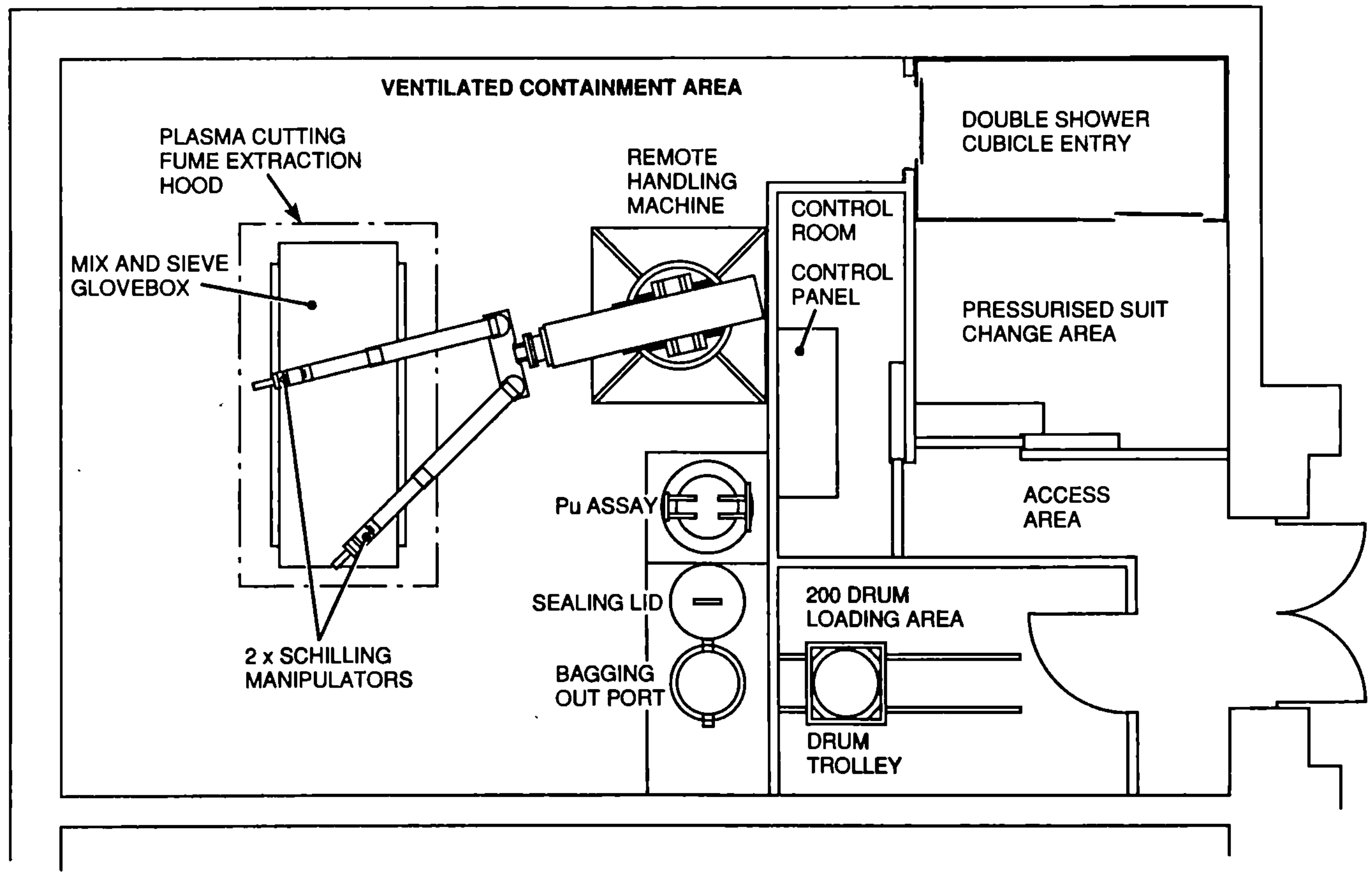


Fig. 2 Ground Floor Plan

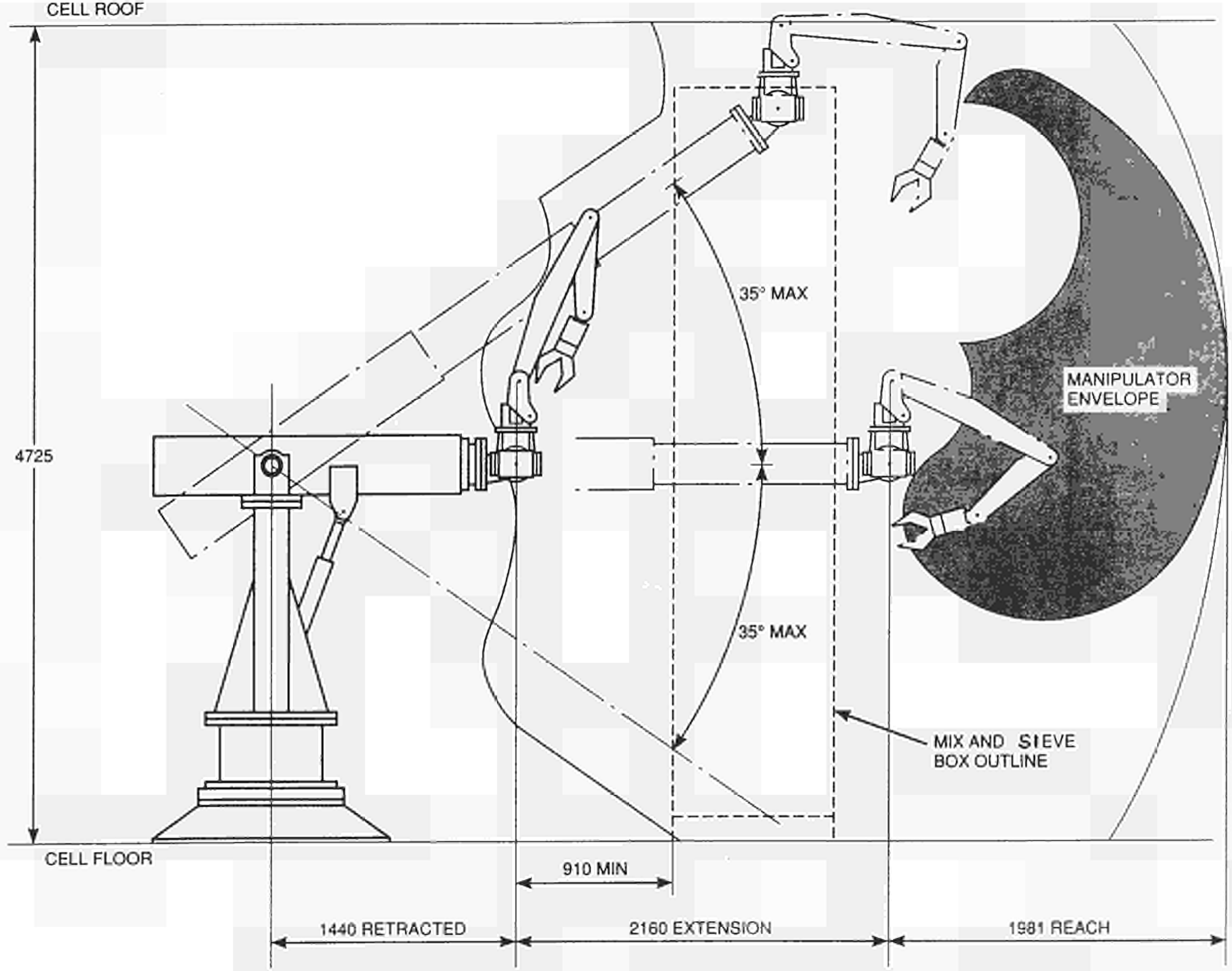


Fig. 3 Remote Handling Machine For DGPP Decommissioning

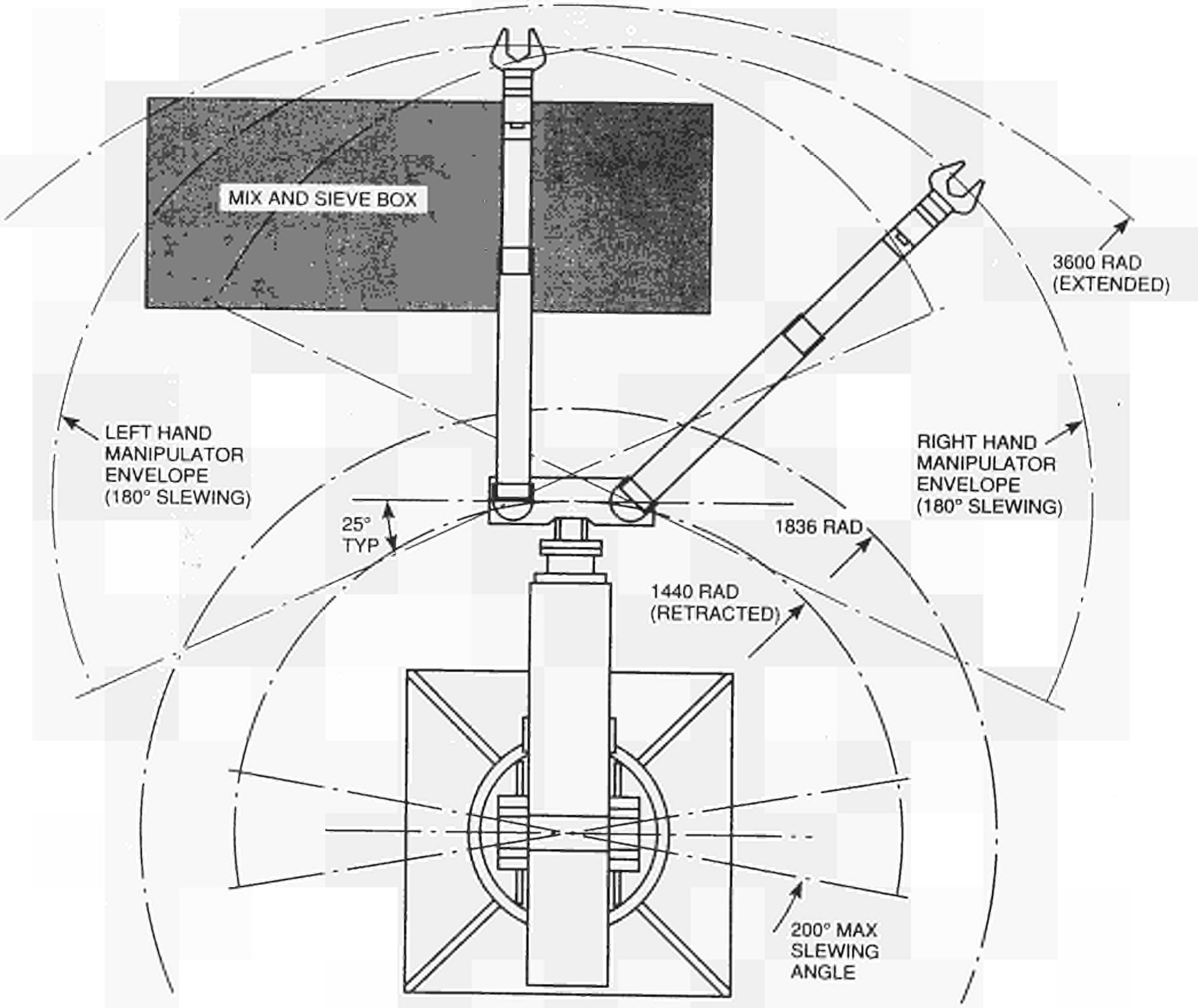


Fig. 4 Plan of Remote Handling Machine for DGPP Decommissioning

8.24. REMOTELY OPERATED UNDERWATER DISMANTLING BY PLASMA ARC AND CABLE SAW OF A STRUCTURE IN A COGEMA STORAGE POND

Contractors: COMEX NUCLÉAIRE, COGEMA, DIAMANT BOART
Contract No.: FI2D-0072
Work Period: January 1992 - December 1993
Coordinator: J BLIGHT, COMEX Nucléaire, Marseille
Phone: 33/91 23 50 00 Fax: 33/91 40 12 80

A. OBJECTIVE AND SCOPE

The objective of this project is to demonstrate the radiological, technical and economic feasibility of using divers to install remotely controlled cutting and decontamination equipment in a nuclear environment.

The demonstration consists in the underwater dismantling by plasma arc or mechanical tools of a number of stainless steel anti-shock mattresses installed in the bottom of the Pond N° 900 at COGEMA, La Hague after initial decontamination. The specific radioactive inventory is in the order of $3.7.10^3$ Bq/l, specific contamination is about 200 Bq/cm² beta and the dose rate is estimated in the range of 1-30 mSv/h. The contractual work will consist of the preparation of equipment and procedures followed by an industrial-scale dismantling operation.

The project will provide the information required to judge whether the use of divers represents a technical and economically viable alternative to either "dry" manual or telemanipulation techniques for the decommissioning of nuclear installations and will permit a definition of the conditions in which the use of divers would represent the optimal solution. The project will also result in a validation of diving equipment and procedures for nuclear environments.

The results of the study should be used for both decommissioning and maintenance operations for nuclear power generation and fuel reprocessing plants. The techniques to be demonstrated could be used for installations which are normally immersed or which could be immersed specifically to permit the use of divers during maintenance or decommissioning.
Typical decommissioning tasks would include decontamination, dismantling and assistance during removal.

Water is an effective biological shield and as such, intervention under water will permit a reduction in dose uptake when compared to manual intervention in an air atmosphere. Real time monitoring systems will ensure that the diver is subjected neither to contamination nor to excessive dose rates.

The project will be carried out by COMEX NUCLÉAIRE (CxN), COGEMA and DIAMANT BOART and will produce specific data on subjects such as dose levels (individual and total) and times and costs for specific tasks, which can be used in the planning of larger-scale decommissioning operations.

B. WORK PROGRAMME

B.1. Preparatory work

B.1.1. Modification of diver equipment, including safer leak-tightness and provision of real-time dose rate measurement with on-line calculation of integrated job dose at exposed parts of the diver, review and upgrade of existing procedures and regulations for divers in ionizing environment.

B.1.2. Implementation of comparative underwater cutting tests on flat plates with plasma arc and mechanical sawing, including the assessment of systems for collection of cutting waste and for an easier decontamination (CxN).

B.1.3. Assessment of cable sawing for metallic structures, also aiming at an application to components of the KRB-A reactor (CxN, Diamond Boart).

B.1.4. Development and testing of systems for the decontamination/cleaning of walls and bottoms of the storage pond by jetting or brushing (CxN).

B.1.5. Preparation of documents for the licensing and operation of all equipment at the storage pond for approval by COGEMA (CxN, COGEMA).

B.2. Execution of the dismantling work including the installation of the equipment, radiological mapping and pre-decontamination of the pool, and the conditioning and removal of dismantling waste and final decontamination of the pond (CxN, COGEMA).

B.3. Evaluation of results obtained with equipment for divers and for applied tools and procedures (all).

B.4. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings from the execution of items B.1., B.2. and B.3. (all).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

During 1992, all the activities relating to the preparation programme (B.1.) have progressed normally, different systems have been designed, while a few of them are still under construction.

The pilot operation (Section B.2.) has been delayed and will not take place in La Hague as initially planned; the operation will certainly be transferred to Marcoule.

Progress and results

1. Preparation (B.1.)

1.1. Diving-related activities (B.1.1.)

1.1.1. Modification of diving equipment

All the modifications to the diving helmet have been completed. The hard hat helmet equipped with the semi-closed gas circuit (Figure 1) has been tested in real conditions during diving operations on several nuclear sites (EdF-CEA), totalling more than forty hours of use, and all the results obtained meet the specifications in terms of security, efficiency and comfort.

Main advantages of the system:

- Total absence of risk of water ingress in the gas circuit and consequently, absence of risk of internal contamination for the diver. The air is rejected on the surface and not in the water through a relief valve located on the helmet shell.
- The air is delivered through a pressure regulator piloted by the water pressure, avoiding the noise generated by the ordinary free flow system which disturbs communications and fatigues the diver after a long time period.

It furthermore allows the diver to control more precisely his buoyancy when working in a lying position on the pool bottom.

Helmet main characteristics:

- Hard shell made of woven fibre glass and polyester resin,
- All metallic parts are made of stainless steel and chrome-plated brass,
- Equipped with two earphones and one microphone
- Weight in air: 10.6 kg.

Buoyancy in water: slightly negative.

1.1.2. Dose monitoring system

All the specifications of the dose monitoring system have been defined, including:

- Type and number of detectors
- Surface interface
- Structure of software
- Display arrangement.

Preliminary tests carried out with traditional GM detectors have not given satisfactory results. The detectors: key elements of the system are now made of silicon diodes. Only they are able to satisfy the requirements in terms of minimizing the size of the components and resistance to a harsh environment and demanding utilisation.

The detector has a response time that is now sufficiently rapid during exposure to measure low dose rates (2 seconds at 200 μGy/h).

Range of measurement: 0.01 Mgy/h to 50 Gy/h

Energy response: 60 KeV to 1.2 MeV.

Size of detector: 20 mm diameter x 85 mm length.

Subsequent work will be the development of the different interfaces, and the completion of the software programme to be integrated in the surface computer.

1.2. Cutting of flat plates sections (B.1.2.)

The choice of cutting tools for segmenting the thick metallic plates has been reconsidered. The arc saw was chosen for its advantages in terms of safety for the diver: low voltage, capability to cut very thick metal plates, and a possible extension for the cutting of loose and non-homogeneous structures, eg: small diameter tubes of heat exchangers.

The arc saw performances will be compared to those of mechanical cutting equipment. The 1200 A arc saw has been designed and its manufacturing is completed.

The carriage unit supporting the cutting tools and the rest of the ancillary equipment are now being fabricated (Figure 2). Welding generators and a hydraulic milling machine are to be delivered by mid-February and the preliminary tests are to begin in March 1993.

1.3. Cable cutting (B.1.3.)

Several types of cable and cutting methods were tested in 1992 for the cutting of metallic structures. Parts of these tests have been performed jointly with the development work required for the KRB-A project. Ordinary steel cable has been evaluated as a cutting tool: the system required a substantial length of cable (300 m) and the cable itself deteriorated quickly. This type of cable does not cut stainless steel components.

Cable using diamond beads has been evaluated during tests in Marseille by COMEX and in Belgium by DIAMANT BOART. The use of synthetic diamonds is the only way to cut stainless steel structures. Of the two types of beads tested, the model using diamonds attached to the bead by galvanic coating is the most effective. Of the two cutting methods tested, straight cable (band saw) or the wrap-around mode, the latter seems the most suitable for cutting structures with a limited site clearance. The drawback of this method as demonstrated during the tests is the great stress sustained by the beads when passing over the sharp edges of the structure to be cut. A new cable, able to cut without damaging the beads, is being developed in DIAMANT BOART's facilities.

A cable-cutting machine able to test the various cables and cutting modes on a mock-up representing the structure to be cut was fabricated (Figure 3).

1.4. Decontamination system (B.1.4.)

A decontamination machine used on pool side walls and bottom has been developed and constructed (Figure 4). The machine is operated by a diver, but can also be installed on a remotely-controlled vehicle.

The basic element is a rotating arm contained in a square metallic housing which is moved across the wall to be cleaned. At each end of the rotating arm, two HP-jetting nozzles directed toward the wall remove the thin layer of radioactive deposits, while by reaction two associated nozzles provide the rotation of the arm.

The inside of the box is kept de-pressurised by means of a pump, avoiding deposits from being dispersed in the surrounding water.

The whole system is supplied with energy from an HP water line at 350 bars. The tests of the machine in the inactive pool have been delayed due to mechanical problems on the HP water pump. Tests will resume in March 1993.

1.5. Preparation of documents for the pilot operation (B.1.5.)

This part of the programme will commence as soon as the nature of the pilot operation will be defined.

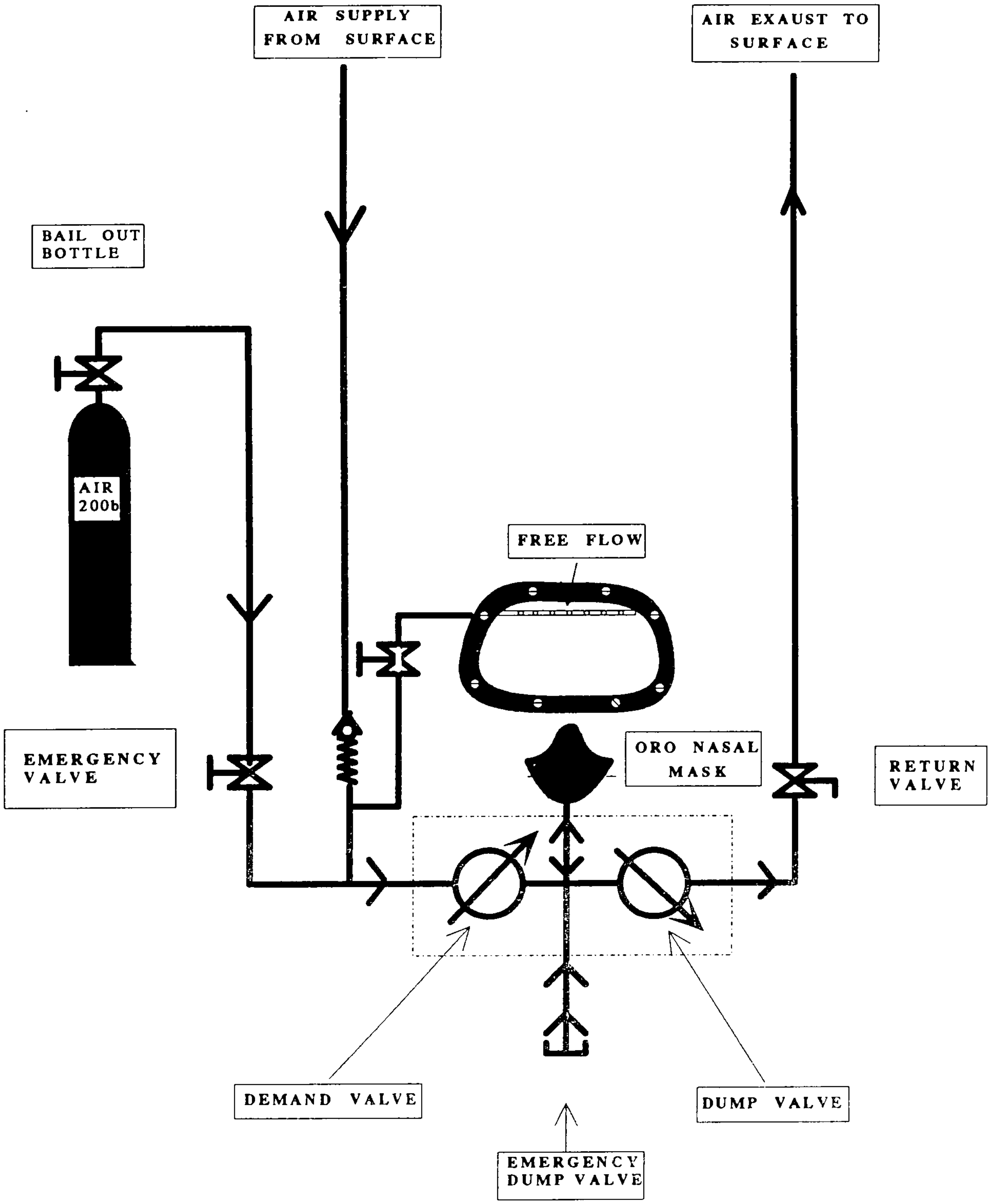
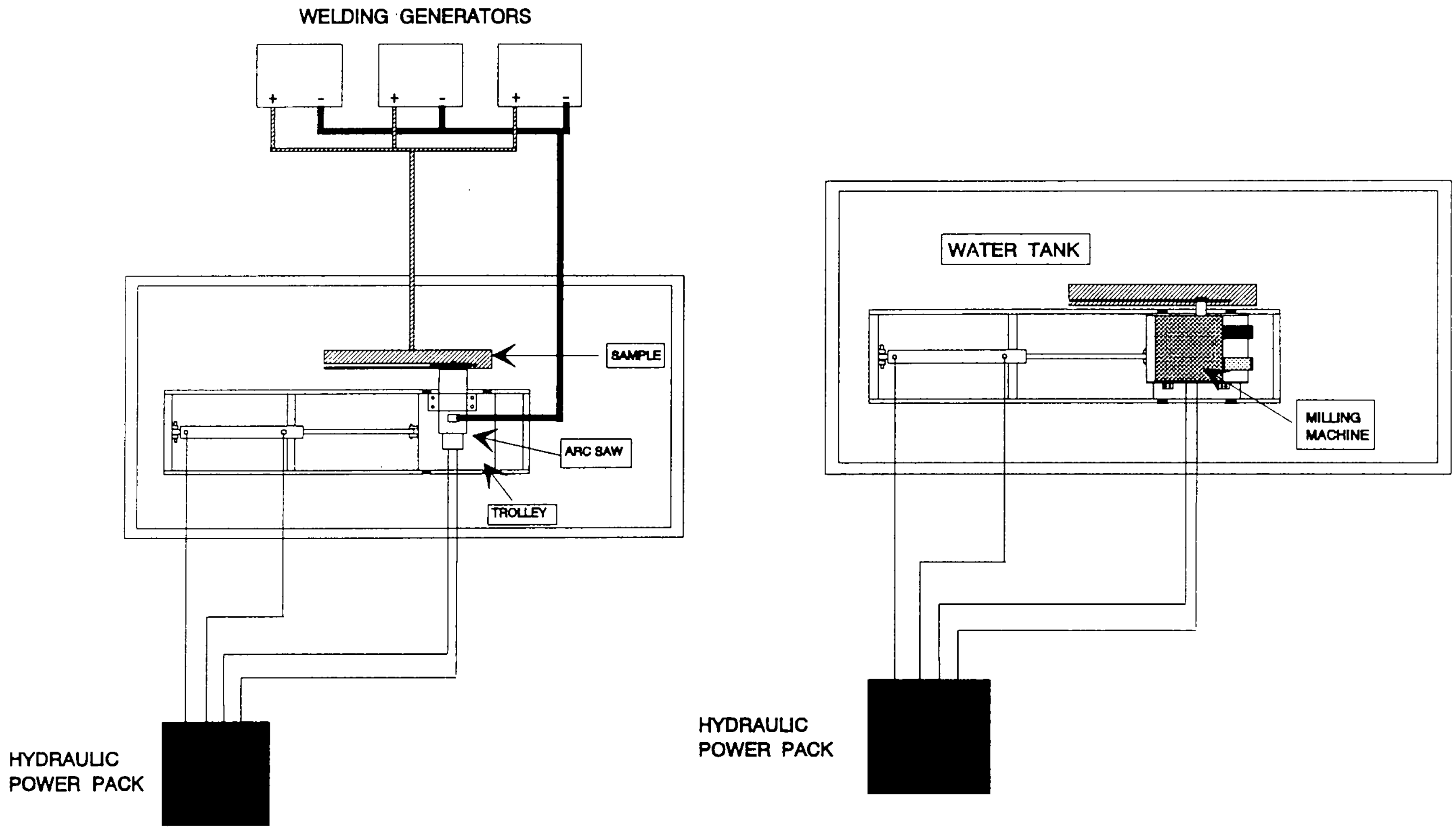

AIR SUPPLY FROM SURFACE
AIR EXAUST TO SURFACE
BAIL OUT BOTTLE
AIR 200b
FREE FLOW
EMERGENCY VALVE
ORO NASAL MASK
RETURN VALVE
DEMAND VALVE
DUMP VALVE
EMERGENCY DUMP VALVE

FIGURE 1


WELDING GENERATORS
+ -
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SAMPLE
ARC SAW
TROLLEY
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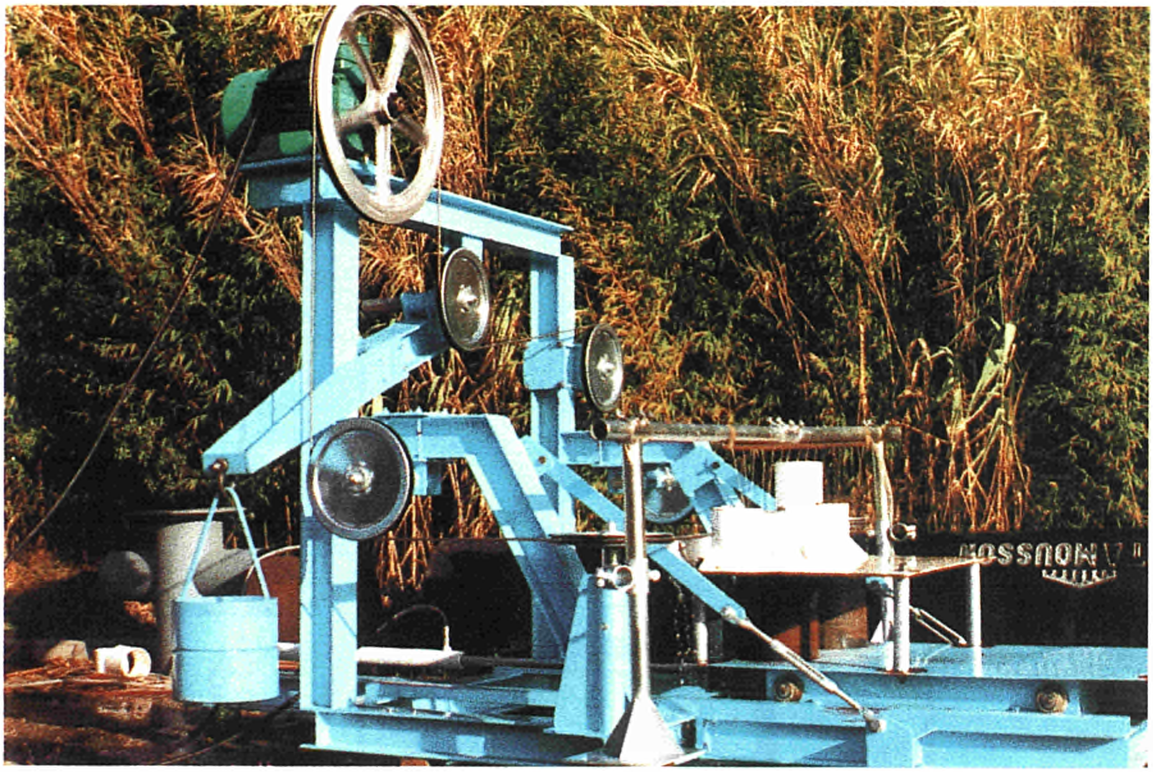
WATER TANK
MILLING
MACHINE
HYDRAULIC
POWER PACK

FIGURE 2

CABLE CUTTING MACHINE



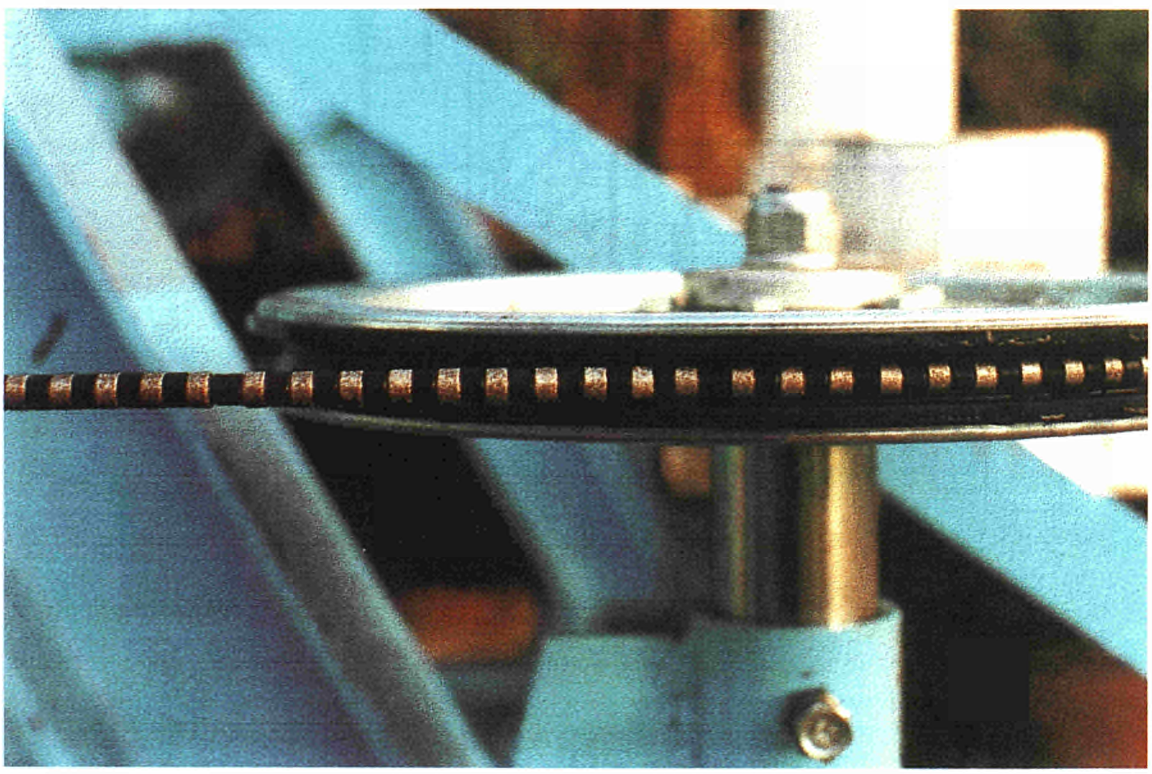


FIGURE 3

HP DECONTAMINATION MACHINE

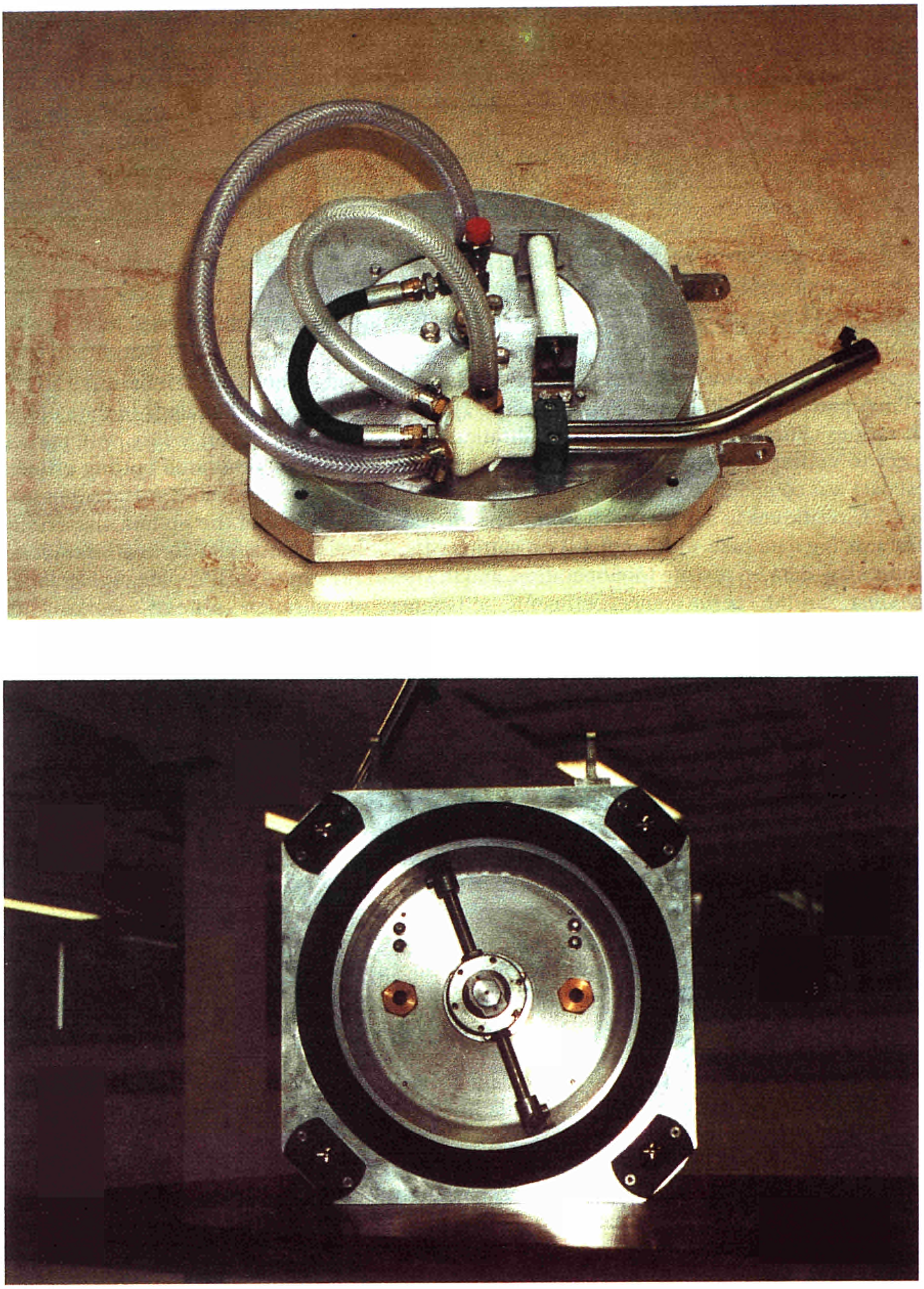


FIGURE 4

8.25. FURTHER DEVELOPMENT OF A DATA BASE ON CUTTING TOOLS AND ASSOCIATED FILTER SYSTEMS FOR DISMANTLING (Joint study)

Contractors:	Uni. Hannover, CEA-Valrhô, AEA Windscale
Contracts Nos.:	FI2D-0056, -0057, -0058
Work Period:	October 1991 - June 1994
Coordinators:	1) G SCHRECK, Uni. Hannover 2) J P RAVERA, CEA-Valrhô 3) S WHITE, AEA Windscale
	Phone: 1) 49/511/762 43 15 Fax: 49/511/762 52 45 2) 33/66 79 63 12 Fax: 33/66 79 64 32 3) 44/9467/72437 Fax: 44/9467/72409

A. OBJECTIVE AND SCOPE

In the framework of the 1984-1988 R&D programme, Universität Hannover (UH) and the CEA performed a joint study (FI1D-0070/71) for the collection and analysis of data obtained with various cutting tools and associated filtration systems in air and under water, with particular respect to cutting and filtration performance, type and amount of generated cutting effluents. Most data was compiled from experimental results obtained on non-radioactive metal components. The CEC continues to support such work. The envisaged work aims at the update and extension of the existing data base with emphasis on cutting of radioactive metal and concrete components including e.g. data on remote tool operation, and the development of an EC-wide usable database (including cutting performance, effluents, efficiency of filtration systems, working time, occupational doses [if relevant]).

B. WORK PROGRAMME

B.1. Development of an appropriate software for data storage and processing **for an EC user-friendly data base, based on commercial software (Uni. Hannover assisted by the partners)**
B.2. Collection of new data on cutting tools and associated filter systems
B.2.1. Assessment of the existing data sets and possible adjustment with a view to practical application, as well as definition of data sets for supplementary tools (all).
B.2.2. Collection and analysis of data produced in former and current EC research contracts (Uni. Hannover).
B.2.3. Collection and analysis of available data produced in France, not considered under B.2.2., and from Japan (CEA)
B.2.4. Collection and analysis of available data produced in the UK, not considered under B.2.2., and from the USA and Canada (AEA)
B.2.5. Collection and analysis of available data not considered under B.2.2. - B.2.4. (all)
B.3. Collection of relevant data relating to remote tooling applications on radioactive components **(AEA assisted by partners)**
B.4. Updating, treatment of collected data and incorporation into the data base (Uni. Hannover)
B.5. Data input (delivery to Uni. Hannover) starting on 01.03.92. **Data base updating will be two-monthly starting on 01.04.92.**
B.6. Definition of a self-supporting system for continuous and systematic updating of the data base **after the end of the present contract (all).**

C. Progress of work and results obtained

Summary of main issues

A second version of EC-DB-TOOL has been developed and distributed to the project partners. This version contains a structured database and an application for data input and retrieval. The database is designed to give maximun flexibility. This allows the addition of new cutting techniques or special demands without changing the basic structure of the database. The application uses the same screens for data input and retrieval. Work has also commenced on the improvement of data retrieval by the addition of screens for query design and the ability to compare results from a number of independent queries.

The collection of new data is continuing. The data are derived from progress and final reports of EC-contracts, worldwide literature reviews and from directly addressing experts by means of data questionnaires. An improved version of the sheets has been produced. These will be distributed to the C-group members to provide the actual data.

AEA has produced a video presentation to highlight the aims and objectives of the project. In support of this two posters have been designed to promote the EC databases on Tooling and Cost under development.

Progress and results

1. Development of the database structure (B.1.)

The basic structure of EC-DB-TOOL is complete. The conceptional view is divided into two levels. The first level contains data common to a series of tests, e.g. the equipment and references. The second level incorporates data relating directly to a cutting task, divided into different sections. Whereas one section deals with common data, referring to any cutting process, like cutting speed, other sections include more specific data and parameters for each cutting technique.

A detailed classifying thesaurus system has been developed to ensure that data is comparable and the complete query results will be achieved.

If the amount of some data is varying or unknown, e.g. gas supply, lists are used to have as much data as needed.

The chosen structure of the database allows the database to be developed by adding new cutting techniques using existing common tables and defining their special parameters. In this respect the application contains a screen for proposals to new data that will be considered in the next update.

The implementation of the EC-DB-TOOL is based on the Oracle relational database management system. The EC-DB-TOOL has been developed as a user-friendly application containing the same screens for input and retrieval of data (Fig. 1). For advanced retrieval, work has also been commenced on development of screens for query design.

EC-DB-TOOL has been distributed to the project partners. Version 6.0 of Oracle has been installed on a PC at AEA. The database has been successfully installed onto the computer and familiarisation work is currently in hand.

2. Collection of new data (B.2.,B.3.)

The collection of new data will be carried out from three sources:

1. Data produced in the former and current CEC R&D programme on decommissioning of nuclear installations will be analysed and incorporated into the data base. It

will be mainly taken from the progress reports and the final reports of each contract.

2. A worldwide literature review shall give information on current dismantling activities and will be done by the partners. This includes data relating to testing of tools, techniques and handling systems which are of interest for decommissioners.

3. Experts and investigators will be addressed directly by data questionnaires . The addressees are asked to fill in these forms as thoroughly as possible, having the opportunity of making comments beyond the fixed questions, and then to send them back to one of the three contractors involved in developing the data base for incorporation into EC-DB-TOOL.

These data questionnaires have been revised and new sheets produced e.g. electro-chemical cutting and electro-discharge machining have been added. To guide the users, an introduction page with instructions for the completion of the data sheets has been written and produced.

There are two types of data sheets:

1. Data sheets of general type independent of the cutting technique used e.g. the description of the dismantling task (Fig. 2) or the contact person and literature. Some sheets of this type are optional, e.g. the sheets to describe data concerning to emissions and filters.

2. Data sheets on specific cutting techniques. For each cutting technique a sheet containing the parameters of this technique has been developed.

The common layout of the sheets is made of sections which allow the brief and homogeneous description of the different cutting techniques. These sections describe the cutting equipment, the power supply and cutting parameters. Additionally the comment field offers the possibility to make remarks or add additional data.

At the top of all sheets is the sheet number to identify the sheet and mark the type, the title (e.g. "G-1 Dismantling task " for the first general type data sheet) and a field for the task reference number. This can be any number to identify the task and put all related sheets together. The revised sheets will be distributed to the C-group shortly. Comments on the presentation and parameters (contents) will be checked and integrated in the next sheet distribution.

The collection of new data is continuing with a significant quantity of material from the USA being collected from AEA. Data relating to hot dismantling operations carried out in AT -1 facility (CEA) has also been collected. This data deals with the remote cutting of a reinforced concrete wall by means of a diamond saw and the superficial layer removal of concrete walls using a steel shot blasting device. Detailed data concerning to generalities like working site, environment, handling and filtration are included together with specific data of the cutting process e.g. cutting speed, temperature.

As a part of the programme for obtaining information, AEA has produced a short video presentation which highlights the aims and objectives of the project. It gives an overview of the current software package and takes the user through a selection of screens and inquiries.

Modification to the two CEC posters have been completed and comments from project partners taken into account prior to final production of the posters.

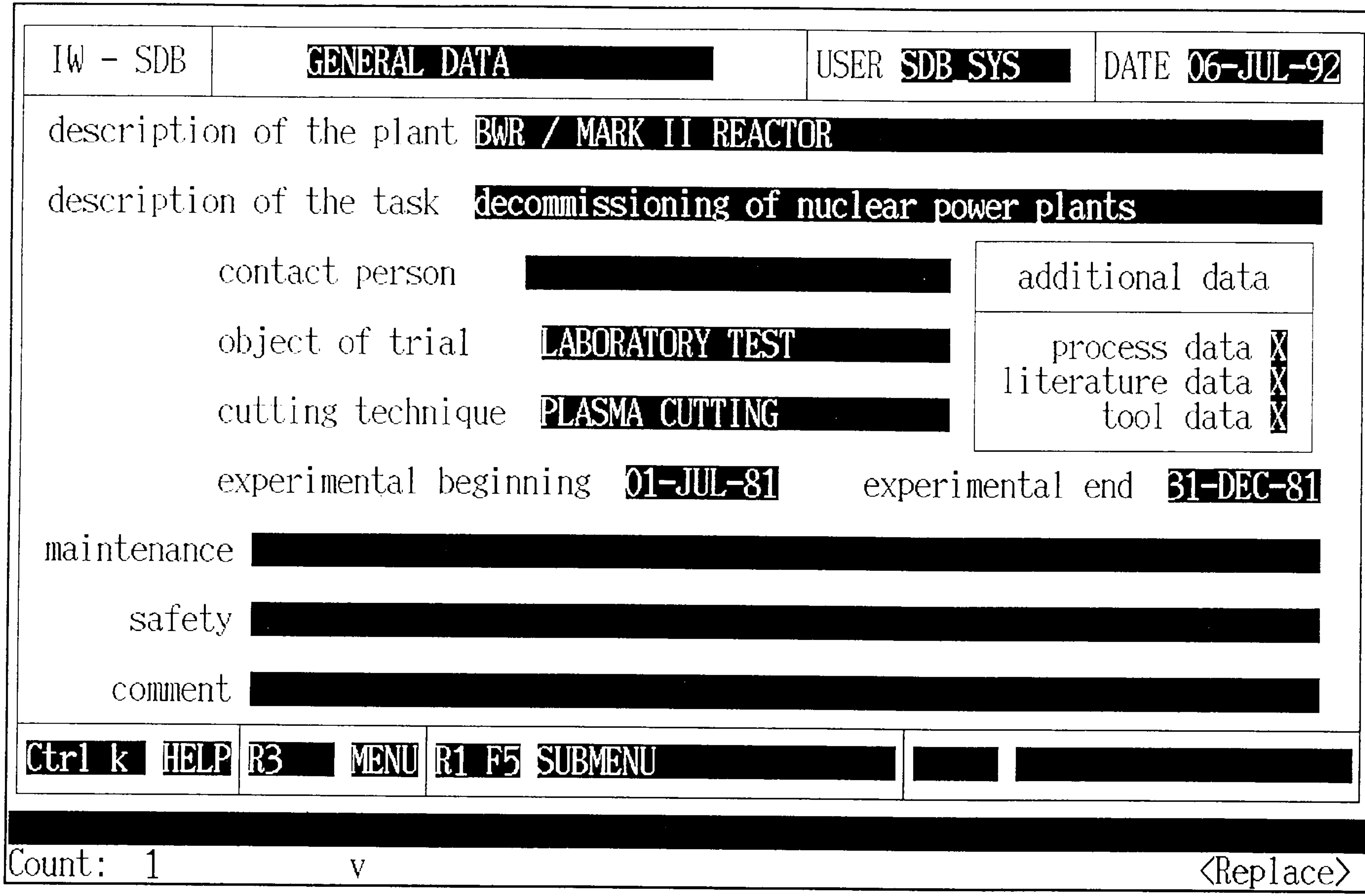


Fig. 1. Sample screen of the EC-DB-TOOL

G-1	DISMANTLING TASK	Task ref. No.:

Workpiece:

mock up [. .]   real component [. .]

origin: . . . .
radiation load (value {unit}): . . . .

time for installation of cutting equipment:. . . . . [man h]
material (standard and/or composition): . . . .

plating:	no [. .]	yes [. .]	plating material:.	thickness:. . . . . [mm]
coating:	no [. .]	yes [. .]	type of coating:	thickness:. . . . . [mm]
reinforced	no [. .]	yes [. .]	reinf. material:.	thickness:. . . . . [mm]
			mesh size:.	no. of layers:. . . . .

form of component (plate, tube etc.):. . . . .
dimensions:
material thickness:. . . . . [mm]
diameter of plate:. . . . . [mm x mm] (l xb)
diameter of tube: . . . . [mm] (outer) . . . . [mm] (inner)

other details of the workpiece / sketch (optional) on separate page [. .]:

Working site:

laboratory [. .]   in situ [. .] , (where ?). . . . .

environment:

atmosphere: [. .]   under water: [. .]

size of operating room:	L. . . . . [m]		size of pool/basin:	L. . . . .[m]
	W . . . . [m]			W. . . . .[m]
	H. . . . . [m]			H . . . .[m]
	V. . . . . [m^3]			V . . . .[m^3]
	temperature . . . . [^{0}C]		operational waterdepth	. . . .[m]
	pressure . . . . [Pa]		pressure:	. . . .[Pa]
water quality:	before cut:		after cut:	. . . .
conductivity:	before cut: . . . . [μS],		after cut:	. . . . [μS]
ph:	before cut:		after cut:	. . . .

other details of the working site / sketch (optional) on separate page: [. .]

Fig. 2. Data sheet of general type (Dismantling Task)

8.26. COLLECTION AND TREATMENT OF SPECIFIC DATA GENERATED IN LARGE-SCALE DISMANTLING OPERATION WITH A VIEW TO THEIR USE FOR COST AND DOSE ESTIMATES (Joint Study)

Contractors: NIS, CEA-Valrhô, BNF
Contracts Nos.: FI2D-0059, -0060, -0061
Work Period: October 1991 - June 1994
Coordinators: 1) P PETRASCH, NIS
2) J M DUFAUD, CEA-Valrhô
3) C L WALTERS, BNF Sellafield
Phone: 1) 49/6181/10 94 58 Fax: 49/6181/12 00 33
2) 33/66 79 63 14 Fax: 33/66 79 64 32
3) 44/9467/76427 Fax: 44/9467/27383

A. OBJECTIVE AND SCOPE

Taking advantage of the orientation of the present R&D programme to large-scale dismantling operations in different types of installations and considering the need to dispose of data obtained under realistic conditions for cost calculations concerning large-scale dismantling operations, the CEC supports work concerned with the collection and treatment of specific data on costs for unit operations, on associated radioactive job doses, on working time and on waste arisings to be derived mainly from all contracts in Section C, where the generation of specific data is a mandatory task (particularly from pilot projects).
Work will take advantage of the methodology developed by NIS and CEA in former joint research contracts FI1D-0074/75 (EC programme 1984-88).

B. WORK PROGRAMME

B.1. Development of an EC-user-friendly data base package based on commercial software for data storage and processing (NIS assisted by the partners)
B.2. Collection and analysis of data on costs and working time for unit operations, on radioactive job doses and on waste arisings generated during former and in current EC contracts (mainly in Section C and Pilot projects) in Germany and elsewhere (NIS)
B.3. Collection and analysis of relevant data not considered under B.2. and generated in France (mainly data related to GCRs of UNGG-type and fuel cycle installations) and from Japan (CEA).
B.4. Collection of other relevant data not considered under B.2. and generated in the UK (mainly data related to AGRs, Magnox and fuel cycle installations) and from the USA and Canada (BNF).
B.5. Continuous updating, treatment and incorporation into the data base of all collected data (NIS) including identification of relevant cost indexes.
B.6. Definition of a self-supporting system for continuous and systematic updating of the data base after the end of the present contract (all).

C. Progress of work and results obtained

Summary of main issues

The data base on decommissioning cost and radiation exposure, entitled DB - COST, is being developed and implemented by NIS Ingenieurgesellschaft mbH in Hanau assisted by Commissariat à l'Energie Atomique in Marcoule and British Nuclear Fuels at Sellafield.

The development started in December 1991 with the selection of the database system ORACLE. The selection considers other existing data bases in the EC R + D programme.

The main issues on the database development are:
- specification of the needed information
- collecting decommissioning data from EC contractors
- development of a data base structure
- development of a user concept and a concept for updating and continuous enlargement
- collection of decommissioning data out of the EC R + D programme.

Progress and results

B.1. Development of an EC-user-friendly data base package based on commercial software for data storage and processing.

The selected data base system ORACLE was installed in the Version 6.0.31 for MS-DOS on a PC with the specification:
- type 486
- 8 MB working memory
- 120 MB hard disk.

The structure of the DB-COST is shown in Fig.1. It consists of four main parts:

Part 1 includes the detailed decommissioning work information organized in working packages and working steps. The main items are:

- Content of a working step	- description of the work
- Source of information	- real work, theoretical study, publication, congress
- For real projects	- name of nuclear installation, type, power or capacity
- For publication	- author, meeting, congress, registration
- The working data	- personnel requirements, duration, radiation data, equipment, consumables
- The waste data	- component, kind of component, mass at the beginning of the work, masses after dismantling and treatment, radioactive waste, reusable material, secondary waste

Part 1 will be handled on four screen menus.

Part 2 includes the decommissioning work information related to a reference value. The reference value could be the mass of a component, a volume of waste or the surface area of a wall. For each working step specified in part 1 a reference value will be defined. The related information will be given in the following table:
- Personnel requirements in man-hours per kg, m^3 or others
- Dose rates in man-hours per kg, m^3 or others
- Equipment cost in ECU per kg, m^3 or others (or other national currency)
- Consumable cost in ECU per kg, m^3 or others (or other national currency)
- Amount of secondary waste, cost for treatment and disposal of secondary waste per kg
- Partitiation of the dismantled component if the reference is a component
- Amount of waste containers for disposal or final storage

Part 3 includes general information about the decommissioning in the different countries. This general information is necessary to compare and extrapolate cost data coming from one country to another country. Such general data are:

- Boundary conditions for decommissioning, e.g.
  - . requirements of final storage facilities (limits,containers)
  - . personnel dose limits for work in nuclear areas
  - . limit values for release of radioactive materials
  - . limit values for the recycling of radioactive material
  - . licensing requirements, rules, acts
  - . daily, monthly working time
- National decommissioning projects
  - . name of project
  - . type of decommissioned nuclear installation
  - . power or capacity
  - . status of decommissioning project (decommissioning stage, date of shut-down, date of finishing the project, removed masses and activity, manpower and cost up to now, collective dose up to now, location of the project and contact address)
  - . information of the nuclear installations of the different countries, number and types of nuclear installations
  - • Currency and economic data of a country, inflation rate values, for extrapolation of past data
  - . Contacts and addresses if any EC-research projects are performed in a country

Part 3 will be handled on three screen menus.

Part 4 includes a handling system for querying the detailed working step information in part 1. This thesaurus system is based on the EC study " Methodology to collect data on decommissioning cost and occupational radiation exposure ".

The main concept of the database part 4 is to list all specified and stored working steps in DB-COST. If the needed information is selected the detailed information coming from part 1 will be displayed.

B.2/3/4. Collection and analysis of data on costs and working time for unit operations, on radioactive job doses and on waste arisings generated during former and in current EC contracts

Mainly the EC research contractors are the source of the detailed decommissioning data. Three data collection forms have been generated, each dealing with specific types of decommissioning operation.

Type 1 - working steps without radiation exposure e.g. planning, documentation, licensing procedure etc.
Type 2 - decommissioning tasks within the controlled area e.g. dismantling, decontamination etc.
Type 3 - tasks relating to final storage, reuse or recycling of wastes

Copies of the data sheets have been distributed to all contractors of Section C. The filled in data sheets will be received by the EC with the semiannual reports on the second half year of 1992.

Information coming from other decommissioning projects are also of interest. The DB-COST group contacted the projects listed in TABLE 1.

B.5. Continuous updating, treatment and incorporation into the data base of all collected data including identification of relevant cost indexes

Task B.5. is relevant after development of the DB-COST and related to B.6. An issue is given in the year 1993.

B.6. Definition of a self-supporting system for continuous and systematic updating of the data base after the end of the present contract

Several concepts for using the DB - COST were discussed in the database group. The proposals are:

- Central concept: The DB - COST will be operated at a central station. An one-line connection is possible. An alternative is to contact the operator by letter or by phone and he will reply by a hardcopy or a short report. Additionally a mobile PC (Notebook or Laptop) could be used.
- Decentral concept: Several databases work at different places, e.g. on the decommissioning project sites.

The main problem of the future operation is the responsibility for:

- continuous and systematic updating or inputting new data
- updating and improving the handling system
- instructing and training of "user-operators" in case of a decentralised concept, because in a central concept a minimum of four central DB - stations is needed:
  - EC in Brussels
  - NIS in Hanau
  - BNFL in Sellafield
  - CEA/UDIN in Marcoule

  If a decentralised concept is preferred, the number of databases in operation would increase.

  The discussion is not concluded, NIS prefers a central DB-COST station.

TABLE I: Contacts to collect decommissioning data

Germany	United Kingdom	France
NPP Niederaichbach Research reactors MZFR, KNK II, HDR NPP Lingen NPP Rheinsberg NPP Greifswald THTR 300 Research reactor AVR	University Research Reactor Windscale Pile Chimneys Storage Pond Windscale Piles Fuel Storage Pond Dry Waste Silo Sludge Settling Tank Fuel Fabrication Laboratory PU Recovery Plant PU Fuels Development Facility Winfrith Uranium Processing Plant Fuel Examination Facility Full Scale Reprocessing Facility Caesium Production Facility Enrichment Plant Capenhurst	Atelier de Retraitement AT1 La Hague Laboratoire de Radio-Metallurgie "RM2" - Fontenay-aux-Roses Réacteur à Neutrons Rapides "Rapsodie" - Cadarache Réacteurs Graphite-Gaz "G2-G3" - Marcoule Cellules chaudes "ORIS" - Saclay

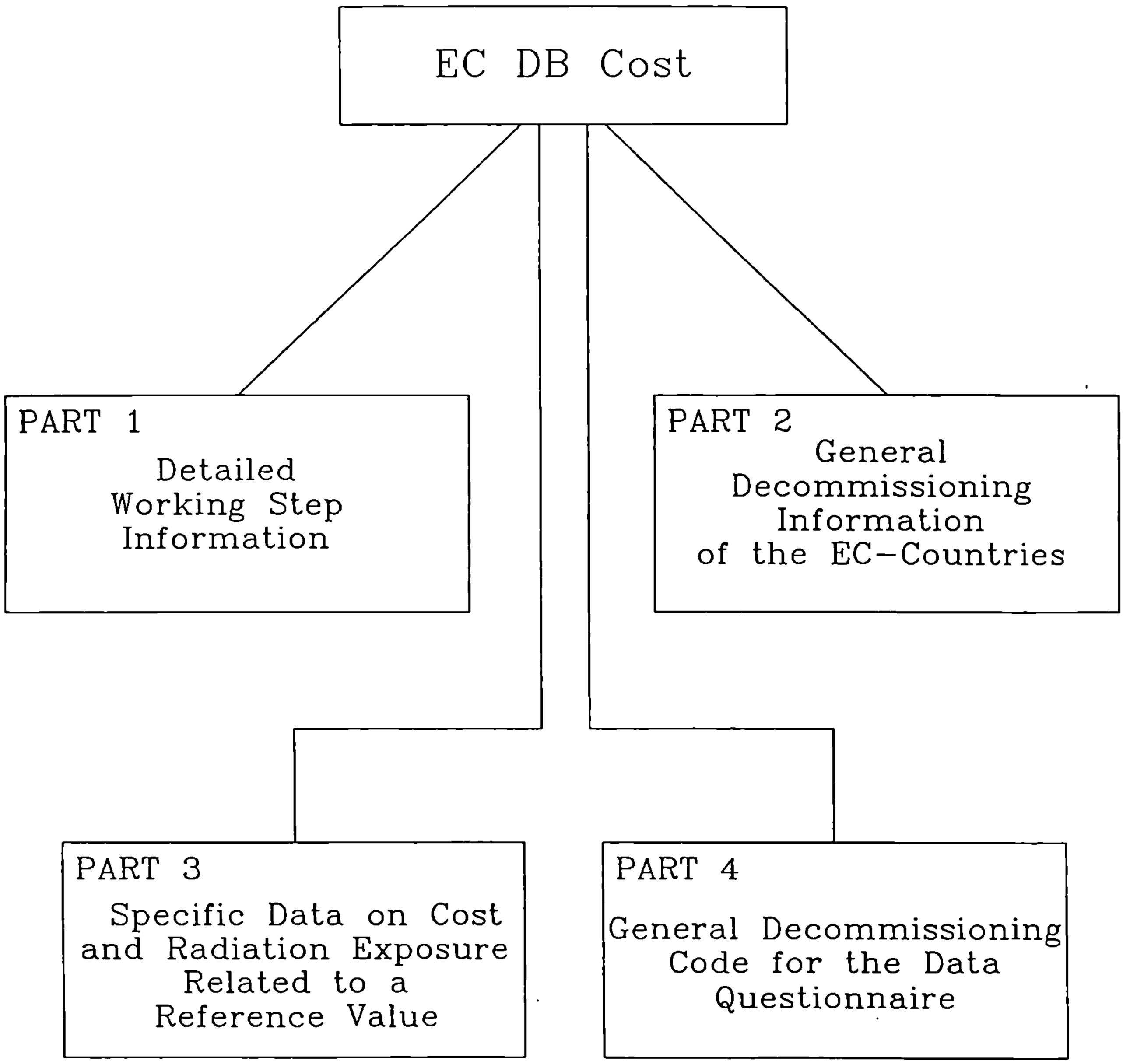


Figure 1: Organisation of EC DB-COST

8.27. SURVEY OF DECOMMISSIONING REQUIREMENTS FOR VVER REACTORS

Contractors: EWN, IND
Contracts Nos.: FI2D-0082
Work Period: December 1992 - February 1994
Coordinators: C FRANK, EWN
Phone: 49/38354/48 80 10 Fax: 49/38354/220 20

A. OBJECTIVE AND SCOPE

The present study should establish a sound basis to judge future R&D needs for the decommissioning of East European nuclear facilities.

Emphasis should therefore be put on VVER reactors due to large numbers of reactors of this type in Germany and several Eastern European countries, and the recognition that lack of certain safety features leads to the requirement to decommission the older type VVER 440 V 230 as soon as possible. Four of these reactors have already been closed down in Germany.

B. WORK PROGRAMME

B.1. Review of East-European nuclear facilities and selection of reference plant (IND)
B.2. Technical description of reference plant (EWN, IND)
B.3. Decommissioning of reference plant (Strategy, decontamination, dismantling procedures) (EWN, IND)
B.4. Comparison with West European experience (Strategy, decontamination, dismantling, spent fuel cycle) (IND, EWN)
B.5. Identification of R&D requirements (EWN, IND)

C. PROGRESS OF WORK AND RESULTS OBTAINED

No significant work was performed in this just starting contract.

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**.

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 Schilds im stillgelegten Kernkraftwerk Lingen. **EUR 8801**.

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

Eickelpasch, W, et al. (1984). Die Aktivierung des biologischen Schilds 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.

Boulitrop, 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 J 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.

Larcombe, M H E, Halsall, D R (1984). Robotics in nuclear engineering. Graham & Trotman Ltd., London. EUR 9312.

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 Schilds. 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.

Barody, 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). Mise au point et essais d'une méthode pour le revêtement de déchets métalliques contaminés, par des résines thermodurcissables. **EUR 9666.**

Migliorati, B, et al. (1985). Smantellamento di componenti metallici e di strutture in calcestruzzo mediante raggio laser. **EUR 9715.**

Fleischer, C C (1985). A study of explosive demolition techniques for heavy reinforced and prestressed concrete structures. **EUR 9862.**

Wieling, N, Hofmann, P J (1985). Erosionskorrosionsversuche mit kobaltfreien Werkstoffen. **EUR 9865.**

Antoine, P, et al. (1985). Intégrité à long terme des bâtiments et des systèmes. **EUR 9928.**

Lewis, G NH (1985). Degradation of building materials over a lifespan of 30-100 years. **EUR 10020.**

Hasselhoff, H, Seidler, M (1985). Anlage zum Einschmelzen von radioaktiven metallischen Abfällen aus der Stillegung. **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). Systems for remotely-controlled decommissioning operations. Graham & Trotman Ltd, London. **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.**

Bargagliotti, A, et al. (1986). Plasma arc and thermal lance techniques for cutting concrete and steel. **EUR 10402.**

Lasch, M (1986). Entwicklung von wirtschaftlichen Dekontaminationsverfahren. **EUR 10519.**

Hulot, M, et al. (1986). State-of-the-art review on technology for measuring and controlling very low-level radioactivity in relation to the decommissioning of nuclear power plants. **EUR 10643.**

ANNEX II

LIST OF PUBLICATIONS RELATING TO THE RESULTS OF THE 1984-88 PROGRAMME ON THE DECOMMISSIONING OF NUCLEAR INSTALLATIONS

A. Annual Progress Reports

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - First Annual Progress Report (year 1985)", **EUR 10740**, 1986.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Second Annual Progress Report (year 1986)", **EUR 11112**, 1987.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Third Annual Progress Report (year 1987)", **EUR 11715**, 1987.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Fourth Annual Progress Report (year 1988)", **EUR 12338**, 1989.

B. 1989 European Conference

Pflugrad, K., et al (ed). Decommissioning of nuclear installations. Proceedings of an international conference held in Brussels, 24-27 October 1989, Elsevier, London, UK. **EUR 12690.**

C. Final Contract Reports

Janberg, K. (1987). Economic comparison of management modes for contaminated metal scrap. **EUR 11149.**

Allibert, M. et al. (1987). Séparation par transport en phase vapeur des constituants d'aciers inoxydables. **EUR 11296.**

Ahlfänger, W. (1988). Vollständige Dekontamination einer Primärdampfleitung des Kernkraftwerks Lingen. **EUR 11435.**

Ebeling, W. et al. (1989). Untersuchung und Optimierung von Filtersystemen zur Abscheidung von Stäuben und Aerosolen bei der Dekontamination von Betonoberflächen. **EUR 11995.**

Gray, P W C (1989). The assessment of low-level contamination from gamma-emitting radionuclides. **EUR 12183.**

Hills, D L (1989). The removal of concrete layers from biological shields by microwaves. **EUR 12185.**

Lewis, C J A, Vibert, C J T (1989). Adaptation of a robot and tools for dismantling of a gas-cooled reactor. **EUR 12186.**

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"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - First Annual Progress Report (year 1990)", EUR 14227, 1991.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Second Annual Progress Report (year 1991)", EUR 14498, 1992.

B. Seminars

Skupinski, E, et al. Second Seminar on practical decommissioning experience with nuclear installations in the European Community, Sellafield-Windermere, 25-26 September 1991, EUR 14363.

Skupinski, E, et al. Third Seminar on practical decommissioning experience with nuclear installations in the European Community, Gundremmingen-Günzburg, 24-25 June 1992, EUR 14879.

Pflugrad, K, et al. Fourth Seminar on practical decommissioning experience with nuclear installations in the European Community, Mol, 6-7 May 1993, EUR 15099.

C. Final Contract Reports

Adler, D, Petrasch, P (1993). Kosten für die Stillegung von Kernkraftwerken mit Leichtwasserreaktoren in Deutschland. EUR 14687.

Jouan, A, Roudil, S (1993). Assainissement final de l'installation prototype de vitrification PIVER - Décontamination de la cellule chaude. EUR 14764.

Adler, D, Petrasch, P (1993). Decommissioning costs of Light Water Nuclear Power Plants in Germany. EUR 14798.

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European Communities — Commission

EUR 15262 — The Community's research and development programme on decommissioning of nuclear installations (1989-93) — Annual progress report 1992

Luxembourg: Office for Official Publications of the European Communities

1993 — X, 393 pp., num. tab., fig. — 21.0 x 29.7 cm

Nuclear science and technology series

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This is the third annual progress report of the European Community's programme (1989-93) of research on decommissioning of nuclear installations. It shows the status of the programme on 31 December 1992.

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